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

Deliverable 3.3

Design bases and safety analyses for system and components

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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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1. List of Acronyms

Abbreviation / Acronym	Description / meaning
AAC	Alternate Alternating Current source
ALARA	As Low As Reasonably Achievable
ALT	Alternativní prostředky (alternative components)
AOO	Anticipated operational occurrence
ASG	Emergency Feedwater System
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire (Nuclear Safety Authority)
BNI	Basic Nuclear Installation
BSR	Basic Safety Rules
BT	Bezpečnostní Třída (safety class)
CBSS	Cooling Basins with Sprinkler System
CHX	Compact Heat Exchanger
DBA	Design Basis Accident
DBC	Design Basic Conditions
DBE	Design Basis Earthquake
DBH	Design Basis external Hazards
DEC	Design Extension Conditions
DG	Diesel Generator
DiD	Defence in Depth
DIV	Diverzní prostředky (diverse components)
DPS	Diverse Protection System
DSA	Deterministic Safety Analysis
DUHS	Diverse Ultimate Heat Sink
EDF	Électricité de France
EIP	Important Elements for Protection
EOP	Emergency Operating Procedures
EPR	Evolutionary Power Reactor (or European Pressurized Reactor)
EPRI	Electric Power Research Institute
ESPN	Équipement Sous Pression Nucléaire (Nuclear Pressurized Equipment)

Abbreviation / Acronym	Description / meaning
ESWS	Essential Service Water System
HEPR	High Energy Pipe Break
HFE	Human Factors Engineering
HVAC	Heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ICRP	International Commission on Radiological Protection
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronic Engineers
INB	Installation Nucléaire de Base (Basic Nuclear Installation)
IRSN	Institut de radioprotection et de sûreté nucléaire (Radioprotection and Nuclear Safety Institute)
ISO	International Standard Organization
KTA	Kerntechnischer Ausschuss (Nuclear Safety Standards Commission)
LOOP	Loss of Offsite Power
LUHS	Loss of Ultimate Heat Sink
MDE	Maximum Design Earthquake
MSIS	Macro Seismic Intensity Scale
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OTS	Operating Technical Specifications
PAMS	Post-Accident Monitoring 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	Système de Borication de Sécurité (Emergency Boration System)
RCS	Reactor Coolant System
RFS	Règles Fondamentales de Sûreté (Fundamental Safety Rules)

Abbreviation / Acronym	Description / meaning
RHRS	Residual Heat Removal System
RHWG	Reactor Harmonization Working Group
RIS	Safety Injection System
SAM	Severe Accident Management
SBO	Station Black-out
SC	Safety Class
SFC	Single Failure Criterion
SFSP	Spent Fuel Storage Pool
SG	Steam Generator
SIE	Single Initiating Event
SL	Seismic Level
SSB	Systémy Související s Bezpečností (safety related systems)
SSC	Systems, Structures and Components
SUJB	Státní Úřad pro Jadernou Bezpečnost (State Office for Nuclear Safety)
TCS	Turbo Compressor System
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
VDA	Main Steam Atmospheric Dump System
VIV	Main Steam Isolation Valve
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 (Basic Safety Functions)

2. Executive Summary

Within the framework of the sCO2-4-NPP project, the consortium has planned to establish a roadmap to inform and prepare the regulatory aspects related to the developed sCO2 system. For this purpose, a multi-stage process has been initiated. After presenting the regulatory texts to be used as a basis in deliverable D3.1, the consortium has laid the regulatory basis for the licensing and acceptance of the modification of a nuclear power plant in deliverable D3.2. In this deliverable (D3.3), we present in detail the regulatory requirements for the development of the sCO2 system, analyzing the expectations from the design phase to the operation phase, including the expectations for qualification by the relevant authorities.

The purpose of this deliverable is to provide detailed requirements for the design and operation of the sCO2 system in Czech and French Nuclear Power Plants (NPP).

In chapter 3, the European standards regarding the main functions of the sCO2-4-NPP system (decay heat removal and heat transfer to the ultimate heat sink) are given, based on the IAEA documents. The RCC-M, AMSE and KTA standards are also considered. After that, the specific safety classification of the sCO2-4-NPP system in Czech and French legislation is presented.

Chapter 4 lists all the requirements for the design basis of Systems, Structures and Components (SSC), such as functions to be performed, internal and external hazards, reliability, monitoring and control, etc. In general, it is much easier to clearly define the design basis for the SSC in Czech Nuclear Power Plants than in French NPPs. Czech Republic is a smaller country with no seacoast and very similar meteorological conditions on the entire area. There are only two NPPs and one of them (Temelín NPP) was taken as the example for establishing the conditions for the sCO2 system. On the other hand, France is a much bigger country with varied meteorological and seismic conditions and the largest number of nuclear reactors in Europe (see chapter 3 in D3.2.). Therefore, the postulated loads and hazards differ depending on which site is considered. In most cases, the legislation and appropriate documents were described in the following chapters instead of specific values and parameters.

Chapter 5 deals with the requirements for the qualification of the SSC. It gives the idea on how to prove the system’s conformity with the requirements listed before (tests, safety analysis etc.).

In chapter 6, some of the most important requirements regarding the operation are presented.

3. sCO2 system safety classification

The purpose of this paragraph is to present different categories to which the sCO2 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 presented in deliverables D3.1 and D3.2 and will be discussed here in more detail.

3.1. System and component safety classification

From the D3.1, we assume that if we refer to the short list of requirements of the International Atomic Energy Agency (IAEA) presented in SSR-2/1, Rev. 1 [1], sCO2-4-NPP is a passive safety system for residual heat removal. The typical generic design requirements for systems are described in IAEA TECDOC-1787 [2]. IAEA SSR-2/1, Rev. 1 [1] document deals with considerations on the application of the IAEA safety requirements for the design of Nuclear Power Plants.

The list of typical generic design requirements for systems is shown in Table 1.

Table 1. Typical generic design requirements for systems (sCO2-4-NPP)

Typical generic design requirements for systems	Requirement applicability to safety category (SC 2 --> safety class 2; SC3 --> safety class 3)
Single failure criterion	Not required for SC2 and SC3
Physical & electrical separation	Yes, for redundant SC2 and SC3 equipment
Emergency power supply	Yes, for SC2 and SC3
Periodic tests	Yes, for SC2 and SC3
Protected against or designed to withstand hazard loads	Yes, for SC2 and SC3
Environmental qualification	Yes, for SC2 and SC3

The IAEA specific safety requirements [1] also concerns the requirements for residual heat removal[1] from the reactor core and heat transfer to the ultimate heat sink[2].

According to IAEA, residual heat removal systems are associated to reactor coolant system (RCS), therefore Section 3 of IAEA SSG-56 [3] is used as design basis. However, specific considerations in the design of associated systems for PWR technology in Section 6 of IAEA SSG-56 [3] further explains the recommendations for heat removal. The heat removal systems are described separately for each operational state. The plant

[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. "

state dependent heat removal systems for ensuring heat removal system are shown in Table 2. By IAEA definition [46] **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**. The system associated to reactor coolant system are called residual heat removal systems, which could be also passive (the KTA 3301 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 secondary side, what according to IAEA terminology used in [3] is 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 [2]. The importance of the barrier role is reflected by assigning the SSC to a barrier safety class.

Table 2. Plant state dependent heat removal systems for ensuring heat removal function [3]

Plant state dependent type of heat removal	Systems for ensuring heat removal function
Operational states	
Heat removal in power operation and hot shutdown modes	Main feedwater system and main steam system
Residual heat removal mode (e.g. hot shutdown conditions, cold shutdown conditions and refuelling conditions)	Residual heat removal system
Accident conditions (excluding design extension conditions with core melting)	
Residual heat removal in hot shutdown modes for design basis accidents	Emergency feedwater system and the steam dump to atmosphere system
Long term removal of residual heat in design basis accidents	Several redundant safety systems, each of which includes a pump and a heat exchanger with the intermediate cooling system (residual heat removal system takes suction water from the reactor coolant system and injects water back into the reactor coolant system after being cooled by the heat exchanger)
Residual heat removal in hot shutdown modes for design extension conditions without significant fuel degradation	Additional design provisions: • Extended autonomy of the emergency feedwater system, with on-site refilling capabilities; • Maintaining the capabilities of the emergency feedwater system and the operation of the steam dump valves to atmosphere in the event of prolonged station blackout; • Implementation of a secondary side passive heat removal system; • Removal of decay heat from the core by operating a primary feed and bleed strategy; • Implementation of a passive system for residual heat removal.

On the other hand, Section 4 of IAEA SSG-56 [3] deals with the ultimate heat sink and residual heat transfer systems (recommendations for safety features for design extension conditions are applicable to diverse ultimate heat sink and heat sink heat exchanger). As the requirements for the above two systems are different, they are treated separately (especially requirements 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 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 [3], "*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 [3] cooling system directly associated with the ultimate heat sink (open loop). Size and characteristics of the Diverse Ultimate Heatsink (DUHS) are crucial for the implementation and operation of the planned sCO2-4-NPP heat removal system in a Nuclear Power Plant. However, in the scope of the project is only design of the heat sink heat exchanger.

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

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 3 requirements for mechanical equipment are given, related to codes and standards like RCC-M or ASME Boiler and Pressure Vessel Code (BPVC), Section III or KTA standards (hereafter term 'ASME code' is used for ASME BPVC).

Table 3. Selection of code or standard with requirements for pressure retaining equipment (Table 18 of [2])

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

Following the IAEA Glossary [53] **the design basis is the range of conditions and events taken explicitly into account in the design of structures, systems and components and equipment of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits**. 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 [19], 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 [19] 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 blow-down from the steam generators.

3.2. Czech plants: Other classification

The specific requirements regarding the sCO2 system will base on the classification of the system parts. The general classification of SSC in a nuclear power plant according to Czech legislation was discussed in D.3.2. The specific division of sCO2 components is 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).**

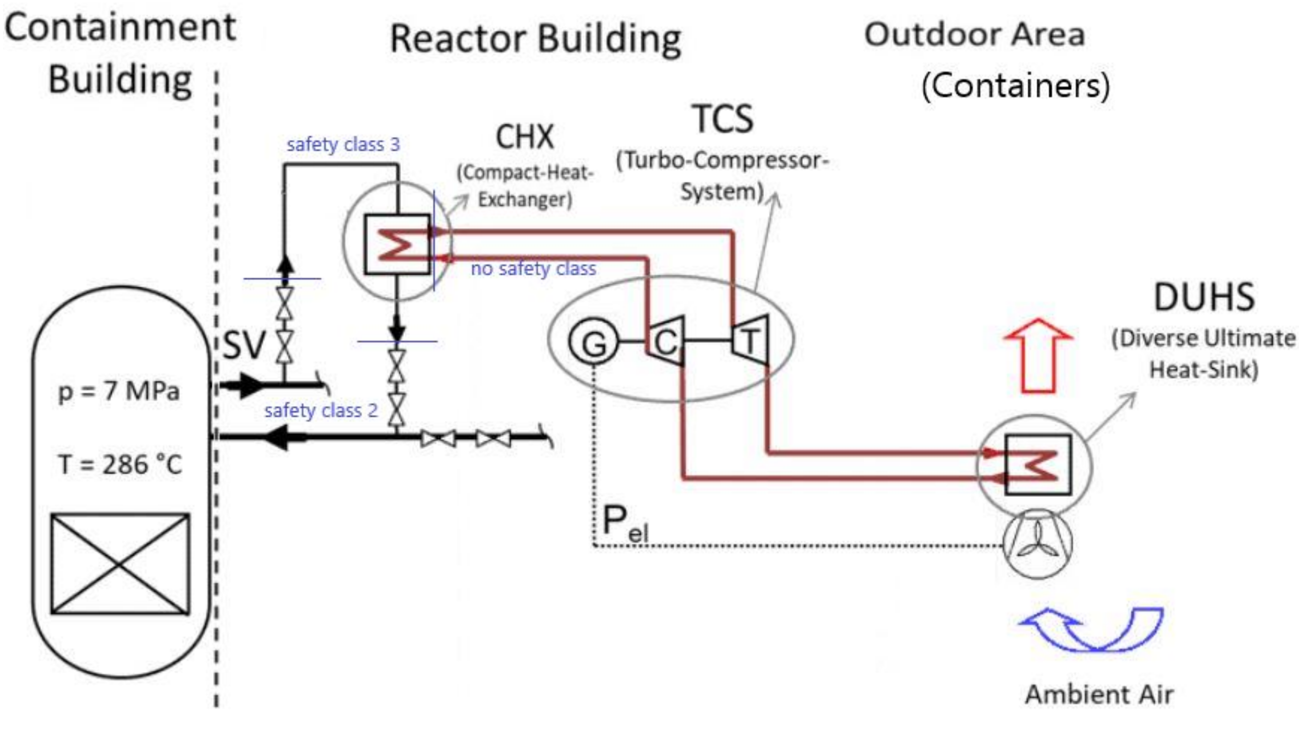

Figure 1. Scheme of sCO2 system [Venker 2015 [30]]

In Czech Atomic Act, selected equipment is defined as **system, structure, component or other part of a nuclear installation affecting nuclear safety and the performance of safety functions**.
The requirements for this kind of equipment are given in Decrees 358/2016, 329/2017, 132/2008.

Also, the whole sCO2 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 system is an innovative solution, different from those used previously in Czech NPPs. Therefore, the classification can vary 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 358/2016.

Regarding seismic classification, the whole sCO2 system belong 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.

3.3. French plants: Other classifications

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.

3.3.1. Mechanical quality classification

In addition to the initial design requirements detailed above, designers of equipment for nuclear power plants must meet mechanical quality levels. These mechanical quality levels are 3 (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 ASG systems (including downstream of pumps), safety injection system (RIS) and safety boration system (RBS) (including downstream of pumps): these equipment respect 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.

3.3.2. Pressurized equipment classification

Pressure equipment is subject to the provisions of Chapter VII of Title V of Book V of the Environmental Code [22], which incorporates the principles of the "New European Approach". New equipment must therefore be designed and manufactured by its manufacturer in compliance with the essential safety requirements set by the regulations and be subject to a conformity assessment by a body authorized by the ASN.

These provisions are supplemented by requirements applicable to in-service monitoring of equipment, which are set out in section 14 of Chapter VII of Title V, Book V of the Environmental Code [22].

Pressure equipment specially designed for NPPs known as "nuclear pressure equipment" (NPE) is subject to both the INB order and pressure equipment regimes. Specific decrees specify, for this equipment, the provisions defined by the environmental code. The decree in force is the decree of December 30, 2015 relating to ESPN, amended by the decree of September 3, 2018 amending certain provisions applicable to nuclear pressure equipment and certain safety accessories intended for their protection.

4. Requirements for design basis

The design and construction of the sCO2 system must meet certain requirements and regulations. In this paragraph, we discuss these processes and regulations for France and for Czech Republic, according to the IAEA SSG-56 recommendations.

To satisfy the IAEA requirements for design [1, 3], the sCO2-4-NPP should be designed to fulfil the following requested functions, as specified in Paragraph 2.1 (d) of IAEA SSG-56 [3].

4.1. Functions to be performed by the system

The sCO2 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 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.

4.2. Postulated initiating events that the system must cope with

The Horizon 2020 project sCO2-HeRo develops and proves the concept of a heat removal backup technology based on sCO2 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-systems, attached to the existing heat removal system, to remove decay heat from the reactor.

A station blackout (SBO) [5] is defined as a plant condition with complete loss of all alternate current (AC) power from off-site sources, from the main generator and from standby AC power sources important to safety to the essential and nonessential switchgear buses.

A loss of normal access to the ultimate heat sink [5] involves the loss of ability to provide a forced flow of water to key plant systems.

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.

4.3. Loads and load combinations the system is expected to withstand

4.3.1. Overview

The sCO2 system and its components must be able to handle various loads and load combinations, according to the procedures that will be established for its operation. Part of the manufacturers' job will therefore be to design components capable of withstanding these combinations. In this paragraph, we will try to discuss the expectations for the French and Czech plants.

From sCO2-4-NPP deliverable D3.1, we can consider that

- Requirement 17 of IAEA SSR-2/1, Rev. 1 [1] for internal and external hazards also requires 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.
- IAEA SSG-56 guide [3] in paragraph 3.78 explains that loads should be identified and analyzed. The type of load and the timing of each load is taken into account. These types of loads are static and permanent loads, or dynamic transients, global or local. The timing of each load should be considered to avoid the unrealistic superposition of load peaks if such peaks cannot occur coincidently.
- IAEA SSG-56 guide [3] in paragraph 3.79 explains that design basis loading conditions, including internal and external hazard loads, should be assigned to different categories that correspond to different plants states or service conditions according to their estimated frequency of occurrence or in accordance with the requirements of accepted codes and national regulations.

The categories of service conditions to consider are (for detailed definitions refer to IAEA SSG-56 [3]):

- Normal service conditions (normal operation)
- Upset conditions (anticipated operational occurrences)
- Emergency conditions (accidents of low frequency)
- Faulted conditions (accidents of very low frequency)

IAEA SSG-56 guide [3] in paragraph 3.80 explains that appropriate acceptance criteria (e.g. design pressure and temperature, and stress limits) to be met for ensuring integrity should be defined and should be appropriate to each load combination, with account taken of the load combination category. IAEA SSG-56 guide [3] in paragraph 3.82 states that meeting the criteria given by internationally recognized codes and standards provides reasonable assurance that structures, systems and components are capable of performing their intended functions.

IAEA SSG-56 guide [3] in paragraph 3.86 states:

"Structures, systems and components that are designed to fulfil their functions in emergency conditions and faulted conditions should be designed to meet adequate service limits, to ensure the necessary integrity and operability of these items while subjected to sustained loads resulting from the occurrence of the postulated initiating events for which they are designed to respond."

4.3.2. Requirements for Czech Plants

In the potential location of sCO2 Compact Heat Exchanger in Temelín NPP, the normal operational conditions are:
Temperature: 35°C
Humidity: 60%
Pressure: 0.1 MPa

The design loads are:
Temperature: 110°C
Humidity: steam-gas mixture
Pressure: 0.12 MPa

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. 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.

High winds
Assessment of the load is based on the measured annual maximum values of wind speed. The values determined on the basis of measurements carried out at the Prague-Ruzyně station, i.e. 49 m/s for a frequency of once in 100 years and 68 m/s for a frequency of once in 10000 years, were taken as the input value for determination of the wind load.

Heavy snowfall and ice
Snow load is expressed in terms of the water equivalent of snow, i.e. the height of the corresponding water column in mm. The input values for snow load and water precipitation are: 92 mm for a frequency of once in 100 years and 157 mm for a frequency of once in 10000 years.

Maximum and minimum temperatures
The effects of outdoor temperatures are assessed on the basis of the measurements of the outdoor air temperatures in the Temelín, Tábor and České Budějovice stations. As a conservative outcome value, the value determined from measurements in Tábor station was used. The input values for the assessment of load due to temperatures were following:

- 39.0 °C as the maximum temperature of air and -32.3 °C for the minimum air temperature (frequency of once in 100 years) and
- 45.6 °C as the maximum air temperature and -45.9 °C as the minimum air temperature (frequency of once in 10000 years).

Table 4. Design loads from external hazards for Temelín NPP

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

Characteristics of the design basis earthquake (DBE)

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 and NS-G-1.6),
- DE (Design Earthquake, referred to also as SL-1 earthquake according to the IAEA Safety Standards Series No. NS-G-3.3 and NS-G-1.6).

Table 5. Seismic levels for Temelín NPP

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, section 2.6) for minimum value of acceleration in horizontal direction PGA_{hor} = 0.1 g.

4.3.3. Requirements for French Plants

4.3.3.1. Mechanical equipment

The French regulations concerning the loads and load combinations that the sCO2 system must meet are presented in the RRC-M [23] for the mechanical equipment of the system.

The guide allows to determine, for the system:

Table 6. Summary for Class 2 Mechanical Equipment (heat recovery heat exchanger, turbo-machine)

Settings	Section
Type of loads	RCC-M, Section I, Subsection C, Table C 3133
Definition of operating conditions	RCC-M, Section I, Subsection C, C 3120
Loading rules	RCC-M, Section I, Subsection C, C 3130
Levels of criteria	RCC-M, Section I, Subsection C, C 3140
Minimum criteria levels	RCC-M, Section I, Subsection C, C 3150
Requirements for stress report	RCC-M, Section I, Subsection C, C3160
Specials considerations (corrosion, cladding, …)	RCC-M, Section I, Subsection C, C3170

Table 7. Summary for Class 3 Equiment

Settings	Section
Design Rules	RCC-M, Section I, Subsection D, D3100
Pump design	RCC-M, Section I, Subsection D, D3400
Piping design	RCC-M, Section I, Subsection D, D3600
Valves design	RCC-M, Section I, Subsection D, D3500
Overpressure protection	RCC-M, Section I, Subsection D, D6000

4.3.3.2. Electrical equipment

For the electrical equipment (I&C included), manufacturers will have to apply RRC-E [24]. The RCC-E comprises a set of technical rules to be applied and implemented by a contractor, manufacturer or supplier in the design and construction of electrical equipment.

It is the responsibility of the prime contractor and the contractor to define the list of electrical equipment and systems to be produced in accordance with the RCC-E. As a minimum, this list must include all safety classified electrical systems and equipment.

The requirements of the code will be referred to in the contractual documents of Plant operator or main suppliers, especially in the technical specifications of electrical and I&C equipment.

As the electrical equipment for the sCO2 system are not finished to be define, it is difficult to provide the good sections and sub-sections needed for the partner. But we can, already, highlight the following sub-sections:

- Volume III: Automation and Control Systems
  - Sub-sections 3000 and 4000: Hardware aspects for Class 1, 2 and 3 systems
  - Sub-sections 5000 and 6000: Software aspects for Class 1, 2 and 3 computer-based systems

This part is a supplement to the standard IEC 60880 regarding software aspects for computer-based systems carrying out category A functions. It retains the structure of IEC 60880 and brings precisions/clarifications or adds requirements/ recommendations when necessary.

- Volume V: Equipment Engineering
  - Sub-sections 2000 and 3000: Qualification and qualification procedures

4.4. Protection against the effects of internal hazards

4.4.1. Overview

Requirement 17 of IAEA SSR-2/1, Rev. 1 [1] for internal and external hazards is:

> "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."

Items important to safety shall be designed and located, with due consideration of other implications for safety, to withstand the effects of hazards or to be protected (this protection should also consider the consequences of the effects of the failure of non-protected structures, systems and components on protected structures, systems and components), in accordance with their importance to safety, against hazards and against common cause failures generated by hazards.

The design methods, as well as the design and construction codes and standards used, should provide adequate margins to avoid cliff edge effects in the event of an increase in the severity of the internal hazards.

Paragraph 5.16 of IAEA SSR-2/1, Rev. 1 [1] states that 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 guide [3] provide recommendations on meeting Requirement 17 and paragraph 5.16 of SSR-2/1 (Rev. 1) [1]. Structures, systems and components important to safety should be protected against the effects of internal hazards. This protection should also consider the consequences of the effects of the failure of non-protected structures, systems and components on protected structures, systems and components. The plant layout and the means for protection of the redundancy provisions of the safety systems should be adequate to provide assurance that an internal hazard cannot represent a common cause failure for the total loss of a safety function to be fulfilled by the reactor coolant system and associated systems. Finally, the design methods, as well as the design and construction codes and standards used, should provide adequate margins to avoid cliff edge effects in the event of an increase in the severity of the internal hazards.

4.4.2. Requirements for Czech Plants

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).
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.

The Temelín NPP design considers following internal hazards (Decree 329/2017, IAEA NS-G-1.7, IAEA NS-G-1.11):

- internal fires,
- internal explosions,
- internal floods,
- missiles generated by pipe ruptures,
- falling objects,
- collapse of structures,
- pipe failures,
- jet effect.

4.4.3. Requirements for French Plants

The sCO2 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.

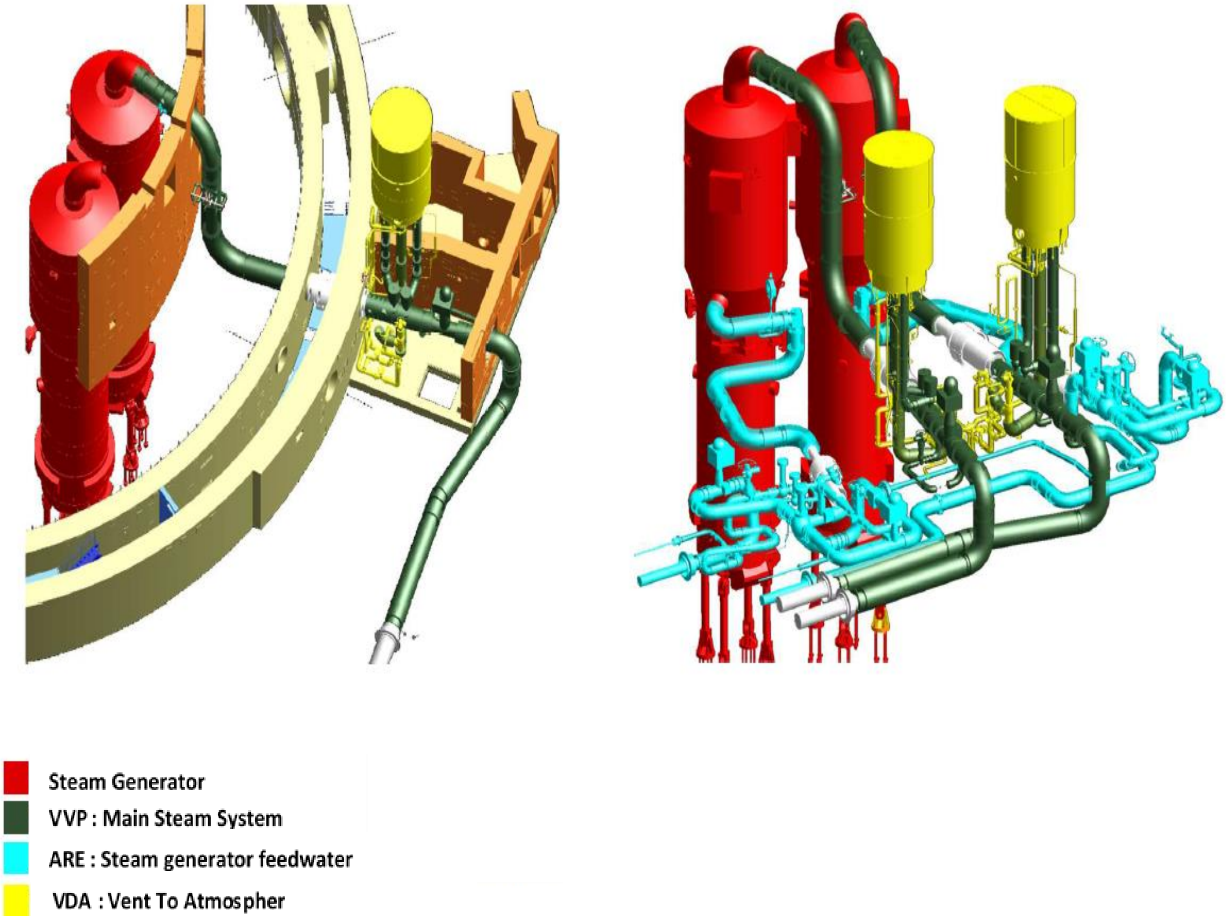

Figure 2. Secondary Loop

In addition, it should be noted that:

- the main piping of the VVP (Vapeur Vive Principale - Main Steam System) system of a power plant is considered high energy, however the concept of rupture exclusion applies here. If the sCO2 system is placed on this piping, the rupture of the main piping of the VVP system will therefore not be considered as an aggressor of the sCO2 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 VIV bypass line or the VDA – Vent To Atmosphere- system. The complete analysis will have to be carried out during the HEPR studies.
- the sCO2 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 High Energy Piping Rupture (HEPR) studies.

Rules can be found in the RCC-CW guide [25]. The next table summarizes the paragraph to consider:

Table 8. Summary for Civil Engineering

Settings	Section
Standards Referred	RCC-CW, GREFD
Design Requirements	RCC-CW, DGENR 2000
Characteristics or Design Values (permanent, accidental…)	RCC-CW, DGENR 3310 RCC-CW, DGENR 3330
Safety requirements for nuclear island building excluding Reactor building	RCC-CW, Table DG-2

4.5. Protection against the effects of external hazards

4.5.1. Overview

Protection against the effects of external hazards will depend greatly on the location of the various modules of the sCO2 system. The buildings where the modules will be installed will have to be able to meet the necessary protection regulations. Similarly, if a temporary installation is envisaged, it will have to meet certain criteria.

Paragraph 5.21 of IAEA SSR-2/1, Rev. 1 [1] states that the design of the plant shall provide for an adequate margin to protect equipment important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site, and to avoid cliff edge effects.

Paragraphs 3.19–3.26 of IAEA SSG-56 guide [3] provide recommendations on meeting Requirement 17 and paragraphs 5.17–5.21A of SSR-2/1 (Rev. 1) [1]. With regard to the effects of external hazards, protection should be applied to the extent possible to prevent damage to the reactor coolant system and associated systems that are important to safety. Protection can rely on an adequate layout of the plant and on protection measures for the buildings at the site. When protection measures are not effective, structures, systems and components should be designed to withstand the hazard loads and the loads associated with likely combinations of hazards.

4.5.2. Requirements for Czech Plants

The requirements regarding the protection against external hazards base on the IAEA DS498: External Events Excluding Earthquakes in the Design of Nuclear Installations.

External influences of natural origin include, for example, geological influences (e.g. seismicity), meteorological influences (wind, snow, temperature extremes, heavy rains) and hydrological influences.

External influences of natural origin must be considered as an integral part of the safety assessment of the power plant. Risks associated with external influences must be minimized for all NPP conditions. The design must consider the extreme values of loads of natural origin.

All external influences that may occur at the site must be identified and analyzed, including the associated risks.

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.

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 apropriate 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.

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.

Individual hazards are assessed in terms of their influence on nuclear safety, radiation protection, technical safety, monitoring of the radiation situation, management of radiation emergency and security during the life cycle of a nuclear power plant. It is a qualitative evaluation criterion, which indicates the importance of the property and the finding from its assessment for the location, construction, but also the operation of Nuclear Power Plants.

4.5.3. Requirements for French Plants

As a system participating in the mitigation of DEC operating conditions, the sCO2 system 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 D3.2).

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 analyzed at a later stage of the design (depending on the plant).

Requirements for the building could be find in the RCC-CW [25], Table 8 presents the different sections to apply.

4.6. Design limits and acceptance criteria applicable to the design of SSC

4.6.1. Overview

The limit acceptance criteria for the sCO2 system will have to be determined from the power plant where the system is installed (e.g. the criteria will not be the same for two power plants of different power or architecture). Nevertheless, it is possible to determine some general rules that manufacturers will have to follow in the further development of the sCO2 system.

Requirement 15 of IAEA SSR-2/1, Rev. 1 [1], entitled Design limits, is:

> "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."

Paragraph 5.4 of IAEA SSR-2/1, Rev. 1 [1] requires that 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.

Requirement 28 of IAEA SSR-2/1, Rev. 1 [1], entitled Operational limits and conditions for safe operation, is:

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

Paragraph 3.44 of IAEA SSG-56 guide [3] provides recommendations on meeting Requirements 15 and 28 of SSR-2/1 (Rev. 1) [1]. The performance of the reactor coolant system and associated systems should be specified to meet a 'well defined and accepted' (accepted by member state regulatory bodies or proposed by international organizations) set of design limits and criteria. Design limits and criteria are required to be specified for each plant state.

In accordance with IAEA-TECDOC-1791 [5] 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.

4.6.2. Requirements for Czech Plants

The design limit is defined in Decree 329/2017:
Design limit means an acceptability criterion that is used to evaluate the ability of a nuclear installation or a system, structure or component thereof to perform its function assumed by the nuclear installation design; a design limit is especially a limit stipulated by legislation or an acceptability criterion derived from it that corresponds to the method used to evaluate the capability of a nuclear installation to perform its function assumed by the nuclear installation design.

The acceptance criteria generally regard:

- the temperature and pressures inside the containment vessel,
- tightness of the containment vessel and
- tolerable deformation of the containment vessel structure.

The acceptance criteria for different NPP states were already discussed in D3.2.

The design limits for the sCO2 system will be derived from the final detailed design of the system. After the safety analysis prove 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.

4.6.3. Requirements for French plants

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 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.

4.7. Reliability

4.7.1. Overview

Paragraphs 3.47–3.56 of SSG-56 [3] provide recommendations on meeting Requirements 21–26, 29 and 30 of SSR-2/1 (Rev. 1) [1].

To achieve the necessary reliability to remove residual heat from the core and to transfer residual heat to the ultimate heat sink, the following factors should be considered:

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

4.7.2. Requirements for Czech Plants

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. Their backup is not directly required in Czech legislation, but in some cases, backup is applied within the design basis (due to availability, accessibility, actuation time etc.). The possibility of substitutability is envisaged for mobile devices. Their backup is not required, but it must be possible to replace it with another available mobile means (e.g. a means from a neighboring unaffected unit).

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.

4.7.3. Requirements for French Plants

ASN Guide No. 22 [22] 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 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 CO2 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).

4.8. Provisions against common cause failures within a system and between systems belonging to different levels of defense in depth

As indicated in report D3.2, the sCO2 system will have to be integrated into the defense-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.

Requirement 24 of IAEA SSR-2/1, Rev. 1 [1] entitled Common cause failures requires:

"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."

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. No specific guide is given in SSG-56 [3].

4.9. Safety classification

4.9.1. Overview

The implementation of the concept of defense in depth leads in particular to the establishment of classifications of mechanical equipment, electrical systems, structures and civil engineering works that provide a rational basis for determining the severity of the requirements to be met for them from their design to their operation.

Classifications based on several principles may be defined a priori, depending on the nature of the equipment concerned, and on the nature of the requirements that will be deduced from these classifications.

4.9.2. Requirements for Czech plants

Safety function of sCO2-4-NPP is to remove decay heat from the core and to transfer residual heat from the reactor coolant system to the ultimate heat sink in accident conditions.

According to Czech legislation (decree 329/2017),

“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.”

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.

Selected equipment performing Category II safety functions, which are safety functions with the highest reliability requirements, shall be classified as **safety class 2**. In the case of nuclear installations with a nuclear reactor, such equipment includes selected equipment performing the passive function of a system, structure or component that is a physical barrier and selected equipment with guaranteed high reliability in performing active safety functions of the safety systems. These SSC 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.

Selected equipment not classified in safety class 1 or 2 performing Category III safety functions, which are safety functions that can be substituted by design measures while achieving the same safety objective shall be classified as **safety class 3**. In the case of nuclear installations with a nuclear reactor, these include safety functions for the removal of heat from safety systems into the surrounding environment.

4.9.3. Requirements for French plants

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 system will have to be classified as safety equipment.

The different safety classes are defined as follows for mechanical equipment carrying a pressurized fluid:

- Safety class 1.

Safety class 1 includes the equipment which forms the pressure boundary of the primary fluid cooling the reactor and whose failure, during normal operation, prevents the reactor from being shut down by the normal systems, i.e. equipment subject to pressure from the main reactor cooling system up to and including the safe isolation devices located on the pipes connected to it and which have an equivalent diameter greater than that which limits any leakage to a value that can be compensated for by the reactor auxiliaries used in normal operation.

- Safety class 2.

Belong to safety class 2:

  - the materials of the primary coolant jacket which are not of safety class 1;
  - the materials of the systems or parts of systems necessary to contain radioactivity in the event of an accident;
  - the materials of the systems or parts of systems necessary to introduce anti-reactivity into the reactor under accident conditions;
  - the equipment which ensures the maintenance of the inventory of emergency coolant;
  - 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;
    - the equipment of the secondary water and steam circuits from the steam generators up to and including their isolation devices.
- Safety class 3:
    - Safety class 3 includes pressure vessels, which, according to the rules defined above, do not belong to safety classes 1 or 2.

In the case of our sCO2 system, we can define that the mechanical components of the sCO2 system belong to safety class 2 by virtue of their role in extracting heat in accidental situations. In report D3.2, we established a first classification in category S3, 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, we recommend continuing the development of the components by applying the most conservative approach: a class 2 for the most critical components (secondary loop/sCO2 exchangers and the turbocompressor). As the piping is classified as HEPR, the associated regulations will have to be applied to them.

4.10. Environmental conditions for qualification

4.10.1. Overview

The sCO2 system components 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. Environmental qualification should be carried out by means of testing, analysis and the use of experience, or through a combination of these. Environmental qualification should include the consideration of such factors as temperature, pressure, humidity and radiation levels.

Requirement 30 of IAEA SSR-2/1, Rev. 1 [1] entitled Qualification of items important to safety requires:

"*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.*"

The environmental conditions shall include the variations in ambient environmental conditions. The qualification programme 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.

These conditions will have to be studied during the equipment qualification process.

4.10.2. Requirements for Czech Plants

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.

Input data for climatic parameters with a return period of 100 years and 10000 years are determined from the statistical processing of annual extremes of values -of relevant meteorological variables, measured over a period of at least 30 years at the Temelín NPP site.
The return period of 10000 years is based on Decree 329/2017 and WENRA Issue T: Natural hazards. The return period of 100 years is based on the expected maximum lifetime of the power plant. For equipment not important for nuclear safety, the use of design standards with a return period of 50 years is permitted. The measurement requirement for at least thirty years is based on IAEA document SSG-18 [36].
The methods of statistical processing are based on the IAEA document SSG-18 Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations from 2011 [36]. The loads of individual hazards were determined from the values of meteorological parameters, using an appropriate statistical distribution (Gumbel) in accordance with the recommendations of the IAEA guidelines.
Meteorological stations are located in the Temelín NPP site, where data has been collected since 1989. The data listed in the table were determined in 2015. The input meteorological data were taken over from The Czech Hydrometeorological Institute from suitable stations in the vicinity of the plant with a sufficiently long recording time.
The determination of seismic hazard of the NPP was prepared in accordance with IAEA standards NS-R-3 rev.1 [33], SSR-1 [35], SSR-2/1 rev.1 [1] and SSG-9 [34], using the Probabilistic Seismic Hazard Assessment (PSHA) approach.

Table 9. Meteorological hazards for Temelín NPP

Meteorological hazard / Parameter	Expected frequency 1/100y		Expected freq. 1/ 10 000y	
	Value	Load	Value	Load
Wind speed **Peak value** **10 sec. measurement** **10 min. measurement** **Basic wind load** **Basic dynamic wind load**	48.00 m/s 38.90 m/s 26.80 m/s	0.94 kN/m² 0.45 kN/m²	65.00 m/s 52.70 m/s 36.30 m/s	1.73 kN/m² 0.82 kN/m²
Snow (equivalent water column)	109.00 mm	1.09 kN/m²	189.00 mm	1.89 kN/m²
Precipitation (water column; 24h)	105.00 mm		180.00 mm	
Maximum temperature **Peak value** **6 h measurement** **24 h measurement** **7 days measurement**	42.00 °C 38.50 °C 31.60 °C 27.70 °C		52.00 °C 46.20 °C 38.80 °C 34.50 °C	

Meteorological hazard / Parameter	Expected frequency 1/100y		Expected freq. 1/ 10 000y	
	Value	Load	Value	Load
Minimum temperature				
Peak value	-35.60 °C		-47.00 °C	
6 h measurement	-30.40 °C		-46.40 °C	
24 h measurement	-24.30 °C		-37.30 °C	
7 days measurement	-20.40 °C		-33.10 °C	
Seismicity	SL-1		SL-2	
Horizontal acceleration	0.05g		0.1 g	

4.10.3. Requirements for French Plants

ASN Guide No. 22 [20] 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. We discuss these provisions in chapter 5.

4.11. Monitoring and control capabilities

4.11.1. Overview

Monitoring and control of the sCO2 system will be carried out by the Instrumentation and Control (I&C) architecture, which consists of several sub-systems and their associated electrical and electronic equipment. The overall design of the I&C architecture and its associated equipment must comply with process, nuclear safety, and operational requirements.

Paragraph 4.2 of SSR-2/1 (Rev. 1) [1] states: “Means of monitoring the status of the plant shall be provided for ensuring that the required safety functions are fulfilled.”

IAEA SSG-56 guide [3] in paragraphs 3.133–3.136 deals with instrumentation. Paragraph 3.133 of IAEA SSG-56 guide [3] states:

"The systems should be provided with adequate instrumentation for the following purposes:

a) Monitoring of the process parameters (e.g. pressure, temperature, water level and flow rate) that indicate whether the system or component is being operated within the range specified for its normal operation;
b) Early detection of abnormal operating conditions;
c) Automatic operation of systems necessary for the mitigation of the consequences of an accident;
d) Providing the main control room and the technical support centre with appropriate and reliable information for accident management;
e) Periodic testing of systems and components;
f) Supporting an understanding of the maintenance state of structures, systems and components."

4.11.2. Requirements for Czech Plants

Control and management system regards the SSC used for measuring, evaluating and displaying the nuclear installation parameters for the needs of the nuclear installation operators and for nuclear installation control, including initiation and management of interventions necessary to ensure nuclear safety, radiation protection, radiological emergency management and security.

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).

PRPS - Primary Reactor Protection System

The task of the PRPS system is to prevent core damage and to ensure the integrity of the containment by using the following functions:

- Rapid shutdown of the reactor in case of emergency,
- Safety systems actuation,
- other important control functions, including technical means for monitoring plant parameters and operator control from the control room.

PCS - Plant Control System

The control of operating states is enabled by the PCS. It provides non-essential protection and control functions for the management of primary and secondary systems.

PAMS - Post Accident Monitoring System

The PAMS is used for post-accident monitoring functions.

- This system, which is one of the important control systems, provides monitoring according to the requirements of NRC Regulatory Guide 1.97.
- The purpose of this system is to provide the operator with important information necessary in emergencies to decide on subsequent manual interventions, on the operation of systems, the

condition of the unit, the performance of basic safety functions and the radiation situation; the system also records selected parameters.

- It is implemented in two divisions, so that resistance to a single failure is ensured to an extent sufficient for the monitoring system. The system provides information from all three safety systems divisions. The backup method of monitoring is enabled by the PRPS and DPS,
- The PAMS provides information to the operator in control room via its displays, as well as by recording for permanent storage of the course of parameters and data transfer to the I&C system.

DPS - Diverse Protection System

In case of failure of PRPS, the Diverse Protection System:

- provides automatic protection in response to PIEs, to prevent reactor core damage and to maintain the integrity of the primary circuit and the integrity of the containment.
- enables operating personnel to safely monitor and manage the emergency conditions of the power plant (achieving stable conditions and subsequently a safe state), including the necessary protective actions in case of AOO.

Non-essential protection and control systems provide automatic and manual control functions (for SSC classified as SSB, DIV, ALT and VyDiD):

- for the operation of SSC during normal operation and prevention of abnormal or emergency conditions,
- in the event of abnormal operation or emergency conditions, providing support functions (technological support interventions, power supply, air conditioning, etc.) and initiating the automatic actuation of safety-related and support systems,
- ensuring safe shutdown of the unit (hot state) and conversion to a cold state,
- providing monitoring and measurements for maintenance and prevention purposes,
- providing communication functions for the transmission of information between operational staff,
- providing additional functions for SSC classified as DIV and ALT (e.g. AAC DG control, mobile monitoring and control means).

Non-essential protection and control systems must have acceptable functional reliability, the level of which results from the set of functions provided. In order to achieve this reliability, the systems are designed with varying degrees of redundancy and independence. In particular, it should be ensured that no failure of one component, which can be expected during normal operation, leads to abnormal operation or emergency conditions and does not require shutdown of the unit.

The sCO2 system will have to be equipped with such control means to meet these objectives.

4.11.3. Requirements for French Plants

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 [22].

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.

The instrumentation shall provide the necessary information for:

- apply procedures or operating guides;
- make decisions concerning the management of events in the reference design basis and the extended design basis.

In particular, the instrumentation must make it possible to make a decision on a possible breakthrough of the vessel and on the presence of hydrogen in the containment.

Requirements about the I&C and monitoring equipment are defined in the RCC-E guide (cf. 4.3.3.2). These requirements depend mainly on the safety class of the components and the system to which the I&C will be connected. The following table summarizes these rules according to the safety class.

Table 10. Summary for I&C equipment

Equipment Safety Class	Classification requirements applicable to I&C Components

Class 2	• RCC-E and relevant standards (e.g. IEC standards).
	  ○ RCC-E, Volume III, Subsection III-3000
	  ○ RCC-E, Volume III, Subsection III-6000
	• Quality Assurance Program must be applied to the overall life cycle activities of the system.
	• Qualification for operating conditions.

Class 3	• RCC-E and relevant standards (e.g. IEC standards)
	  ○ RCC-E, Volume III, Subsection III-4000
	  ○ RCC-E, Volume III, Subsection III-6000
	• Quality Assurance Program must be applied to the overall life cycle activities of the system.
	• Qualification for operating conditions.

4.12. Materials

4.12.1. Overview

The materials used for the sCO2 system must be chosen taking into account their chemical composition and the phenomena to which they are likely to be subjected in order to limit the risks related to the operation of the system, to the use of CO2 in the supercritical state, and to the pressures and temperatures to which they will be subjected.

4.12.2. Requirements for Czech Plants

The specific requirements for selected equipment and pressure equipment in the NPP are given in Decree 358/2016.

Pressure equipment materials

- 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.
- Basic and auxiliary materials used must be suitable for the given use for the entire expected service life of the pressure equipment.

Materials of parts of pressure equipment exposed to pressure

- The basic materials affecting the technical safety of pressure equipment must meet the requirements of technical specifications for pressure equipment, both alone and in a structure together with suitable auxiliary material, especially requirements for suitable properties under all operating conditions under which the pressure equipment is to perform its function.
- Parts of selected equipment exposed to pressure are always considered to be parts that make up a pressure interface or that are joined to these parts in a permanent manner.
- 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:
    - the same as the material of the original part listed in the technical specifications of the pressure equipment;
    - on the list of materials permitted for the given use; or
    - another material, if the above cannot be used.
- 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, an authorized person must arrange the specific assessment of the proposed material.
- Suitable measures must be implemented during manufacture, installation, repair, or modification to ensure that the material used complies with the requirements of the technical specifications for the pressure equipment. In particular, there must be documentation for all basic and auxiliary materials used available confirming the compliance of the materials used with the material's technical specifications.
- Only material that has undergone an assessment of its conformity with technical requirements for material may be used to manufacture, repair, or modify pressure equipment.
- The material assessment must be certified in terms of its compliance with technical material specifications.

4.12.3. Requirements for French Plants

General material provisions are covered in **chapters 2000 of the various sub-sections of RCC-M, Section I [23]**.

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 codes concern essentially the use of complementary analyses and additional restrictions, which are required in order to improve the following properties:

- Inter-granular corrosion resistance,
- limitation on carbon content,
- increase of chromium minimum content,
- control of delta ferrite content,
- product toughness (limitation on S, P and Si content): a minimum KV notch impact energy is required by the RCC-M, which necessitates low inclusions content,
- weldability of stainless steels, through a limitation of Boron content.

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.

Additionally, Charpy KV tests are also specified for stainless steels. As the RCC-M is dedicated to specific applications. It includes provisions, which would be expected to be specified, in the US, by contractors in equipment specifications. In particular, RCC-M it is the only code where product procurement specifications are accompanied by dedicated specifications for parts: a precise correspondence between the parts and the applicable specifications is given in the Chapter 2000 tables of the applicable sub-sections of the section I.

4.13. Provisions for testing, inspection, maintenance and decommissioning

4.13.1. Overview

In order to guarantee an adequate level of reliability during reactor operation, the sCO2 system shall be maintained under suitable conditions in order to be available and ready to operate correctly.

This implies being able to determine the periodic tests, preventive maintenance operations (and decommissioning if necessary) to be set up for the sCO2 system.

In the state of development of the system, it is difficult to determine the maintenance operations that will depend on the characteristics of the final equipment. However, it is possible to determine whether periodic testing or inspections will be required.

4.13.2. Requirements for Czech Plants

The main requirements are given in Decree 358/2016.

Final assessment of pressure equipment must include

1) a final test

- During the final test, a visual inspection of the pressure equipment and a check of the accompanying technical documentation for the selected equipment must be performed to assess whether the selected equipment and related quality assurance records are in mutual accord and comply with all requirements stipulated in technical documentation or compliance verification documentation.
- Checks performed during the manufacturing of the pressure equipment can also be taken into account in performing the final test.
- During the final test, every part of the pressure equipment must be visually inspected internally as well as externally in terms of technical safety, if necessary. If this inspection cannot be arranged during the final test, especially in cases where the nature of the pressure equipment makes inspection of its individual parts impossible without the need for disassembly, this inspection can be performed during checking operations preceding the final test, and the final test will simply involve a check of the accompanying technical documentation.
- The final test must primarily verify the following:
    - identification markings on the pressure equipment, including information on equipment labels and information embossed on pressure parts and markings on materials, castings, and intermediate products;
    - the main dimensions of the equipment, the location of orifices, access holes, gear, feet, supports, and the assembly of individual parts according to drawings;
    - the results of checks of welded joints via an external or internal inspection, including the results of prescribed checks during the performance of special processes, welders’ marks, welding supervision records, thermal treatment records, and materials certification of materials and intermediate products used; and
    - compliance with welders' marks placed on selected equipment with lists of welders, identifying their qualifications.

2) a pressure test, a tightness test, or other equivalent check;

- During a pressure test or tightness test, it must be verified that the pressure equipment does not exhibit significant deformation or leakage exceeding stipulated acceptability criteria.
- If a pressure test or tightness test is unsuitable for or impossible to perform on the given pressure equipment, other equivalent checks must be performed that can be used to verify the strength and tightness of the pressure equipment.
- A pressure test and tightness test must be performed using hydraulic pressure prescribed in technical specifications for the given pressure equipment. The test pressure must be stipulated in relation to the calculated or highest permitted pressure, taking into account evaluation of geometric and material

properties and test conditions during manufacturing and operation in accordance with requirements specified in technical regulations or technical specifications for the manufacture of pressure equipment.

3) a check of safety equipment and equipment ensuring the functionality of pressure equipment;

- for safety equipment (Point 12 of Part A of Annex 1, Decree 358/2016)
- for electrical equipment (Point 13)
- for hydraulic and pneumatic equipment providing control, regulation, signaling and measurement during operation (Point 15)

Supervision by an authorized person

- Supervision 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.
- A manufacturer, importer, or person installing selected equipment after manufacture must grant the authorized person access to manufacturing, inspection, test, and storage areas in order to perform supervision, and provide him with all needed information.
- For purposes of supervision, an authorized person has implemented a system of checks that specifies the type and frequency of checks performed on the premises of the manufacturer, importer, or person installing selected equipment after manufacture.
- Within the scope of supervision, an authorized person must perform regular checks to make sure that the manufacturer, importer, or person installing selected equipment after manufacture maintains and applies the management system as it was approved. He must choose the frequency of regular checks so that a new complete audit takes place at least once every 12 months.
- Within the scope of supervision, an authorized person must perform unannounced checks on the premises of a manufacturer, importer, or person installing selected equipment after manufacture. The authorized person must stipulate the type and frequency of unannounced checks primarily taking into account:
    - the safety class of the selected equipment;
    - the results of previous checks performed as part of supervision;
    - the need to monitor adherence to corrective measures; and
    - significant changes in the organization of manufacturing or manufacturing concept or technology.
- During these checks, the authorized person may perform checks (or have them performed) to verify whether the management system is working correctly.
- Based on performed checks, an authorized person must create reports on the results of supervision and give them to the manufacturer, importer, or person installing selected equipment after manufacture.

4.13.3. Requirements for French plants

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, VDA circuit for the EPR, etc.).

5. Requirements for Qualification

5.1. Requirements for Czech plants

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. Conformity assessment must be documented in conformity assessment documentation pursuant to requirements stipulated in Decree 358/2016. 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.

An authorized person performs conformity assessment for selected equipment with safety class 1, 2 and 3. 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.

Verification of conformity documentation must contain:

- work orders and guidelines for verification of conformity;
- internal rules, if they contain information regarding checks of selected equipment;
- plans for operating checks;
- an operating check programme;
- a managed ageing operating programme;
- accompanying technical documentation for selected equipment pursuant to Annex 4 to to Decree 358/2016;
- documentation applicable to the preparation and performance of repairs and maintenance of selected equipment; and
- records of operating checks and checks performed during repairs, maintenance, or modifications to selected equipment.

5.1.1. Conformity assessment procedure

The manufacturer or importer must ensure, in compliance with this procedure, that the selected equipment design meets the requirements of Decree 358/2016.

The manufacturer or importer must submit a request for a conformity assessment to the chosen authorized person. The request must contain:

- identification information of the manufacturer or importer, as follows:
- a written declaration that a conformity assessment contract has not been concluded with a different authorized person;
- technical documentation for the selected equipment;
- the selected equipment design; and

- other information on the selected equipment necessary for the conformity assessment, in particular its safety class.

An authorized person must:

- review technical documentation for the selected equipment, including an assessment of whether it meets requirements stipulated in Annex 3 to Decree 358/2016;
- perform assessment of materials used, including assessment of material certificates, if they were not already assessed by a different authorized person;
- check technological procedures for the creation of permanent joints and approve these procedures, if they were not already approved by a different authorized person;
- check that personnel performing special processes and welding supervisors have valid qualification certificates, and approve these personnel pursuant to Point 6.5, 7.2, and 8.4 of part A of Annex 2 to to Decree 358/2016;
- perform checks or have them performed in order to determine whether technical standards or technical specifications have been used properly;
- check that the selected equipment design is in compliance with the requirements of Decree 358/2016;
- create an inspection report.

5.2. Requirements for French plants

Component and system qualification are two important points for the justification and validation of the use of a sCO2 system. Indeed, this qualification will feed the modification file and must be included in the operator's integrated management system.

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.

5.2.1. Testing Qualification Strategy

The whole 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 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.

5.2.1.1. Test Requirements

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.

5.2.1.2. Test Sequence

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,
- Tests of functional limits of use,
- Tests of robustness in time,
- Accidental tests.

5.2.1.3. Reference Tests

Reference tests are performed at the beginning and end of the test sequence. This may include:

- Visual examinations,
- Electrical tests,
- Functional tests.

These tests are used as a reference during qualification. In addition, intermediate functional tests are performed during the sequence to verify the proper operation of the equipment.

5.2.1.4. Tests of functional limits of use

The equipment is also tested within its operating limits (functionally, electrically and environmentally). Generally and depending on the equipment, the tests carried out are:

- Voltage variation,
- Climatic tests (between -70°C and +55°C),
- EMC humidity (Electromagnetic Compatibility).

5.2.1.5. Tests of robustness in time

The tests carried out are based on a qualified service life assumption which depends on the equipment and is between 10 years and 60 years. This qualified lifetime cannot be guaranteed if the sequence is not scrupulously followed, in particular with regard to the severity and order of the tests.

The equipment undergoes a series of tests simulating ageing and must show its functional and mechanical robustness over time:

- Climatic tests (temperature variations, hot, cold, humidity, ...),
- Mechanical tests (vibrations, prolonged operation, ...),
- Irradiation tests for equipment classified K1 and K2.

5.2.1.6. Accidental tests

The purpose of these tests is to determine the resistance of the system and its components to accidental hazards.

Earthquake:

IEC 60980 recommends practices for seismic qualification. The seismic test is, in the majority of cases, performed on material "aged" by the previous test sequence. The equipment is mounted on a vibrating table and undergoes accelerations according to a specific spectrum depending on its location in the plant.

The technical parameters (earthquake force, duration, etc.) are defined according to the safety expectations at the risk of earthquakes in the power plant.

Accidental irradiation

Depending on its location in the plant and its function, the equipment may receive radiation doses and must continue to operate. Functional tests are carried out after an irradiation test.

Thermodynamic accident

In the event of a thermodynamic accident (e.g. steam line rupture), the equipment must withstand a particular Pressure/Temperature profile (e.g. 156 °C and 5.5 bars for a K1 qualification).

5.2.2. Numerical Qualification Strategy

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 qualify the system as a whole, using 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.

5.2.2.1. Probability Safety Assessment

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 particular, probabilistic analyses and PSAs must be used in order to:

- evaluate the overall frequency of fuel meltdown and the frequency of releases, which contribute to the assessment of the safety level of the BNI. In practice, the probabilistic safety objective can be broken down into several probabilistic targets, which refer to a reduced scope of initiating events or operating states of the installation.
- to shed light on the extremely unlikely nature of accident situations;
- to highlight possible scenarios having a largely preponderant contribution to the calculated frequency of fuel melting or to the calculated release frequencies;
- to assess the robustness of the installation in the face of aggression, when this is feasible;
- to confirm and complete if necessary the list of scenarios to be retained for the study of DEC-A conditions and to verify the adequacy of the measures implemented following these studies to prevent accidents involving fuel melting;
- to assess the adequacy of the provisions adopted to limit the consequences of accidents involving fuel melting.

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

5.2.2.2. Computers Codes

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.

5.2.3. 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.

5.2.3.1. Traceability

All the steps during a qualification must be traced in order to ensure that the requirements are met. This requires a follow-up during the phases of:

- Design (specifications),
- Testing/Qualification (manufacture of test specimen, test monitoring, qualification validation),
- Mass production (production of the test specimen),
- Installation (follow-up of on-site tests),
- In operation (periodic tests, maintenance, ...).

Any evolution and any anomaly must be traced and validated in order to guarantee the qualification and its maintenance during the life of the nuclear power plant.

5.2.3.2. Qualification specimen

The qualification specimen must be representative of the equipment assembled on site. It must be perfectly described in order to know exactly which elements will be qualified. In addition, a follow-up of its manufacture must be carried out in order to be able to manufacture the serial equipment identically. The manufacturing process is traced in a file called "Reference file". The specimen does not undergo any modification during the qualification phase (except in special cases which must be traced) and includes, in general, 3 specimens of each component to be qualified.

5.2.3.3. Testing and accreditation

The tests are based on international standards, codes and norms. A level of requirement is required for qualification testing. The tests must be controlled with an irreproachable quality system management. This is why it is requested to carry out the tests under a COFRAC (Name of the French committee for accreditation) accreditation (or equivalent to ISO 17025). A qualification program defines the strategy for demonstrating that the equipment meets the required requirements. Each test must be perfectly described (performances and criteria to be respected, environmental conditions, ...). A test sequence can last between 6 months and 18 months depending on the classification of the tested equipment. Failure to comply with the defined criteria leads to the management of a non-conformity.

5.2.3.4. Management of non-conformities

During the tests, random events may occur. In this case, it is requested that testing should not be continued until a causal analysis has been carried out to understand the origin and reasons for the failure. A non-conformity sheet is then opened including the analysis and the associated actions to close this hazard. Any non-conformity is traced and included in the qualification file to provide full transparency and demonstration of the qualification. Depending on the impact of the non-conformity, the tests can be either resumed, continued or even completely stopped because a new design is then necessary. In this case, it leads to a new qualification.

6. Requirements for Operation

6.1. Requirements for Czech Plants

Most of the corresponding requirements can be found in Decree 358/2016.

Requirements for ensuring compliance when operating selected equipment and parts thereof

- Selected equipment must be operated so that its technical safety is maintained during operation.
- During operation, accompanying technical documentation of the selected equipment must be supplemented with additional documents regarding the performance of repair, maintenance, or modifications to this equipment. A system of maintaining accompanying technical documentation must be implemented to make it possible to verify the fulfilment of technical requirements for selected equipment.
- Selected equipment can be installed and de-installed only under predetermined safe conditions and in compliance with rules for installation, de-installation, and re-commissioning.
- Selected equipment must be operated in compliance with the requirements of internal rules and other documentation for the operation of a nuclear facility. Rules for the maintenance and operation of selected equipment must include technical requirements and recommendations of the manufacturer of the selected equipment.
- The selected equipment may be operated and used only for purposes and under conditions for which it is intended, and in compliance with the nuclear installation's design. Technical and organizational measures must be put in place that shall ensure that selected equipment is operated under conditions for which it has been designed and does not threaten human health or present an inadmissible risk of damage to property.
- During the operation of selected equipment, a system of tracking and documenting discrepancies from normal operation that could lead to defects and a reduction of the technical safety level of the selected equipment must be implemented.
- During the operation of selected equipment, within the scope of an implemented managed ageing process for selected equipment, its condition must be tracked systematically, the impact of ageing and the effect of degradation mechanisms that could lead to defects and a reduction of the technical safety level of the selected equipment must be determined.
- 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.
- Maintenance, repair, or modification of selected equipment that is in operation must be performed in compliance with compliance assurance requirements for design, manufacturing, installation, and commissioning of selected equipment specified in Part A to D; if special processes are performed during maintenance, repair, or modification of selected equipment that is in operation, they must be performed in compliance with requirements for permanent joints, non-destructive checks, and thermal treatment specified in Part A Points 6 to 8.
- During maintenance, repairs, or modifications of selected equipment, contractor supervision must take place involving verification that activities occurring during maintenance, repair, or modification of selected equipment are performed in compliance with documentation applicable to the preparation and performance of maintenance, repairs, or modifications of selected equipment.

- Activities taking place on selected control equipment may be performed only by personnel qualified pursuant to Decree No 50/1978, on professional electrical engineering qualifications, as amended.

6.2. Requirements for French Plants

The regulations to be applied for the sCO2 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 system.

6.2.1. Human Factors

Relevant international Human Factors standards and guidelines could been applied to the design of the sCO2 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 been 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 (RCC-E, see Sub-chapter 3.8).

6.2.2. Operating Technical Specifications

The OTS form part of the operating documentation that must be developed for the plant operation. 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 system in a power plant, these specifications will have to be re-evaluated.

This sub-chapter will address inoperability conditions (or events); however, the associated corrective measure and completion time will not be presented as they must be defined in agreement with the Licensee.

6.2.3. Emergency plans

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

Conclusions

The work done in this deliverable has identified, for each stage of the development of the sCO2 system studied in the sCO2-4-NPP project, the various regulatory expectations.

As far as possible, after having given an overview of a given deliverable, we have presented its declination with regard to Czech and French regulations. Like deliverable D3.2, the present one highlights process differences between the two countries. These differences mainly result from the fact that the regulatory approach between the two countries is not the same, although both are based on the same international texts.

This difference is mainly reflected in the establishment of certain safety criteria, mainly related to environmental risks. For the Czech Republic, these criteria can be easily identified, while for France, these same criteria will be specific to each power plant, although they will be calculated using the same method because they take into account the diversity of power plant locations (seaside, riverside....). Similarly, the number of reactors in France has made it possible to set up an identified process for component qualification (with the establishment of reference guides by AFCEN - Association française pour les règles de conception, de construction et de surveillance en exploitation des matériels des chaudières électro-nucléaires- or standards) based on international texts, while the Czech Republic will rely directly on international texts.

Further project documents aiming to discuss the requirement for the sCO2 equipment will be D3.4: *Requirements for testing and operation* and D3.5: *Independent review of requirements.*

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