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

Deliverable 3.2
Requirements for reference plant modifications for installation of sCO2-4-NPP

Work programme topic addressed: NFRP-2018-10: Encouraging innovation in nuclear safety for the benefit of European citizen

Type of action: Innovation action

Grant Agreement number: 847606

Start date of project: 1 September 2019 Duration: 36 months

Lead beneficiary of this deliverable: NRI

Due date of deliverable: 31/10/2020 Actual submission date: 30/10/2020

Version #: R1.0

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

Release	Date	Reason for Change	Author	Distribution
R0.1	16/10/20	Version for Review	NRI (UJV), EDF	USTUTT, JSI
R1.0	30/10/20	Final version	NRI (UJV), EDF	Public

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1. List of Acronyms

Abbreviation / Acronym	Description / meaning
ALARA	As Low As Reasonably Achievable
AOO	Anticipated operational occurrence
ASN	Autorité de sûreté nucléaire (Nuclear Safety Authority)
BNI	Basic Nuclear Installation
BS	Bezpečnostní Systémy (Safety Systems)
BSR	Basic Safety Rules
ČEZ	České Energetické Závod (the largest Czech energy company)
DBA	Design Basis Accident
DBC	Design Basic Conditions
DEC	Design Extension Conditions
DG	Diesel Generator
DiD	Defence in Depth
DSA	Deterministic Safety Analysis
EDF	Électricité de France
EIP	Important Elements for Protection
EOP	Emergency Operating Procedures
EPR	Evolutionary Power Reactor (or European Pressurised Reactor)
ESWS	Essential Service Water System
HVAC	Heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IRSN	Institut de radioprotection et de sûreté nucléaire (Radioprotection and Nuclear Safety Institute)
ISO	International Standard Organisation
I&C	Instrumentation and Control
LUHS	Loss of Ultimate Heat Sink
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PIE	Postulated Initiating Events

Abbreviation / Acronym	Description / meaning
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RFS	Règles Fondamentales de Sûreté
RHWG	Reactor Harmonisation Working Group
SAM	Severe Accident Management
SBO	Station Black-out
SFC	Single Failure Criterion
SFSP	Spent Fuel Storage Pool
SIE	Single Initiating Event
SSB	SSB - Systémy Související s Bezpečností (Safety-related systems)
SSC	Systems, Structures, Components
SÚJB	Státní úřad pro jadernou bezpečnost (State Office for Nuclear Safety)
TMI	Three Mile Island
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USSR	Union of Soviet Socialist Republics
VVER	Vodo-Wodyanoi Energetichesky Reaktor (Water-Water Energy Reactor)
WENRA	Western European Nuclear Regulators Association
ZBF	Základní Bezpečnostní Funkce (Basic Safety Functions)

2. Executive Summary

The purpose of this deliverable is to describe the requirements in Czech Republic and in France in order to approve nuclear power plant modification and installation of the heat recovery system. The licensing requirements depend on the country regulation, design requirements and other factors that should be considered in the analysis. In parallel with D3.1. [1] and D3.3. [2], this report provides the set of requirements for the implementation of the system in the nuclear power plant in selected reactor types (VVER in Czech Republic and PWR in France).

The previous deliverable, D3.1. ("Report on identification of the regulatory elements for design of components and system") [1] provided the identification of the nuclear regulatory elements to be considered in the design of components and system for passive decay heat removal, called sCO2-4-NPP. The starting point for identification of the nuclear regulatory elements was the setup of a hierarchy of regulatory requirements proposed to be used for the sCO2-4-NPP project. Here, in D3.2, the more specific guidelines are presented, both for Czech Republic and France. The nuclear regulatory framework in these countries bases on the international requirements, mainly by IAEA and WENRA, as described in D3.1. However, there are different ways of incorporating these rules into national legislation.

Chapter 3 of this report provides a brief introduction of the nuclear industry in Czech Republic (subsection 3.1) and in France (subsection 3.2). The part regarding the Czech Republic case bases on the example of VVER-1000 reactor and Temelín NPP. The French part uses the example of EPR reactor.

Next, the nuclear regulatory framework in Czech Republic (subsection 4.1) and in France (subsection 4.2) is described. The implementation of the international requirements is presented as well as country specific legislation and hierarchy of the rules.

Chapter 5 includes the description of general approach to nuclear safety. The most important aspects are presented, such as Defence in Depth concept, classification of NPP states, safety analysis and acceptance criteria. The concepts and rules are similar in Czech and French legislation and some links between the two parts are given, where appropriate.

In chapter 6 the requirements and classification of NPP systems, structures and components are presented, with the special emphasis on the place of sCO2 system in the classification.

Finally, chapter 7 describes the process of NPP modification in Czech Republic and in France. All main steps and requirements regarding the process are presented briefly.

The information included in this report gives the general view of the nuclear regulatory framework, approach to safety and requirements regarding the modifications of NPPs in Czech Republic and in France. It allows to understand the rules that must be considered in sCO2-4-NPP system design assessment and implementation. More specific guidelines will be included in the next deliverable, D3.3 ("Design bases and analyses for system and components") [2].

3. Introduction to nuclear industry in Czech Republic and in France

3.1. Czech Republic

As for 2019, the Nuclear Power Plants provided about 34.5 % of the electricity needs in the Czech Republic [3]. There are two NPPs, one at Dukovany in South Moravia and another at Temelín in South Bohemia. The Dukovany Nuclear Power Plant has four operational units (VVER-440/213) each of which has a thermal power of 1 375 MWt (510 MWe), resulting in total installed capacity of 2 040 MWe. The Temelín Nuclear Power Plant has two operational units (VVER-1000), the first of which began trial operation in mid-2002. Temelín 2 began full operation in October 2004. The two units have each a thermal power of 3 000 MWt (1 082 MWe) resulting in total installed capacity of 2 164 MWe. In addition, the Czech Republic has three research reactors, several radioactive waste storage facilities and interim spent fuel storage facilities (operated at Temelín and at Dukovany) and a medium and low-level institutional radioactive waste repository (operated at Dukovany).

Figure 1. Nuclear facilities in Czech Republic [4]

3.1.1. Temelín NPP

The origin of the Temelín Nuclear Power Plant can be dated back to 1979 with the publication of the Investment Plan. Based on this plan, a government decision was taken in 1980 on the construction of four units of the NPP with reactors of the newly developed type V 320, VVER-1000 in the Temelín locality. In 1982, a contract was signed with the former USSR for the supply of a technical project. This project included the main production unit, the auxiliary plant building and the diesel generator station. The other parts of the power plant were designed by the Czech partner on the basis of a contract [5].

Introductory project IV. A. of the Temelín NPP was completed by ENERGOPROJEKT PRAHA in 1985. It was the part needed to start excavation work. Introductionary project for the 1st and 2nd block construction IV. B was completed in 1986 and the building permit was issued in the same year. The actual construction was started in February 1987. After 1989, the need for 4 000 MWe of installed capacity was re-evaluated and at the same time, a new safety evaluation of the project was carried out. In March 1993, the Czech government has again reconsidered the construction of the Temelín NPP and definitively approved its completion, at the same time deciding that only two of the originally planned four NPP units should be completed.

The investor of the construction of the Temelín nuclear power plant was the power company ČEZ, a.s.; the general contractor of the construction part was Vodní Stavby; the general supplier of the technological part was ŠKODA Praha; the general designer was Energoprojekt Praha. The choice and evaluation of the location was made by the state organization TERPLÁN in 1980. It was confirmed by the government.

The site is located in southern Bohemia, about 25 km north of České Budějovice and 45 ÷ 50 km from the state borders with Austria and Germany, at the altitude of 510 m. The nearest town is Týn nad Vltavou, located 5 km northeast of the plant. The selected locality was satisfactory in terms of the analysed criteria (geological, hydrogeological, seismic, water management and geographical requirements, settlement, etc.). Less favourable were the technical requirements for local infrastructure (transport connections, connection to the transmission system, etc.).

The nuclear power plant currently consists of two nuclear units, each of which has an output of 3 120 MWt (1 250 MWe) - after increasing the capacity of the units and other modifications. The original output was 3 000 MWt (1,111 MWe).

The cessation of the construction of units 3 and 4 affected in particular the systems connecting power units with external buildings. The main changes were reflected in the part of the external connecting pipeline and the implementation of systems for the operation of Units 1 and 2. These impacts were addressed by separate amendments to the introductory project IV.

In accordance with Government Resolution 121/1997, a dry storage of spent nuclear fuel was built in the years 2009 - 2011 at the Temelín NPP site.

Each power unit consists of:

- Reactor pressure vessel with airtight containment and reactor building,
- the engine room with the secondary circuit, turbogenerator systems and heat exchanger station,
- systems for ensuring electrical supply for own needs

The primary circuit consists of a reactor, pressurizer and four primary coolant loops, each with a main circulation pump and a horizontal type steam generator. The primary circuit components are placed in an airtight containment vessel made of concrete. The containment vessel consists of a cylindrical structure with an inner diameter of 45 m, closed by a hemispherical cover. The inner surface of the containment is covered with a hermetically sealed steel lining. Spent fuel storage pools are located inside the containment, where spent fuel is removed from the reactor core. The reactor core is cooled and moderated by the light water of the primary circuit, which is pumped through the core by the main circulation pumps. The heat accumulated in the coolant after passing through the reactor is transferred in the steam generators to the water of the secondary circuit. The pressure of the primary circuit is maintained by a pressurizer.

For emergency cooling of the reactor and reduction of the pressure in the containment, a safety system is used, consisting of three structurally and electrically independent units. The activity of one unit will suffice to

mitigate the transient. Each unit includes emergency core cooling tanks, emergency boric acid storage tanks, emergency cooling spray pumps, high- and low-pressure emergency cooling pumps, and other components.

The primary part of the nuclear unit also consists (in addition to the primary circuit and safety systems) of the following **auxiliary systems**, e.g.:

- primary circuit volume control system and boric acid control,
- primary circuit drainage collecting system,
- continuous primary coolant cleaning system,
- hydrogen recombiners,
- spent fuel pool cleaning and cooling system,
- HVAC systems.

The secondary circuit consists of four loops of the secondary side of steam generator, uniaxial full-speed steam turbine (3 000 rpm) with one two-stream high pressure part and three low pressure parts, two parallel separators with reheater, three surface condensers, four condensate pumps, four-stage regenerative condensate heating, feed tank with thermal degassing, two-branch high-pressure feedwater heating, three feed pumps driven by condensing steam turbine, alternator with accessories.

The steam turbine drives a synchronous two-pole generator (water-cooled stator, hydrogen rotor).

The output electrical power is driven by a block transformer 1 200 MVA, 420 / 24kV from three single-phase units and a simple 400 kV line to the Kočín substation. The NPP own power consumption is provided by a 24kV line.

The **spent fuel storage** is built on the site of the Temelín NPP on the basis of Government Resolution No. 121/1997. After reducing the residual power in the spent fuel pool, the spent fuel is transferred to the packaging and taken to the spent fuel storage facility. Heat dissipation from stored packages is a natural ventilation of the air. It is designed for 60 years of operation.

The **active safety systems** have 3 x 100% redundancy; they are independent and physically separate. The **passive safety systems** (hydro accumulators inside the containment) have 2 x 100% redundancy. The seismic resilience of all redundant safety systems, including power supply, control systems and all auxiliary systems, is ensured. The backup power sources and control systems are independent, physically separate and seismically resilient (subject to qualification as safety systems). There are also backup, non-seismically resilient power sources for safety-related systems. The project uses diverse systems to guarantee three basic safety functions: 1) shutdown of the reactor (sub-criticality), 2) transfer of heat (cooling) and 3) preventing radioactive leaks (barriers and insulations for the containment) [6].

The **power supply sources** for all systems providing heat transfer from the reactor core have the following defence-in-depth structure:

- Operating power sources (operating transformers for own power consumption), or
- Backup power supply from the grid, or
- Secure power supply from systematic and common DG and from batteries.

External objects for the operation of the secondary circuit and other necessary NPP objects:

The tertiary cooling circuit consists of a cooling water pumping station and cooling towers with natural draft. The pumping station also includes fire water pumps, service water pumps and diesel generators cooling pumps.

Other systems ensuring reliable operation of the Temelín NPP are, for example, water management systems ensuring the supply of raw water, Hněvkovice raw water pumping station, reservoir treatment of additional water to external cooling circuits [5].

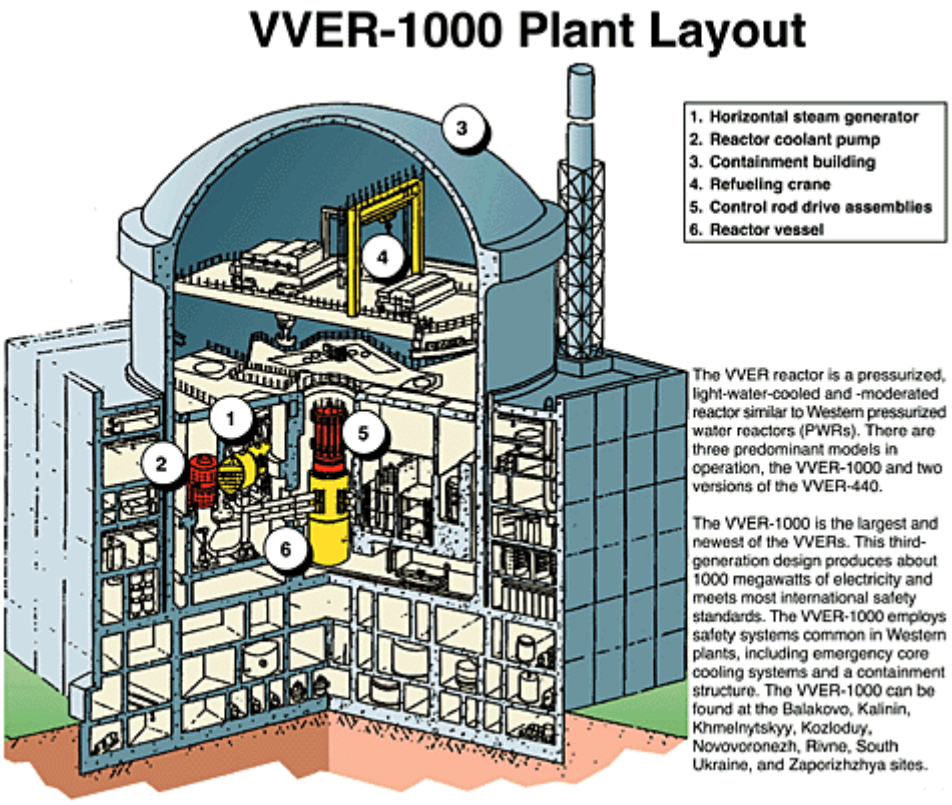

Figure 2. VVER-1000 Plant layout [International Nuclear Safety Project]

3.1.2. Potential need and criteria for the modification

Although the Temelín NPP has already existing safety systems to mitigate different accident conditions, there is room for improvements. In Stress Test for Dukovany and Temelín NPPs (2012) [6] it was concluded that alternative methods for the residual heat removal in accident conditions are highly recommended:

There is only one ultimate heat sink within the Temelín NPP for the transfer of heat from the safety systems, and that is the surrounding atmosphere. The means of transferring residual heat from the core, from spent fuel stored in the SFSP and from the components of safety systems into the ultimate heat sink is the ESW system. During SBO events, the ESW pumps have no power supply. Because it is the ESW system that transfers heat into the atmosphere, an SBO implies the loss of the forced heat transfer from the primary circuit and SFSP into the atmosphere. An SBO event will automatically lead to a loss of the ultimate heat sink in the given unit, as a consequence of the loss of the power supply for the pumps of the ESW system. If the ultimate heat sink is lost and, at the same time, the power supply from the operating and backup sources are lost, the failure of the DG

cooling will lead to an SBO situation within the unit. The reason for this is the dependency between the DG and the ESW – the failure of one means the failure of both. The transfer of heat from the core via the secondary circuit (SG) can be used for transferring residual heat only in hot or semi-hot states of the units and only until the water supply in the SG has been consumed. However, there is no backup for heat transfer from spent fuel stored in the SFSP. The above facts imply that the operability of the ESW system transferring heat into the ultimate heat sink and the operability of emergency power supply sources are interconnected.
Although full loss of the capacity to transfer heat into the ultimate heat sink would require combined failures within defence-in-depth levels, the potential consequences of such a situation require additional measures that could help increase the already very high sturdiness of the project in terms of the capacity to transfer heat into the atmosphere as the ultimate heat sink.

The aim of the proposed measures is to strengthen the levels of defence-in-depth in case of initiating events beyond the design basis (earthquake, floods, extreme conditions, human factor, etc.), which could lead to the loss of safety functions during an SBO:
1. Propose and implement diversified methods of cooling and heat transfer from the core and the SFSP, including the possibility of their connection to existing systems.
2. Propose and implement alternative methods for securing the cooling of the I&C necessary for monitoring the situation and controlling selected components.
3. Describe the use of alternative and diversified means (proposed according to Points 1 and 2) – "emergency plans", with the aim of securing cooling and heat transfer from the core and the SFSP.

The sCO2 emergency cooling loop would serve the above purposes, significantly strengthening the plant safety. Installing this new safety system would be a meaningful modification of the plant and as such it would require several steps:

- Careful assessment of the design, including potential benefits and costs, safety analysis, impact on other SSC, compliance with current Emergency Operating Procedures (EOPs) and Severe Accident Management (SAM) guidelines etc.
- Obtaining permission for the modification from the regulatory authority (SÚJB).
- Preparing and updating appropriate documentation regarding the NPP, safety procedures, safety analyses etc.
- Assessment of the need for additional personnel for constructing and operating the system; preparation of the training for existing and new personnel.
- Installation of the new system during the plant outage.
- Test and evaluation of the system operation.
- etc.

The existing legislation significant for the process, classification of the SSC, requirements for the safety analysis and plant modifications is described in the following chapters.

3.2. France

French nuclear program expanded after the Second World War, initially to develop military nuclear power. The construction of the first nuclear power plants began in parallel, in order to produce plutonium and electricity.

At the time of the oil shocks in the 1970s, the French government opted for "all-nuclear" in order to guarantee France's energy independence for the production of electricity (65% of electricity was then produced by fossil fuel power plants). Following this decision, in the space of 20 years, 58 nuclear reactors were built. The share of nuclear power in the production of electricity reached 75%. These reactors are all based on a Pressurized Water Reactor concept.

Since the end of the construction of these 58 reactors, France has focused on increasing the share of renewable energies in its energy mix, but French power plants continue to provide the majority of electricity and 56 of the 58 reactors are still in operation.

The following figure shows the distribution of nuclear reactors in operation in France. Since 2020, the Fessenheim reactors have been shut down due to their age.

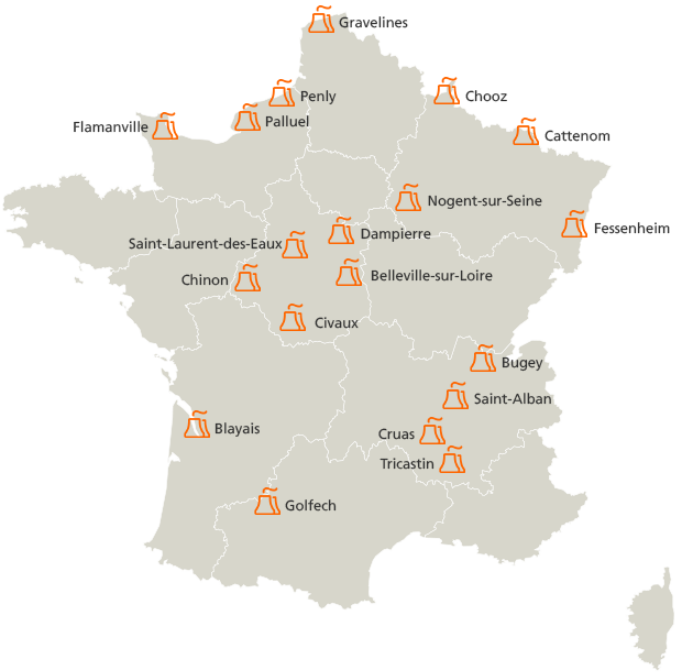

Figure 3. Nuclear reactors in operation in France (2018) (Source EDF)

As the French nuclear fleet has an average age of more than 30 years, the French government has decided to relaunch the French nuclear industry in order to renew part of this fleet in the coming years, with new reactor designs such as the EPR.

3.2.1. EPR Technology

The EPR (Evolutionary Pressurized Reactor) is a 1600 MW pressurized water reactor.

Compared to previous PWR plants built in France, the EPR is a more complex and powerful project (1650 MW compared to 1450 MW for the N4 and Konvoï reactors). Designed to meet the standards set by the German

and French safety authorities in the 1990s, the EPR is technically based on the French N4 and German Konvoï reactor concepts, from which it takes over certain characteristics.

The changes compared to the previous system, requested by the nuclear safety authorities that certified it, are intended to limit the risk of accidents, reduce radiation doses likely to affect personnel, and reduce radioactive emissions into the surrounding environment. The level of personnel exposure to radiation is reduced by a factor of two, and the activity level of releases by a factor of ten compared with the most recent facilities in operation.

In terms of competitiveness, the EPR has a higher power output, better availability, better thermal efficiency and a longer operating life than Generation II reactors.

From a technical point of view, the design of the EPR is characterized in particular by its containment, composed of two 1.3 m thick concrete walls, the inner face of the inner wall being completely covered by a metal skin (the liner), and by a new device called a "corium catcher" designed to collect the part of the molten core (corium) that is likely to pass through the vessel (without this, the corium could sink into the ground and contaminate the environment, if it managed to pass through the vessel and the concrete apron of the containment).

The EPR reactor has several active and passive protections against nuclear accidents:

- four independent emergency cooling systems, each capable of cooling the reactor after its shutdown;
- containment heat removal system;
- a containment made of two separate layers, in total 2.6 m thick;
- a corium catcher (in case of perforation of the vessel by the molten core).

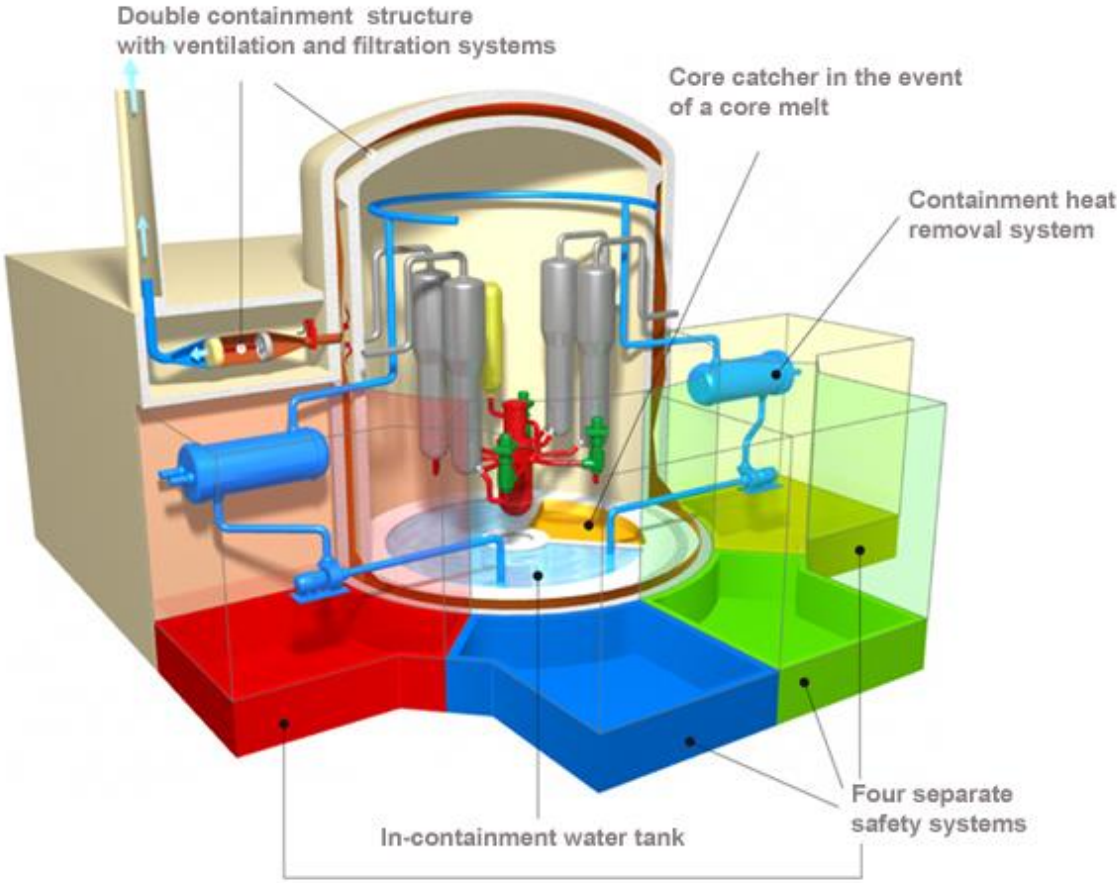

Figure 4. EPR diagram

3.2.2. Potential need and criteria for the modification

After the Fukushima accident, the French power plants, like all European power plants, underwent stress tests so that the safety authorities could establish modification criteria if necessary.

At the end of this complementary safety tests, in its opinion of January 3, 2012, the ASN emphasized that it would impose on operators a set of measures to strengthen the prevention of natural risks (earthquakes and floods) or risks related to other industrial activities and ensure control of accident situations that could result (loss of power supplies, loss of cooling resources, serious accidents). These requirements imposed in particular on EDF :

- the implementation of a "hard core" consisting of a limited number of pieces of equipment to ensure safety functions in extreme situations. This "hard core" must be as independent as possible of existing devices, in particular as regards control-command and power supply. On January 21, 2014, the ASN college adopted 19 decisions specifying the objectives and elements making up this "hard core". EDF proposed to set up this hard core in two phases:
    - a first phase involving the installation of the fundamental elements of the hard core, in particular, for each reactor, a high-capacity ultimate back-up diesel requiring the construction of a dedicated building, a dedicated ultimate water source and an ultimate water make-up, and for each site the construction of a local crisis center capable of withstanding extreme

external hazard. The implementation of these measures will be gradual, beginning in 2015 and will be mostly completed by 2022;

  - a second phase to complement the first to improve the coverage rate of potential accident scenarios, taking into account in particular the potential consequences of serious accidents. These means include the finalization of the connections of the ultimate back-up to the reactor, the installation of an ultimate control command system and the definitive instrumentation of the hard core, the installation of an ultimate containment cooling system to prevent the opening of the filtered vent of the containment, the installation of a solution for flooding the reactor vessel shaft to prevent the corium from crossing the reactor floor. These means were also defined by EDF with a view to the continued operation of the reactors since they correspond to the objectives set by the ASN in this context. EDF therefore plans to put them in place within the framework of the next safety reviews.

- the deployment of the "Nuclear Rapid Action Force" (FARN), which makes it possible to provide assistance to a site with 4 accident reactors by providing specialized teams that can supplement those of the power plant concerned and mobile equipment providing additional water and electricity. Within this framework, several modifications have been implemented on the reactors in order to facilitate the connection of this equipment provided by the FARN. The ASN ensured that it was operational by the end of 2014.
- the implementation of a set of temporary or mobile provisions aimed at reinforcing the consideration of situations of total loss of the heat sink or loss of transient power supplies. These provisions include, for example, the installation of medium-power generators on each reactor, the reinforcement of local crisis resources (pumps, generators, hoses, etc.), the installation of connection points for mobile resources, especially those of the FARN, and the reinforcement of the resistance to earthquakes (SMS) and flooding (millennium flood) of crisis management premises. The implementation of these measures is well advanced and should be completed in 2015.

EDF complied with all the regulatory deadlines for these decisions and, in particular, implemented the modifications required by the decisions of June 26, 2012, which expire on December 31, 2014, in particular with regard to seismic risk, flood risk, limiting discharges in the event of an accident, maintaining the water inventory of swimming pools in situations of exceptional external hazard, and improving instrumentation.

4. Nuclear regulatory framework

The general hierarchy of rules in the frame of sCO2-4-NPP was presented in the sCO2-4-NPP deliverable D3.1. The purpose of this deliverable is to specifically focus on the nuclear regulatory framework in Czech Republic and in France.

4.1. Nuclear regulatory framework in Czech Republic

The requirements can be arranged into a 5-level "pyramid", that represents their hierarchy. First level contains the most important, legal documents and the last one - normative and technical documentation [5].

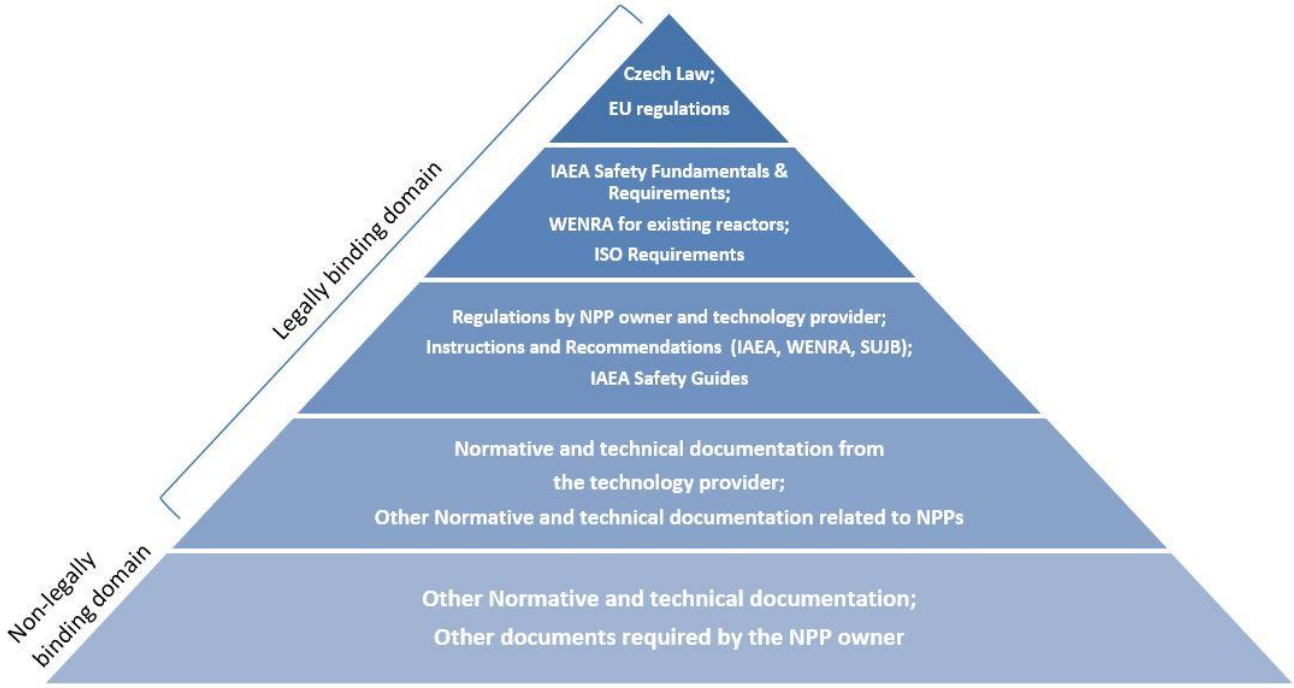

Figure 5. Hierarchy of nuclear regulations in Czech Republic

4.1.1. Nuclear regulatory framework in Czech Republic – level 1

This level includes:

- Czech legislation of all degrees currently in force,
- International conventions and agreements which are bounding for the Czech Republic,
- Directly bounding EU regulations.

All these are generally binding documents. They were issued by Czech government, Czech national nuclear regulator (SÚJB), EU Commission, Euratom and other international institutions.

Some of the most important documents are listed below:

- 18/1997, Act on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act),
- 263/2016, the new Atomic Act,

- 183/2006, Act on Spatial Planning and Building Rules (the Building Act),
- 225/2017, Act amending Act No. 183/2006 (Building Act),
- 458/2000, Act on business conditions and public administration in the energy sectors (Energy Act),
- 133/1985, Act on Fire Protection, as amended,
- 174/1968, Act on the state expert supervision of occupational safety, as amended,
- 162/2017; SÚJB of 26 September 2017 on requirements for safety assessment pursuant to the Atomic Act,
- 329/2017; SÚJB on requirements for the design of a nuclear facility,
- 408/2016; SÚJB on requirements for the management system,
- 132/2008; SÚJB on Quality Assurance System in carrying out activities connected with utilization of nuclear energy and radiation protection and on Quality assurance of selected equipment in regard their assignment to classes of nuclear safety;
- 195/1999; SÚJB on basic design criteria for nuclear installations with respect to nuclear safety, radiation protection and emergency preparedness,
- Decree 422/2016 on Radiation Protection and Security of a Radioactive Source (implementing the respective Euratom regulations),
- Decree 408/2016 on Management System requirements (implementing the respective Euratom regulations),
- 2016/631, EU Commission Regulation of 14 April 2016 establishing a network code on requirements for grid connection of generators.

4.1.2. Nuclear regulatory framework in Czech Republic – level 2

This level includes:

- IAEA Safety Fundamentals and IAEA Safety Requirements,
- WENRA Safety Reference Levels for Existing Reactors,
- ISO Quality Assurance and Environmental Requirements.

Most of the documents on this level were issued by IAEA and WENRA. As for 2019, Czech Republic has implemented all WENRA requirements.

Some of the most important documents are listed below:

- SF-1, IAEA Fundamental safety principles (2006),
- SSR-2/1 Rev.1, IAEA Safety of Nuclear Power Plants: Design (2016),
- SSR-2/2 Rev.1, IAEA Safety of Nuclear Power Plants: Commissioning and Operation (2016),
- SSR-1, IAEA Site evaluation for nuclear installations (2019),
- GSR Part 3, IAEA Radiation Protection and Safety of Radiation Sources: International basic safety standards (2014),
- GSR Part 4, IAEA Safety Assessment for Facilities and Activities (2014),
- Safety of new NPP designs 01/2013, WENRA,
- WENRA/RHWG Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons (2013).

4.1.3. Nuclear regulatory framework in Czech Republic – level 3

This level includes:

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

This group contains documents issued by SÚJB, IAEA, WENRA as well as the NRC, Russian nuclear regulator (Gosatomenergonadzor), Russian nuclear company (Atomenergoprojekt), Czech nuclear operator (ČEZ), Czech Nuclear Research Institute (UJV) etc.

Some of the most important documents are listed below:

- Post Fukushima National Action Plan (NAcP) on Strengthening Nuclear Safety of Nuclear Facilities in the Czech Republic, SÚJB (2013),
- Requirements for the design of nuclear facilities to ensure nuclear safety, radiation protection, physical protection and emergency preparedness, SÚJB, BN-JB-1.0 (2011),
- Probabilistic Safety Assessment, SÚJB, BN-JB-2.5 (2018),
- Modifications of structures, systems, components and processes of nuclear facilities, SÚJB, BN-JB-1.10 (2010),
- Selection and evaluation of design and beyond design basis accidents and risks for Nuclear Power Plants, SÚJB, BN-JB-1.7 (2010),
- WENRA Report: Safety of new NPP designs (2013),
- IAEA NS-G-2.3, Modifications to Nuclear Power Plants (2001),
- IAEA Safety Standards Series No. SSG-2 (Rev.1), Deterministic Safety Analysis for Nuclear Power Plants (2019),
- IAEA-TECDOC-1791, Consideration on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants (2016),
- Gosatomenergonadzor of Russia, OPB-88/15, General Safety Provisions for Nuclear Power Plants (2015).

4.1.4. Nuclear regulatory framework in Czech Republic – level 4

This level includes:

- Normative and technical documentation of the country of origin of the technology related to nuclear energy,
- Other normative and technical documentation related to nuclear energy.

These documents were issued by A.S.I. (Association of Mechanical Engineers, Czech Republic), ANSI (American National Standards Institute), IEEE (Institute of Electrical and Electronics Engineers), Gosgortechnadzor, NRC, KTA (Kerntechnischer Ausschuss - German Nuclear Standard Committee) etc.

Some of the most important documents are listed below:

- 603-2018 - IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,
- NRC, Code of Federal Regulations, Part 50 - Domestic Licensing of Production and Utilization Facilities (10CFR 50),
- NRC, REGULATORY GUIDE 1.155 Tables 1, (Task SI 5014) 5, and 6. STATION BLACKOUT.

4.1.5. Nuclear regulatory framework in Czech Republic – level 5

This level includes:

- Other normative and technical documentation,
- Other documents required by the NPP owner.

All the above documentation will be useful in the process of plant modification. Especially the national legislation, international agreements and applicable requirements from level 1 and 2 (5.1, 5.2) must be considered and fulfilled.

4.2. Nuclear regulatory framework in France

4.2.1. Regulatory authority: ASN

Nuclear legislation in France was developed in successive stages alongside technological advances and growth in the atomic energy field. Therefore, many of the enactments governing nuclear activities are to be found in the general French legislation on environmental protection, water supply, atmospheric pollution, public health and labour.

However, the French Parliament also adopted a number of specific enactments, now mostly embodied in the Title IX of Book V of the Environment Code (Environment Code) [7].

French nuclear regulation and law are characterized by many sources, as in other countries with nuclear energy capacities, the original features of this legislation derive chiefly from international recommendations or regulations.

French nuclear legislation began to develop in the 60’s and 70’s with the launch of the construction of the French nuclear park. This development passed several landmarks: an authorization requirement for major nuclear installations was introduced, setting Government responsibility in matters of population and occupational safety (Decree of December 11th, 1963) [8]. Prior to this, procedures concerning the licensing and control of industrial activities were dealt with by the prefect for each department. In 1973, this system was expanded to cover the development of the nuclear power programme, and better define the role of Government authorities. Finally, the decree of 20 June 1966 included Euratom directives as part of the French radiation protection regulations.

In 13 June 2006, Act No. 2006-686 of 13 June 2006 on Transparency and Security in the Nuclear Field (Act 2006-686) [9] created the ASN. ASN ensures the oversight of nuclear safety and radiation protection in order to protect people and the environment. It informs the public and contributes to enlightened societal choices. ASN is an independent administrative agency, consulted before decisions concerning nuclear safety, nuclear security and radioprotection are taken by decrees.

The ASN is also responsible for the following:

- Organizing and directing the control of nuclear installations (designation of inspectors, delivery of permits, etc.);
- Monitoring radioprotection over national territory;
- Proposing and organizing public information, especially on nuclear safety;
- Establishing the procedures for licensing large nuclear installations (licences for setting up, commissioning, disposal, shutdown, etc.);
- Helping manage the emergency situation in the event of an accident involving radioactive exposure.

The role and responsibility of ASN could be resumed in this figure

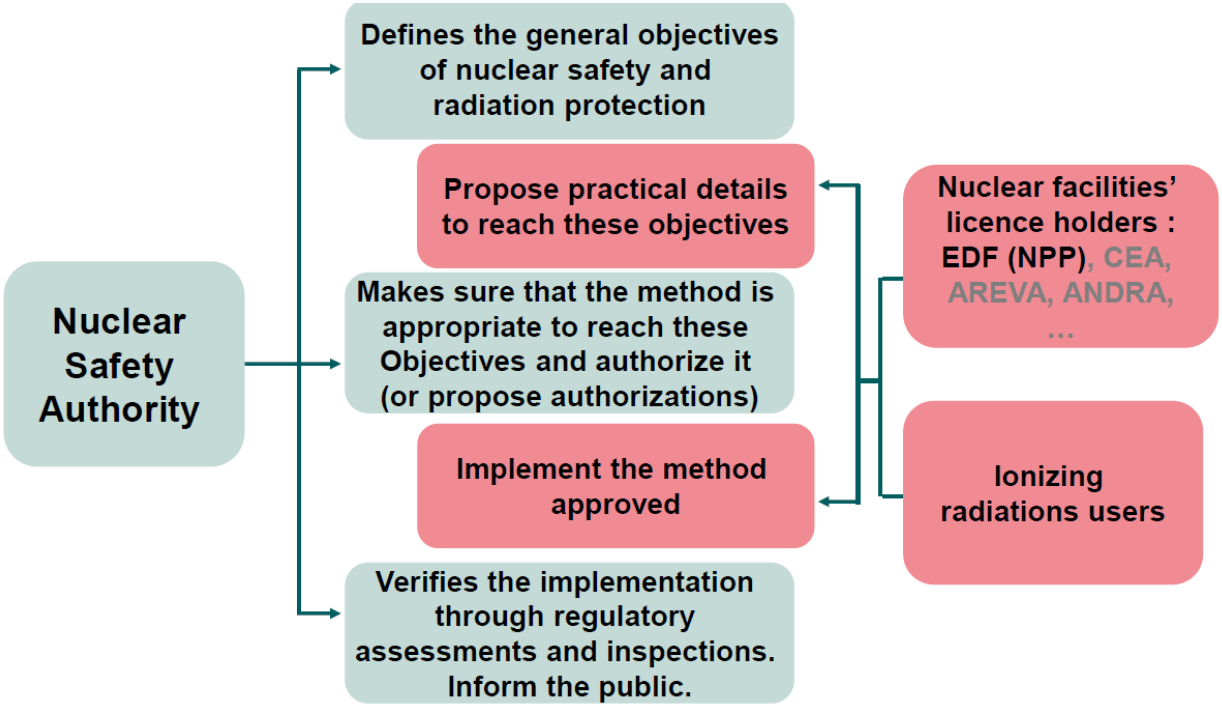

Figure 6. Regulatory Framework of France for NPPS (copyright : IRSN)

4.2.2. Bases of French regulations for nuclear activities

The legal framework specific to nuclear activities finds its source in norms, standards or recommendations established by various international bodies. In particular, we can cite:

- International Atomic Energy Agency (IAEA), which publishes and regularly revises "standards" in the fields of nuclear safety and radiation protection.
- International Organization for Standardization (ISO, International Standard Organisation), which publishes international technical standards that are references in the field of radiation protection.
- At the European level, the Euratom Treaty, defines some procedures for nuclear activities and specifies the powers and obligations of the European Commission with regard to their application.

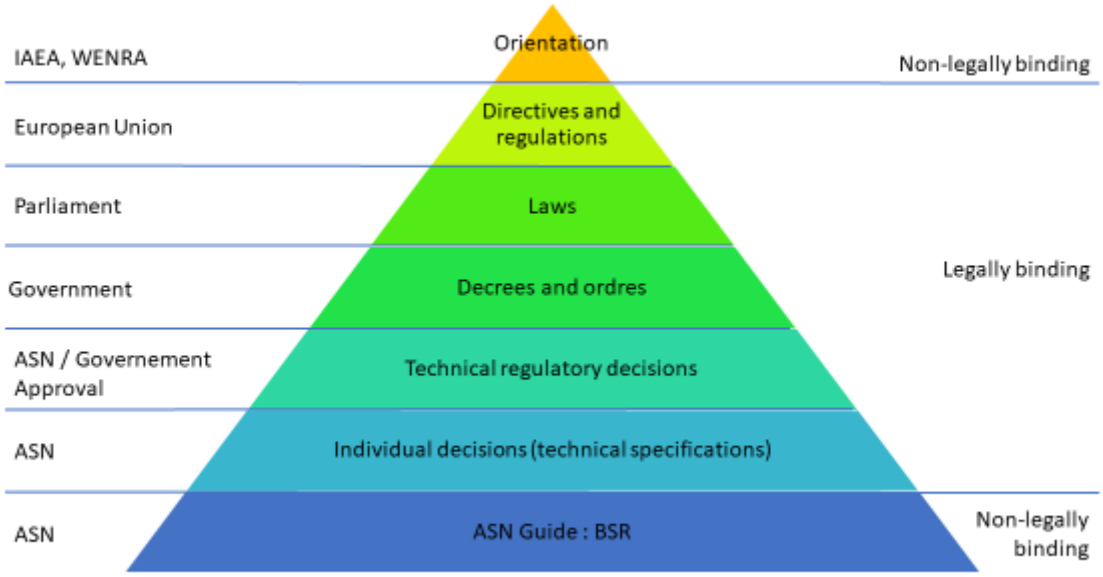

Figure 7. Levels of regulation in the local nuclear field in France

Based on these recommendations and directives, the French government can then issue the decrees and orders necessary to regulate nuclear activities in France. The following figure summarizes these steps at the national level.

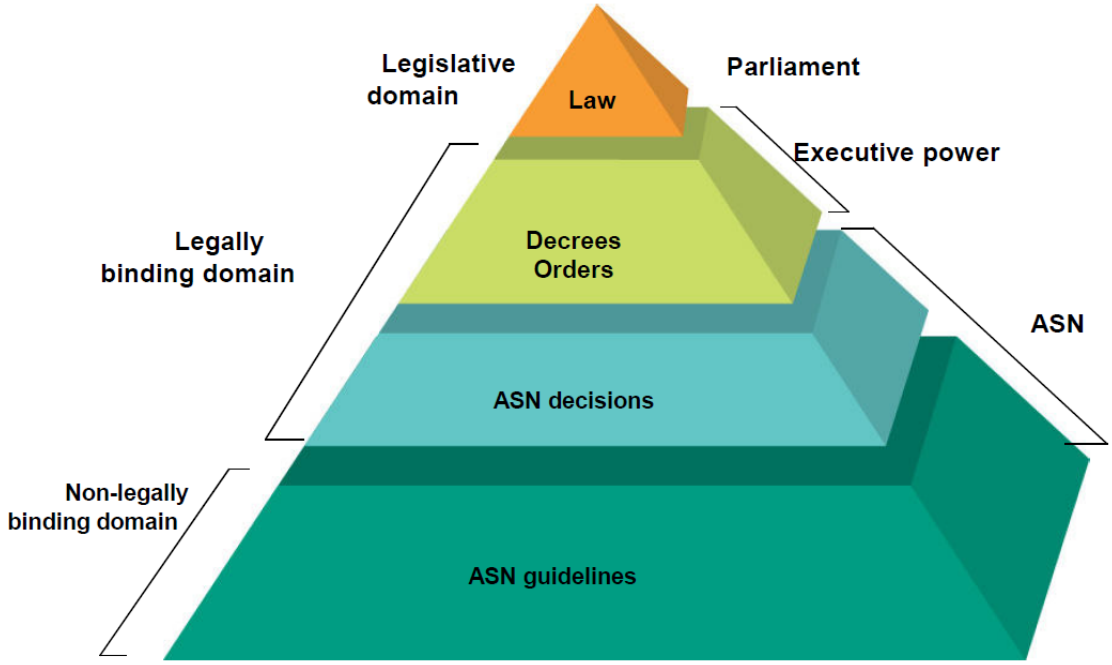

Figure 8. French nuclear regulation process

The responsibilities belong to the French Government. Decrees and orders are taken at ministerial level. ASN proposes or gives advice on these ministerial decisions.

The ASN issues regulatory decisions (mainly technical rules) have to be endorsed by the French Government.

Afterwards, the ASN publish Guides that explain how to consider the corresponding regulation. These Guides are not prescriptive.

During the entire process of creating and voting on the decrees, the transparency of exchanges with the ASN guarantees that all points of discussion, all advices are taken into account.

4.2.2.1. Environmental code

The Environmental Code (Article L. 591-1) [7] defines the main concepts for nuclear activities in France. Nuclear security includes nuclear safety, radiation protection, prevention and fight against malicious acts as well as civil security actions in case of accident. However, the expression "nuclear security" is still, in some texts, limited to the prevention of malicious acts and the fight against them.

Nuclear safety is "all technical provisions and organizational measures relating to the design, construction, operation, shutdown and dismantling of plants and the transport of radioactive substances, taken with a view to preventing accidents or limiting their effects".

Radiation protection is "protection against ionizing radiation, i.e. all the rules, procedures and means of prevention and monitoring aimed at preventing or reducing the harmful effects of ionizing radiation produced on people, directly or indirectly, including by damage to the environment".

Article L. 593-42 of the Environment Code specifies that "The general rules, prescriptions and measures taken in application of the present chapter and of chapters V and VI for the protection of public health, when they concern the radiation protection of workers, relate to collective protection measures which are the responsibility of the operator and are such as to ensure compliance with the principles of radiation protection defined in article L. 1333-2 of the Public Health Code. They apply to the design, operation and dismantling phases of the installation and are without prejudice to the obligations incumbent on the employer in application of Articles L. 4121-1 et seq. of the Labor Code."

Transparency in nuclear matters is "the set of measures taken to guarantee the public's right to reliable and accessible information on nuclear security as defined in Article L. 591-1".

Article L. 591-2 of the Environment Code sets out the role of the State in nuclear security and provides that it defines nuclear security regulations and implements the controls necessary for their application.

This article specifies that the State "shall ensure that nuclear safety and radiation protection regulations, and its control, are evaluated and improved, if necessary, taking into account experience gained in the course of operation, lessons learned from nuclear safety analyses carried out for nuclear installations in operation, technological developments and the results of research into nuclear safety, if available and relevant". In accordance with Article L. 125-13 of the Environment Code, "the State shall ensure that the public is informed of the risks associated with nuclear activities as defined in the first paragraph of Article L. 1333-1 of the Public Health Code and their impact on the health and safety of persons and the environment". The general principles applicable to nuclear activities are mentioned successively in Articles L. 591-3 and L. 591-4 of the Environmental Code.

Chapter V of Title II of Book I of the Environment Code deals with public information on nuclear safety.

Chapter II of Title IV of Book V of the Environment Code sets the framework for the management of radioactive materials and waste. It requires plants operators to set aside provisions for the management of their waste and spent fuel and the dismantling of their facilities.

Chapter II of Title IX of Book V of the Environmental Code establishes the ASN, defines its general mission and attributions and specifies its composition and operation.

Other codes or laws containing provisions specific to nuclear activities:

The Labour Code defines specific provisions for the protection of workers, whether salaried or not, exposed to ionizing radiation.

Finally, the Defence Code contains various provisions relating to protection against malicious acts in the nuclear field or the control of nuclear activities and facilities of interest to defence.

4.2.2.2. International standard for radiation protection

The legal framework specific to radiation protection finds its source in norms, standards or recommendations established by various international bodies. In particular, we can cite:

- International Commission on Radiological Protection (ICRP), a non-governmental organization made up of international experts from various disciplines, which publishes recommendations on the protection of workers, the population and patents against ionizing radiation, based on the analysis of available scientific and technical knowledge, particularly that published by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR).
- International Atomic Energy Agency (IAEA), which publishes and regularly revises "standards" in the fields of nuclear safety and radiation protection.
- International Organization for Standardization (ISO, International Standard Organisation), which publishes international technical standards that are references in the field of radiation protection.
- At the European level, the Euratom Treaty, defines the procedures for drawing up Community provisions on protection against ionizing radiation and specifies the powers and obligations of the European Commission with regard to their application.

4.2.3. Bases of French Regulation for nuclear plants

4.2.3.1. International conventions and standards

IAEA develops, reference texts called "Safety Standards", describing safety principles and practices. They cover the safety of installations, radiation protection, the safety of waste management and the safety of transport of radioactive substances. Although these documents are non-binding, they nevertheless constitute references that very largely inspire the national legal framework.

Several legislative and regulatory provisions relating to BNIs (Basic Nuclear Installation) are derived from or incorporate international conventions and standards, in particular those of the IAEA [10].

4.2.3.2. European Community texts

Several community texts are applicable to BNI. The most important of these are detailed below.

- **Euratom Treaty:**

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

- **Directive of June 25, 2009 establishing a Community framework for the nuclear safety of nuclear installations, amended by Directive 2014/87/Euratom of July 8, 2014:**

This Directive establishes a Community framework for nuclear safety and paves the way for a common legal framework in the field of nuclear safety for all Member States.

This Directive defines the fundamental obligations and general principles in this field. It strengthens the role of national regulatory bodies, contributes to the harmonization of safety requirements between Member States for the development of a high level of safety of installations and promotes transparency on these issues. It includes requirements in the areas of cooperation between safety authorities, including the establishment of a peer review mechanism, staff training, control of nuclear installations and transparency to the public. As such, it reinforces the cooperative action between Member States. Finally, it takes into account the harmonization work carried out by the Western European Nuclear Regulators Association (WENRA).

The 2014 amendment made the following substantial improvements:

- Concepts convergent with those of the IAEA (incident, accident, etc.);
- Emphasis on the principles of "defense in depth" and "safety culture";
- Clarification of responsibilities for monitoring the safety of nuclear installations;
- Safety objectives for nuclear installations directly derived from the safety standards used by the WENRA association;
- Reassessment of the safety of each nuclear installation at least every ten years;
- Implementation, every six years, of examinations by European counterparts on specific safety topics, based on the principle of resistance tests conducted after the Fukushima accident;
- Obligation for the operator of an installation and the safety authority to inform the public and stakeholders.

4.2.3.3. French Law Text: Environmental Code

The provisions of Chapters III, V and VI of Title IX of Book V of the Environment Code provide the basis for the authorization and control regime for BNIs.

The legal regime for BNIs is said to be "integrated" because it aims at preventing and controlling all the risks and nuisances that a BNI is likely to create for people and the environment, whether or not they are of a radioactive nature.

The provisions of Chapter II of Title IV of Book V of the Environment Code establish a coherent and integrated legislative framework for the management of all radioactive waste.

Chapter III of Title IX of Book V of the Environment Code (regulatory part) defines the framework within which procedures relating to BNI are conducted and deals with the entire life cycle of a BNI, from the definition of

safety options, its creation authorization and commissioning, to its final shutdown and dismantling, and then its decommissioning. It sets out the relationship between the Minister in charge of nuclear safety and the ASN in the field of BNI safety.

These provisions define the procedures applicable for the adoption of general regulations and the taking of individual decisions relating to BNIs. It also defines the specific conditions for the application of certain administrative regimes within the perimeter of BNIs.

Chapter VI of this Title V of Book V sets out the modalities for the application of the law with regard to inspections, police measures and administrative and penal sanctions.

The provisions relating to the right to information, transparency in nuclear matters are contained in section 2 of Chapter V of Title II of Book I of the Environment Code.

4.2.4. General Technical Regulations

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

4.2.4.1. "INB order"

In 2012, the French government implemented the order of February 7, 2012 (NBI Order) [11] setting the general rules relating to basic nuclear installations, known as the "INB order".

It defines the requirements applicable to basic nuclear installations for the protection of the interests listed by law: public safety, health and hygiene, protection of nature and the environment. These requirements are applicable to the design, construction, operation, decommissioning, dismantling, maintenance and monitoring of BNIs.

The order recalls the principles:

- Protection of all the interests mentioned in Article L. 593-1 of the Environmental Code (public safety, health and hygiene or the protection of nature and the environment), beyond the mere prevention of accidents;
- Reaching, taking into account the state of knowledge, practices and vulnerability of the environment, of a level of risk as low as possible under economically acceptable conditions;
- "Graded approach", i.e. the graduated nature of the requirements and control which must be proportionate to the issues at stake.

The order also recalls "the priority given to the protection of the above-mentioned interests, first and foremost through the prevention of accidents and the limitation of their consequences in terms of nuclear safety".

4.2.4.2. ASN regulatory decisions

In application of Article L. 592-20 of the Environment Code, the ASN may take regulatory decisions of a technical nature to specify decrees and orders issued in matters of nuclear safety or radiation protection.

Among these ASN decisions, some of them concern particularly the case of modifications on a nuclear power plant and others will have to be applied to complete the modification file. The most important in sCO2-4-NPP are the following:

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

4.2.4.3. Basic Safety rules and ASN guides

The Basic Safety Rules (BSR) are recommendations that specify safety objectives and describe practices that the ASN considers satisfactory.
As part of the current restructuring of the general technical regulations applicable to basic nuclear installations, the BSR are gradually being replaced by ASN guides.
The collection of ASN guides currently includes more than thirty guides of a non-prescriptive nature that are intended to:

- Accompany certain regulatory changes, by promoting knowledge of the regulations and the expectations of the ASN;
- Propose the methods for achieving the objectives set by the texts;
- Present the methods and best practices resulting from feedback from significant events.

The ASN guide that interests us most in the sCO2-4-NPP project will be Guide n°22 (ASN) [12]. It presents ASN and IRSN recommendations for the "Design of Pressurized Water Reactors" (PWR).

This guide, whose major field of application is the design of new PWRs, deals mainly with the prevention of accidents and incidents of a radiological nature and the limitation of their consequences, but also addresses other aspects related to the management of risks of a non-radiological nature or the inconveniences that will result from the operation of the installation. It recalls the regulatory requirements to be taken into account at the design stage and presents the recommendations that make it possible to comply with them, both on technical aspects and on relevant organizational and human factors, aimed at protecting the interests mentioned in the first paragraph of Article L. 593-1 of the Environmental Code. These recommendations relate in particular to what should be the objectives, requirements and criteria that the industrialists set for the design of the installation in order to respect the general objectives set by the regulations.

Although the recommendations in this guide are not prescriptive in nature, compliance with them is considered to be a satisfactory way of meeting the regulatory requirements concerning nuclear safety.

5. Safety general approach

5.1. Czech Republic

5.1.1. Safety analysis

Necessary analyses must be performed during the design and operation of the Nuclear Power Plant. Safety analyses are a key part of the safety assessment process. Safety Assessment is a systematic process that is performed throughout the life of a unit to ensure that all relevant safety requirements are met. Safety analyses should:

- demonstrate the fulfilment of basic safety functions by the intervention of safety systems and operators of nuclear facility,
- consider the occurrence of the most-limiting single failure of safety systems with an active actuation,
- verify the effectiveness of the intervention of the safety systems in the least favourable conditions envisaged by the design after the considered PIE,
- demonstrate that the influence of initial parameters and calculation procedures uncertainties was taken into account in the NPP evaluation,
- be performed both with deterministic and probabilistic approach.

Safety analyses must be performed at the stage of:

- design,
- commissioning,
- operation,
- modifications, upgrades or other changes to the design,
- periodic safety review,
- life extension.

The safety analyses used in the plant documentation must be subject to a quality assurance program and must be independently verified to an appropriate extent.

The requirements are based on:

- IAEA Safety Standards, Deterministic Safety Analysis for Nuclear Power Plants, Safety Standards Series No. SSG-2, Rev. 1 (2019) [13],
- 329/2017 Sb. SÚJB of 26 September 2017 on requirements for the design of a nuclear facility [14],
- 162/2017 Sb. SÚJB of 26 September 2017 on requirements for safety assessment pursuant to the Atomic Act. [15]

The main safety assessment of the project can be divided into:

- deterministic safety analyses (DSA), best estimate and conservative,
- probabilistic safety analysis (PSA).

5.1.1.1. Deterministic safety analysis

Deterministic safety analyses must prove the acceptability of the consequences of any PIE (postulated initiating event) in terms of nuclear safety, radiation protection, technical safety and management of a radiation emergency.

Deterministic safety analyses must assess:

- the ability of the nuclear installation to ensure compliance with the principles of safe use of nuclear energy,
- the resilience, reliability and efficiency of safety systems and other systems, structures and components affecting nuclear safety under the accident conditions,
- the ability of operators to ensure the performance of the basic safety functions of a nuclear installation by undertaking the required activities.

Deterministic safety analyses are one of the key demonstrations of the safety of a given nuclear installation as a whole. Their deterministic character is given both by a deterministically defined set of PIEs and by deterministically applied assumption of a most-limiting single failure.

The most important safety objectives are determined as acceptance criteria for safety analyses, given for individual categories of initiating events. The criteria are usually different depending on the frequency of occurrence of the given initiating event, more restrictive criteria are set for events with a higher frequency of occurrence. The criteria are divided into radiation (ensuring the acceptability of radiation consequences) and technical (ensuring the functionality of physical barriers against the release of radioactive substances). Radiation acceptance criteria are set by the regulator. The technical acceptance criteria are set by the designer of the nuclear power plant and must be accepted by the regulator.

Deterministic analyses of Anticipated Operational Occurrences (AOO) and Design Basis Accidents (DBA)

Analyses of anticipated operational occurrences and design basis accidents are performed by conservative computational analyses using realistic codes. Conservatism can be ensured by choosing input data, boundary and initial conditions (including possible uncertainties in parameter settings) and accepting additional limiting assumptions (e.g. application of single failures). In this case, the input data are selected in such a way that the plant conditions are as unfavourable as possible for the chosen scenario. For each initiating event analysed, it must be determined which acceptance criteria are relevant and which physical parameters are limiting.

Deterministic analyses for design extension conditions (DEC)

Analyses of DEC are performed by realistic calculations, given that it is important to get the best possible approximation to the actual response of the power plant. Realistic assumptions about the configuration and behaviour of the equipment and mitigated acceptance criteria compared to design accident analyses can be used.

For DEC B (DEC with postulated severe fuel damage) analyses, the appropriate assumptions are made according to the individual sets of measures for mitigating the consequences of severe accidents.

5.1.1.2. Probabilistic safety analysis

The probabilistic safety assessment must include two levels of analysis:

- Level 1 of a probabilistic safety assessment is performed to identify the sequence and likelihood of events that could lead to damage to the nuclear fuel or other SSC resulting in radioactive releases; The results of PSA level 1 analyses should be used as one of the tools for the selection of new or modified SSC configurations,
- Level 2 of the probabilistic safety assessment includes the analysis of scenarios identified in PSA Level 1 as leading to damage to nuclear fuel or other important SSC; a quantitative assessment of the resulting phenomena is performed.

The basis for the evaluation of probabilistic safety assessment (PSA) is a complex model of a given nuclear facility structured into event trees and fault trees. Unlike deterministic safety analyses, PSA studies are not limited to analyses of a defined PIE spectrum, but concern all realistically possible emergency situations and their development scenarios, including situations with simultaneous multiple component and system failures where important safety features are not fully available or not available at all.

The numerical result of these probabilistic analyses in the final phase is usually the probability of core melting (PSA level 1), and the probability of release of radioactive substances into the environment (PSA level 2).

The aim of the analyses is to demonstrate that no SSC, external hazard or PIE increase the risk of a radiation accident disproportionately and that the risk of breaking physical safety barriers is acceptably low.

Probabilistic safety analyses of nuclear power plants must be performed according to IAEA SSG-3 in order to:

- Provide a systematic analysis demonstrating that the project will be in line with the general safety objectives,
- demonstrate that a design is balanced in such a way that no hazard or PIE makes a disproportionately high contribution to the overall risk,
- ensure that, after a small deviation of one of the parameters, there is no rapid deviation from normal conditions resulting in an emergency ("Cliff edge effect"),
- provide an assessment of the probability of DEC B conditions and an assessment of the risks of large and early releases of radioactive substances,
- provide an assessment of the probability and consequences of external risks, in particular those unique to the site,
- identify SSC for which design improvements or modifications of operating procedures could reduce the likelihood of severe accidents or mitigate their consequences,
- assess the adequacy of procedures for dealing with emergencies,
- verify compliance with the probabilistic objectives of the project, if any.

5.1.2. Defence in depth

Defence in depth (DiD) is the basic safety principle of nuclear facilities.

In general, the design of a nuclear installation must set out requirements for systems, structures and components and procedures to implement safety functions to protect the integrity and functionality of

physical safety barriers at various levels of DiD so that these barriers prevent a radiological release. In accordance with law 263/2016 Coll. [16] and decree 329/2017 Sb. [14] and with international requirements (IAEA SSR-2/1 rev.1 [17] , WENRA Safety Reference Levels for Existing Reactors [18]) compliance with the basic objectives of nuclear safety must be ensured. The DiD strategy must also cover the role and influences of the human factor and must be included in the procedures, regulations and instructions for the normal and emergency conditions of the NPP.

DiD takes into account the links between all SSC (with and without influence on nuclear safety) and the human factor in order to ensure plant safety. It has two main goals:

- accident prevention,
- mitigation of the accidents' consequences.

It is a fundamental principle and philosophy of safety applied to Nuclear Power Plants, which includes all activities related to the location studies, design, construction, commissioning, operation and decommissioning.

Requirements for the DiD principle must be ensured for all technical activities related to the use of nuclear energy in connection with the NPP project:

- the creation of a series of back-up physical safety barriers between radioactive substances and the vicinity of a nuclear installation,
- systems, structures and components and procedures for the application of safety functions to protect the integrity and functionality of physical safety barriers at various levels of in-depth protection,
- prevention of the occurrence of a radiation emergency by means of physical safety barriers.

Proper in-depth protection ensures that no single technical, human or organizational failure can lead to significant adverse effects and that combinations of failures that could lead to significant adverse effects are very unlikely.

Other necessary requirements for the DiD principle are:

- the greatest possible robustness of physical barriers as well as the instrumentation and control of individual levels of DiD,
- the greatest possible degree of independence of DiD levels.

The requirements for robustness and degree of independence complement each other and support each other in terms of ensuring the effectiveness of the protective barrier functions.

Below are the goals for each level of DiD:

- the aim of the DiD1 level is to prevent deviations from normal operation and to prevent system failures. This means that NPPs must be properly and conservatively designed, constructed, operated and maintained in accordance with the required level of quality and engineering practices, such as the redundancy, independence and diversity,
- the aim of the DiD2 level is to detect and correct deviations from normal operation in order to prevent the anticipated operational occurrences from developing into emergency conditions, i.e. to ensure that abnormal operation is controlled and leads to transition to normal operation. This level of

protection requires the establishment of adequate systems and the definition of operating procedures that prevent or mitigate such deviations,

- the aim of the DiD3a level is to ensure the fulfilment of basic safety functions by activating safety and safety-related systems in order to limit the possible consequences of DBA, if previous levels of protection did not prevent DBA conditions or more serious PIEs leading to DBA. This requires that these systems can first provide the stable and controlled and then the safe conditions and maintain at least the integrity of one barrier against the release of radioactive substances,
- **the aim of the DiD3b level is to manage DEC A and prevent the transition to DEC B. The fulfilment of basic safety functions is ensured by additional measures, including certain safety systems (technical means) and DEC management procedures (organizational measures),**
- **the aim of the DiD4 level is to mitigate the consequences of the DEC B events; the means of this level should prevent the further development of the event and ensure that there is no early or large radiation accident. The most important task of this level is to ensure the containment integrity, to prevent its failure and long-term heat removal from the containment;**
- The aim of the DiD5 level is to limit and mitigate the radiation consequences of potential releases of radioactive materials through an emergency response (emergency planning zones and appropriate procedures). It is implemented mainly outside the NPP.

5.1.3. NPP normal and accident conditions

For nuclear facilities with a reactor (NPP unit including auxiliary equipment) there is a classification of the operating conditions, according to the requirements of decree 329/2017 Coll. [14] the following states are considered:

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

A conservative design of a nuclear power plant (and its units) must ensure that conditions and events outside the design framework (if they lead to early or large releases of radioactivity) are practically eliminated or pose an acceptable level of risk.

DEC A (DEC without fuel damage)

Managing DEC A prevents worsening the plant conditions into a severe accident (DEC B) and reduces potential radiation consequences.
The transients classified as DEC A can be further divided into two groups:

- local complex transients typical for one block, e.g. the abnormal conditions or design basis accident with subsequent failure of one or more safety systems. In these cases, it is possible to use other systems of the given unit as well as the neighbouring block and external means (e.g. electricity system),

- **global complex transients that may affect more units or the entire NPP. These are typically SBO, LUHS and combinations of the above. Mutual assistance of blocks is limited or excluded in these cases. The same applies to external means. The availability of mitigating measures strongly depends on the type of phenomenon (hazard), its impact on NPP units, the surrounding area (infrastructure) and the time factor (infrastructure renewal time). In these cases, the procedures include the deployment of DIV and ALT systems, which cooperate with still available safety and safety-related systems (BS, SSB). As part of ensuring the ability of mitigate the DEC A without significant damage to nuclear fuel, the management of postulated initiating events (PIEs) and scenarios for beyond design conditions must be technically addressed. Design extension conditions without significant fuel damage are solved within the level of protection to the depth of DiD 3b. These kinds of accidents are particularly important from the point of view of sCO2-4-NPP project.**

In current state of VVER reactor operated in Temelín NPP there are several means (DiD levels) to cope with the loss of offsite power. The first and preferred method of maintaining the decay heat removal from the core would be the use of main emergency Diesel Generators. If they fail, the second choice is the activation of auxiliary Diesel Generators. In case that they are also unavailable, there is a possibility to depressurize the secondary circuit of the reactor (steam generators) and use the fire trucks to provide the water for cooling.
The sCO2 cooling loop would be an alternative method of residual heat removal, providing high reliability. According to the design, it would be use in the case of station black-out, e.g. if all the onsite power sources fail.

5.1.4. Acceptance criteria

The determination of these criteria is based on the IAEA TECDOC-1791 (Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants) [19]. The specific acceptance criteria for radiation protection in Czech Republic are given in the Annex to the SÚJB letter no. SÚJB / RO / 9326/2019, (The acceptance criteria for radiation protection submitted by SÚJB for Dukovany and Temelín NPPs) [20].

Table 1. Acceptance criteria for different NPP conditions in Czech Republic [5,20]

Plant conditions and frequency [1/reactor-year]	Basis criterion	Criteria regarding integrity of safety barriers	Criteria regarding radiation safety	Criteria regarding reactivation (or decommissioning) of the plant
Normal operation	Maintaining the normal operation and electricity generation; avoiding the abnormal conditions	No threat to the safety barriers	-	Normal operation is maintained (including start-up, shut-down, maintenance, refueling)
Anticipated operational occurrence (AOO), $>10^{-2}$	Control of abnormal conditions; prevention of unnecessary safety systems operation	No threat to the safety barriers	0.1 mSv/1y	After managing the event, eliminating its causes and consequences the normal operation is restored
Design basis accidents (DBA), $10^{-2} – 10^{-6}$	Control and mitigation of event; transition to stable and safe condition with the use of safety systems and operator action	No subsequent failure for reactor cooling, maintaining containment integrity, limited core damage – active reactor core must remain coolable and removable	1 mSv/1y	Possible return to the normal operation is dependent of the extent of the damage, elimination and liquidation of the consequences
Beyond design basis accidents (BDBA) / DEC A, $10^{-4} – 10^{-6}$	Mitigating DEC A and prevention of DEC B (severe accident); limiting the radiation consequences	No subsequent failure for reactor cooling, maintaining containment integrity, limited core damage – active reactor core must remain coolable and removable	10 mSv/2days	Possible return to the normal operation is dependent of the extent of the damage, elimination and liquidation of the consequences
Severe accidents (SA)/ DEC B, $< 10^{-6}$	Mitigating the consequences of severe accident, preventing the large/ early radioactive releases	Maintaining containment integrity	100 mSv/event	Return to the normal operation is not possible; necessary decommissioning
Unlikely Extreme events	Mitigating the consequences of radiation releases	Not applicable (containment integrity not maintained)	Not specified	Return to the normal operation is not possible; necessary decommissioning

5.2. France

In this paragraph, we describe the approach adopted by French operators to develop and submit the construction and start-up of a nuclear power plant. We also present the regulatory texts to which operators must comply. This approach will be the same to be applied for the sCO2 system, which will constitute a significant modification for the operator and will therefore be subject to authorization by the safety authorities.

5.2.1. Safety Objectives to be achieved

Article 1.2 of the "INB Order" [11] states that the operator of a BNI must ensure that the provisions adopted for the exercise of the activities mentioned in Article 1.1. (i.e. the design, construction, operation, decommissioning, dismantling, maintenance and surveillance of basic nuclear installations) make it possible to achieve, taking into account the state of knowledge, practices and environmental vulnerability, a level of risks and inconveniences [...] as low as possible under economically acceptable conditions.

These provisions, which apply for the entire lifetime of the plant, must therefore also be complied with when applying for a modification. For this reason, we recommend that the same approach be applied to the design of the sCO2 system as to the design of a power plant.

According to IAEA safety standard SF-1 (IAEA, 2006) [10], the fundamental safety objective is to protect people and the environment against the effects of ionising radiation. This fundamental safety objective of the IAEA as well as the Fundamental Safety Principles that must be applied to achieve it must be retained in all stages of the design of the sCO2 system.

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

In the following table, we summarise the declination adopted by EDF for WENRA's safety objectives. This declination has been used in the preparation and design of the latest French power plant concepts (EPR and EPR New Model).

Table 2. WENRA safety objectives in French nuclear regulations

WENRA Objective	Description	Explanation
O1	Normal operation and prevention of incidents and accidents	Reduce the frequency of incidents by enhancing the ability to maintain the facility within the normal operating range. Reduce the risk of the normal situation deteriorating into an accident situation by increasing the facility's ability to control incidents.
O2	Accidents without core or fuel meltdown	Ensure that accidents without core or fuel meltdown do not result in off-site radiological impact or only minor radiological impact. Reduce under economically acceptable conditions: • the core or fuel meltdown frequency taking into account all types of credible failures and hazards, as well as plausible cumulative events; • the release of radioactive substances from all sources. Pay particular attention to site selection and design to reduce the impact of external hazards.
O3	Accidents with core or fuel meltdown	Reduce potential radioactive discharges into the environment, including in the long term, in accordance with the qualitative objectives below: • core or fuel meltdown accidents that would lead to large or early releases must be made extremely unlikely with a high degree of confidence; • or other core meltdown accidents, design provisions must be made to ensure that only spatially and temporally limited protective measures are required for the public and that there is sufficient time to implement these measures.
O4	Sufficient independence between levels of defence in depth	Ensure sufficient independence between the different levels of defense in depth, in particular by resorting where necessary to diversification, to ensure an overall strengthening of defense in depth.
O5	Interfaces between nuclear safety and security	Ensuring that nuclear safety and nuclear security arrangements are designed and implemented in an integrated manner.
O6	Radiation protection and waste management	Reduce, by design provisions, for all normal reactor operating states as well as during the final shutdown and dismantling of the reactor: • the doses received by workers; • discharges of radioactive substances into the environment; • the quantity and activity of radioactive waste.
O7	Safety management	Ensuring effective safety management from the design stage: • Safety management set up by the operator for the entire duration of a new nuclear installation project; • All organizations involved in the site selection, design, construction, operation and decommissioning of a new installation demonstrate the ability of their staff to address nuclear safety issues in relation to their work and their role in the safety organization.

5.2.2. Safety requirements

French operators opt for a deterministic approach based on the principle of defense in depth supplemented by a probabilistic verification for their safety approach at the design stage. The design of the sCO2 system and the modification request will have to follow the same approach. In this paragraph, we will first present the several requirements for the safety of the plant.

5.2.2.1. Defense in depth

The principles adopted by the French operators for defense in depth are the same as those for the Czech plants. We therefore invite the reader to refer to paragraph 5.1.2.

5.2.2.2. Basis Safety Rules

According to the “INB Order”, the basis safety functions, which must not be degraded by the sCO2 system are the following:

- Control of nuclear chain reactions;
- Evacuation of thermal power from radioactive substances and nuclear reactions;
- Containment of radioactive substances;
- Protection of people and the environment against ionising radiation.

Compliance with the first three fundamental safety functions makes it possible to ensure the fourth.

These functions must be ensured for the different levels of defence in depth. To this end, for each operating condition, the Structures, Systems and Components (SSCs) required to perform the fundamental safety functions are defined.

5.2.2.3. Barriers

The function of containment of radioactive substances is ensured by the interposition, between these substances and people and the environment, of one or more successive barriers sufficiently independent of each other.

Thus, three containment barriers are provided between the fuel located in the reactor core and the environment:

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

5.2.2.4. Safety general approach at design stage

As already stated, French operators opt for a deterministic approach based on the principle of defense in depth supplemented by a probabilistic verification for their safety approach at the design stage.

It requires determining the events likely to affect a barrier or a safety function and then defining the measures to be implemented on the installation to prevent these events and limit their consequences if they are

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

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

Consistent with the WENRA and IAEA texts, the deterministic design approach is articulated around two distinct domains:

- Reference design domain for sizing SSCs for prevention and mitigation of reference operating conditions,
- Extended design domain for the sizing of SSCs for the mitigation of operating conditions with multiple failures as well as for the mitigation of operating conditions with core meltdown.

The dimensioning of SSCs in these two areas is completed by taking into account, at the design stage, internal and external hazards and plausible accumulations of events, which can lead to:

- Sizing additional SSCs necessary for the prevention or mitigation of these hazards;
- Defining additional requirements applicable to the design of SSCs.

In the remainder of the paragraph, we present a summary of the main elements that feed the reflexion for the general safety approach for a nuclear power plant in its entirety and not only applied to the sCO2 system. From this general approach, we can deduce the constraints relating to the sCO2 system and detail them in Chapter 6.

The deterministic and probabilistic part of the safety general approach applied to the sCO2 system will be detailed in deliverable D3.3. They will constitute the bases for the design of the components of the sCO2 system.

5.2.3. NPP normal and accident conditions

The reference design area includes the reference operating conditions. These reference operating conditions are defined so as to be representative of the consequences of single initiating events (SIEs) due to simple internal failures that may occur in the installation.

The reference operating conditions DBC are classified into five categories:

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

DBC1 operating conditions represent transients that may occur frequently or regularly during normal operation. These conditions are not subject to the nuclear safety demonstration but contribute to the definition of the loading conditions of the pressure boundary of the primary and main secondary circuits. They also make it possible to dimension the regulations and any limitations which keep the main boiler parameters within an authorised range. They are associated with levels 1 and 2 of the defence in depth (see paragraph 5.1.2.).

The DBC2-4 operating conditions are used to dimension the Structures, Systems and Components (SSCs) for the mitigation of these incidents and accidents (in particular the so-called "back-up" systems) including the SSCs corresponding to their support functions (electrical source, control-command, cold source, ventilation, etc.). They are associated with level 3a of defence in depth and ensure that the design and dimensioning results in a robust and stable boiler with respect to the major families of phenomena and accidents that may affect it. In the reference design field, the dimensioning is based on conservative rules, hypotheses and methods. These are the DBC categories for which the sCO2 system has been developed.

The safety objective associated with the DBC reference operating conditions is the absence of core meltdown and zero or minor radiological impact. The radiological consequences of DBC2 operating conditions are compared to the dose limit levels defined for incidents while the radiological consequences of DBC3-4 operating conditions are compared to the dose limit levels defined for accidents without core meltdown or fuel meltdown.

The extended design domain includes multiple failure operating conditions as well as core meltdown operating conditions.

5.2.3.1. Operating conditions with multiple failures

The **DEC A** multiple failure operating conditions are defined to handle more complex situations than those of the reference sizing. They are representative of the consequences of multiple internal failures by common cause. They are associated with level 3b of defense in depth.

These operating conditions allow the design or verification of diversified design provisions known as "DEC A provisions".

This dimensioning uses rules and hypotheses that are less penalizing than those used in the reference design domain.

The safety objective associated with DEC-A multiple failure operating conditions is the absence of core meltdown and no or minor radiological impact. The radiological consequences of DEC-A operating conditions are compared to the dose limits defined for accidents without core meltdown or fuel meltdown.

5.2.3.2. Operating conditions with core meltdown

The operator must establish a core meltdown operations list DEC B to dimension complementary design provisions known as "DEC B provisions" to ensure in particular the containment of radioactive substances in the event of a core meltdown accident. These operating conditions are associated with level 4 of defence in depth.

In this case, the design uses less penalizing rules than for levels 3a and 3b of defence in depth, realistic hypotheses and physical methods that take into account the uncertainties associated with the modelled phenomena.

The safety objective associated with DEC B core meltdown operating conditions is that public protection measures remain limited in space and time. The radiological consequences of the DEC B operating conditions are compared to the dose limits defined for core meltdown accidents.

5.2.3.3. Protection against internal hazard

The different types of internal hazard listed below are to be distinguished:

- Internal hazards constituting the initiator:
  - internal hazard that does not induce a DBC type event;
  - internal hazard inducing a DBC event, for example DBC2;
- Internal hazards resulting from an operating condition:
  - internal hazard directly linked to a DBC3 or DBC4 operating condition: some DBC3-4 accidents themselves constitute an internal hazard, such as Primary Coolant Loss Accidents (PLA) or Steam Pipe Ruptures (SPR);
  - Internal hazard resulting from a DEC A or DEC B operating condition.

The internal hazards to be taken into account, in coherence with the "INB order":

- failure of pressure equipment (pipes, tanks, pumps and valves);
- flooding of internal origin;
- the emission of projectiles, in particular that induced by the failure of high-energy or rotating equipment;
- collision and falling loads;
- fire of internal origin;
- explosion of internal origin;
- electromagnetic interference;
- emission of hazardous substances.

The design approach implemented for the protection of Structures, Systems and Components (SSCs) against the effects of internal hazard consists, firstly, in defining design provisions aimed at detecting and preventing the occurrence of hazard and, secondly, in spatially limiting the effects of hazard by physical (barriers) or spatial (distance) separation. The possible consequences of internal hazard must be limited and the failure of equipment as a result of internal hazard must not jeopardize compliance with the safety objectives. In particular, if an internal attack induces a DBC type event, the SSCs used in the nuclear safety demonstration to manage the consequences must remain available in sufficient number despite the consequences of the attack and in compliance with the rules for the study of internal attacks.

5.2.3.4. Protections against external hazard

Defense in depth applied to protection against external hazards is based above all on the choice of a suitable site, taking into account in particular the risks of natural or industrial origin weighing on the installation of a

nuclear power plant. As the sCO2 system is a modification of an existing power plant or can be integrated into a future power plant, this site selection condition is either already respected (for an existing power plant) or will be studied (for the site selection of a future power plant).

The equipment can be designed to support itself the case of load or accumulation of loads, or it can be protected by the structure that houses it or benefit from the principle of geographical separation if it is designed to be redundant.

The list of events of external hazards to be retained, in coherence with the INB Order, is as follows:

- risks induced by industrial activities and communication channels (explosions, aircraft crashes);
- the earthquake;
- lightning and electromagnetic interference;
- meteorological or climatic conditions (cold or hot air temperatures, cold or hot water temperatures, snow, winds and tornadoes, induced projectiles);
- floods originating outside the perimeter of the installation (overflow of the cold source, rain, etc.);
- external hazards affecting the main safety cold source.

Within the framework of the sCO2-4-NPP project, the levels associated with external hazards will have to be established in accordance with:

- RFS 2001-01 (earthquake),
- ASN guide n°13 (external floods),
- RFS-I.2.a. (aircraft crash),
- AFNOR NF EN 62305-1 standard (lightning risk and electromagnetic interference).

5.2.4. Application to sCO2 System

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

It must:

- Be part of one of the levels of defense in depth set up by the operator,
- Be developed in order to improve the response of operators to the second fundamental function and thus participate in the fourth function,
- Do not interfere with containment barriers. Particular attention will be paid to its installation in relation to the containment in order not to weaken it.
- Be developed following the same principles of the general operator safety approach.

The details of the general safety approach applied to the sCO2 system will be detailed in deliverable D3.3.

In order to compile the modification request file, the operator will have to implement a safety general approach including the sCO2 system. This implementation will be detailed in deliverable D3.3 of the project, as it is a step that will identify the design bases, standards and analyses required for the qualification of the sCO2 module.

6. Requirements for the SSC

6.1. Czech Republic

The basics for the Temelin NPP project was the Soviet design from 1986 with all its amendments. Later, with growing experience in NPP operation and two severe accidents (TMI, Chernobyl) there was further development of legislation and implementation of requirements to the design. Many adjustments and improvements were implemented by the authors of the design. Another big milestone for the development of the project was the possibility of applying Western technology after year 1989, e.g. improvement of I&C, fuel arrangement, new backup power sources etc.

The corresponding goals and principles of the project are summarized in the document "Safety Philosophy for NPPs operated by ČEZ, a. s., ÚJV-EGP, 06/2009" [21]. The document summarized the improved design, the requirements set by Czech legislation and international standards (IAEA etc.).

Another significant development of requirements for the Temelin NPP occurred after the accident in Fukushima NPP. Performing of the stress tests required for the European NPPs have resulted in national action plans for the implementation of measures to increase safety, taking into account Design Extension Conditions (DEC) as well as unlikely extreme events. Gradually implemented improvements of the project and safety procedures at the Temelin NPP were mainly based on international and European regulations (IAEA, WENRA), which arose or were amended in this context. The next stage of development was the new Czech nuclear legislation (new Atomic Act, 263/2016 [16] and its implementing decrees, especially 329/2017 [14]).

6.1.1. Classification of safety functions

The main assumption of the Temelín NPP project in terms of safety is to fulfil the following fundamental safety functions in all states of the plant. This concept is in line with the Atomic Act of the Czech Republic 263/2016, the IAEA standard SSR-2/1 rev.1 [17] and the WENRA report - Safety Reference Levels for Existing Reactors [18].

The fundamental safety functions (ZBF = Základní Bezpečnostní Funkce) for nuclear facilities with a nuclear reactor are:

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

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

- 329/2017; SÚJB on requirements for the design of a nuclear facility [14],
- 408/2016; SÚJB on requirements for the management system [22],
- IAEA SSR-2/1 rev.1 Safety of Nuclear Power Plants: Design, 2016 including IAEA - TECDOC-1791 (2016), IAEA SSG-30 (2014), IAEA SRS 46 (2005), IAEA SSR-2/1 (2016),

- Safety philosophy for NPPs operated by ČEZ, a. s., Archive number EGP 5010-F-090427, 2009 (also includes SÚJB decrees 195 and 132 etc.),
- SÚJB safety instructions (JB-1.0, 11/2011 [23], JB-1.7, 12/2010 [24]).

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.

6.1.2. Classification of structures, systems and components (SSC)

According to the "new legislation" (263/2016, Atomic Act and its implementing decrees in accordance with § 8 in 329/2017; SÚJB on requirements for the design of a nuclear facility), the basic safety classification is updated as follows.

Within the framework of ensuring the fulfilment of safety functions in accordance with their categorization, the design of a nuclear facility must divide SSC into:

- SSC not affecting nuclear safety,
- SSC with an impact on nuclear safety, which are not selected facilities
- selected equipment (SSC), with the impact on nuclear safety, namely:
  - selected devices that are not safety systems,
    - systems related to nuclear safety, which are not safety systems,
    - various safety means and measures, which are not safety systems,
  - safety systems.

Table 3. SSC classification in Czech Republic [5]

Basic safety classification All plant SSC							Other classifications			
Important / not important to safety	**Impact/ no impact on safety**	**Defence in depth**	**Functions**	**Type of device**	**Group of SSC**		**Importance for safety**	**Safety classes**	**Seismic categories**	**Other**
Not important to safety; not VZ	No impact on safety	No	PF N	ZPP	SNB		PF, VF, BF; more detailed categorization according to SSG-30 or CEZ methodology ME 0901	BT1, BT2, BT3, not classified	1a, 1b, 1c, NC	Electric power, instrumentation and control etc.
		Yes	PF DiD	ZPP	VyDiD					
	Impact on safety		VF	Alternative	ALT					
				ZPP	NEP					
Important to safety (selected SSC); VZ			BF (safety functions)	Various	VZ, but no BS	DIV				
				ZPP		SSB				
				ZPP	BS (safety systems)					

VZ - vybraná zařízení (selected equipment)
VyDiD – Významné z hlediska DiD (significant for Defence in Depth)
PF - provozní funkce (operational functions)
VF - Funkce s vlivem na jadernou bezpečnost (functions with impact to nuclear safety)
BF - bezpečnostní funkce (safety functions)
ZPP - základní projektové prostředky (basic design components)
SNB – Nedůležité z hlediska bezpečnosti (not important for nuclear safety)
ALT - alternativní prostředky (alternative components)
DIV - diverzní prostředky (various components)
SSB - systémy související s bezpečností (safety-related systems)
BS – bezpečnostní systémy (safety systems)
BT - bezpečnostní třída (safety class)

6.1.2.1. Safety classes

Next, the SSC with the impact on nuclear safety are divided into three safety classes (BT = bezpečnostní třída):

- Safety class 1 (BT1) includes SSC fulfilling the safety functions of category I., which is the passive function of the SSC in the pressure boundaries of the primary circuit. Selected devices belonging to the pressure boundaries of the primary circuit, the damage of which does not threaten the plant safety, do not have to be included in safety class 1.
- Safety class 2 (BT2) includes SSC fulfilling safety functions of category II. These SSCs are selected equipment fulfilling the passive function and selected equipment with guaranteed high reliability of fulfilling the active safety function. These selected devices are:
  - fuel rods cladding,
  - parts of the pressure boundaries of the primary circuit which do not belong to safety class 1; Selected devices belonging to the pressure boundaries of the primary circuit, the damage of which does not threaten the plant safety, do not have to be included in safety class 2,
  - containment vessel,
  - other selected devices fulfilling safety functions
- Safety class 3 (BT3) includes SSC not included in safety class 1 or 2, fulfilling safety functions of category III.

6.1.2.2. Seismic classification

According to the IAEA standards, seismic classification is performed according to the following principles.

Two levels are considered:

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

SSC are divided into the following seismic categories:

- Category 1a must retain full functionality, including integrity during and after the seismic event, up to the level of the maximum calculated earthquake (SL-2); Category 1a must contain the following SSC and their supporting structures:
  - SSC, the failure of which could directly or indirectly cause an emergency conditions of the unit,
  - SSC important for providing fundamental safety functions.
- Category 1b requires seismic resistance in the sense of maintaining mechanical integrity; partial malfunctions are possible up to the level of the maximum calculated earthquake (SL-2); Category 1b must include:
  - SSC not included in category 1a, necessary for the prevention or mitigation of emergency situations,
  - SSC, which are not included in category 1a, but represent a potential source of radiation,

  - SSC ensuring the accessibility of the site and necessary for the implementation of the evacuation plan.
- Category 1c requires seismic resistance only in terms of seismic interactions with other structures, systems or partial components of the equipment, partial malfunctions and mechanical integrity are possible up to the level of the maximum calculated earthquake SL-2; Category 1c must include:
  - SSC, which for some reason (e.g. collapse, seismicity) may interact with SSC category 1a or 1b.
- Category NC (not classified):
  - All SSC not included in categories 1a to 1c. The SSC design is carried out according to normative and technical documentation for common industrial buildings and equipment.

6.1.3. sCO2 system in SSC classification

According to Czech legislation, sCO2-4-NPP components would be classified as system fulfilling one of the **fundamental safety function (ZBF2),** namely the long-term heat removal from the containment vessel. It can be assigned to the **safety category III** and **seismic category 1a**.

For passive safety systems (such as sCO2 loop), the required standard is 2 × 100% redundancy. The 2 × 100% principle is based on a Russian technical design and is based on the IAEA recommendation (SAFETY SERIES No. 50-P-1 Application of the Single Failure Criterion [25]). Exceptions to the application of this rule must be justified by design and safety analyses. The passive safety systems rely on simple physical principles and do not require any external control or power supply to activate the safety function. Therefore, their reliability in terms of fulfilling the required safety function is significantly higher.

6.2. France

In accordance with the "INB order", SSCs classified as safety are part of the Important Elements for Protection (EIP).

The safety classification of SSCs ensures that they are designed, manufactured and monitored in operation with a level of quality corresponding to their importance in demonstrating nuclear safety.

The SSCs to be classified are mainly identified on the basis of the functions they perform. These functions are categorized according to their importance for safety.

The following figure shows the main steps to follow to classify SSCs and assign requirements to them:

- a functional analysis is used to categorize the functions according to the conditions under which they are required. This analysis is based on all the studies necessary for the demonstration of nuclear safety, in particular the studies of the DBC reference operating conditions and DEC, as well as internal/external hazard studies, and allows the identification of the functions necessary to meet all radiological safety objectives ;
- the SSCs required to perform a categorized function are identified and classified accordingly;
- requirements are associated with the classified SSCs in order to guarantee that they will be able to satisfy their safety function with the required level of performance and reliability.

6.2.1. Categorization of safety functions

Safety functions, including support functions, are categorized according to their importance, in 3 categories. The highest category is associated with a maximum level of requirements. The categorization of the functions is established in coherence with:

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

Safety Category 1 (Cat1):

- functions required to reach the controlled state of a reference operating condition DBC2-4;
- protection functions of system components required to achieve the controlled state under DBC reference operating conditions and having priority over Cat1 functions;
- functions required in reference operating condition DBC2-4 to ensure static containment of the zone in which the transient is initiated.

Safety Category 2 (Cat2):

- functions required during the first 24 hours after an initiator to reach and maintain the safe shutdown state in DBC reference operating condition;
- functions required in reference operating condition DBC2-4 before 24 hours to ensure static containment from a peripheral zone to the zone in which the transient is initiated;
- functions required in reference operating condition DBC2-4 before 24 hours to ensure dynamic containment (of a part) of a building.

Safety Category 3 (Cat3):

- functions required to limit the consequences of a DEC A multiple failure condition;
- functions required to limit the consequences of an operating condition with DEC B core meltdown including the information necessary to manage the associated accidental conduct;
- Limitation functions set up to prevent Automatic Reactor Shutdown (AAR) trips, in case of exit from the normal operating diagram;
- monitoring functions of the main parameters taken into account as initial conditions for the studies of operating conditions;
- functions designed to monitor the availability of Cat1 or Cat2 functions (except equipment required to perform periodic testing which may be Non-Classified (NC));
- monitoring functions delivering the information needed to verify the achievement and maintenance of the controlled state under operating conditions with DEC B core meltdown and necessary for the initiation of population protection actions by the public authorities;
- functions required in reference operating condition DBC2-4 after 24 hours and before 100 hours to ensure the static or dynamic containment (of a part) of a building;
- functions required under multiple failure operating condition DEC A or core meltdown operating condition DEC B to provide static or dynamic containment (of a part) of a building;

- functions required at the earliest 24 hours after the initiator to maintain the safe state of a HABD reference operating condition until 100 hours after the event;
- functions required for the operation of the local crisis center (CCL);
- functions specifically designed to control an internal or external hazard;
- functions necessary for the prevention of "virtually eliminated" situations;
- functions required during normal operation to monitor the condition of the barriers when this monitoring is not ensured by periodic testing.

Category NC:

Any function not categorized as Cat1, Cat2 or Cat3.

6.2.2. Classification and requirements for SSCs

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

Ranking rules:

The security classification assigned to a EIP is defined by the highest security category of the functions to which it contributes. EIP may also be ranked according to other criteria, such as the significance of the consequences of its failure, regardless of the functions they perform.

The classification of the SSCs is therefore made up of three safety classes (Safety Classes S1, S2 and S3) and one unclassified safety class (NC).

Safety class 1 (S1):

- a system or component required to perform a Cat1 function;
- equipment whose failure is excluded such as the reactor vessel;
- the equipment of the main primary circuit and main secondary circuit, by convention.

Safety class 2 (S2):

- a system or component required to perform a function at most Cat2;
- a system or component that supports a Cat1 function but whose failure does not immediately lead to the loss of the supported function or whose operation is not affected by the initiating event or its consequences;
- a control-command chain whose failure in normal operation would lead directly to a reference operating condition DBC3 or DBC4.

Safety class 3 (S3):

- a system or component required to perform a function up to Cat3;
- a system or component that supports a Cat 2 function but whose failure does not immediately result in the loss of the supported function or whose operation is not affected by the initiating event or its consequences;

- a system or component whose failure in case of external hazard may affect another system or component whose availability is considered for this hazard;
- a system or component carrying activity and whose failure in normal operation would lead to radiological consequences greater than those of normal operation;
- a system or component whose mechanical quality level Q is valued in hazard studies.

Systems or components that are not assigned to a safety class are not classified (NC).

Any exception to the classification rule must be justified, especially for certain equipment of the conventional island such as those ensuring turbine tripping.

6.2.3. Safety requirements

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

Table 4. Decoupled design requirements associated with Cat1, Cat2 and Cat3 safety functions

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

In addition, the security significance of an EIP is reflected in its security class. This safety class helps to define requirements for:

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

This ensures that an EIP, in accordance with the "INB Order", applies proportionate requirements for design, manufacture, controls, construction, testing and maintenance. Thus, depending on the type of EIP:

- a level of mechanical requirement is applied to the fluid-carrying equipment, the internals of the large components and the supports of the EIP;
- a level of control-command requirement is applied to the control-command components;
- an electrical requirement level is applied to the electrical components;
- Dedicated requirements are applied to lifting equipment and the polar bridge;
- Dedicated requirements are applied to the structures.

6.2.4. Application to sCO2 system

In the case of French power plants, as for Czech power plants, the sCO2 system is part of Category 3 (S3) for the classification of SSCs, as it provides a Category 3 security function. This classification will allow us to describe the requirements for the design, qualification and operation of the sCO2 system in deliverable D3.3.

7. Requirements for the plant modification

7.1. Czech Republic

No change made to a nuclear power plant shall affect its ability to be operated safely in accordance with the assumptions and goals of the project. The NPP operator is responsible for the modification management. The proposed changes must be divided into categories according to their impact on nuclear safety, radiation protection, emergency preparedness, technical safety and physical protection. SÚJB must be informed about this process and its expected results before the changes are implemented. If it is necessary to make a specific change, its impacts on the safety of the entire NPP must be assessed. Therefore, an overall review must be performed before the final decision. It is appropriate to include experience from similar changes to other NPPs in the assessment. Changes affecting the NPP configuration and operating limits and conditions must follow the requirements and related regulations. In particular, the operability of the safety systems must be maintained. All related documents must be updated and the NPP staff must be informed about the modifications and trained appropriately [26].

7.1.1. Categorization of changes according to their significance

All proposed changes must be evaluated and categorized according to their influence on nuclear safety, physical barriers, radiation protection, emergency preparedness etc. These items must be properly identified, described and documented.

The classification of the modifications is determined by the operator.

Category 1

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

- technical modifications to SSC classified into safety class 1 or 2 (in accordance with § 12 SÚJB Decree No. 132/2008 Coll. [27]), affecting their safety functions, changes of algorithms and safety system parameters, changes affecting the initiating events and changes interfering with the function of barriers against the radioactive releases,
- changes to the documentation submitted for SÚJB approval,
- changes in the management protocols having direct impact on nuclear safety and activities particularly important from the point of view of the radiation protection.

Category 1 changes may not be implemented before the proposer has obtained a permit from SÚJB.

Category 2

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

- technical changes to selected equipment of safety class 2, e.g. change of the manufacturer, the type of device or component which does not alter its design function,
- changes leading to the elimination of the identified non-compliance and ensuring the correct operation of SSC,
- organizational changes leading to improved management system functionality, such as are changes of staff performing activities related to plant safety,
- changes of suppliers.

Category 2 changes must be notified in writing to SÚJB for the assessment and approval.

Category 3

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

The principles for modification management are the same for all categories. However, it is necessary to determine the category of the modification on the basis of the prepared safety assessment. The classification has a direct effect on the procedure and scope of preparation, execution and documentation. The NPP operator is obliged to prepare documents with detailed requirements and definitions for classification of planned changes to the relevant category.

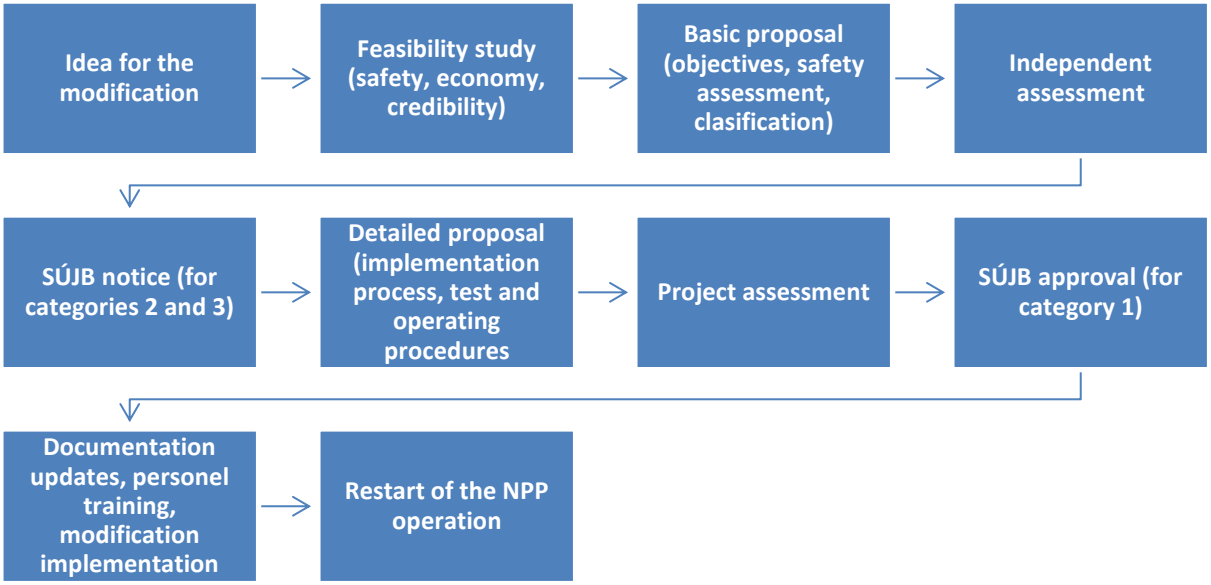

Figure 9. Process of NPP modification [26]

7.1.2. Responsibilities

OPERATOR

The NPP operator (ČEZ) is responsible for the safety aspects of the changes and for obtaining the appropriate assessment and approval of regulatory body in accordance with the state regulations. The operator must establish the process to ensure the appropriate design, assessment, management and implementation of all

permanent and temporary changes. The operator must ensure that appropriate safety analyses are performed with the relevant approach. The operator must ensure a systematic safety assessment which confirms on an ongoing basis that NPP safety analyses are still valid after the modifications. This assessment may be included in the periodic assessment of the NPP.

The scope and impact of the proposed modification must be assessed by independent institutions that are not the proponents or providers of the suggested modification.

The operator must ensure the cooperation of the professional staff who will be involved in the modification process. Appropriate training and education of personnel including the use of the simulator must be provided.

REGULATORY AUTHORITY

The regulatory body having competence in the field of nuclear safety, radiation protection, physical protection, emergency preparedness and technical safety of nuclear power plants in the Czech Republic is The State Office for Nuclear Safety (SÚJB), which proceeds with the assessment and approval of the modifications to the NPPs in accordance with the Atomic Act and other regulations related to Atomic Act (SÚJB decrees). The process of assessing, evaluating and approving changes is subject of SÚJB internal regulations.

OTHER ORGANIZATIONS INCLUDING SUPPLIERS

The operator may outsource the performance of technical, analytical, design or production activities to another organization. Responsibility for meeting the requirements regarding the safety of the NPP is borne only by the operator. The operator must have his own staff with sufficient knowledge and experience to lead and supervise the activities carried out by hired organizations.

If the modification is made by supplier, a professional level must be demonstrated, the experience and qualifications of all the workers who are to perform the work. Supplier staff must be adequately trained in accordance with pre-approved procedures in relation to the part of the NPP in which they are to operate. The supplier organization must have a proven quality assurance system and must be regularly audited by the NPP operator.

7.1.3. Implementation of modifications

ADMINISTRATIVE PROCEDURE

The operator of the nuclear installation is responsible for the management processes of all changes. For major modifications it must include the establishment of objectives and organizational structure, determination of responsibilities, project leader, management and control procedures and allocation of adequate resources. The implementation of changes, including the necessary tests, must be carried out in accordance with the applicable management documentation of the quality system, which includes work management programs and tests.

The operator must ensure that all personnel and staff of contractors who participate in the implementation of change, have been qualified, experienced and trained in the field. Personnel whose activities will be affected by the change must be demonstrably acquainted with the subject of the change.

CONSIDERATION OF SPECIFIC SAFETY ASPECTS

The following safety aspects must be included in the change assessment system:

- Radiation protection, including consideration of the ALARA principle;
- Radioactive waste management, transport, decontamination and decommissioning;
- Measures necessary to limit the spread of contamination;
- Occupational safety during the modification implementation;
- Technical safety;
- Protective equipment for personnel.

At the time of the modification process, the NPP or the implemented system must be in the appropriate operation conditions. If the evaluation of the change, including the operating condition, reveals a possible increase of the risk, it is necessary to consider temporary changes to emergency regulations.

TESTS AND START – UP

The ability of the modified nuclear power plant to continue safe operation must be verified by testing procedures that include control, measurement and evaluation before, during and after the completion of the modification. Trial and start-up, including pre-operational tests of equipment, equipment qualification, must be focused on evidence that the change has met the design requirements for the anticipated operational occurrences and design basis accidents. Pre-operational tests must meet the specific acceptance tests criteria based on operational criteria and test requirements that are part of the design documentation. Test programs must be evaluated by authorized persons or, on request, submitted to the supervisory authority for approval. Validation can be performed by testing on simulators or specially controlled operational tests, which demonstrates operability and confirms compliance with the target requirements. If operating conditions do not allow to perform tests after the implementation of the change, it is necessary to ensure performing tests at a facility designed and equipped for this purpose. The process of testing and quality assurance should be considered already at the stage of proposing a change.

Changes to safety-relevant software must be tested in advance off-line or on-site using a simulator according to pre-prepared programs. Before putting the modified system into operation, a final approval is needed, based on successfully performed activities, tests and verification. Complete implementation of safety-relevant changes and tests must be evaluated by the designers with regard to the project intent. An appropriate report must be prepared to demonstrate the compliance of the test results with the requirements and acceptance criteria. The report must be approved by the persons specified in the system NPP quality, i.e. by appropriate management or a designated committee. The final report is submitted to the supervisor demonstrating the fulfilment of the conditions previously assumed.

OPERATION

Before putting the modified systems into operation, the following must be checked:

- updating of the documentation directly related to the modification (safety reports, limits and operating conditions, drawing documentation of the actual design, emergency regulations, operational control programs, list of selected equipment);
- updating of the main design document of the NPP;
- performing all required tests and updating project requirements;

- training of staff on the modification made;
- completeness and accuracy of all data on design, commissioning, safety assessment, tests performed and implementation.

The completion of the change must include a review of all temporary implemented measures and procedures and their cancellation in time.

Changes to computer systems and software during operation can only be made after the detailed assessment. Changes to the operating parameters (e.g. trip and system actuation setpoints) may be carried out only if they are included in the safety analysis. The effect of changes must be evaluated using the simulator and associated calculation codes.

QUALITY ASSURANCE

Processes of preparation, assessment, implementation, control and final evaluation of all changes must be incorporated into the quality system documentation in accordance with the legislation.

Quality assurance programs, the approval of which is required by the Atomic Act as amended, must contain at least the information required by the decree SÚJB on quality assurance, as well as a list of supporting documents related to it with the modification management process. To document the qualifications of companies involved in the implementation of modifications, the system for their evaluation must be developed, containing requirements, criteria, method of evaluation their ability to perform work on NPPs.

TRAINING

The personnel responsible for operation and maintenance must be properly trained to continue safe plant operation. Attention must be paid to the interconnection of the altered and unaltered parts of the systems. Details on personnel training are given in the safety manual BN-JB-1.3. Appropriate training must take place before the modified system is put into operation.

The effects of the change on staff training requirements must be assessed in accordance with evaluation result. If the staff training programs are to be dependent on planned change, it is necessary to prepare new documents at an early stage of modification process. If a significant modification results in a change in qualification requirements to certain groups of operating personnel, the personnel with the necessary education must be provided [28].

7.1.4. Safety assessment

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

A comprehensive safety assessment must include an evaluation of the impact of the proposed change on radiological risks during its implementation and testing and during further operation. It should also include the effect of the modified part of the NPP and related systems on the nearest adjacent systems and components and interconnected and support systems such as electrical power supply.

A comprehensive safety assessment must demonstrate that after the modification, the NPP may be operated safely and in accordance with approved documentation. Special attention must be devoted to:

- Achieving compliance with the requirements of relevant safety standards at all operating modes.
- New or modified systems should not affect safety characteristics of other safety-relevant elements in all operating modes.
- The change must be made without a significant increase in the radioactive exposure of the personnel and members of the population (ALARA principle) and the risk of an accident.
- The change must be made without significant impact on the safety of the NPP and must not result in new risks.
- Each possible failure of the modified system must be assessed with a relevant method; impact of external events and environmental hazards must be assessed.
- The interaction of the implemented modification with other previous modifications must be evaluated.
- A possible incorrect implementation of the modification must be included in the safety assessment.
- If radioactive isotopes are generated during the implementation, their handling must be ensured in accordance with approved procedures.

A comprehensive safety assessment must include the results of deterministic analyses and probabilistic assessment, if appropriate to the safety significance of the modification.

Typical content of the safety assessment submitted to SÚJB:

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

REVIEW OF THE MODIFICATION PROPOSAL

The scope, safety analysis and consequences of the proposed modification must be assessed by an independent person or institution, not directly involved in the design and implementation of the change. Evaluators should be representatives of operational and technical personnel, project organizations, security experts and other technical and management staff. The latter may be independent external advisers, in particular in the case of major changes, to ensure reliable and comprehensive discussion about all safety aspects of the modification. This assessment should include independent validation and verification of software changes (for significant modifications).

Proposals for amendments submitted for independent assessment must meet the criteria in accordance with quality assurance requirements. The functional and safety requirements according the modification should be

included in the documentation. The level of details included in the documentation depends on the extent and complexity of the change. In any case, the documentation should include the following:

- Description of the project and justification of the proposed change;
- Sketches, drawings and material data;
- Applicable codes, standards and updated parts of the safety report;
- Safety assessment and possible proposal to change of the Limits and conditions of safe operation;
- Analyses of negative impacts on the environment and operating conditions, including production of radioactive waste, contamination and radiation exposure;
- Description of manufacturing processes, implementation and tests, including validation and verification methods;
- NPP operating mode at the time of the modification implementation;
- Quality assurance and quality control requirements;
- Description of the test program to be performed after the modification is completed;
- Description of changes in the operational control plan.

EVALUATION OF THE IMPACT ON PROJECT CHARACTERISTICS

The possible deviations from the basic characteristics of the project resulting from the modification should be minimized. If deviations are unavoidable, they must be assessed from the point of view of safety requirements and their acceptability must be demonstrated. The detailed design of the modification must include requirements for construction, assembly, start-up, testing, including acceptance criteria, and in-service maintenance.

NPP layout changes must meet the set of project safety requirements and related safety instructions. The necessary revisions to the preparation procedures must be included in the modification management process and the use of simulators for the training of NPP personnel. The revised documents include operational regulations for normal operating conditions, emergency regulations, operational control programs and test programs. The operating personnel must be familiarized with the process and significant changes must be covered in staff training with special professional competence. All these accompanying processes require close cooperation between designers, technicians, system administrators, maintenance and training centre.

Special attention must be paid to the development of methodologies that exclude the possibility of accidentally making two or more modifications to the same part or to related parts of the NPP. This means that the drawing documentation and the safety report must be thoroughly checked and controlled. Project changes must be submitted to the plant operator for assessment and determination of any impact on the design before it is fully implemented or cancelled. A procedure should be developed to propose changes and to coordinate the whole change management process.

7.2. France

The regulations relating to modifications to a nuclear power plant are described in Decision No. 2017-DC-0616 of the Nuclear Safety Authority of November 30, 2017 on significant modifications to basic nuclear installations. This decision, which was subsequently published in the Official Journal (and therefore validated by the government), describes the regulations relating to different types of modifications.

7.2.1. Categorization of changes according to their significance

Decision No. 2017-DC-0616 lists the different categories of modifications relating to a nuclear power plant in France. These categories are following:

- documentary modification: modification of one of the documents constituting the files filed with the safety authorities concerning the authorization, the authorization for commissioning, the operation and the safety of the plant;
- physical modification: modification consisting of the addition, alteration or removal of one or more protectively important items (EIPs), or the addition, alteration or removal of one or more items whose presence, operation or failure is likely to affect the operation or integrity of a EIPs;
- significant modification: modification falling under II or III of Article L. 593-14 or Article L. 593-15 of the Environmental Code (change of operator, operating procedures, dismantling);
- organizational modification: modification consisting of the addition, change or deletion of elements of the organizational structure or integrated management system, elements relating to roles and responsibilities, interfaces between entities, assigned resources, control and decision-making processes, computer and document management tools, temporal organization of work;
- substantial modification: modification falling under II or III of article L. 593-14 of the environment code.

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

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

7.2.2. Responsibilities

OPERATOR

The NPP operator is responsible for the safety aspects of the changes and for obtaining the appropriate assessment and approval of regulatory body in accordance with the state regulations.

The operator will be responsible for compiling the modification request file with all the necessary documents. It is possible to be exempted for the operator to present certain documents upon justification on his part and acceptance by the ASN.

The operator's requirements include, in particular, that the following actions be carried out:

1) determine whether or not any proposed modification is significant and, among the significant modifications, those that fall under II or III of Article L. 593-14 of the Environment Code, those that are subject to authorization by the Nuclear Safety Authority and those that are subject to declaration

to the Nuclear Safety Authority, pursuant to Article L. 593-15 of the Environment Code and Articles 26 and 27 of the aforementioned Decree of November 2, 2007, as specified by this decision;

2) give reasons for any significant modification envisaged;
3) design the proposed significant modification and, in this context:
    - take into account the users and their needs for the implementation of the modification and the operation of the modified facility;
    - to take advantage, from the point of view of the protection of interests, of the best available techniques and feedback for the design, conditions of implementation and future operation of the modification;
    - evaluate the possible negative consequences of the envisaged modification on the protected interests, taking into account the initial state of the INB, and limit and compensate for these consequences, as far as reasonably possible;
4) to define the possible actions to be implemented at the end of the process carried out in 3) in the field of:
    - manufacturing requirements, implementation of modified or newly installed elements;
    - training of the participants concerned and, where appropriate, the evolution of any driving simulators or installation processes;
    - organization and work environment, including the implementation of the change by the stakeholders;
    - collective radioprotection of the workers, in application of the article L. 593-42 of the environment code, for the implementation of the concerned modification and the operation of the modified installation;
5) to determine the possible tests to be carried out in order to guarantee that the EIP subject to any significant modification are subject, during the implementation and then the exploitation of the modification, to the qualification and the durability of this qualification, mentioned in article 2.5.1 of the aforementioned decree of February 7, 2012;
6) analyse the compatibility with the regulatory requirements and the individual prescriptions of the Nuclear Safety Authority taken in application of article 18 or article 25 of the decree of November 2, 2007;
7) Analyse the impact of any significant changes on:
    - the documents constituting the files mentioned in articles 8, 20 and 37-1 of the decree of November 2, 2007;
    - operating documents required by the operator's integrated management system for normal operation, degraded operation, incidents and accidents;
    - the documents used for the training of the concerned intervening parties and possible driving or installation process simulators;
8) to determine the possible provisions allowing the control of the implementation of a planned modification, prior to any decision of implementation by the operator;
9) formalize the operator's decision to implement any significant changes;
10) prepare the documentary changes required by the implementation of the significant change;
11) to implement, in accordance with the elements resulting from actions 1) to 10), and under conditions compatible with the integrated management system of the operator and with the constituent parts of the files mentioned in articles 8, 20 and 37-1 of the above-mentioned decree of November 2, 2007 in their applicable versions :
    - significant change;

- modifications to any driving simulators or installation processes if necessary;
- any tests associated with the implementation of this modification;

12) implement the amended documents to ensure consistency between the documentary and physical state of the facility following implementation of the significant change;
13) monitor the completion of the significant change and its compliance, as implemented, with the defined requirements applicable to it;
14) to control the effective training of the persons having to know about the significant modification;
15) draw and take into account the feedback from the implementation of the significant change.

REGULATORY AUTHORITY

Any significant change is subject to systematic verification of the defined requirements covering the achievement of the defined requirements for the management of significant changes. This verification is prior to any decision by the operator to implement the modification in question. It is carried out by an internal control body made up of persons with the appropriate skills to examine the modification in question. The organization to ensure the independence of this verification from the persons directly responsible for the operation or modification is proportionate to the issues that the modification is likely to present for the protection of interests.

OTHER ORGANIZATIONS INCLUDING SUPPLIERS

The operator may outsource the performance of technical, analytical, design or production activities to another organization. Responsibility for meeting the requirements regarding the safety of the NPP is borne only by the operator. The operator must have his own staff with sufficient knowledge and experience to lead and supervise the activities carried out by hired organizations.

If the modification is made by supplier, a professional level must be demonstrated, the experience and qualifications of all the workers who are to perform the work. Supplier staff must be adequately trained in accordance with pre-approved procedures in relation to the part of the NPP in which they are to operate. The supplier organization must have a proven quality assurance system and must be regularly audited by the NPP operator.

7.2.3. Implementations of modifications

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

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

The operator must also assess and reduce as far as possible the possible consequences of any physical modification of the installation which could affect public safety, health and hygiene or the protection of nature

and the environment. The operator must justify the acceptability of the modification, in order to prepare and then carry out the modification.

The file to be presented to the ASN is highly codified and aims to demonstrate that the operator has control over the physical modifications that the plant will undergo and that he can justify control over the level of safety of the installation. As in the case of an installation application, the file for a substantial plant modification should include the following elements:

- If the operator is a legal entity, its corporate name or denomination, its registered office and the capacity of the signatory of the application;
- A document describing the nature of the modification of the installation, its technical characteristics, the principles of its operation, the operations that will be carried out and the different phases of its realization;
- A detailed plan of the installation on a scale of at least 1/2,500; however, this scale may be reduced due to the size of the installation:
- The impact study;
- The updated version of the safety report;
- The updated risk management study;
- Updating of the decommissioning plan, which presents the methodological principles and the steps envisaged for the dismantling of the installation as well as the rehabilitation and subsequent monitoring of the site;
- Updating the general operating rules that the operator plans to implement,
- In the event of modification of the internal emergency plan, the opinion of the Health, Safety and Working Conditions Committee in accordance with the French Labor Code.

In addition to the file, the operator must also ensure the management of physical changes (“INB order”). Via the integrated management system (obligatory for any operator of a nuclear power plant), the operator must put in place provisions to ensure that the physical modifications are designed, validated and implemented in particular in compliance with the regulations applicable to the INB. Concerning these provisions, they must include elementary actions. The operator shall ensure the traceability of these actions, keep the corresponding documents in such a way that they remain easily accessible and legible, protected, in good conditions, and archive them for an appropriate and justified period of time.

We then detail the contents of some of the supporting documents in the file, so that the various project partners can integrate into their development roadmap the processes that will guarantee the operator the supply of the necessary documents for the integrated management system.

7.2.4. Description of the modifications

The description of the modifications desired by the operator is part of the regulatory file. In compiling the file, the operator must ensure that he has responded to the following basic actions:

1) Motivations for the material modification envisaged and justification, from the point of view of the protection of the interests mentioned, of the design chosen and the future operating methods, in particular with regard to the best available techniques and the feedback of experience;
2) Determination of the possible limited duration of the effect of the material modification;

3) Analysis, then limitation of the possible consequences of the envisaged material modification, taking into account the initial state of the INB, and determination of the implications of this analysis in terms of:
    a. training of the personnel concerned, prior to the implementation of the material modification;
    b. the evolution of possible simulation tools for the piloting of the installation;
    c. organization and work environment;
    d. manufacturing requirements, implementation of modified or newly installed elements;
    e. radiation protection of workers, for the phases of realization of this modification and operation of the modified facilities.
    f. This analysis must be verified by persons who are not directly involved in the design of the physical change or its implementation;
4) Determination of any tests to be performed to ensure that the modified EIPs are qualified for continued operation of the plant upon completion of the modification;
5) Analysis of the impact of the hardware change on:
    a. the parts that make up the installation file and any previous modification files;
    b. the operating documents required by the operator's integrated management system for normal operation, degraded operation, incidents and accidents;
    c. training documents and possible driving or process simulators;
6) Implementation, under conditions compatible with the integrated management system of the operator of the INB and with the constituent parts of the files already constituted for the plant:
    a. of the hardware modification on the INB;
    b. testing associated with the implementation of this change;
    c. Modifications of possible driving or process simulators when necessary;
7) Updating, if necessary, at the time of implementation of the material change, of the documents referred to in 5;
8) Control of the completion of the physical modification and its conformity "as implemented" to the defined requirements applicable to it, as well as control of the effective training of the persons having to know about this modification.

In addition, the operator shall maintain an updated status report on the implementation of physical changes to each NBI, which shall be submitted to the ASN annually.

7.2.5. Impact Study

The impact study is an important part of the file and must be as exhaustive as possible in order to answer all possible questions from the ASN and other assessors of the file (IRSN).

It thus makes it possible to describe:

- water withdrawals and planned liquid or gaseous effluent discharges. It specifies the different types of effluents to be treated and their respective origin, their quantity, their physical characteristics, their composition, both radioactive and chemical, the treatment process used, the conditions under which discharges will be made into the receiving environment and the composition of the effluents to be discharged.
- the waste that will be produced by all the facilities and equipment located within the perimeter of the facility, whether radioactive or not, as well as its volume, nature, harmfulness and the disposal

methods envisaged. It describes the provisions adopted by the operator to ensure that the management of this waste complies with the legislation in force.

- the radiological status of the environment on the site and its surroundings,
- the significant impact that the project is likely to have on the environment, the different phases of construction and operation of the facility. It takes into account seasonal and climatic variations.
- the impact of the facility on the water resource, the aquatic environment, water flow, level and quality, including runoff,
- aerosol or dust fallout and deposition; it indicates the impact of the facility on air and soil quality.
- optimizing the management of liquid and gaseous effluent and waste discharges, particularly in view of the overall impact of all these emissions on the environment and human health.
- exposure of the public to ionizing radiation as a result of the installation, taking into account in particular irradiation caused directly by the installation and transfers of radionuclides by the various vectors, including food chains.

The impact study must also present the measures envisaged to meet the regulatory requirements, in particular by justifying the use of the best available techniques:

- The expected performances, in particular, with regard to groundwater protection, purification, evacuation, management and monitoring of residual water and gaseous emissions;
- The conditions for the supply of materials to the plant for processing, the transport of the manufactured products and the rational use of energy;
- The measures chosen by the operator to control water withdrawals, discharges from the facility and to monitor the effects of the facility on the environment;
- The solutions chosen to minimize the volumes of waste produced and their radiological, chemical and biological toxicity.

In our case, the use of CO2 will therefore have to be included in this impact study, with an assessment of the quantities used and/or stored, the risks of release and the options considered for the transport, evacuation and management of CO2.

7.2.6. Safety report

The proposed modification must not have a negative impact on the safety level of the plant. For this reason, the operator will be requested to provide an update of the safety report in effect at the time of the change request.

This safety report is made up as follows:

- It includes an inventory of the risks presented by the sCO2 module (alone and coupled with the power plant) as well as an analysis of the measures taken to prevent them and a description of the measures to limit the probability of accidents and their effects. Its content is related to the importance of the hazards presented by the installation and their foreseeable effects.
- With regard to accidents of external origin, the operator must take into account the impact of installations which, whether or not they are under his responsibility, are likely, by their proximity or their connection with the planned installation, to increase the risks of accidents and their effects. The report must therefore describe:

  - Accidents that may occur, whether their cause is of internal or external origin, including the nature and extent of the consequences of malicious acts studied in application of Chapter III of Title III of Book III of Part One of the Defense Code;
  - The nature and extent of the effects that an accident could have;
  - The provisions envisaged to prevent such accidents or to limit their probability, with the exception of those relating to the prevention and control of malicious acts, or to limit their effects.
- It sets out the radiological risks presented by the installation and the collective radiation protection provisions for which the operator is responsible, including those arising from the design, so as to ensure compliance with regulatory radiation protection principles, including under normal operating conditions.
- It justifies that the project achieves, taking into account the state of technical knowledge, practices and the vulnerability of the facility's environment, a level of risk that is as low as reasonably practicable under economically acceptable conditions.
- It includes a section entitled "Study of the dimensioning of the internal emergency plan". This study deals with the accidents mentioned in the preceding paragraphs that require protective measures on or off site. The study describes the various accident scenarios and their consequences with regard to the safety of the installations and the protection of persons. It presents the organization planned by the operator of its own emergency means to combat the effects of a possible disaster.
- It describes and justifies the arrangements for the management of radioactive sources held in the basic nuclear facility, including the transport of such sources, in order to ensure the protection of workers, the public and the environment against the risks of irradiation and contamination.

This last point does not concern the sCO2 module, but other requests relating to the safety report will have to be studied by the power plant operator in consultation with the developer of the sCO2 module. The main issue will be to demonstrate that the addition of the sCO2 modules does not constitute a safety risk both by its installation and by possible accidents.

7.2.7. Risk Management Study

The purpose of the risk control study is to present the inventory of risks as well as the analysis of the measures taken to prevent these risks and the measures to limit the probability of accidents and their effects as set out in the preliminary version of the safety report. This study will be necessary to carry out the local consultations as well as the public inquiry if the requested modification requires it. In our case, the use of CO2 in a supercritical state could be a factor influencing the need for public consultation.

Its content is related to the importance of the dangers presented by the installation and their foreseeable effects, in the event of a disaster, on the power plant and its surroundings. As such, the risk control study includes:

- An inventory of the risks presented by the installation, both internal and external;
- An analysis of feedback from similar installations;
- A presentation of the methods used for risk analysis;
- An analysis of the consequences of possible accidents for people and the environment;
- A presentation of the measures envisaged for risk control, including the prevention of accidents and the limitation of their effects;

- An overview of the monitoring systems as well as the emergency devices and means;
- A non-technical summary of the study intended to facilitate the public's familiarity with the information it contains.

The risk control study justifies that the project makes it possible to achieve, taking into account the state of knowledge, practices and the vulnerability of the facility's environment, a level of risk that is as low as reasonably possible under economically acceptable conditions.

8. Conclusions

The present report covers few main aspects of nuclear safety, namely the nuclear regulatory framework, the general approach to safety and the requirements for systems, structures and components in an NPP. Finally, the requirements for the plant modifications are discussed. All these topics are presented for both Czech Republic and France cases. Many regulations regarding nuclear safety are similar in these countries as they are based on the same international rules (IAEA, WENRA, ICRP etc.). This fact makes it easier for the sCO2-4-NPP system designers to make it possible to implement the system in more European countries.

According to the previous studies described here, it will be highly beneficial to install alternative system for the emergency residual heat removal in Czech NPPs. However, it would require the regulatory body (SÚJB) approval along with the careful preparation of design documentation, safety assessment, personnel training etc. All these steps should be consistent with the applicable national and international regulations currently in force.

Concerning the requirements for licensing in France, the role of the operator of the power plant on which the sCO2 system will be installed is crucial. Indeed, the French legislative texts are not very prescriptive, so it is up to the operator to design a modification application that meets the level expected by the ASN and IRSN (which, at the request of the ASN, will study the application from a technical point of view). It is the operator's experience that will determine the right level of justification and thus guide the manufacturers towards the necessary qualification stages, in particular for the reports in the modification application relating to safety and risk control.

The next step will be D.3.3 “Design bases and analyses for system and components” with more detailed analysis and identification of relevant design bases and safety analyses for system and components.

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