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

Deliverable 3.1

Report on identification of the regulatory elements for design of components and system

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DEC	Websites, patents filing, press & media actions, videos, etc.	
OTHER	Software, technical diagram, etc.	
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Deliverable Contributors

Authors

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

Contributors

Partner	Name

Internal Reviewers

Partner	Name
EDF	ALBANNIE CAGNAC
USTUTT	JÖRG STARFLINGER

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1.1 List of acronyms

Abbreviation / Acronym	Description / meaning
ACI	American Concrete Institute
AFCEN	Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaire
AGR	Advanced Gas cooled Reactors
ANSI	American National Standards Institute
API	American Petroleum Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire
ASTM	American Society for Testing and Materials
AVN	Association Vinçotte Nuclear
AWS	American Welding Society
BPV	Boiler and Pressure Vessel
BPVC	Boiler and Pressure Vessel Code
BWR	boiling water reactor
CC	concrete containment
CEN	Comité Européen de Normalisation (European Committee for Standardization)
Cenelec	Comité Européen de Normalisation Electrotechnique (European Committee for Electrotechnical Standardization)
CNRA	Committee on Nuclear Regulatory Activities
CS	core support structures
CSN	Consejo de Seguridad Nuclear (Spanish Nuclear Safety Council)
CSWG	Codes and Standards Working Group
CVR	Centrum Vyzkumu Rez
DNB	Dimensionering av nukleära bygg-nadskonstruktioner (in English 'Design Guide for Nuclear Civil Structure')
EDF	Électricité de France
EM	Evaluation Model

Abbreviation / Acronym	Description / meaning
EMDAP	Evaluation Model Development and Assessment Process
EN	European Standard
EPR	European Pressurised Reactor
ETSI	European Telecommunications Standards Institute
GDC	General Design Criteria
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
GSR	general safety requirement
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
ISO	International Organization for Standardization
LWR	light water reactor
MC	metallic containment
MDEP	Multinational Design Evaluation Programme
MSS	Manufacturers Standardization Society
NDE	non-destructive examination
NEA	Nuclear Energy Agency
NF	Norme Française
NKe	Normenausschuss Kerntechnik (Nuclear Technology Standards Committee)
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
PED	Pressure Equipment Directive
PIE	postulated initiating event
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RCC	Rules for Design and Construction of Components of PWR Nuclear Islands
RCC-CW	Règles de Conception et de Construction du Génie Civil REP (in English 'Design and construction rules for civil works in PWR nuclear islands')

Abbreviation / Acronym	Description / meaning
RCC-E	Règles de Conception et de Construction des matériels Electriques des îlots nucléaires REP (in English 'Design and construction rules for electrical equipment of PWR nuclear islands')
RCC-M	Règles de Conception et de Construction des Matériels mécaniques des îlots nucléaires REP (in English, 'Design and Construction Rules for the Mechanical Components of PWR Nuclear Islands')
RFS	Règles Fondamentales de Sûreté (Basic safety rules)
RL	reference level
RSK	Reaktor-Sicherheitskommission (Reactor Safety Commission)
RT	radiographic testing
SAR	Safety Analysis Report
SRL	safety reference level
SSR	specific safety requirement
TÜV	Technische Überwachungsvereine (in English 'Technological Surveillance Associations')
UDE	Universität Duisburg-Essen (in English 'University of Duisburg-Essen')
UK	United Kingdom
USTUTT	Universität Stuttgart (in English 'University of Stuttgart')
UT	ultrasonic testing
VVER	Vodo-Vodjanoj Energetičeski Reaktor (in English 'water-water energetic reactor')
WENRA	Western European Nuclear Regulators Association

1.2 Nomenclature

Symbol	Description / meaning
A	additional thickness
c_1	absolute value of the minus tolerance
c_2	value that accounts for wall thickness reduction due to wear
D_0	outside diameter of pipe
d_a	outside diameter
d_i	inside diameter
p	design pressure
P	internal design pressure

Symbol	Description / meaning
s_0	wall thickness
s_{0n}	nominal wall thickness of the shell excluding allowances
S	design stress intensity
S_m	design stress intensity (Class 1)
y	parameter to adjust the Boardman equation to the Lamé equation

2 Executive Summary

The objective of this deliverable is 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 design of components within the framework of nuclear licensing is an important step to enable the adoption of sCO2-4-NPP by nuclear authorities and nuclear power plant (NPP) operators. The detailed design of the sCO2-4-NPP components (turbomachinery, heat exchangers and auxiliary systems) will therefore be specified taking into account regulatory requirements provided in this deliverable.

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. It consists of five levels of rules, where the first two levels are equivalent, consisting of European harmonized requirements for existing reactors and internationally established requirements for design of nuclear power plants. A few examples of national nuclear regulatory pyramids are also given showing that the hierarchy of regulatory requirements is similar, with the exception that instead of the two levels at the top, there is one level, representing country law. Before identified specific requirements are described, the regulatory definitions of passive system and Organisation for Economic Co-operation and Development (OECD) regulatory practice to assess passive safety systems in new nuclear power plant designs are given.

The main results are identification and description of the five levels of regulatory rules (Level I through V regulatory rules) for sCO2-4-NPP design of system and components. Level I high level requirements of Western European Nuclear Regulators Association (WENRA) and the Level II requirements for nuclear power plant design of International Atomic Energy Agency (IAEA) are equivalent levels (highest requirements like country legislation), with the difference that WENRA presented harmonized European requirements for existing reactors and are therefore at the top (the report focuses on design requirements), while IAEA presented internationally established standards for design of nuclear power plants, but the scope is broader than that of WENRA and was therefore included as complementary. Namely, the WENRA document for existing reactors has been established for greater harmonization within WENRA countries. The areas and issues they address were selected to cover important aspects of nuclear safety where differences in substance between WENRA countries might be expected. They do not seek to cover everything that could have an impact upon nuclear safety.

Level III documents deal with process oriented documents (quality assurance, regulatory guides on design, modification, etc.). For quality assurance a few standards satisfying specific nuclear requirements may be used, including IAEA management system. For design processes IAEA or U.S. Nuclear Regulatory Commission (NRC) again provide acceptable guidance, if national regulatory guides of a selected European country are not available. For nuclear civil structures the design guide of the Swedish Radiation Safety Authority is given. Finally, plant modification process guides are also described.

Level IV presents documents which are component-oriented for design and operation. Regulatory guides and nuclear codes and standards like ASME, KTA, and RCC are described. For nuclear codes and standards for mechanical component design it was identified (based on literature) that although the French RCC-M and ASME Section III codes may contain different sets of requirements, they result in components of an equivalent level of quality. Similar conclusions could be drawn for the German KTA standard for the selected example. Also it was identified that the pressure boundary codes and standards are very large, complex and detailed documents. Therefore, it is difficult for non-code specialists to appreciate the important requirements of pressure boundary codes and standards and this should be taken into account during design of components. Nuclear codes and standards for civil structures and electrical equipment are also described.

Finally, Level V deals with the codes and standards used for conventional facilities. It is expected, that primarily Level IV nuclear component oriented documents will be used for design of sCO2-4-NPP components. Also it is expected that industry designers of components are familiar with conventional codes and standards. The Pressure Equipment Directive (PED) 2014/68/EU is expected to be followed when the need would be identified to select conventional codes and standards due to non-availability of nuclear codes and standards for innovative sCO2-4-NPP component design.

The regulatory requirements (this deliverable D3.1 [9M, JSI]), according to nuclear regulatory restrictions, will be used in the sCO2-4NPP project as the basis for a conceptual design of a turbomachine for the sCO2-4-NPP cycle (developed by UDE and NP TEC in WP4 task 4.2 [M1-M36]) and proposing the best optimised design solutions for the heat sink exchanger (by USTUTT, CVR and FIVES in WP4 task 4.4 [M12-M36]). Finally, in WP4 task 4.5 FIVES will perform a complete mechanical study in order to improve the mechanical integrity of the heat recovery exchanger, according to both initial and boundary conditions of the sCO2-4-NPP (WP2) and regulatory requirements from WP3 (present deliverable D3.1 [9M, JSI]). Then, FIVES, USTUTT and CVR will perform together the final design of the heat recovery exchanger, also according to the regulatory requirements from WP3 (present deliverable D3.1 [9M, JSI]).

The future key issue is that according to WENRA the current safety approach relies primarily on active safety systems. Therefore, achieving the same reliability as for active safety systems may challenge the existing safety strategy. Also, the safety demonstration of reactor designs relying on passive safety features need to be developed to ensure safe operation of those designs in the future.

To conclude, the present deliverable allows the successful completion of task 3.1 and provides the necessary inputs for WP4. Thus, the goals of deliverable D3.1 are attained.

3 Introduction

The objective of this deliverable is to identify requirements on the sCO-4-NPP system arising from licensing and regulation and to ensure that they are included into the entire design process of components and system architecture in the frame of sCO2-4-NPP project. The design of components within the framework of nuclear licensing is an important step to enable the adoption of sCO2-4-NPP by nuclear authorities and nuclear power plant (NPP) operators. Namely, the final industrial version will be adaptable to most reactor types in Europe (BWR, PWR, VVER, HTR…), and could be retrofitted to existing plants or included in the future plants. The detailed design of the sCO2-4-NPP components (turbomachinery, heat exchangers and auxiliary systems) will therefore be specified taking into account regulatory requirements provided in this deliverable.

The licensing basis is a set of regulatory requirements applicable to nuclear installation (in addition it may also include agreements and commitments made between the regulatory body and the licensee). A hierarchy of regulatory requirements, proposed to be used for sCO2-4-NPP project, is presented first in Section 4. Some examples of three European country nuclear regulatory frameworks follow. The nuclear regulatory requirements consist of national or international laws and regulations. The top level national legislation of European countries, (acts, decrees), typically includes also ratified international agreements and European Union legislation. These international acts represent an important legal basis in the field of nuclear safety. Western European Nuclear Regulators Association (WENRA) documents, which represent harmonized levels of nuclear safety, are considered in the legislation. Therefore, WENRA requirements are taken here, rather than legislation and regulation of specific countries.

In Section 5 regulatory requirements for passive safety systems are described first. Definitions of passive system and Organisation for Economic Co-operation and Development (OECD) survey on the regulatory practice to assess passive safety systems in new nuclear power plant designs are given Subsection 5.1, followed by OECD survey on the regulatory practice to assess passive safety systems in new nuclear power plant designs Subsection 5.2.

Then, the sCO2-4-NPP hierarchy Levels I through V are presented in Section 6:

- Level I deals with the WENRA design (the focus of this document), operation and safety management requirements related to sCO2-4-NPP and some country specific requirements.
- Level II deals with the International Atomic Energy Agency (IAEA) design requirements, which is an international consensus of minimum requirements that should be fulfilled.
- Level III documents deal with the process oriented documents (quality assurance, regulatory guides on design and operation etc.).
- Level IV presents the documents which are component-oriented (regulatory guides and nuclear standards like KTA, RCC, ASME).
- Level V deals with the codes and standards used for conventional facilities (generally recognized codes and standards that need to be evaluated to determine their applicability, adequacy, and sufficiency for a new passive system qualification).

Finally, in Section 7 the main results of the work on the identification of nuclear regulatory elements to be considered in the design process of sCO2-4-NPP components and system for passive decay heat removal are summarized.

4 Hierarchy of regulatory requirements

4.1 Hierarchy of rules in the frame of sCO2-4-NPP

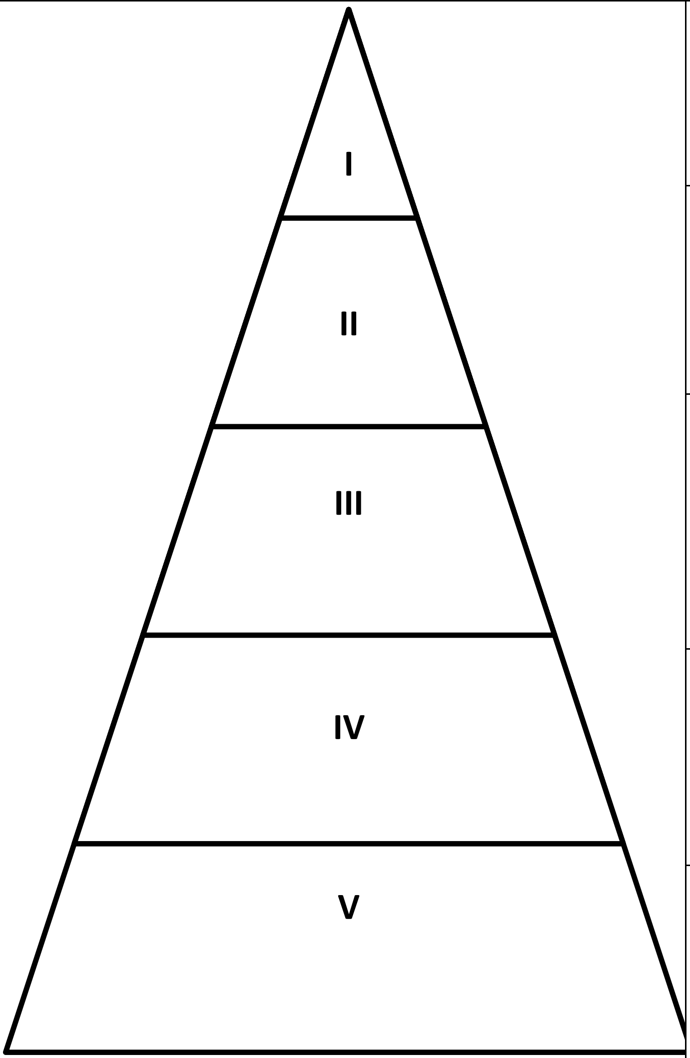
For the purpose of this document regarding requirements for the sCO2-4-NPP system and components the hierarchy of rules is shown in Table 1. Namely, the sCO2-4-NPP will be designed for implementation on a pressurized water reactor (PWR) type of nuclear power plant (NPP), which represents almost the entire fleet of large European NPPs. At the highest level (Level I) are legislation and regulation. Besides selected country legislation, Western European Nuclear Regulators Association (WENRA) documents are also considered. WENRA is the independent association of European national nuclear regulators recognised for establishing, implementing, and disseminating harmonized model levels of nuclear safety. The WENRA safety reference levels (SRLs) are a key driver for developing nuclear safety by a continuous improvement and harmonization of regulatory approaches in Europe [4]. National legislation in each country is also at the highest level (see Table 1). Legislation is typically applied in nuclear power plants to the structures, systems and components (SSC). As national legislations in European countries varies, the WENRA safety reference levels for existing nuclear power plants (NPP) [4] and its safety objectives for new NPPs are taken as representative of EU countries legislation [5]. These documents have been designed to be technology neutral.

The International Atomic Energy Agency (IAEA) safety standards are not legislation but reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which have three categories: safety fundamentals, safety standards and safety guides. For the purpose of sCO2-4-NPP at Level II, the IAEA safety requirements, which are further divided into general safety requirement (GSRs) and specific safety requirements (SSRs) are considered. GSRs are applicable to all facilities and activities, while SSRs are applicable to specified facilities and activities. At Level III are process oriented nuclear documents. Level III constitute the guidance documents of countries, including IAEA general specific guides (GSG) and specific safety guides (SSGs), applicable to all facilities and activities and to specified facilities and activities, respectively. Component oriented nuclear documents are at Level IV. Level IV also consists of guidance documents and nuclear codes and standards. Finally, Level V consists of conventional codes and standards of domestic and international organisations.

The above structure for Level I and Level II documents is also in-line with European Utility Requirements (EUR), which is not legislation, but consists of comprehensive NPP specifications written by a group of potential investors in electricity generation in Europe under the European Utility Requirements (EUR) Organisation. The purpose of the EUR Organisation is to actively develop and promote harmonised requirements for new mid- and large-size light water reactor (LWR) NPPs that are proposed for construction in Europe [22]. The harmonisation, which is sought by the 14 (fourteen) member utilities of the EUR Organisation, aims at delivering the safest and most competitive designs based on common requirements shared across Europe. As stated in paper [22], one of the priority objectives assigned to the EUR organisation in the 2013-2015 roadmap was to launch a new major revision of the EUR document including all the new updated international standards and lessons learned from the Fukushima Dai-Ichi accident, including safety requirements. The revision of the structure of the chapter results in a document which is more easily usable for the bidding and licensing purposes. The new EUR chapter 2.1 "Safety Requirements" systematically proposes functional requirements which are organised in a structure similar as other international standards (in particular IAEA SSR-2/1 Rev. 1 [11]). This new EUR chapter specifies a set of requirements that take the IAEA (SSR-2/1 [11], SSR-2/2 [12]) and WENRA high-level requirements and apply them in the European context (WENRA 2013 [5], WENRA 2014 [4]).

The above WENRA documents were used as Level I documents and IAEA as Level II documents in the proposed sCO2-4-NPP hierarchy of rules. On the other side, because EUR requirements are not regulations and they are not publicly available, they will not be considered in the frame of sCO2-4-NPP.

Table 1: Hierarchy of rules

Levels of rules	Description
 I	**Level I: Legislation & safety regulations** country legislation, WENRA [4]
II	**Level II: IAEA Safety Standards** safety fundamentals [8], safety requirements (generic [34] and specific [11])
III	**Level III: Nuclear process oriented documents** QA: ISO 19443:2018 [39], EN ISO 9001:2015 [38], KTA 1401 [49], 10CFR50 App. B [84], ASME NQA-1-2019 [48], IAEA GS-G-3 [37] Design and operation: IAEA NS-G (e.g. [30]), IAEA SSG (e.g. [24]), KTA [50], RG [85], RFS [14] and ASN guides [15], CSN [16]
IV	**Level IV: Nuclear component oriented documents** DIN [41], KTA [50], ANSI [42], ASME III Div 1, class 1, 2 or 3 [63], ISO [43], IEEE [44], RG [85], RCC-M [45] RCC-E [46]
V	**Level V: Conventional codes and standards** usually applied to the structures, systems and components of conventional facilities

4.2 Hierarchy of rules in the national legislation

To put it simply, the nuclear licensing requirements consist of legislation, guidelines, and codes and standards. The highest level national legislation of European countries (acts, decrees), typically contains also ratified international agreements. European Union legislation also represents an important legal basis in the field of nuclear safety. The overview of the nuclear legislation for OECD and Nuclear Energy Agency (NEA) countries, which include also European countries, can be found in https://www.oecd-nea.org/law/legislation/ [18]. All project partner's countries are represented (Czech Republic, France, Germany, Italy and Slovenia). A full range of nuclear law topics is given, including nuclear installations, which is of interest in the design of sCO2-4-NPP system. Each profile is complemented by reproductions of the primary legislation regulating nuclear activities in the member country.

Guidelines at the second level are not legal documents or requirements. However, they represent a public repository of the methods, which describe implementation of the specific parts of legislation and are acceptable for the regulatory bodies. Typically, codes and standards (nuclear as well as conventional ones) are endorsed by these guidelines.

4.2.1 Example of German nuclear regulatory framework

In Figure 1, an example for Germany is given [2]. The German nuclear regulatory framework presents a hierarchically structured pyramid, the so-called regulatory pyramid. At the top of the regulatory pyramid, there are the Basic Law, the Atomic Energy Act and a series of ordinances. They are generally binding and contain general requirements. The General Administrative Provisions are located in the central block of the regulatory pyramid. They regulate the actions of the authorities. At the footing of the regulatory pyramid there are the publications of the Federal Environment Ministry, standards of the Nuclear Safety Standards Commission (KTA Standards) and conventional technical standards. These regulations are not generally binding but contain specific requirements. For example, for the operator they can only become binding after they have been included in the license with respective provisions.

Länder Ministry is in charge of licensing, supervision and inspection of nuclear installations. The Technological Surveillance Associations (TÜV) are autonomous economic bodies in the form of private registered associations. They exist in all the Länder and may be entrusted by the competent official bodies to act on their behalf with respect to the implementation of nearly all control and surveillance measures required by law in relation to technical equipment and installations. In the nuclear technology field the licensing authorities also as a rule entrust the TÜV with the implementation of detailed safety inspections and the preparation of opinions and reports.

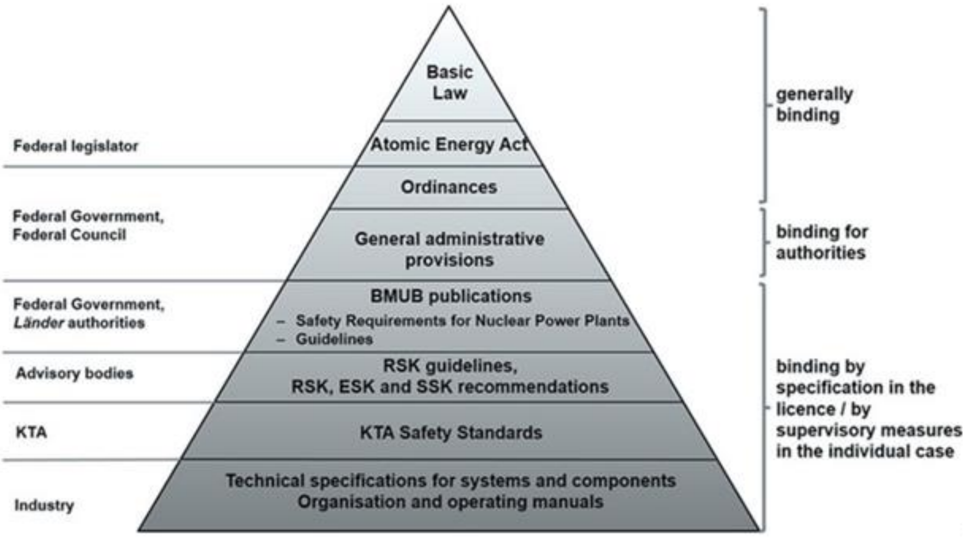


Figure 1: Nuclear regulatory pyramid in Germany [2]

4.2.2 Example of French nuclear regulatory framework

In Figure 2, an example for France is given [3]. Responsibility belongs to the state. Decrees and orders are taken at the ministerial level. However, the nuclear regulatory body (ASN) proposes or gives advice on ministerial decisions and issues technical rules and prescriptions. ASN decisions have to be endorsed by the government before enforcement. Finally, guides are not prescriptive but provide an interpretation and explain how to consider the corresponding regulation.

Control and regulation of nuclear activities is a fundamental responsibility of ASN. ASN is in charge of verifying the implementation of practices through regulatory assessments and inspections of nuclear operators/actors. Inspection is the key means of monitoring available to ASN. It consists in performing spot checks on the conformity of a given situation with regulatory or technical baseline requirements. ASN's regulatory actions are also carried out by other means such as examination of authorisation applications and analysis of significant events. The inspection is proportionate to the level of risk presented by the installation or the activity and the way in which the licensee assumes its responsibilities. If the results of ASN inspections are not satisfactory, there is a requirement to shut down or not restart (with obligation to follow the requirement) the plant until the situation is resolved.

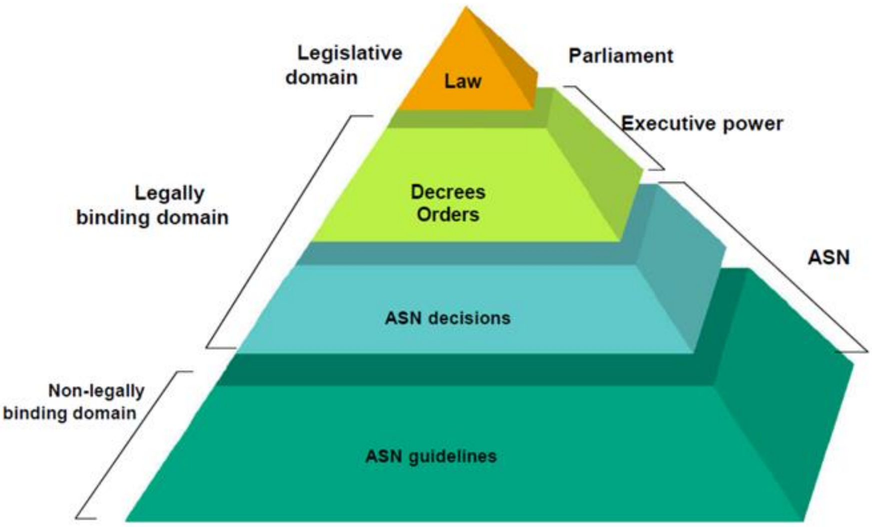


Figure 2: Nuclear regulatory pyramid in France [2]

4.2.3 Example of Slovenian nuclear regulatory framework

The situation in other European countries is similar like in Germany and France. For example, in Slovenia the first level is ionising radiation protection and nuclear safety act. It represents a basis for second-level decrees and regulations. In case that EU regulations would not be included in the national legislation, they should be respected. At the third level are recommendations and other non-legally binding documents of the European Union. At the fourth level are standards, valid in Slovenia. If not available in Slovenian legislation, the vendor country legislation may be used as reference. For example United States acts: 10CFR, Regulatory Guides, Generic Communications, Administrative Letters, Bulletins, Circulars, Information Notices, Security Advisories. IAEA standards recommendations may also be used. Nevertheless, vendor country legislation or IAEA standards are non-binding.

Slovenian Nuclear Safety Administration (SNSA) is an agency within the framework of Ministry of the Environment and Spatial Planning and is competent for performing administrative and technical affairs related to nuclear and radiation safety of nuclear facilities. SNSA performs specialised technical and development administrative tasks and activities of inspection control in the areas of radiation and nuclear safety.

The Ministry of the Environment and Spatial Planning issues the licence for use of the facility after it verifies that parameters regarding environmental impact from the trial operation meet the prescribed limits. The operator applies to the SNSA for an operating licence after receiving a licence for use of the facility. The application for the operating licence shall contain an updated safety analysis report (SAR), an opinion from an approved expert in radiation and nuclear safety and other prescribed documentation. The SAR must be updated with changes that occurred during the trial operation.

In accordance with act, inspection and enforcement of nuclear and radiation safety rest with the SNSA. The inspection powers include control over implementation of provisions of the act, regulations and decrees and other terms of the licences.

The act contains a requirement that the operators of radiation or nuclear facilities must obtain the opinion of approved experts on specific modifications in the facilities. Approved experts provide professional support to the Krško NPP by preparing independent expertise. An important part of the work focuses on an independent review and assessment of plant modifications. Only legal entities can be appointed as an approved expert for radiation and nuclear safety.

5 Overview of regulatory requirements for passive system in new nuclear power plants

The use of passive safety systems, nowadays, is one of the trends in many new reactor designs [13]. However, existing regulations oftentimes were developed for nuclear power plants (NPP) with mainly active safety systems. The existing reactors in addition require backfitting of complementary passive or active safety systems for prevention of severe accidents.

Hereafter, the information on national approaches to define and regulate the use of the passive safety systems is given. First, the review of existing definitions of passive systems is given, in order to distinguish them from the active systems. Then, the OECD survey on the current regulatory practice to assess the passive safety systems in the new nuclear power plant designs is presented [13]. With the participation of 6 out of 9 countries from Europe the survey reflects the European passive safety system requirements. Both systems intended to operate in the design basis accidents and the non-severe accident design extension conditions are subjects of the survey. The proposed novel sCO2-4-NPP passive safety system is intended to be used as backup passive cooling system for the reactor core in the case of a station blackout and loss of ultimate heat sink, which are design extension conditions. Therefore, the current regulatory practice to assess the passive safety systems in the new nuclear power plant designs is applicable for considering in the design of sCO2-4-NPP.

5.1 Definition of passive system

The WENRA report on regulatory aspects of passive systems from 2018 [1] recognized that international standards do not establish a clear definition of passive system. For example, neither IAEA nor Nuclear Regulatory Commission (NRC) glossaries include such definition. According to the WENRA document from 2018 [1] a quite flexible definition is the IAEA one from 1991 [9]: "*either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation*."

The limitations are further clarified that a component or system can be called passive when all three of the following considerations are satisfied in a self-contained manner:

- there must be "intelligence" such as a signal or parametric change to initiate action;
- there must be power and potential difference or motive force to change states; and
- there must be the means to continue to operate in the second state.

Conversely, according to IAEA technical document [9] a system is considered active if external inputs are needed. Four categories of passive system were defined to distinguish the different degrees of passivity [9].

Category A is characterized by:

- no signal inputs of "intelligence", no external power sources or forces,
- no moving mechanical parts,
- no moving working fluid.

Category B is characterized by:

- no signal inputs of "intelligence", no external power sources or forces,
- no moving mechanical parts, but
- moving working fluids.

Category C is characterized by:

- no signal inputs of "intelligence", no external power sources or forces; but
- moving mechanical parts, whether or not moving working fluids are also present.

Category D:

This category addresses the intermediary zone between active and passive where the execution of the safety function is made through passive methods as described in the previous categories except that internal intelligence is not available to initiate the process. In these cases, an external signal is permitted to trigger the passive process. To recognize this departure, this category is referred to as "passive execution/active initiation".

American Nuclear Society (ANS) glossary [17] provides the following definition for passive structure, system and component (SSC): "*an SSC that performs one or more safety functions either fully or partially via passive means (i.e., relying on natural physical processes such as natural convection, thermal conduction, radiation, gravity, or pressure differentials, or depending on the integrity of a pressure boundary or structural component). Examples include piping systems that are used to maintain an inventory of fluid and deliver flow along a fluid path, and structural supports for SSCs*."

The WENRA position paper from 2018 [1] comments that European Utility Requirements (EUR) provide also a quite flexible and more accurate definition than IAEA of a passive system: "*a system which is essentially self-contained or self-supported, which relies on natural forces, such as gravity or natural circulation, or stored energy, such as batteries, rotating inertia, and compressed fluids, or energy inherent to the system itself for its motive power, and check valves and non-cycling powered valves (which may change state to perform their intended functions but do not require a sub-sequent change of state nor continuous availability of power to maintain their intended functions)*".

On the basis of IAEA and EUR definition (provided in document [1]) WENRA Reactor Harmonization Working Group (RHWG) identified two characteristics of passive systems: actuation (if any) and performance of safety function [1].

Actuation. In case actuation is ensured by use of components that need to change state, the limited number of components is used and these components:

- Change state only once: when actuation of the function is necessary,
- Only rely on stored energy or are self-actuated to change state,
- Do not rely on continuous function on support features.

Performance of safety function. With respect to performance of safety function, the driving force is an important characteristic. The driving forces belong to the following list:

- gravity, including density difference,
- pressure difference,
- thermal exchanges,
- internal heating phenomena (e.g. nuclear decay heat),
- internal chemical phenomena,
- phase changes (e.g. from steam to liquid water or from liquid water to steam),
- any combination of the above forces.

There are also some other characteristics linked to the above, including the absence of the need for:

- component movement,
- support features, unless they could be considered as passive,

- human action,
- instrumentation and control (I&C).

Finally, in the framework of OECD/NEA/Committee on Nuclear Regulatory Activities (CNRA) [13] a survey on regulatory practices to assess passive safety systems used in new nuclear power plants has been conducted. Nine countries have participated in the survey, including Finland, Germany, United Kingdom and United States.

German representative provided the following answer [13]: "*Passive equipment of the safety system is equipment, that carries out a safety function without actuators or without auxiliary equipment, e.g. the primary circuit, the containment and shielding are called passive equipment of the safety system*."

Not all countries' regulatory frameworks have a formal definition of passive safety system. Nevertheless, the common understanding of what is the passive system is similar in the OECD/NEA/CNRA survey [13]: "*Usually, a passive system is understood as a system that either is composed entirely of passive components and structures or that uses active components in a very limited way to initiate subsequent passive operation. Passive operation typically implies the reliance on natural forces (e.g. convection) and (or) stored energy (e.g. gravity flow*)."

5.2 OECD/NEA/CNRA survey on the regulatory practice to assess passive safety systems in new nuclear power plant designs

The nuclear regulations cover all nuclear facilities and activities. In the case of sCO2-4-NPP, the focus is on the passive safety system for the decay heat removal. Currently in Europe the heat removal systems of existing boiling water reactors and pressurised water reactors during accidents mostly rely upon active safety systems for emergency core cooling. Basic elements of an active safety system are pumps driven by motor(s), which deliver water from reservoir(s) for core cooling. The motors are usually powered by electricity, with several back-up systems. Having the safety principles of redundancy, diversity, fail safe, autarchy, spatial separation, etc. applied, all active systems fulfil the safety and regulatory requirements and provide the current state of the art.

The OECD/NEA/CNRA survey in 2019 [13] covered the following topics:

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

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

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

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

What has been recognized as a difference is application of the single failure criterion. In some countries the application of single failure is the same for both active and passive systems (e.g. Poland, Slovak Republic, U.S.), while in a number of countries the approaches on the application of single failure criteria are different for active and passive systems (e.g. in Germany the criteria does not need to be postulated if it is demonstrated that the equipment is designed in accordance with certain requirements).

Survey respondents were asked what safety principles must be demonstrated through testing and analyses. Respondents were also asked about their expectations for the validation of computer codes and the conduct of testing used to demonstrate safety performance.

There is no significant difference in the approaches applicable for active and passive systems. Nevertheless, some countries indicate that passive safety systems as a rule require more emphasis on experimental substantiation than on analytical approaches.

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

What is important for the sCO2-4-NPP project is that the regulatory frameworks of a number of countries participating in the survey favour the use of passive systems over active ones and that these countries have explicit requirements for passive systems. Other countries participating in survey not having explicit requirements encourage the usage of passive systems. This means that the safety expectations for passive and active systems are similar. What may be different are the approaches to implement them.

6 Rules to be considered in the design of sCO2-4-NPP components and system

Rules are groups of laws, decrees, regulations and codes and standards which are required. As explained in Section 4, the rules were grouped into five levels. Level I and II rules representing WENRA and IAEA requirements specify the legal requirements. Level III rules are nuclear documents to be used for design of sCO2-4-NPP system, while Level IV rules are nuclear codes and standards to be considered for design of sCO2-4-NPP components. Finally, Level V rules are widely accepted conventional codes and standards from industry. In the following subsections 6.1 through 6.5 the Level I through Level V rules are described.

6.1 Level I rules - WENRA design requirements for passive safety system

6.1.1 WENRA RHWG report on regulatory aspects of passive systems

A specific WENRA document dealing with the passive systems is a position paper of Reactor Harmonization Working Group (RHWG) from 2018 [1]. The report on regulatory aspects of passive systems is based on the passive heat removal systems being an example of a passive system. Therefore, this report is very relevant for the sCO2-4-NPP project. It addresses innovative passive heat removal systems implying low driving forces (please note that in sCO2-4-NPP the driving forces are high). The report draws attention to the attributes of passive systems that are worthwhile to be considered with regards to safety in view of current regulatory practices in Europe (see Section 5). The key features of passive systems have been reviewed in the report. The safety assessment of any system (active or passive) should consider:

a) actuation of a passive system,

b) performance of safety function,

c) operating experience feedback.

In general, safety assessment is the assessment of all aspects of a practice that are relevant to protection and safety (this includes siting, design and operation of the facility). In the following the above three aspects a) to c) dealt with in the WENRA-RHWG report are described [1].

6.1.1.1 Actuation of a passive system

Very often the passive system has a very limited number of components that need to change state and do not rely on support features. This could lead to lower actuation failure rate. However, this lower failure likelihood need to be:

- demonstrated by a comprehensive analysis,
- ensured by verification of the components' operational availability,
- ensured by availability of necessary instrumentation and control (I&C) and support systems needed for actuation, if any.

The inadvertent actuation of a passive system should also be studied for possible negative consequences.

The actuation of passive systems requires an in-depth case-by-case safety assessment similar to active systems. These are covered by existing framework.

6.1.1.2 Performance of safety function

Specific range of conditions and consequences on safety analysis

For the performance of safety function of systems relying on low driving forces the range of conditions necessary to perform a safety function could be narrow. Therefore to demonstrate that such system can ensure a safety function with a **high level of reliability**, comparing to other systems, the following should be addressed when relevant:

- the failure mode analysis could be different comparing to active systems,
- impact of environmental conditions on system performance need to be considered,
- application of concept of margins, especially to ensure distance to cliff edge effects,
- it is necessary to consider that passive system performance may show a dynamic behaviour, and
- evaluation of potential adverse system interactions that could be different.

In addition, computer codes should be able to simulate the phenomena in the range of conditions, relevant for the performance of passive system. The codes should be validated and this may require specific experimental tests.

Failure mode analysis requires comprehensive knowledge of phenomena and parameters that could influence the performance or failure of passive system. Some phenomena, usually neglected for active systems, may jeopardise the safety function (e.g. non condensable gases). Passive systems may be sensitive to environmental conditions, internal and external. To apply margins to ensure distance to cliff-edge effect could be more demanding considering the uncertainties in performance. Namely, the range of conditions to perform a safety function could be narrow, therefore a limited change may be more or less challenging. A dynamic behaviour should be considered, as operation of passive system can change the boundary conditions. This, in turn, influences the driving forces during natural circulation.

General items for safety demonstration also include performance demonstration, including the use of computer codes used for modelling, consideration of hazards, consideration of human actions, and probabilistic safety assessment. These aspects are considered in the following sub-chapters.

Performance demonstration

Phenomena and parameters that influence the performance of a passive system can be different from an active system due to a specific range of operating conditions. Therefore, the performance demonstration may be different from active systems.

- The list of phenomena influencing the passive system is needed. This list can be obtained by a specific failure mode analysis.
- The influence of active systems should also be considered, also non safety, which could challenge the performance of passive systems.
- When all influencing factors are identified, well-known and understood, a set of representative parameters should be established (including their ranges to define boundary conditions) to demonstrate the performance of passive systems. As the range of operating conditions for the passive systems may differ from the active systems, the range of conditions can be out of the range for which the computer codes used for demonstration have been validated.
- The validation may require additional experimental tests.
- If the range of conditions to perform a safety function is narrow, integral test should be used to study reciprocal influences (e.g. temperature influences the volume of non-condensable gases) and scaling effects need to be considered.

- The capability of bringing plant to stable long term condition in timely manner should also be considered.
- Safety function performed by the passive system should be ensured for the plant lifetime, as for the active systems.
- Commissioning and periodic test programs should be defined and justified.
- Parameters necessary to justify the operability should be followed daily and integrated into operational limits and conditions (OLC).

Internal and external hazards consideration for passive systems

WENRA reference level (RL) T5.3 [4] required that "*the protection concept [for natural hazards] shall ensure that measures to cope with a design basis accident remain effective during and following a design basis event*". The hazards in general modify the environmental conditions that system has to cope with. For active systems, technological choices can ensure that components withstand these changes. On the other hand, the efficiency of passive systems relies more on a specific range of boundary conditions. So, for passive safety systems sensitive to the environmental changes resulting from hazards, the sensitivity should be evaluated, e.g.:

- environmental conditions that change air temperature, moisture and concentration of particles in the air for systems that use atmosphere as heat sink;
- fire that could modify the necessary temperature distribution in a system that uses buoyancy for fluid circulation;
- pipe deformation due to seismic deformation or load drop in a system that uses natural fluid circulation.

For DEC conditions it will probably be even more complicated. Namely, WENRA RL T6.3 [4] requires "*when assessing the effects of natural hazards included in the DEC analysis, and identifying reasonably practicable improvements related to such events, analysis shall, as far as practicable, include demonstration of sufficient margins to avoid "cliff-edge effects" that would result in loss of a fundamental safety function*". Due to potentially narrow range of operating conditions such demonstration may be challenging.

Consideration of human errors

The passive safety systems do not rely on operator actions. The RHWG recognized reduced potential for human error. Nevertheless, sensitivity to human errors has to be considered in the design phase, construction phase and operation phase (e.g. maintenance).

The benefits or needs of operator actions have to be anticipated during accidental conditions. Monitoring instrumentation is needed to provide status of the performance of passive system. Emergency operating procedures and severe accident management guidelines should be established for passive systems in the same way as for active systems. Finally, feasibility of human actions and monitoring should be ensured, which requires source of power (for monitoring, lightning, ventilation).

Probabilistic safety assessment

Probabilistic safety assessment (PSA) model is needed for analysis of all initiating events. The reliability assessment of human actions and SSCs is performed. The reliability of active systems relies on the failure probability of components. For the passive systems, the phenomena to actuate and/or maintain may be ineffective, leading to the failure probability of passive function. For the passive systems functional analysis and the set of representative parameters is needed. This can lead to identification of root causes which may prevent the parameters to be within the operating range. To include phenomenological causes, the PSA model should include also these root causes.

6.1.1.3 Operating experience

Operating experience is one of the pillars of safety assessment. However, obtaining such operating experience may be a challenge for passive systems. The deployment of new reactors with passive systems will for sure provide some operating experience. In case of limited feedback, full scale commissioning tests and periodic testing could complement operating experience feedback.

6.1.2 WENRA reference levels for existing reactors

As was already mentioned the WENRA safety reference levels (SRLs) are a key driver for developing nuclear safety by a continuous improvement and harmonization of regulatory approaches in Europe [4].

One important aspect is that quality assurance (management system) should be used in the design and all other processes, described in Issue C (area safety management) [4].

The design area includes Issue E, which provides design basis for existing reactors; Issue F for design extension conditions of existing reactors; Issue G, which sets requirements for safety classification of SSC, Issue Q for plant modifications and Issue T for natural hazards. During design also provisions for maintenance, testing and inspection should be considered (in Issue K). In the following sub-sections focus is on the safety systems and components for decay heat removal.

6.1.2.1 Issue C: Management System

According to Issue C [4] the main aim of the management system shall be to achieve and enhance nuclear safety by ensuring that other demands on the licensee are not considered separately from nuclear safety requirements, to help preclude their possible negative impact on nuclear safety. The following Issue C RLs are specified [4]:

- C2. General requirements
- C3. Management commitment
- C4. Resources
- C5. Process implementation
- C6. Measurement, assessment and improvement
- C7. Safety culture

After the Fukushima Dai-Ichi accident, the RLs relevant to safety culture have been introduced, while the other issues are practically unchanged from WENRA RLs 2008 [52]. Therefore, for brevity reasons only WENRA RLs for safety culture are given in the following.

WENRA RLs C7.1 through C7.3 for safety culture are [4]:

> *“C7.1 Management, at all levels in the licensee organization, shall consistently demonstrate, support, and promote attitudes and behaviours that result in an enduring and strong safety culture. This shall include ensuring that their actions discourage complacency, encourage an open reporting culture as well as a questioning and learning attitude with a readiness to challenge acts or conditions adverse to safety.*
>
> *C7.2 The management system shall provide the means to systematically develop, support, and promote desired and expected attitudes and behaviours that result in a strong safety culture. The adequacy and effectiveness of these means shall be assessed as part of self-assessments and management system reviews.*
>
> *C7.3 The licensee shall ensure that its suppliers and contractors whose operations may have a bearing on the safety of the nuclear facility comply with C7.1 and C7.2 to the appropriate extent.”*

6.1.2.2 Issue E: Design Basis Envelope for Existing Reactors

The Issue E requirements shall be fulfilled only if sCO2-4-NPP will be used for design basis envelope (i.e. not design extension conditions). WENRA RL E3.1 [4] requires that "*during normal operation, anticipated operational occurrences and design basis accidents, the plant shall be able to fulfil the fundamental safety functions [including] removal of heat from the reactor core and from the spent fuel*".

WENRA RL E5.1 [4] requires consideration of internal events in the design of the plant:

"*E5.1 Internal events such as loss of coolant accidents, equipment failures, maloperation and internal hazards, and their consequential events, shall be taken into account in the design of the plant.*"

WENRA RL E5.2 [4] requires consideration of external hazards in the design of the plant:

"*E5.2 External hazards shall be taken into account in the design of the plant. In addition to natural hazards, human made external hazards – including airplane crash and other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the nuclear power plant – shall as a minimum be taken into account in the design of the plant according to site specific conditions.*"

This means that sCO2-4-NPP system and its components should consider internal and external hazards.

For passive systems according to WENRA RL E8.2 [4] it is not necessary to assume the failure of a passive component, if it is very unlikely and its function remains unaffected during postulated initiating event (PIE):

"*E8.2 The worst single failure shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE*."

WENRA RL E8.3 [4] requires that sCO2-4-NPP system need to be suitably classified:

"*E8.3 Only systems that are suitably safety classified can be credited to carry out a safety function.*"

WENRA RLs E9.1 through RL9.5 [4] deal with design of safety functions. In the following the general requirements are listed:

"*E9.1 The fail-safe principle shall be considered in the design of systems and components important to safety.*

E9.2 A failure in a system intended for normal operation shall not affect a safety function.

E9.3 Activations and control of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes of the initiating event. Any operator actions required by the design within 30 minutes of the initiating event shall be justified.

E9.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components, redundancy, diversity, physical and functional separation and isolation.

E9.5 For sites with multiple units, appropriate independence between them shall be ensured."

For heat removal system the WENRA RL level requirement for existing reactors is WENRA RL E9.9 [4]:

"E9.9 Means for removing residual heat from the core after shutdown and from spent fuel storage, during and after anticipated operational occurrences and design basis accidents, shall be provided taking into account the assumptions of a single failure and the loss of off-site power."

This means that when sCO2-4-NPP will be used for design basis accidents, it shall fulfil the single failure criterion. However, for passive systems refer also to RL E8.2 [4].

When passive system relies on instrumentation the following WENRA RL [4] requirements should be fulfilled:

> "*E10. Instrumentation and control systems*
>
> *E10.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, the containment, and the state of the spent fuel storage. Instrumentation shall also be provided for obtaining any information on the plant necessary for its reliable and safe operation, and for determining the status of the plant in design basis accidents. Provision shall be made for automatic recording of measurements of any derived parameters that are important to safety.*
>
> *E10.2 Instrumentation shall be adequate for measuring plant parameters and shall be environmentally qualified for the plant states concerned.*"

6.1.2.3 Issue F: Design extension of existing reactors

WENRA RL F4.1 [4] deals with ensuring safety functions in design extension conditions. It requires that "*in DEC A, it is the objective that the plant shall be able to fulfil, the fundamental safety functions [including] removal of heat from the reactor core.*"

WENRA RL F4.2 [4] deals with capacity and capability:

> "4.2 It shall be demonstrated that SSCs (including mobile equipment and their connecting points, if applicable) for the prevention of severe fuel damage or mitigation of consequences in DEC have the capacity and capability and are adequately qualified to per-form their relevant functions for the appropriate period of time."

The above requirement for SSCs includes their support functions and related instrumentation.

WENRA RL F4.5 [4] deals with autonomy:

> "*F4.5 The NPP site shall be autonomous regarding supplies supporting safety functions for a period of time until it can be demonstrated with confidence that adequate supplies can be established from off site.*"

For heat removal system the WENRA RL level requirement for existing reactors WENRA RL F4.7 [4] requires:

> "*F4.7 There shall be sufficient independent and diverse means including necessary power supplies available to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.*"

For instrumentation and control WENRA RLs F4.15 and F4.16 [4] require:

> "*F4.15 Adequately qualified instrumentation shall be available for DEC for determining the status of plant (including spent fuel storage) and safety functions as far as required for making decisions.*
>
> *F4.16 There shall be an operational and habitable control room (or another suitably equipped location) available during DEC in order to manage such situations.*"

6.1.2.4 Issue G: Safety Classification of Structures, Systems and Components

WENRA RLs G2 [4] deal with requirements for classification process:

> "*G2.1 The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment.*
>
> *G2.2 The classification shall identify for each safety class:*
>
> - *The appropriate codes and standards in design, manufacturing, construction and inspection;*
> - *Need for emergency power supply, qualification to environmental conditions;*
> - *The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;*
> - *The applicable quality requirements"*

The designer shall select appropriate codes and standards, system and components shall be environmentally qualified and the quality requirements shall be specified.

WENRA RLs G3 [4] deal with the requirements for the reliability:

> "*G3.1 SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.*
>
> *G3.2 The failure of a SSC in one safety class shall not cause the failure of other SSCs in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.*"

WENRA RLs G4 [4] deal with selection of materials and qualification of equipment:

> "*G4.1 The design of SSCs important to safety and the materials used shall take into account the effects of operational conditions over the lifetime of the plant and, when required, the effects of accident conditions on their characteristics and performance.*
>
> *G4.2 Qualification procedures shall be adopted to confirm that SSCs important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions over the lifetime of the plant and when required in anticipated operational occurrences and accident conditions.*"

Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electro-magnetic interference, irradiation, humidity, and combinations thereof.

6.1.2.5 Issue K: Maintenance, In-Service Inspection and Functional Testing

Issue K belongs to operation and deals with implementation [4]. With respect to design the WENRA RL K3.1 [4] requires:

> "*K3.1 SSCs important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures*."

6.1.2.6 Issue Q: Plant Modifications

Issue Q belongs to operation. With respect to procedure for dealing with plant modifications the WENRA RL Q2.2 [4] requires:

"For modifications to SSC, this process shall include the following:

- *Reason and justification for modification;*
- *Design;*
- *Safety assessment;*
- *Updating plant documentation and training;*
- *Fabrication, installation and testing; and*
- *Commissioning the modification*."

With respect to requirements on safety assessment and review of modifications the WENRA RLs Q3 [4] require:

"*Q3.1 An initial safety assessment shall be carried out to determine any consequences for safety.*

Q3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

Q3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

Q3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by personnel not immediately involved in their design or implementation."

With respect to implementation of modifications the WENRA RLs Q4 [4] require:

"*Q4.1 Implementation and testing of plant modifications shall be performed in accordance with the applicable work control and plant testing procedures.*

Q4.2 The impact upon procedures, training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated.

Q4.3 Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated."

6.1.2.7 Issue T: Natural Hazards

WENRA RLs T5 [4] deal with protection against design basis events:

"*T5.1 Protection shall be provided for design basis events. A protection concept shall be established to provide a basis for the design of suitable protection measures.*

T5.2 The protection concept shall be of sufficient reliability that the fundamental safety functions are conservatively ensured for any direct and credible indirect effects of the design basis event."

Additional requirements for protection concept are given in WENRA RL T5.3 [4].

WENRA RLs T6 [4] deals with considerations for events more severe than the design basis events:

"*T6.1 Events that are more severe than the design basis events shall be identified as part of DEC analysis. Their selection shall be justified. Further detailed analysis of an event will not be necessary, if it is shown that its occurrence can be considered with a high degree of confidence to be extremely unlikely.*

T6.2 To support identification of events and assessment of their effects, the hazards severity as a function of exceedance frequency or other parameters related to the event shall be developed, when practicable.

T6.3 When assessing the effects of natural hazards included in the DEC analysis, and identifying reasonably practicable improvements related to such events, analysis shall, as far as practicable, include:

(a) demonstration of sufficient margins to avoid "cliff-edge effects" that would result in loss of a fundamental safety function;

(b) identification and assessment of the most resilient means for ensuring the fundamental safety functions;

(c) consideration that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;

(d) demonstration that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

(e) on-site verification (typically by walk-down methods)."

6.1.3 WENRA input to IAEA safety strategy

The WENRA position paper [7] states that WENRA highly appreciates the work of the IAEA and is grateful for its important contributions to enhance nuclear safety worldwide. It recognized the IAEA's vital importance in continuing to strengthen nuclear safety worldwide. Therefore, they highlighted some issues for the IAEA safety strategy. For safety approach of reactors with passive safety features according to WENRA the IAEA should consider that:

"*1. some new reactor designs apply more and more passive safety features;*

2. the current safety approach relies primarily on active safety systems. Achieving the same reliability as for active safety systems may challenge the existing safety strategy as for example defined in SSR 2/1;

3. the safety demonstration as well as review and assessment of new reactor designs relying on passive safety features need to be developed to ensure safe operation of those designs in the future;"

The above has not yet been included into IAEA SSR 2/1 document, whose Rev. 1 has been released in 2016, i.e. before the WENRA position paper from 2017 [7]. This should be kept in mind, when referring to Level II rules described below.

6.2 Level II rules - IAEA design requirements for passive safety system

Document IAEA SSR-2/1 [11] has general plant design requirements and design requirements for specific plant systems.

6.2.1 Management of safety in design

For management safety in design three requirements are given. However, for the purpose of sCO2-4-NPP project the Requirement 3 [11] dealing with safety of the plant design throughout the lifetime of the plant is not relevant.

Requirement 1 [11] deals with responsibilities in the management of safety in plant design:

"*An applicant for a licence to construct and/or operate a nuclear power plant shall be responsible for ensuring that the design submitted to the regulatory body meets all applicable safety requirements*."

Requirement 2 [11] deals with management system for plant design:

"*The design organization shall establish and implement a management system for ensuring that all safety requirements established for the design of the plant are considered and implemented in all phases of the design process and that they are met in the final design.*"

6.2.1.1 Leadership and management for safety

Requirements on the management system are established in IAEA general safety requirements standard GSR Part 2 [34]. It comprises of the following 14 requirements:

- Requirement 1: Achieving the fundamental safety objective
- Requirement 2: Demonstration of leadership for safety by managers
- Requirement 3: Responsibility of senior management for the management system
- Requirement 4: Goals, strategies, plans and objectives
- Requirement 5: Interaction with interested parties
- Requirement 6: Integration of the management system
- Requirement 7: Application of the graded approach to the management system
- Requirement 8: Documentation of the management system
- Requirement 9: Provision of resources
- Requirement 10: Management of processes and activities
- Requirement 11: Management of the supply chain
- Requirement 12: Fostering a culture for safety
- Requirement 13: Measurement, assessment and improvement of the management system
- Requirement 14: Measurement, assessment and improvement of leadership for safety and of safety culture

For further details for each requirement the reader can refer to [34]. It should be noted that WENRA Issue C has been developed based on IAEA GS-R-3 from 2006 [36], while IAEA GSR Part 2 [34], which supersedes the IAEA GS-R-3 [36], has been published in 2016.

IAEA GSR Part 2 [34] clarifies the application to all types of installations, lifecycles and sizes of organisations. It clarifies the requirements on leadership for safety and recognises the key influence of senior management. It contains enhanced details of the safety culture requirements and has links to emergency preparedness & security standards. It includes more explicit supply chain requirements. It emphasizes that a systemic approach to safety is one essential element to foster a strong safety culture.

Most things stated above were mentioned in IAEA GS-R-3 [36] but IAEA GSR Part 2 [34] are more detailed or slightly changed perspective due to the lessons learned over the years. For example, lessons arising from the Fukushima-Daiichi event were incorporated.

6.2.2 General plant design requirements

Requirement 4 [11] deals with the fundamental safety functions:

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

Requirement 9 [11] deals with proven engineering practices:

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

Requirement 11 [11] deals with provision for construction:

"Items important to safety for a nuclear power plant shall be designed so that they can be manufactured, constructed, assembled, installed and erected in accordance with established processes that ensure the achievement of the design specifications and the required level of safety."

Requirement 13 [11] deals categories of plant states:

"Plant states shall be identified and shall be grouped into a limited number of categories primarily on the basis of their frequency of occurrence at the nuclear power plant."

Requirement 14 [11] deals with design basis for items important to safety:

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

This requirement is very comprehensive and specifies which information shall be documented to operate the plant safely.

Requirement 15 [11] deals with design limits:

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

Requirement 16 [11] deals with postulated initiating events:

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

Requirement 17 [11] deals with internal and external hazards:

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

Requirement 18 [11] deals with engineering design rules:

"*The engineering design rules for items important to safety at a nuclear power plant shall be specified and shall comply with the relevant national or international codes and standards and with proven engineering practices, with due account taken of their relevance to nuclear power technology*."

Requirement 19 [11] deals with design basis accidents:

"A set of accidents that are to be considered in the design shall be derived from postulated initiating events for the purpose of establishing the boundary conditions for the nuclear power plant to withstand, without acceptable limits for radiation protection being exceeded."

Requirement 20 [11] deals with design extension conditions:

“A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”

Requirement 21 [11] deals with physical separation and independence of safety systems:

“*Interference between safety systems or between redundant elements of a system shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication (data transfer), as appropriate*.”

Requirement 22 [11] deals with safety classification:

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

Requirement 23 [11] deals with reliability of items important to safety:

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

Requirement 24 [11] deals with common cause failures:

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

Requirement 25 [11] deals with single failure criterion:

“The single failure criterion shall be applied to each safety group incorporated in the plant design.”

Requirement 26 [11] deals with fail-safe design:

“*The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety*.”

Requirement 29 [11] deals with calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety:

“*Items important to safety for a nuclear power plant shall be designed to be calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions specified in their design basis*.”

Requirement 30 [11] deals with qualification of items important to safety:

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

Requirement 33 [11] deals with safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant:

> "*Each unit of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions*."

Requirement 40 [11] deals with prevention of harmful interactions of systems important for safety:

> "*The potential for harmful interactions of systems important to safety at the nuclear power plant that might be required to operate simultaneously shall be evaluated, and effects of any harmful interactions shall be prevented*."

Requirement 42 [11] deals with safety analysis of the plant design:

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

6.2.3 Design of specific plant systems

Requirement 51 [11] deals with removal of residual heat from the reactor core:

> "*Means shall be provided for the removal of residual heat from the reactor core in the shutdown state of the nuclear power plant such that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded*."

Requirement 53 [11] deals with heat transfer to an ultimate heat sink:

> "*The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states*."

Requirement 59 [11] deals with provision of instrumentation:

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

6.2.4 Comparison between WENRA and IAEA design requirements

The WENRA document with RLs for existing reactors [4] states that given the various regulatory regimes and range of types of plants (PWR, BWR, CANDU and gas-cooled reactors) in operation in WENRA countries, the RLs do not go into legal and technical details. Guidance on specific issues is also available on the WENRA website www.wenra.org. There are significant interactions between some of the issues and hence each issue should not necessarily be considered self-standing and the RLs need to be considered as a whole set. In WENRA document [4] it is also stated that when needed, a reference to a relevant IAEA publication is inserted.

The study for UK European Pressurised Reactor (EPR) safety assessment in 2012 [33] indicates that the WENRA organisation took into account the IAEA standards and guidelines, which have been produced in order to establish a common reference basis amongst European regulators. They recognized that the scope of the WENRA work from 2007 is narrower than the scope of the IAEA guidelines (before 2012). In the report [33] it is stated that at that time only a small number of reference levels were not yet implemented on the French fleet of reactors. Discrepancies with one design related RL dealing with the single failure criteria (E 8.2) was also identified. There is no public document specifically comparing the WENRA 2008 and IAEA requirements before 2012.

Because both WENRA and IAEA standards were updated after 2011 and due to the fact that the scope of IAEA requirement is broader, Table 2 with comparison between WENRA RL from 2014 [4] and IAEA design requirements from 2016 [11] has been prepared.

Table 2: Comparison of WENRA and IAEA design requirements

WENRA [4]	IAEA SSR-2/1 [11]	IAEA Requirement explained in paragraphs of IAEA SSG-56 [23]
E4. Establishment of the design basis	Req. 13: Categories of plant states Req. 14: Design basis for items important to safety Req. 16: Postulated initiating events	Paragraphs 3.10-3.12, 3.44, 5.54-5.56
E5. Set of design basis events	Req. 17: Internal and external hazards	Paragraphs 3.14-3.26
E6. Combination of events		
E7. Definition and application of technical acceptance criteria	Req. 14: Design basis for items important to safety Req. 15: Design limits	Paragraph 3.44 (see also paragraphs 4.1-4.5, 4.12-4.13, 4.15-4.18 of IAEA SSG-2 [25])
E8. Demonstration of reasonable conservatism and safety margins	Req. 19: Design basis accidents Req. 25: Single failure criterion Req. 42: Safety analysis of the plant design	Paragraphs 3.34-3.35 Paragraph 3.35 N.A. (see paragraphs 5.71-5.74 of IAEA SSG-2 [25])
E9. Design of safety functions	Req. 4: Fundamental safety functions Req. 26: Fail-safe design Req. 51: Heat transfer to ultimate heat sink	Paragraph 3.8 Paragraphs 3.48-3.52, 6.75-6.88, 7.20
E10. Instrumentation and control systems	Req. 59: Instrumentation	Paragraphs 3.133–3.136
F2. Selection of design extension conditions	Req. 20: Design extension conditions	Paragraphs 3.37–3.42
F3. Safety analysis of design extension conditions	Req. 20: Design extension conditions Req. 42: Safety analysis of the plant design	Paragraphs 3.37–3.42 N.A. (see paragraphs for criteria 4.1-4.5, 4.12-4.13, 4.15-4.18 and paragraphs 5.71-5.74 of IAEA SSG-2 [25] for safety analysis)
F4. Ensuring safety functions in design extension conditions	Req. 20: Design extension conditions Req. 33: Safety features for DEC Req. 51: Heat transfer to ultimate heat sink Req. 53: Instrumentation	Paragraphs 3.37–3.42 Paragraphs 3.137 Paragraphs 4.29-4-40, 6.90-6.91, 7.22 Paragraphs 4.2-4-19

WENRA [4]	IAEA SSR-2/1 [11]	IAEA Requirement explained in paragraphs of IAEA SSG-56 [23]
G2. Classification process	Req. 18: Engineering design rules Req. 22: Safety classification	Paragraph 3.139 Paragraphs 3.63–3.66
G3. Ensuring reliability	Req. 23: Reliability of items important to safety	Paragraphs 3.47–3.56 (cover R21-R26, R29 and R30)
G4. Selection of materials and qualification of equipment	Req. 30: Qualification of items important to safety	Paragraphs 3.68–3.75
T5. Protection against design basis events	Req. 17: Internal and external hazards	Paragraphs 3.14–3.17, 3.19-3.26, 5.16, 5.17–5.21A
T6. Considerations for events more severe than the design basis events	Req. 17: Internal and external hazards	Paragraphs 3.14–3.17, 3.19-3.26, 5.16, 5.17–5.21A

6.3 Level III rules – Nuclear process oriented documents

Level III rules constitute the relevant process oriented guidance documents and standards for management system (quality assurance) and design and operation processes related to the sCO2-4-NPP system.

Managements systems include IAEA GS-G-3 [37], RG 1.28 [87], ISO 9001:2015 [38], ISO 19443:2018 [39], ASME NQA-1-2019 [48], and supplementation of ISO 9001:2015 [38] with nuclear requirements.

Design and operation process oriented documents and standards include mostly IAEA general specific guides (GSG) and specific safety guides (SSGs), applicable to all facilities and activities and to specified facilities and activities, respectively. The IAEA documents are guidance documents to IAEA requirements presented in this report as Level II rules. In addition U.S. based documents, which seem to be the most complete and have been already used for assessment of several passive systems of advanced reactors, are also included. It should be also noted that these documents endorse several codes and standards. WENRA also issue a few guidance documents and the one relevant for the sCO-4-NPP project is included.

6.3.1 Quality assurance guidance and standards

The quality management system like ISO 9001:2015 is not accepted by nuclear regulators for items important to safety. Namely, improving safety is a key objective of most industries and boosting the quality of the products and services that contribute to safety is necessary to achieve it. Therefore, it must be complemented by nuclear requirements. In the following are described the following documents related to management systems: IAEA GS-G-3 [37], RG 1.28 [87], ASME NQA-1-2019 [48], KTA-1401 standard [49], ISO 9001:2015 [38], ISO 19443:2018 [39] and supplementation of ISO 9001:2015 [38] with nuclear requirements.

6.3.1.1 IAEA GS-G-3

IAEA GSR Part 2 establishes requirements [34] and its guidance is not yet prepared (proposed title is Leadership, Management and Culture for Safety and target date for publication is end of 2022). Currently, document IAEA GS-G-3 [37] is used, which supports the Safety Requirements publication on The Management System for Facilities and Activities (GS-R-3 [36]), which has been superseded by IAEA GSR Part 2 [34]. This guide incudes safety (of nuclear facilities), health, environmental, security, quality and economic elements and other considerations such as social responsibility. A robust and effective management system should support the enhancement and improvement of safety culture and the achievement of high levels of safety performance.

6.3.1.2 Regulatory guide 1.28

Regulation 10CFR50, Appendix B [84], “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants” establish the 18 Criteria of Appendix B are applied in the siting, design, construction and operation of a nuclear facility. The pertinent requirements of 10CFR50, Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components; including designing, purchasing, fabricating, handling, shipping, storing, operating, maintaining, repairing, refueling and modifying.

Regulatory Guide 1.28 [87] describes methods that the staff of the U.S. NRC considers acceptable for complying with the provisions of 10CFR50 and 10CFR52, which refer to 10CFR50, Appendix B, for establishing and implementing a quality assurance (QA) program for the design and construction of nuclear power plants and fuel reprocessing plants. The Revision 5 of RG 1.28 [87] endorses, with certain clarifications and regulatory positions, various versions of the ASME NQA-1 standard. The NRC staff determined that the NQA-1-2015 provide the most current guidance for QA. In the view of harmonization the NRC staff reviewed guidance from the International Atomic Energy Agency (IAEA) and did not identify any standards that provided useful guidance to NRC staff, applicants, or licensees.

The Part I and Part II requirements included in the NQA-1-2015 (and some earlier versions) for the implementation of a QA program during the design and construction phases of nuclear power plants and fuel reprocessing plants are endorsed by the NRC staff, and provide an adequate basis for complying with the requirements of Appendix B to 10 CFR Part 50, subject to the exceptions and clarifications identified in RG 1.28 [87]. While the 18 criteria spelled out what requirements had to be met, they provided little guidance on how to accomplish the tasks. NQA-1 standard is intended to provide a method for users to meet the requirements of 10CFR50 Appendix B.

6.3.1.3 ASME NQA-1-2019

The latest ASME NQA-1-2019 standard [48] provides requirements and guidelines for the establishment and execution of quality assurance programs during siting, design, construction, operation and decommissioning of nuclear facilities. It reflects industry experience and current understanding of the quality assurance requirements necessary to achieve safe, reliable, and efficient utilization of nuclear energy, and management and processing of radioactive materials. It focuses on the achievement of results, emphasizes the role of the individual and line management in the achievement of quality, and fosters the application of these requirements in a manner consistent with the relative importance of the item or activity.

Part I contains requirements for developing and implementing a Quality Assurance Program for nuclear facility applications. Essentially these are the 18 criteria of 10CFR50, Appendix B [84] with additional details and requirements. Part II contains additional quality assurance requirements for the planning and conduct of specific work activities under a Quality Assurance Program developed in accordance with Part I. Part III contains guidance for implementing the requirements of Parts I and II. Finally, Part IV contains guidance for application of NQA-1 and comparisons of NQA-1 with other quality requirements.

6.3.1.4 KTA-1401 (2017-11)

The KTA-1401 standard [49] provides general requirements for the quality assurance and applies to the quality assurance during:

- safety-related conceptual design,
- planning and design,
- procurement,
- fabrication and assembly of product forms, parts, components and systems,
- manufacture or the providing of products,
- erection and subsequent work on building structures,

as well as

- commissioning

including the tests and inspections performed with special regard to those quality characteristics important to the precautionary measures against damage of the safety-related parts and services for stationary power plants during construction, operation and until decommissioning.

6.3.1.5 ISO 9001:2015

ISO 9001:2015 [38] specifies requirements for a quality management system when an organization:

- needs to demonstrate its ability to consistently provide products and services that meet customer and applicable statutory and regulatory requirements, and
- aims to enhance customer satisfaction through the effective application of the system, including processes for improvement of the system and the assurance of conformity to customer and applicable statutory and regulatory requirements.

All the requirements of ISO 9001:2015 are generic and are intended to be applicable to any organization, regardless of its type or size, or the products and services it provides. Therefore, ISO 9001:2015 [38] is not sufficient to fulfil management system requirement for nuclear sector. However, it could be complemented by specific nuclear requirements (see sub-section 6.3.1.7).

6.3.1.6 ISO 19443:2018

The nuclear sector is set to benefit with a new ISO 19443:2018 standard [39] that applies the principles of ISO 9001:2015, to the nuclear sector, combining best practice in quality with the specific requirements of the nuclear industry. ISO closely cooperated with the IAEA. The standard ISO 19443:2018 [39] will help to increase the safety culture in the sector and harmonize supplier assessments such as auditing.

ISO 19443:2018 standard [39] applies the principles of ISO 9001:2015 [38] to the nuclear sector and also takes due cognisance of the IAEA GSR Part 2 [34]. Comparing to ISO 9001:2015 the following clauses are completely new [40]:

- nuclear safety culture;
- determination of services important to nuclear safety (ITNS) items and activities;
- graded approach to the application of quality requirements.

Besides new clauses slight adaptation to complement ISO 9001:2015 with nuclear specifications has also been done in several clauses.

6.3.1.7 ISO 9001 complemented by nuclear quality requirements

As ISO 19443:2018 standard [39] was created just recently, one option is also to complement ISO 9001 by nuclear quality requirements. For example, in France Quality Order of August 10, 1984 “Concerning Basic Nuclear Installation design, construction and operation quality”-OFFICIAL GAZETTE OF THE FRENCH REPUBLIC SPECIAL ISSUE September 22, 1984, page 8652 is based on ISO 9001 and IAEA management system standard. In the U.S. the ISO 9001 requirements must be complemented by ASME NQA-1 nuclear requirements.

6.3.2 Design and operation documents

The scope of this deliverable is to provide harmonized legislation at the European (using WENRA) and international level (using IAEA). Therefore Level III documents described in the following are mainly WENRA, IAEA and the U.S. Regulatory Guides as country example of design and operation process documents. Other country specific nuclear documents and standards are generally not considered in the following. This may be German RSK [51] and KTA documents [50], DIN Nuclear Standards Committee (NKe) documents (most standards approved by KTA are also published as DIN standards [41]), French RFS [14] (the RFS are intended to be gradually replaced by ASN guides) and ASN guides [15], Spanish CSN guides [16] etc. The nuclear documents of two countries, France and Czech Republic, will be covered in the sCO2-4-NPP deliverables D3.3 [20] and D3.4 [21].

6.3.2.1 WENRA guidance Issue F: Design Extension of Existing Reactors

WENRA-RHWG Guidance on Issue F [6] provides all RLs of Issue F with explanations of the intent of each RLs of Issue F, to contribute to a consistent interpretation and to permit insights into the considerations which have led to their formulation. In addition, some background information is provided for easy reference. In the following a few more relevant guidances for the sCO2-4-NPP system are described.

Guidance on WENRA RL F4.2 explains that for demonstration of the ability of SSCs to perform safety function accessibility to critical SSCs in station black out conditions should be considered. It is also explained that the "appropriate period of time" refers to the time after the event which is required to reach and sustain and end state according to RL F3.1 (i). In guidance for point (i) of RL F3.1 it is explained that the end state could be a "safe state" as defined by IAEA SSR 2/1, Rev. 0.

For heat removal functions the guidance is given that if there is an alternative ultimate heat sink, it should be independent as far as practicable from the primary ultimate heat sink (for example, water from river/water from pond, or seawater/air).

Also, the alternative ultimate heat sink should be able to secure the cooling of the core for an extended period of time in case of a design extension condition (beyond the point at which a defined end state - see guidance to RL F3.1 (i) - has been reached).

6.3.2.2 IAEA specific safety guides and technical documents

The guides and documents cover guidance on design requirements of passive safety system and application of these requirements; guidance on classification of structures, systems and components and its application; document on reliability assessment and guidance on deterministic safety analysis.

IAEA SSG-56

According to IAEA SSG-56 [23] systems for core cooling and residual heat removal in accident conditions are designed to remove decay heat from the core in accident conditions with or without a loss of the integrity of the reactor coolant system, to cool the reactor coolant system in the accident conditions until the safe shutdown conditions are reached and to transfer residual heat from the reactor coolant system to the ultimate heat sink. Finally, they are designed to maintain the long term safe shutdown conditions.

The ultimate heat sink is a medium into which the transferred residual heat can always be accepted. The ultimate heat sink is usually a body of water (including groundwater) or the atmosphere [23].

According to IAEA SSG-56 [23] the guide for design basis is:

"*A design basis should be defined for every structure, system and component and should specify the following:*

(a) Function(s) to be performed by the structure, system or component;

(b) Postulated initiating events that the structure, system or component has to cope with;

(c) Loads and load combinations the structure or component is expected to withstand;

(d) Protection against the effects of internal hazards;

(e) Protection against the effects of external hazards;

(f) Design limits and acceptance criteria applicable to the design of structures, systems and components;

(g) Reliability;

(h) Provisions against common cause failures within a system and between systems belonging to different levels of defence in depth;

(i) Safety classification;

(j) Environmental conditions for qualification;

(k) Monitoring and control capabilities;

(l) Materials;

(m) Provisions for testing, inspection, maintenance and decommissioning."

In the following, information is given, how specific safety guide SSG-56 [23] supports the IAEA design requirements in SSR-2/1 (Rev. 1) [11]. Link is given to paragraphs in SSG-56 [23] to get further guide.

<u>Postulated initiating events</u>: Paragraphs 3.10–3.12 of SSG-56 [23] provide recommendations on meeting Requirement 16 of SSR-2/1 (Rev. 1) [11].

<u>Internal hazards</u>: Paragraphs 3.14–3.17 of SSG-56 [23] provide recommendations on meeting Requirement 17 and paragraph 5.16 of SSR-2/1 (Rev. 1) [11] in relation to internal hazards.

<u>External hazards</u>: Paragraphs 3.19–3.26 of SSG-56 [23] provide recommendations on meeting Requirement 17 and paragraphs 5.17–5.21A of SSR-2/1 (Rev. 1) [11] in relation to external hazards.

<u>Accident conditions</u>: Paragraphs 3.30–3.32 of SSG-56 [23] provide recommendations on meeting Requirement 18 of SSR-2/1 (Rev. 1) [11]. Paragraphs 3.34 and 3.35 of SSG-56 [23] provide recommendations on meeting Requirements 19 and 25 of SSR-2/1 (Rev. 1) [11] for DBAs. Paragraphs 3.37–3.42 of SSG-56 [23] provide recommendations on meeting Requirement 20 of SSR-2/1 (Rev. 1) [11] for DEC without significant fuel degradation.

<u>Design limits and acceptance criteria</u>: Paragraph 3.44 of SSG-56 [23] provides recommendations on meeting Requirements 15 and 28 of SSR-2/1 (Rev. 1) [11].

<u>Reliability</u>: Paragraphs 3.47–3.56 of SSG-56 [23] provide recommendations on meeting Requirements 21–26, 29 and 30 of SSR-2/1 (Rev. 1) [11].

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

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

<u>Safety classification</u>: Paragraphs 3.63–3.66 of SSG-56 [23] provide recommendations on meeting Requirement 22 of SSR-2/1 (Rev. 1) [11]. The recommendations provided in IAEA SSG-30 [24], should also be considered.

<u>Codes and standards</u>: Paragraph 3.139 of SSG-56 [23] provides recommendations on meeting Requirement 9 and paragraphs 4.14–4.16 of SSR-2/1 (Rev. 1) [11].

Proven and widely accepted codes and standards are required to be used for the design of the reactor coolant system and associated systems. The selected codes and standards should be applicable to the particular design and should form an integrated and comprehensive set of standards and criteria. For design and construction, the latest editions of the applicable codes and standards should preferably be considered.

Codes and standards have been developed by various national and international organizations, covering areas such as:

(a) Materials;
(b) Manufacturing (e.g. welding) and construction;
(c) Civil structures;
(d) Pressure vessels and pipes;
(e) Instrumentation and control;
(f) Environmental and seismic qualification;
(g) Pre-service and in-service inspection and testing;
(h) The management system;
(i) Fire protection.

It should be noted that civil structure are not in the scope of sCO2-4-NPP system and components design, therefore they are not included in Sections 6.4 and 0 describing nuclear and conventional codes and standards, respectively. Also, the fire protections codes and standards are not included.

<u>Systems for residual heat removal in accident conditions (for PWRs)</u>:

Paragraphs 6.43–6.48 of SSG-56 [23] provide recommendations on meeting Requirements 7, 19 and 29 of SSR-2/1 (Rev. 1) [11] and supplement the generic recommendations in paragraphs 3.33-3.42 and 3.48-3.52 of SSG-56 [23]. They provide recommendations for the design of system necessary to remove residual heat from the reactor coolant system in all accident conditions except for the design extension conditions with core melting.

The needs for different, independent and diverse systems depend on the necessary reliability of the safety systems and on potential vulnerabilities to common cause failures among their redundancies. Systems designed for cooling the core in design basis accidents or design extension conditions without significant fuel degradation should be independent, to the extent possible, from those for operating conditions and design extension conditions with core melt. Safety systems should be designed to meet the regulatory criteria specified for DBAs. The performance of safety features for DEC should be adequate to prevent accident conditions without significant fuel degradation from escalating to design extension conditions with core melting. For design, the same engineering criteria as those applied for DBAs can be used, but less conservative hypotheses and conditions are generally considered.

Paragraphs 6.85–6.88 of SSG-56 [23] provide recommendations on meeting Requirement 51 of SSR-2/1 (Rev. 1) [11] for the removal of residual heat from the reactor core in DBAs. The system should be designed in accordance with the recommendations provided in paragraphs 3.47–3.52 of SSG-56 [23] for safety systems. The system should be designed to remove the decay heat and cool the reactor coolant system (RCS) to safe shutdown conditions. Pressure retaining equipment should be designed and manufactured using proven codes and standards widely used in nuclear industry (e.g. ASME Section III, RCC-M or JSME).

Paragraphs 6.90 and 6.91 of SSG-56 [23] provide recommendations on meeting Requirement 51 of SSR-2/1 (Rev. 1) [11] for the removal of residual heat from the reactor core in DEC without significant fuel degradation. Additional design provisions should be considered in case of loss of normal and safety systems for residual heat removal, including implementation of a secondary side passive heat removal system (note of authors: this statement support the design of sCO2-4-NPP passive decay heat removal system).

Systems for residual heat removal in accident conditions (for BWRs):

Paragraphs 7.20 and 7.22 of SSG-56 [23] provide recommendations on meeting Requirement 51 of SSR-2/1 (Rev. 1) [11] for the removal of residual heat from the reactor core in DBAs and DEC without significant fuel degradation.

The design of the plant should include additional systems to remove residual heat from the reactor coolant system in the event of DBAs where the systems operated in normal shutdown conditions are not designed to meet the engineering design requirements applicable to safety systems. The need for DEC additional safety features to ensure the emergency cooling of the core in the event of a loss of coolant accident combined with multiple failures in the emergency core cooling system should be evaluated, and appropriate measures should be implemented as necessary.

IAEA TECDOC-1791

The main purpose of IAEA TECDOC-1791 technical document [27] is to provide insights and approaches in support of the practical application of the new crucial requirements (i.e. design requirements for safety of nuclear power plants) in IAEA SSR-2/1 [11] and subsequently reinforced in SSR-2/1 [11]. The IAEA TECDOC-1791 [27] also identifies some terms that need to be explained consistently with the requirements. In document IAEA TECDOC-1791 [27] it is also stated that this publication could also be used as the basis for a future Safety Guide.

A technical discussion on the following selected topics is given:

- Categories of plant states
- Concept of defence in depth
- Concept of independence of the safety provisions at different levels of defence in depth
- Concept of practical elimination
- Cliff edge effects and safety margins
- Design for external hazards
- Use of non-permanent equipment for accident management
- Reliability of the ultimate heat sink

The last topic is of special relevance for sCO2-4-NPP project. It described the issue and provided guidance on the understanding of the ultimate heat sink, the relevant challenges to reliable heat transfer including the need for diversity as well as comprehensiveness of the systems and components to be covered. For design basis the technical information on IAEA SSR-2/1 [11] Requirement 53 (see Section 6.2.3) is also given [27]:

"*The design bases of SSCs accomplishing the heat transfer to the ultimate heat sink need to be defined with sufficient margins against postulated external hazards and with high levels of reliability. Reliability of the heat transfer function can be ensured by a number of safety provisions, including high quality, redundancy, diversity, physical separation, etc. as appropriate*."

As IAEA SSR-2/1 [11], Requirement 53, relates to all plant states:

"*If the loss of the heat transfer chain has been selected as DEC, the safety features to backup the heat transfer chain need to be independent from the systems to remove residual heat used at the 3a level of defence. This may include the need for an alternate ultimate heat sink or connecting point as being currently required in SSR-2/1*".

As stated above, for DEC the backup system for heat removal need to be independent from DBA safety systems, which may require the need for alternate heat sink, which again supports the proposed sCO2-4-NPP passive system for decay heat removal.

IAEA SSG-30

IAEA SSG-30 [24] standard provides recommendations and guidance on how to meet the requirements established in IAEA SSR-2/1 Rev. 1 [11] for the identification of structures, systems and components (SSCs) important to safety in nuclear power plants and for their classification on the basis of their function and safety significance.

<u>Safety category 1</u> is any function that is required to reach the controlled state after an anticipated operational occurrence or a design basis accident and whose failure, when challenged, would result in consequences of ‘high’ severity.

<u>Safety category 2</u> is any function that is designed to provide a backup of a function categorized in safety category 1 and that is required to control design extension conditions without core melt.

<u>Safety category 3</u> is any function that is required to mitigate the consequences of design extension conditions, unless already required to be categorized in safety category 2, and whose failure, when challenged, would result in consequences of ‘high’ severity.

As all existing nuclear power plants already have the heat removal safety function, the sCO2-4-NPP heat removal systems could be used as a backup system for the design basis accidents or for the design extension conditions (e.g. SBO), the safety function is safety category 2 or 3 (only when used for mitigation of design extension conditions).

Once the safety categorization of the functions is completed, the SSCs performing these functions should be assigned to a safety class. All SSCs required to perform a function that is safety categorized should be identified and classified according to their safety significance following a process that takes into account the factors indicated by Requirement 22 of IAEA SSR-2/1 Rev. 1 [11]. These factors are safety functions to be performed by the SSCs, the consequences of failure to perform safety function, the frequency with which the SSC will be called upon to perform safety function and the time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function.

By assigning each SSC to a safety class, a set of engineering, design and manufacturing rules can be identified and applied to the SSCs to achieve the appropriate quality and reliability. An SSC implemented as a design provision should, however, be classified directly:

- Safety class 1 (SC1): Any SSC whose failure would lead to consequences of ‘high’ severity.
- Safety class 2 (SC2): Any SSC whose failure would lead to consequences of ‘medium’ severity.
- Safety class 3 (SC3): Any SSC whose failure would lead to consequences of ‘low’ severity.

The adequacy of the classification should be verified by deterministic safety analysis, taking into account insights probabilistic safety assessment and/or supported by engineering judgement.

By assigning each SSC to a safety class, a set of engineering, design and manufacturing rules can be identified and applied to the SSCs to achieve the appropriate quality and reliability.

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

- Design requirements applied at the system level may include specific requirements, such as single failure criteria, independence of redundancies, diversity and testability.
- Design requirements applied for individual structures and components may include specific requirements such as environmental and seismic qualification, and manufacturing quality assurance procedures. They are typically expressed by specifying the codes or standards that apply (typically those widely used by the nuclear industry).

Once the engineering design requirements have been identified for systems and their individual components, it should be verified that the system can perform its function with the reliability that was assumed in the safety analysis.

In the light of the above, the sCO2-4-NPP decay heat removal system function is suggested to be categorized as safety category 2. This suggests that the sCO2-4-NPP system should be classified as safety class 2. The requirements specified in sections 6.1 and 6.2 for passive safety system apply. Design requirements applied to individual components may include specific requirements such as environmental and seismic qualification, and manufacturing quality assurance procedures.

IAEA TECDOC-1787

Document IAEA TECDOC-1787 [26] describes how to complete the tasks associated with every step of the classification methodology set out in IAEA SSG-30 — in particular, how to capture all the SSCs to be safety classified.

The objective of safety classification is to link the safety significance of functions to design requirements (capability, reliability and robustness) of the SSCs performing these functions. Once the safety categorization of the functions is completed, the SSCs performing these functions are then assigned to a safety class corresponding to the safety category of the function they perform (e.g. safety category 2 to safety class 2).

To determine the classification at the component level, the four factors given in SSG-30 [24] should be considered (safety function, the consequences of failure to perform a safety function, the frequency with which the item will be called upon to perform a safety function and time or the period the item will be called upon to perform a safety function).

Practically, capability and reliability of systems performing a categorized function is achieved by meeting design requirements relevant for the safety class of the system. Document IAEA TECDOC-1787 [26] gives a set of typical generic design requirements for systems:

- Single failure criterion (not required for SC2 and SC3)
- Physical & electrical separation (Yes for redundant SC2 and SC3 equipment)
- Emergency power supply (Yes for SC2 and SC3)
- Periodic tests (Yes for SC2 and SC3)
- Protected against or designed to withstand hazard loads (Yes for SC2 and SC3)
- Environmental qualification (Yes for SC2 and SC3)

Document IAEA TECDOC-1787 [26] gives also requirements applicable to individual structures and components (generic consideration, seismic requirements, environmental qualification, pressure retaining equipment, supports, electrical systems, and I&C equipment).

<u>Generic consideration</u>: By assigning a safety class to every individual SSC, a set of design and manufacturing requirements needs to be established to meet the requested quality and reliability objectives. Generally, design/manufacturing requirements and codes to be used are defined for a type of equipment (civil structure, pressure retaining equipment, electrical or I&C equipment). For specific equipment, requirements may be directly defined in the equipment specification.

<u>Seismic requirements</u>: Document IAEA TECDOC-1787 [26] also stress that in relation to the safety classification proposed by the Safety Guide SSG-30 [24], seismic requirements are expected to be established considering the following:

- *“Safety class 2 systems designed to reach and maintain safe state after design basis accident are expected to keep the operability in place in case of earthquake of level SL-2;*

- *The operability of safety class 2 systems designed as a backup of a safety class 1 system may not be needed, provided that an earthquake is not part of a combination of failures considered as a design extension condition for which the backup is designed;"*

Following NS-G-1.6 [30] two levels of earthquakes (SL-1 level earthquake and SL-2 level earthquake) have been defined based on the severity of the ground motion. SL-2 is associated with the safe shutdown earthquake (SSE) and corresponds to the severity to be considered for licensing the plant. SL-1 corresponds to a less severe, more probable earthquake level that normally has different safety implications.

Environmental qualification: It provides evidence that safety classified equipment is able to fulfil its required function(s) during design basis accidents or design extension conditions, despite the harsh environmental conditions. The location of item and the mission of item need to be considered. The environmental conditions depend on the location. The mission time depends whether the item is needed for a short time or for reaching and maintaining safe plant state.

Pressure retaining equipment: Document IAEA TECDOC-1787 [26] gives as an examples of well-established codes defining design and manufacturing requirements for pressure retaining equipment the following:

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

All the above standards are further explained in Section 6.4.1. Table 3 shows relationship between safety classes SC2 and SC3 and code requirements for pressure retaining equipment.

Table 3: Relationship between safety class and code requirements for pressure retaining equipment (adapted per Table 18 of [26])

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

Note: Cat. 2 is Safety category 2; Cat. 3 is Safety category 3 (see definition in Section 0)

Supports: Design and manufacturing requirements of supports are determined on the principle that support is as important as the component being supported. ASME III Division 1, Section III, subsection NF should be used when ASME code is applied. In case of supports for RCC-M1 and RCC-M2 components the requirements of the dedicated RCC-M subsection are applied (Volume H, requirements for S1 and S2 supports, respectively).

Electrical systems: Electrical equipment includes various types of equipment like alternate current (AC) and direct current (DC) power sources, transformers, switchgears, electrical distribution systems and protection devices. Examples of the correspondence between the safety class of electrical equipment items and codes are provided in the Table 4.

I&C equipment: I&C equipment includes the different I&C systems for the control of the plant in the different plant states, including the monitoring of the plant parameters for accident conditions. The engineering requirements to be applied to I&C systems and components are usually defined in the relevant I&C industry standards (e.g. IEC Standards or IEEE code). Both IAEA SSG-30 [24] and IEC 61226 aim at meeting the overall classification requirements given in IAEA SSR-2/1 [11].

Table 4: Relationship between safety class of electrical equipment items and codes (adapted per Table 19 of [26])

Safety Class	Safety classified electrical equipment items	Code requirement	Example of SSCs
SC2	Electrical equipment supporting Cat. 2 functions in DBAs	IEEE: 1E RCC-E: C1	Electric drives supporting Cat. 2 functions
	Electrical equipment supporting Cat. 2 functions implemented as a backup for a Cat. 1 function	IEEE: Specific requirements RCC-E: C1	Electric drives supporting backup of Cat. 2 functions
SC3	Electrical equipment supporting Cat. 3 functions	IEEE: non 1E RCC-E: C3 + specific requirements	Alternate AC power sources Uninterruptable power supply system for severe accidents Electric drives supporting Cat. 3 functions

IAEA TECDOC-1698

In the IAEA TECDOC-1698 technical report [28] the results of Performance Assessment of Passive Gaseous Provisions collaborative project are described, which scope was to reach a consensus on the definition of reliability of thermal hydraulic passive systems as well as a methodology to assess it, in coordination with the IAEA and other international initiatives on the subject. Reliability evaluation methods are described, system modelling of thermal hydraulic (station blackout scenario and loss of coolant accident), modelling and evaluation of passive system, uncertainty and sensitivity analysis of thermal hydraulic calculation and reliability evaluation.

IAEA TECDOC-1752

In the IAEA TECDOC-1752 technical report [29] specific research objectives were to identify the scope of application and common requirements for a technology neutral methodology for reliability assessment of passive systems for advanced NPPs and to identify a set of common benchmark problems to compare and validate methodologies for reliability assessment of passive systems, including such issues as systematic failure modes and effects analysis (FMEA), component failure rates, treatment of dependencies in fault tree (FT) models, impact from internal and external hazards, etc..

The reliability method has also been presented, consisting of the following steps:

- Definition of accident scenario
- System characterizations (mission of the system, failure mode, success/failure criteria)
- System modelling
- Identification of sources of uncertainty
- Uncertainty quantification
- Sensitivity analysis
- Reliability evaluations
- Integration of passive system reliability in probabilistic safety assessment (PSA)

It may be seen that reliability assessment of passive system requires significant efforts for simulations using thermal hydraulic code, including uncertainty and sensitivity analysis. Reliability has to be also assessed, which may require probabilistic safety assessment (PSA). In the frame of sCO-4-NPP the first step has been performed. Namely, one of the tasks completed was station blackout scenario definition in deliverable D2.1 [32].

IAEA SSG-2

The IAEA specific safety guide SSG-2 [25] provides recommendations and guidance on the use of deterministic safety analysis and its application to nuclear power plants in compliance with the requirements established in IAEA SSR-2/1 (Rev. 1) [11] and GSR Part 4 (Rev. 1) [35]. In the scope sCO2-4-NPP project are deterministic safety analyses for design basis accidents and design extension conditions. Such analyses are primarily required to demonstrate adequate fulfilment of safety functions by the design, to ensure that barriers to the release of radioactive material will prevent an uncontrolled release to the environment for all plant states, and to demonstrate the validity of the operational limits and conditions. The focus of description in the following is on fulfilment of safety functions by the design. IAEA SSG-2 [25] addresses the use of deterministic safety analysis for design or licensing purposes, aimed at demonstrating, with adequate margins, compliance with established acceptance criteria.

The specific safety guide IAEA SSG-2 [25] covers:

- general considerations;
- identification, categorization and grouping of postulated initiating events (PIE) and accident scenarios;
- acceptance criteria for deterministic safety analysis;
- use of computer codes for deterministic safety analysis;
- general approaches for ensuring safety margins in deterministic safety analysis;
- deterministic safety analysis for different plant states;
- documentation, review and updating safety analysis;
- independent verification of deterministic safety analysis by licensee.

In the frame of the sCO2-4-NPP analyses of long term station blackout will be performed for the definition of initial and boundary conditions and for analysis of the performance of sCO2-4-NPP system under accident scenarios based on scaled-up components data. Later, after integration of the heat recovery system, simulation of sCO2-4-NPP loop in a real NPP using real design parameters will be performed. The results obtained will then be used to establish the parameters of the simulations on the glass model and on the control room simulator. Real-time simulations will be performed on PWR simulator for validating sCO2-4-NPP loop in a virtual “KONVOI” PWR. The results will then be used to define regulatory roadmap. This mean that in the frame of sCO2-4-NPP it is not expected to perform real design or licensing analysis. Therefore not all above topics recommended by IAEA are envisaged to be performed.

The analyses are expected to be performed using best estimate computer codes without uncertainty analysis, i.e. Option 2 approach as defined in IAEA SSG-2. Release of radioactive materials will not be considered, as well as identification, categorization and grouping of PIE and accident scenarios. Regarding acceptance criteria, IAEA SSG-2 recommends that the range and conditions of applicability of each individual criterion should be clearly specified. For the use of computer codes verification and validation is expected. The assessment of the individual computer code should include identification of important phenomena, estimation of uncertainties in numerical methods, estimating uncertainties of the main models used in the code and sensitivities of the main parameters. The results should be compared to experimental data for significant phenomena. It should be confirmed that the nodalization and the plant models provide a good representation of the behaviour of the plant, that input data are correct, and the results of calculations are evaluated and adequately and used correctly. The deterministic safety analysis should demonstrate that the associated safety requirements are met and that adequate margins (depending on the plant state) exist.

For the analyses of DEC without core melt the same or similar technical and radiological criteria as those for design basis accidents may be considered to the extent practicable. From the point of availability of systems, only systems shown to be operable for category of DEC should be credited in the analysis. Safety systems that are not affected by the failures assumed in the design extension conditions without significant fuel degradation sequence may be credited in the analysis. For DEC without significant fuel degradation, the single failure criterion does not need to be applied. Furthermore, the unavailability of safety features for category of DEC due to maintenance may not need to be considered. For operator actions best estimate assumptions may be used for DEC. For DEC without significant fuel degradation, in principle the combined approach (Option 2) or the best estimate approach with quantification of uncertainties may be used (Option 3). However, in line with the general rules for analysis of DEC, best estimate analysis without a quantification of uncertainties may also be used. Finally, requirements on documentation are given. The documentation of results should typically include the following information:

- A chronological description of the main events as they have been calculated;
- A description and evaluation of the accident on the basis of the parameters selected;
- Figures showing plots of the main parameters calculated;
- Conclusions on the acceptability of the level of safety achieved and a statement on compliance with all relevant acceptance criteria, including the adequacy of margins;
- Results of sensitivity analyses, as appropriate.

6.3.2.3 US regulatory guides

The Regulatory Guide series provides guidance to licensees and applicants on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. Regulatory guides are issued in the 10 broad divisions, where power reactors being division 1. Each guide is identified by a number composed of the regulatory guide designator (RG), followed by a division number, a period, and a sequential guide number (e.g., RG 1.26). In the following some relevant regulation for design, and codes and standards is given, before relevant guides for the sCO2-4-NPP system design are briefly described.

U.S. regulation for design

U.S. legislation requires that an application for a construction permit must include the principal design criteria for a proposed facility. Also application for a manufacturing license must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

The development of these general design criteria (GDC) is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. These matters also include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety.
(2) Consideration of redundancy and diversity requirements for fluid systems important to safety.

GDC are prescribed in 10CFR50 Appendix A [19]. There are in total 55 such criteria, divided into the following six categories (in the parenthesis the criteria numbers are given):

I. Overall requirements (Criteria 1 to 5)
II. Protection by Multiple Fission Product Barriers (Criteria 10 to 19)
III. Protection and Reactivity Control Systems (Criteria 20 to 29)
IV. Fluid Systems (Criteria 30 to 46)
V. Reactor Containment (Criteria 50 to 57)
VI. Fuel and Radioactivity (Criteria 60 to 64)

For the purpose of sCO2-4-NPP general design criteria covering overall requirements and fluid systems are relevant. They are listed in the following (for more details refer to 10CFR50 Appendix A [19]):

- Criterion 1 - Quality standards and records
- Criterion 2 - Design bases for protection against natural phenomena
- Criterion 3 - Fire protection
- Criterion 4 - Environmental and dynamic effects design bases
- Criterion 5 - Sharing of structures, systems, and components
- Criterion 34 - Residual heat removal.

U.S. regulation for codes and standards

The 10CFR50.55a regulation [84] endorses certain codes and standards by reference; the requirements of those codes and standards become part of the regulations (legally binding), except as modified by the referencing statement. The structure of 10CFR50.55a regulation is:

(a) Documents approved for incorporation by reference

(b) Use and conditions on the use of standards

(c) Reactor coolant pressure boundary

(d) Quality Group B components

(e) Quality Group C components

(f) Preservice and inservice testing requirements

g) Preservice and inservice inspection requirements.

(h) Protection and safety systems

Three key “paragraphs” for ASME codes are:

- (b)(1) for BPV Code Section III
- (b)(2) for BPV Code Section XI

- (b)(3) for OM Code Section IST

where BPV means Boiler Pressure Vessel. The ASME issues Code Cases as an approved alternative to the "code". Since the NRC endorses the "code", clearly they should decide if these code alternatives are acceptable.

NRC uses four RGs to manage this Code Case endorsement process:

- RG 1.84 – NPP Construction (BPV Code Section III)
- RG 1.147 – NPP ISI (BPV Code Section XI)
- RG 1.192 – NPP IST (OM Code Section IST)
- RG 1.193 – Unacceptable Code Cases

where ISI means inservice inspection and IST inservice testing.

The paragraph for IEEE standard is:

- (h) (3) for IEEE Std. 603

The safety criteria of IEEE Std. 603 [105] are important for power, control, and instrumentation systems, so they are incorporated by reference in 10 CFR 50.55a(h).

The components classified Quality Group B and C must meet the requirements for Class 2 and 3 Components in Section III of the ASME BPV Code. Guidance for quality group classifications of components may be found in Regulatory Guide 1.26 [86].

Regulatory guide 1.26

Regulatory guide (RG) 1.26 [86] provides an acceptable approach for identification of Quality Group B, C, and D items on a functional basis. Quality Group A components must meet the requirements for Class 1 components in Section III of the ASME BPV Code [63]. The quality group B, C and D standards are given in Table 5. Quality group B components include also part of systems or portions of systems important to safety that are designed for residual heat removal. Quality Group C components include also part of residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (i) do not operate during any mode of normal reactor operation and (ii) cannot be tested adequately.

Table 5: Quality group standards (adapted per Table 1 of [86])

	QUALITY STANDARDS		
Components	**Quality Group B**	**Quality Group C**	**Quality Group D**
Pressure Vessels	ASME BPV Code, Section III, "Rules for Construction of Nuclear Facility Components," Class 2	ASME BPV Code, Section III, "Rules for Construction of Nuclear Facility Components," Class 3	ASME BPV Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels"
Piping	Class 2	Class 3	ASME B31.1
Pumps	Class 2	Class 3	Manufacturers' standards
Valves	Class 2	Class 3	ASME B31.1
Atmospheric Storage Tanks	Class 2	Class 3	API-650, AWWA D-100, or ASME B96.1
15 psig Storage Tanks	Class 2	Class 3	API-620

Regulatory guide 1.84

This regulatory guide lists the ASME Boiler and Pressure Vessel (BPV) Code, Section III Code Cases that the U.S. Nuclear Regulatory Commission (NRC) has approved for use as voluntary alternatives to the mandatory ASME BPV Code provisions that are incorporated by reference into 10CFR50 [84]. Namely, in the event there is an urgent need for alternative rules concerning materials, construction, or inservice inspection activities not covered by existing Boiler and Pressure Vessel Code rules, or for early implementation of an approved Code revision, ASME may issue a Code Case. Code Cases are effective immediately upon ASME approval and do not expire. Approved Code Cases are published quarterly in two categories: Boiler and Pressure Vessels (CC-BPV) and Nuclear (CC-NUC).

The RG 1.84 [88] contains new Code Cases and revisions to existing Code Cases that the NRC staff has approved for use, as listed in Tables 1 and 2 of this guide. The RG 1.84 [88] also states the requirements that govern the use of Code Cases. Licensees may voluntarily use Code Cases approved by the NRC as an alternative to compliance with the ASME Section III Code (see Section 6.4.1.1) provisions that have been incorporated by reference into 10CFR50.55a.

Regulatory guide 1.192

The RG 1.192 [93] lists Code Cases associated with the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), Code Section IST [64] that the NRC has approved for use as voluntary alternatives to the mandatory ASME OM Code provisions that are incorporated by reference into [84]. The NRC endorsed OM Code Cases through RG 1.192 guide [93] for the first time in June 2003.

Regulatory guide 1.193

The RG 1.193 [94] lists the ASME Code Cases that the U.S. NRC has determined not to be acceptable for use on a generic basis. A brief description of the basis for the determination is given with each Code Case. Applicants or licensees may submit a request to implement one or more of the Code Cases listed below through 10CFR50.55a(z), which permits the use of alternatives to the Code Case requirements referenced in 10CFR50.55a as long as the proposed alternatives result in an acceptable level of quality and safety.

Regulatory guide 1.89

The RG 1.89 [89] (from review in 2018 it was established that revision is needed, e.g. newer standards of IEEE 323 were not endorsed since 1984, including the latest IEC/IEEE 60780-323:2016). This regulatory guide describes a method acceptable to the NRC staff for complying with 10CFR50.49 [84] with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident. IEEE Std 323-1974 provides an acceptable method. For this reason the reader is referred to subsection 6.4.2.4, describing EN 60780-323:2016 standard [102], which for text used IEC/IEEE 60780-323-2016 standard [101].

Regulatory guide 1.136

The regulatory guide 1.136 endorses ASME and the American Concrete Institute (ACI) jointly published the “Code for Concrete Containments,” also known as either the ASME BVPC, Section III, Division 2, or ACI Standard 359-01.

The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of concrete containments are covered by codes, standards, specifications, and guides that are applicable either in their entirety or in part. In addition to this regulatory guide, the following codes and guides are acceptable to the NRC staff:

- ASME, Section III, Division 2 entitled "Code for Concrete Containments";

- ASME, Section III, Subsection NCA entitled General Requirements for Division 1 and Division 2.

Articles CC-1000 through CC-6000 of the Divison 2 subsections CC – "Concrete Containments and Division 2 Appendices" are acceptable to the NRC staff for the scope, material, design, construction, examination, and testing of concrete containments of nuclear power plants subject to the regulatory positions described in Section C of RG 1.136 [90].

Regulatory guide 1.187

Regulatory guide 1.187 [92] explains the changes (including modifications) as required by 10CFR50.59 [84], which objectives are to ensure that licensees (1) evaluate proposed changes to their facilities for their effects on the licensing basis of the plant, as described in the final safety analysis report, and (2) obtain prior NRC approval for changes that meet specified criteria as having a potential impact upon the basis for issuance of the operating license. The RG 1.187 [92], through its endorsement of a guideline document for licensees, provides guidance on complying with the revised requirements of 10CFR50.59. Acceptable is Revision 1 of NEI 96-07 document [97], which provides methods that are acceptable to the NRC staff for complying with the provisions of 10CFR50.59.

Regulatory guide 1.203

The RG 1.203 [95] describes a process that the staff of the U.S. NRC considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

An evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, e.g.:

- procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
- specification of those portions of the analysis not included in the computer programs (alternative approaches are used)
- all other information needed to specify the calculation procedure

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations.

In the Figure 3 the Evaluation Model Development and Assessment Process (EMDAP) is shown. The principles of an EMDAP were developed and applied in a study on quantifying reactor safety margins (NUREG/CR-5249 [96]), which applied the code scaling, applicability, and uncertainty (CSAU) evaluation methodology to a large-break LOCA. The methodology consists of 4 elements. Element 1 establishes requirements for evaluation model capability (phenomena, processes, key parameters,…). Element 2 deals with the scaling methodology that includes acquiring appropriate experimental data relevant to the scenario being considered and ensuring the suitability of experimental scaling (separate effects tests, integral effects tests). Element 3 deals with the development of evaluation model (structure of the computer code, calculational procedures, code models). Finally, adequacy of evaluation model should be assessed.

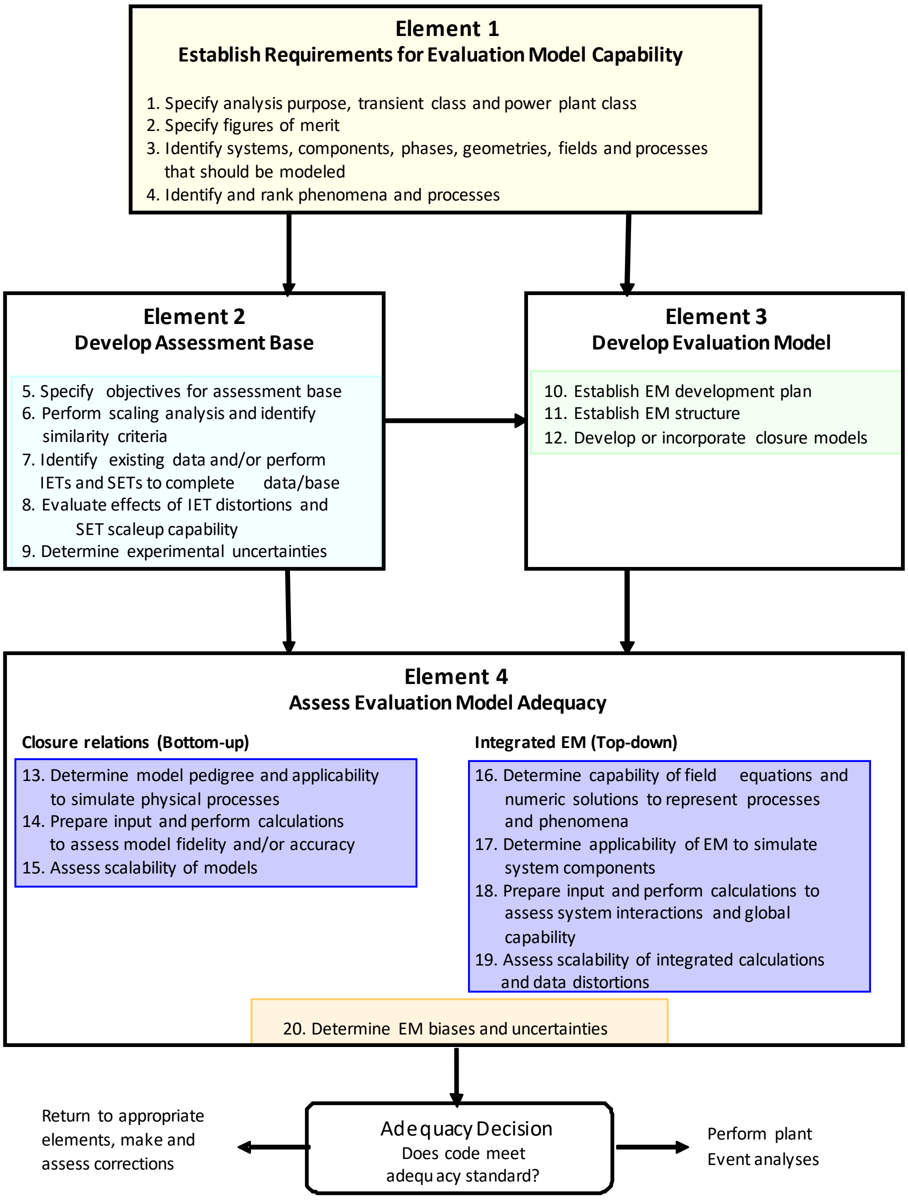


Figure 3: Evaluation Model Development and Assessment Process (EMDAP)

In the context of sCO2-4-NPP evaluation model is needed for simulation of station blackout scenario.

6.3.2.4 Swedish design guide for nuclear civil structures

Swedish Radiation Safety Authority design guide for nuclear civil structure (DNB) [99] describes design provisions for concrete structures at nuclear power plants and other nuclear facilities in Sweden. The scope of DNB includes provisions regarding design and analysis of loadbearing concrete structures covering reactor containments as well as other safety-related structures. Regarding nuclear power plants, DNB can be applied for light-water reactors of type BWR and PWR.

In addition to the Eurocode conventional requirements (see Section 6.5.2.1), additional safety requirements based on laws and regulations valid for nuclear facilities are prescribed. In order to demonstrate that the nuclear safety requirements are fulfilled, other regulations than the Eurocodes need to be referred to, preferably regulations specifically established for nuclear power plants and other nuclear facilities. This is schematically shown in Figure 4, where ASCE 4-98 is entitled "Seismic Analysis of Safety-Related Nuclear Structures and Commentary" and has been developed by American Society of Civil Engineers (ASCE).

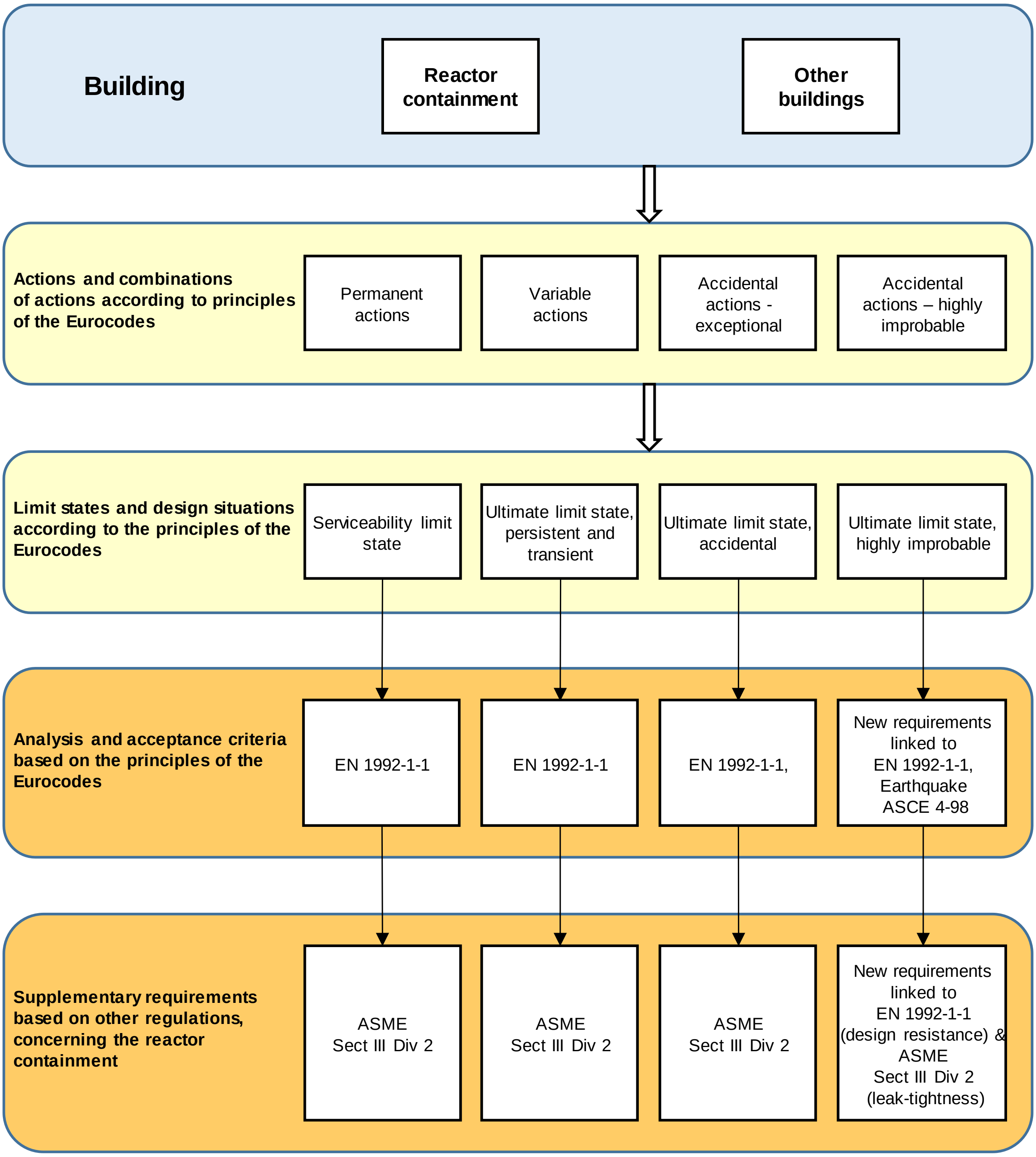


Figure 4: Schematic figure of the arrangement of the design provisions (adapted per [99])

According to DNB [99], the actions and combinations of actions as well as limit states and design situations according to the principles of conventional Eurocode shall be applied to both the reactor containment and other buildings (Eurocode required is EN 1992 [117], one of the ten Eurocodes suite European Standards for structural design, which cover the design of concrete. For details refer to section 6.5.2.1.

Also, requirements, analyses and acceptance criteria according to the Eurocodes are applied in both serviceability limit state and ultimate limit state. Necessary nuclear-related modifications and amendments have been introduced, which is described in general below.

To ensure that the reactor containment function in the event of an accident is not compromised or that its operational life time is not significantly reduced due to normal operation events, the DNB guide provides additional requirements for the reactor containment based on ASME Sect III Div 2.

When combinations of actions for the ultimate limit state are affecting the reactor containment, DNB guide refer to supplementary requirements regarding the containment load-carrying capacity. Following Figure 4 ASME Sect III Div 2 applies to persistent, transient and accidental design situations. Regarding highly improbable design situations, unique requirements based on the Eurocodes have been established in DNB guide since ASME Sect III Div 2 does not cover this type of events.

For details the reader can refer to [99].

6.3.2.5 Plant modification process guides

WENRA RL Q1.1 (see Section 6.1.2.6) requires that licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely. This means that installation of sCO2-4-NPP system should not affect a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.

WENRA RL Q2.2 requires that the process of modification shall include safety assessment. This means also that technical requirements and criteria of existing plant SSC should remain fulfilled.

Guidelines how to treat modifications are described in documents like GRS-AVN-IRSN Safety Assessment Guide [98], IAEA NS-G-2.3 [31] and NEI 96-07 Rev. 1 [97].

GRS-AVN-IRSN Safety Assessment Guide

The purpose of this guide [98] is to provide recommendation to expertise bodies on reviewing and assessing the safety questions raised in nuclear activities. Reviewing and assessing the various safety related issues raised by the nuclear activities concerning nuclear facilities at different stages (siting, design, construction, commissioning, operation and decommissioning or closure) to determine whether the activities comply with the applicable safety objectives and requirements is one of the principal ways to achieve and maintain such a high level of safety in nuclear activities. For example, the assessment also aims at determining whether the proposed modifications to the facility, at whichever stage in its lifetime, have been conceived and their implementation planned so that safety is not compromised.

The assessment method shall satisfy the assessment requirements and shall among other include the acceptance criteria (verification and approval) needed to allow release of the assessment.

While conducting the assessment the expertise body shall verify that the different aspects of the query raised have been properly considered. It shall ensure that the nuclear safety objectives and the safety policy principles are not impaired and that the technical requirements and criteria are fulfilled.

NEI 96-07 Rev. 1

The purpose of NEI 96-07 Rev. 1 document [97] is to provide guidance for developing effective and consistent 10CFR50.59 implementation processes (see RG 1.187 described in section 6.3.2.3).

After determining that a proposed activity is safe and effective through appropriate engineering and technical evaluations, the 10CFR50.59 process is applied to determine if a license amendment is required prior to implementation. This process involves the following basic steps:

- Applicability and Screening: Determine if a 10CFR50.59 evaluation is required.
- Evaluation: Apply the eight evaluation criteria of 10CFR50.59(c)(2) to determine if a license amendment must be obtained from the NRC.

- Documentation & reporting: Document and report to the NRC activities implemented under 10CFR50.59.

In order to perform 10CFR50.59 screenings and evaluations, an understanding of the design and licensing basis of the plant and of the specific requirements of the regulations is necessary. Individuals performing 10CFR50.59 screenings and evaluations should also understand the rule and concepts discussed in this guidance document.

In Section 2 NEI 96-07 Rev. 1 document [97], the relationship between the design criteria established in 10CFR50, Appendix A, and 10CFR50.59 is discussed as background for applying the rule. Section 3 presents definitions and discussion of key terms used in 10CFR50.59 and NEI 96-07 Rev. 1 guideline. Section 4 discusses the application of the definitions and criteria presented in 10CFR50.59 to the process of changing the plant or procedures and the conduct of tests or experiments. Section 4 also includes guidance on the applicability requirements for the rule, the screening process for determining when a 10CFR50.59 evaluation must be performed, and the eight evaluation criteria for determining if prior NRC approval is required. Section 5 provides guidance on documenting 10CFR50.59 evaluations and reporting to NRC.

IAEA NS-G-2.3

IAEA NS-G-2.3 [31] safety guide provides recommendations and guidance on controlling activities relating to modifications to nuclear power plants so as to reduce risk and to ensure that the configuration of the plant is under control at all times, and that the modified configuration conforms to the approved basis for granting an operation licence. The recommendations cover the whole process from conception to completion for modifications to structures, systems and components, operational limits and conditions, procedures and software, and the management systems and tools for plant operation.

6.4 Level IV rules - Nuclear component oriented documents

Nuclear component oriented documents comprise nuclear codes and standards for mechanical and electrical equipment. This level is not covered neither by WENRA nor IAEA documents. An acceptable approach for mechanical equipment is to use ASME Boiler and Pressure Vessel Code, Section III, AFCEN’s RCC-M or KTA 3211 standard for pressure and activity retaining components of systems outside the primary circuit.

For electrical equipment AFCEN’s RCC-E or IEEE Std. 603 [105] for safety systems is an acceptable approach. For electrical equipment qualification European harmonized EN 60780-323:2017 is acceptable (published also as national adoptions by DIN in Germany, NF in France or SIST in Slovenia).

6.4.1 Pressure boundary codes and standards

A nuclear code for design and construction is not only a set of books, but it is also large community. Not only that codes and standards in various countries are different, also the way how codes and standards are applied to structures, systems and components affects the design and construction of nuclear power plants [59]. Codes and standards are important because they play important role in providing technical basis for ensuring nuclear safety. For example, in the OECD/NEA Multinational Design Evaluation Programme (MDEP) participating countries, many safety authorities adopt or approve codes and standards developed by standard developing organisations or equivalent organisations [59]. The development of rules for codes and standards often involves various organisations (utilities, constructors, vendors, academia, regulators etc.). In such manner the complex rules are developed from a broad range of perspectives and voluntary consensus is achieved. These voluntary consensus codes and standards are the part of the framework to establish requirements to structures, systems and components important to safety. The requirements are set for design, fabrication, construction, testing and performance. Participation of regulators provides them opportunity for incorporation of safety views and consistency with regulatory positions and requirements. The report of MDEP Codes and Standards Working Group (CSWG) [59] presented practices in the following MDEP countries: Canada, China, Finland, France, Japan, Republic of Korea, Russian Federation, South Africa, United Kingdom and United States. In general, codes and standards are not enforced. Even if the codes and standards are approved by selected country, the alternatives exist. The MDEP report TR-CSWG-02 [60] recognized that each country's pressure boundary code or standard is comprehensive and living document. The nuclear codes and standards used by countries are the following:

- Canada: CSA (Canadian Standards Association) N285.0 Standard [69] (it provided the approach for adopting the ASME BPVC (Boiler and Pressure Vessel Code) for use in the construction of the CANDU pressure boundary);
- Finland: ASME BPVC Section III [63], RCC-M [70];
- France: RCC- M[70];
- Japan: JSME (Japanese Society of Mechanical Engineers) S NC-1 [71];
- Korea: KEPIC (Korea Electric Power Industry Code) [72] (KEPIC was developed consistent with ASME BVPC layout);
- Russian Federation: PNAE G-7 series [73];
- South Africa: RCC-M;
- UK: ASME BPVC Section III (known), RCC-M (code was unknown before EPR was proposed in UK);
- US: ASME BPVC Section III (over 500 industry standards cited by NRC in regulatory guidance documents in construction of new NPPs).

From above it may be seen that ASME BPVC and RCC-M are widely used standards and are presented in more detail in Sections 6.4.1.1 and 6.4.1.2.

The rules in the pressure boundary codes and standards include comprehensive requirements for the design and construction of nuclear power plant components including design, materials selection, fabrication, examination, testing and overpressure protection. The rules also contain programmatic and administrative requirements such as quality assurance, conformity assessment, welding, qualification of welders and equipment, non-destructive examination (NDE) and qualification of NDE personnel requirements [60].

The report TR-CSWG-02 [60] also provides information that Canada, France, Japan, Korea and Russian Federation originally used the requirements from ASME Boiler and Pressure Vessel Code (BPVC). For Class 1 components comparison of the above five pressure boundary codes with ASME BPVC was also done. The results of the code-comparison project enabled the CSWG to take the next steps towards harmonisation of codes and standards. As for sCO2-4-NPP Class 2 and 3 are in the scope, the harmonization approach for Class 1 components will not be described here and for further information the reader can refer to report TR-CSWG-03 [61]. Nevertheless, some results for Class 1 pressure boundary components are presented, as after determining the usefulness of the code comparison of Class 1 and depending on the industry needs the standard developing organisations will consider expanding this code comparison to Class 2 and Class 3 components.

TR-CSWG-03 report [61] provides the fundamental attributes which have been developed for the codes and standards used in the design and construction of reactor coolant pressure boundary components in nuclear power plants. The fundamental attributes are the basic concepts to be considered in the design, materials, fabrication, installation, examination, testing and over-pressure protection requirements for pressure boundary components. The TR-CSWG-03 report [61] states that IAEA SSR-2/1 [11] includes some requirements in term of design and overpressure protection devices, which can be termed as fundamental attributes for the pressure boundary components, but this is not sufficient. Topics as fabrication, installation, examination and testing are not covered. Table 6 lists all these fundamental attributes, which provide fundamental concepts governing the design and construction of Class 1 pressure boundary components, but approach looks useful also for design and construction of Class 2 and Class 3 components. Namely, the seven topics for fundamental attributes are practically same like titles for specific rules in the pressure boundary codes and standards. The qualitative performance descriptions of the rules and practises from the codes and standards, which can be considered essential, are described in the report TR-CSWG-04 [62]. These essential guidelines can govern most of the pressure boundary codes and standards. They provide guidelines for topics for fundamental attributes: material, design, fabrication and installation, examination, testing and overpressure protection.

The report TR-CSWG-03 [61] also explains that pressure boundary codes and standards of each MDEP country provide specific rules for the design, material, fabrication, installation, examination, testing and overpressure protection. Finally, information is also given that the pressure boundary codes are very large, complex and detailed documents. Therefore, it is difficult for non-code specialist to appreciate the important requirements of pressure boundary codes and standards.

In the following ASME BVPC code, which is an international code, will be first briefly described, followed by description of French RCC-M code.

Table 6: Fundamental attributes per Multinational Design Evaluation Programme (MDEP) Codes and Standards Working Group (CSWG) [61]

Fundamental Attribute	Description
1. General	
Fundamental Attribute 1	Management/quality assurance system for the plant design and construction
Fundamental Attribute 2	Classification and Graded Approach
Fundamental Attribute 3	Service Conditions corresponding to Plant States
2. Design	
Fundamental Attribute 4	Design basis for items important to safety
Fundamental Attribute 5	Provision for inspectability
Fundamental Attribute 6	Design by Analysis
Fundamental Attribute 7	Design by Rule
3. Materials	
Fundamental Attribute 8	Provision for material
Fundamental Attribute 9	Specification of materials
Fundamental Attribute 10	The additional requirements for material
Fundamental Attribute 11	Quality management requirements for materials
4. Fabrication and installation	
Fundamental Attribute 12	Provision for fabrication and installation
Fundamental Attribute 13	Quality management requirements during fabrication
Fundamental Attribute 14	Requirements on fabrication
Fundamental Attribute 15	Requirements on heat treatment
5. Examination	
Fundamental Attribute 16	Provision for examination
Fundamental Attribute 17	Requirements on non-destructive examination
Fundamental Attribute 18	Qualification of non-destructive examination personnel, equipment and procedures
6. Testing	
Fundamental Attribute 19	Provision for testing
7. Over-pressure protection	
Fundamental Attribute 20	Provision for over-pressure protection

6.4.1.1 ASME Boiler and Pressure Vessel Code

ASME has played a vital role in supporting the nuclear industry since its inception in 1963, when ASME codes, standards and conformity assessment programs, originally developed for fossil fuel-fired plants, were applied to nuclear power plant construction. Its widely-adopted BPVC Section III, Rules for Construction of Nuclear Facility Components [66].

Presently, about half of the world's nuclear power plants incorporate all or portions of ASME nuclear codes and standards in their construction, operation, and/ or maintenance. Around sixty nations generally recognize and apply the ASME Boiler and Pressure Vessel Code (BPVC), while two third nuclear nations purchase their nuclear components to specifications contained within ASME's nuclear codes and standards.

Primary nuclear sections of ASME BPVC are [67]:

- Section III, Rules for Construction of Nuclear Facility Components
- Section XI, Rules for Inservice Inspection of Nuclear Power Components
- Common referenced sections:
  - Section II Materials, Parts A through D

- Section V Nondestructive Examination
- Section IX Welding and Brazing Qualification

In addition, Section III referenced Section XI and vice versa.

Section III Material Requirements are:

- Material Specification: Section II: Parts A, B, C, & D
- Control of Material: Section III: NCA-3800
- Special Material Requirements: Section III: NX-2000

where letter “X” in “NX” above is B, C, D, E, F, G. With the construction of commercial nuclear power plants it was recognized that “high” standards for passive component construction needed to be used so that they could operate for their life without attention. Section III was first published in 1963 as already mentioned above and was developed by ASME from the naval reactors program. “Construction” as used in Section III, Division 1 encompasses materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of an item.

ASME Boiler and Pressure Vessel Code, Section III, Division 1 [63] sets rules for construction of nuclear facility components. It provides general requirements which address the material, design, fabrication, examination, testing and overpressure protection of the items specified within each respective Subsection, assuring their structural integrity. Subsection NCA, which is referenced by and is an integral part of Division 1 and Division 2 of Section III, covers requirements for quality assurance, certification, and authorized inspection for Class 1, 2, 3, MC, CS and CC construction [67]. Division 1 subsections are designated by capital letter preceded by letter “N”. Only Class 2 and Class 3 are of interest in the frame of sCO2-4-NPP components design (see shaded boxes in in Figure 5).

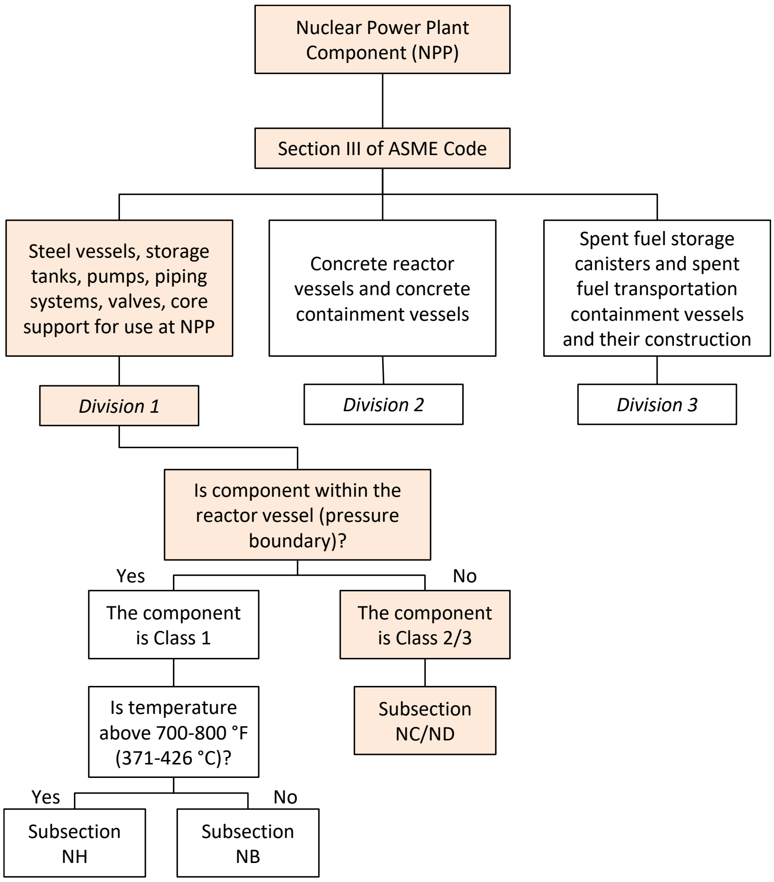


Figure 5: How to go around the ASME code, Section III – construction of components (adapted per [82])

Subsection NCA: Subsection NCA [81] covers general requirements for manufacturers, fabricators, installers, designers, material manufacturers, material suppliers, and owners of nuclear power plants. This Subsection which is referenced by and is an integral part of Division 1, Subsections NB through NG, and Division 2 of Section III, covers quality assurance requirements, ASME Product Certification Marks, and authorized inspection for Class 1, 2, 3, MC (metallic containment), CS (core support structures), and CC (concrete containment) construction. Selective reference of ASME Standard NQA-1, Quality Assurance Program Requirements for Nuclear Facilities, is made in this Subsection (see Subsection 6.3.1.3). NQA-1 provides the programmatic quality assurance requirements for the establishment and execution of a quality assurance program. NQA-1 is not included in the BPVC subscription.

Appendices: There are mandatory and nonmandatory appendices [83].

Mandatory Appendices are invoked within the text of a Code paragraph and are required:

- Appendix-I Design Fatigue Curves
- Appendix-II Experimental Stress Analysis
- Appendix-III Basis for Establishing Design Stress Values
- Appendix-IV Approval for New Material under BPV code
- Appendix-V Certificate Holders Report Forms, Instructions, etc.
- Appendix-VI Rounded Indications
- Appendix VII Charts and Tables for Determining Shell Thickness of Cylindrical and Spherical Components Under External Pressure
- Appendix-XI Rules for Bolted Flange Connections for Class 2, 3, MC
- Appendix-XII Design Considerations for Bolted Connections
- Appendix-XIII Design Based on Stress Analysis
- Appendix-XIV Design Based on Fatigue Analysis
- Appendix-XVIII Capacity Conversions for Pressure Relief Valves
- Appendix-XIX Integral Flat Head with a large Opening
- Appendix XX Submittal of Technical Inquires to BPV Committee
- Appendix-XXI Adhesive Attachment of Nameplates
- Appendix-XXII Rules for Reinforcement of Cone-to-Cylinder Junction Under External Pressure
- Appendix XXIII Qualifications and Duties of Specialized Professional Engineers
- Mandatory Appendix XXIV Standard Units for Use in Equations
- Mandatory Appendix XXV ASME-Provided Material Stress–Strain Data
- Mandatory Appendix XXVI Rules for Construction of Class 3 Buried Polyethylene Pressure Piping

Nonmandatory Appendices are invoked by a footnote to a Code paragraph and provide information or guidance. They use alphabetic designators A through HH.

Subsection NC addresses items which are intended to conform to the requirements for Class 2 construction and Subsection ND addresses items which are intended to conform to the requirements for Class 3 construction. Division 1 subsections are divided into articles. Articles for Division 1 subsections are (where letter “X” in “NX” below is B, C, D, E, F, G):

- Article NX-1000: Introduction
- Article NX-2000 Material
- Article NX-3000 Design
- Article NX-4000 Fabrication and Installation
- Article NX-5000 Examination

- Article NX-6000 Testing
- Article NX-7000 Overpressure Protection
- Article NX-8000 Nameplates, Stamping with Certification Mark, and Reports

Class 2 components subsection NC uses letter C for “X” above and Class 3 subsection ND uses letter “D”. It may be seen that article titles from NX-2000 to NX-7000 are in agreement with the topics for fundamental attributes described in Table 6 and article titles from NX-2000 to NX-6000 match with first five out of seven attributes for all-inclusive term ‘construction’ (comprising materials, design, fabrication, examination, inspection, testing, certification, and pressure relief) as used in Section III.

Subarticles for design are (NX represents Subsections NC and ND) [83]:

- NX-3100 General Design
- NC-3200 Alternate Design Rules for Vessels
- NX-3300 Vessel Design
- NX-3400 Pump Design
- NX-3500 Valve Design
- NX-3600 Piping Design
- NX-3700 Electrical and Mechanical Penetration Assemblies
- NX-3800 Atmospheric Storage Tanks
- NX-3900 Storage Tanks 0-15 psig (1.03 bar gauge)

Design rules for component specific analysis of ASME are defined in Subsections NC and ND 3300-3600. ASME provides alternative design rules for vessels in NC-3200 for Class 2 components in addition to those of NC-3300.

Subarticles for fabrication and installation are (NX represents Subsections NC and ND) [83]:

- NX-4800 Expansion Joints
- NX-4100 General Requirements
- NX-4200 Forming, Fitting and Aligning
- NX-4300 Welding Qualifications
- NX-4400 Making, Examining and Repairing Welds
- NX-4500 Brazing
- NX-4600 Heat Treatment
- NX-4700 Mechanical Joints

Subarticles for examination are (NX represents Subsections NC and ND) [83]:

- NX-5100 General Requirements
- NX-5200 Examination of Welds
- NX-5300 Acceptance Standards
- NX-5400 Final Examination of Components
- NX-5500 Qualification and Certification of NDE Personnel
- NX-5700 Examination Requirements for Expansion Joints

Subarticles for testing are (NX represents Subsections NC and ND) [83]:

- NX-6100 General Requirements
- NX-6200 Hydrostatic Tests
- NX-6300 Pneumatic Tests
- NX-6400 Pressure Test

- NX-6500 Atmospheric & 0-15 psig Storage Tanks
- NX-6600 Special Test Pressure Situations
- NX-6900 Proof Tests to Establish Design Pressure

Subarticles for overpressure protection (NX represents Subsections NB, NC and ND) [83]

- NX-7100 General Requirements
- NX-7200 Overpressure Protection Report
- NX-7300 Relieving Capacity Requirements
- NX-7400 Set Pressures of Pressure Relief Devices
- NX-7500 Operating and Design Requirements for Pressure and Vacuum Relief Valves
- NX-7600 Non-reclosing Pressure Relief Devices
- NX-7700 Certification
- NX-7800 Marking, Stamping With Certification Mark, and Data Reports

Subsection NC: contains rules for the material, design, fabrication, examination, testing, overpressure relief, marking stamping, and preparation of reports for items conforming to the requirements for Class 2 construction. These rules cover the strength and pressure integrity of items the failure of which would violate the pressure retaining boundary. The rules cover load stresses, but do not cover deterioration which may occur in service as a result of corrosion, radiation effects, or instability of materials.

NC-2000 Material: Rules for Class 2 material are similar as for Class 1. More materials are available in Section II, Part D, Subpart 1, Tables 1A (ferrous material), 1B (nonferrous material) and 3 as shown in Figure 6. Plates over 51 mm (2 inch) are not subject to ultrasonic testing (UT) examination. Impact testing is required but it is lower for Class 2 than for Class 1. Dropweight test to show that the lowest service metal temperature is satisfied in accordance with Appendix R to 10CFR is alternate test. Many more materials are also exempted from toughness testing of Class 2 comparing to Class 1.

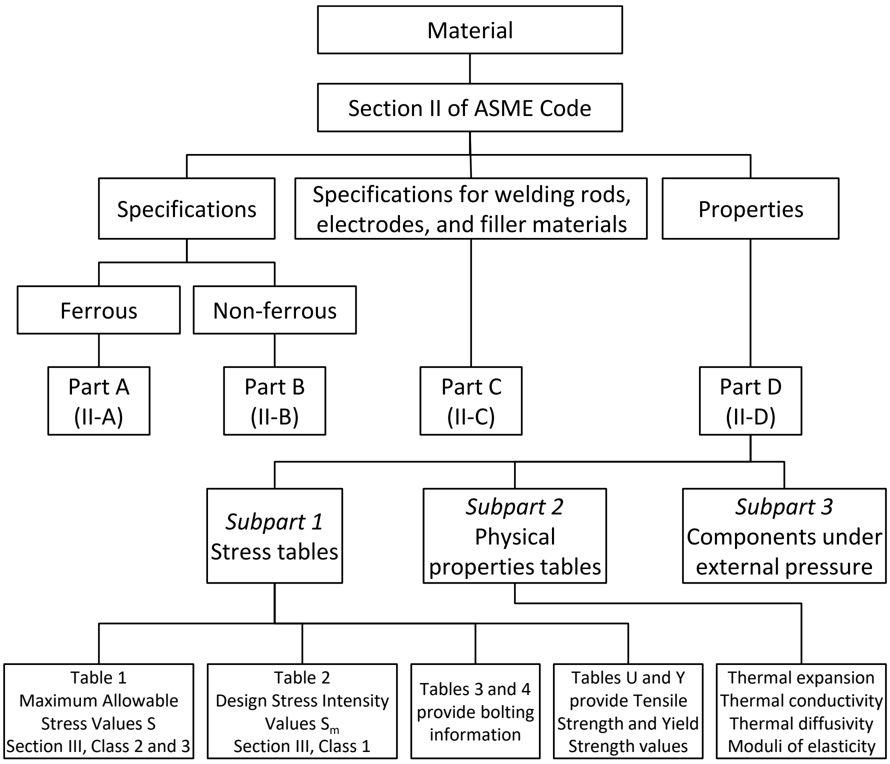


Figure 6: How to go around the ASME code, Section II – materials (adapted per [82])

NC-3000 Design: Design of pumps (ref. [80]) generally use formulas based on vessel design. Design of valves is generally designed to pressure-temperature rating of ANSI B16.34. Piping is designed using simplified formulas and system analysis (similar to ANSI B31.1 Design). Cyclic (fatigue) analysis is optional for Class NC [68]. Design by analysis is permitted for NC components using the Tresca failure theory, but not required.

NC-4000 Fabrication and installation: rules very similar to Class 1 rules (NB-4000). Details of construction are less restrictive. Storage tank fabrication rules are more restrictive than API standards. Qualification of weld procedures and welders are same as Class 1. Requirements for post weld heat treatment (PWHT) are less stringent.

NC-5000 Examination: longitudinal and circumferential welds require radiographic testing (RT). Surface examination of these welds is not required. Personal qualifications for the non-destructive examination (NDE) are the same as Class 1. Penetration welds only require RT when they are butt welds.

Subsection ND: contains rules for the materials, design, fabrication, examination, testing, overpressure relief, marking, stamping, and preparation of reports by the certificate holder for items conforming to the requirements for Class 3 construction. The rules of subsection ND cover the strength and pressure integrity of items the failure of which would violate the pressure retaining boundary. The rules cover load stresses, but do not cover deterioration which may occur in service as a result of corrosion, radiation effects, or instability of materials. The allowable stresses for design for materials are listed in Tables 1A and 1B, section II, Part D, Subpart 1 (see Figure 6). The material shall not be used at metal and design temperatures that exceed the temperature limit in the applicability column for which stress values are given.

ND-2000 Material: Materials are available in Section II, Part D, Subpart 1, Tables 1A (ferrous material), 1B (nonferrous material) and 3 as shown in Figure 6. Other than those in Section II, materials are available for specified applications. Testing is in accordance with material specification. Design specification must state whether impact testing is required.

ND-3000 Design: Design by analysis is not permitted for ND components. Class ND uses allowable stress design rather than design by analysis.

ND-4000 Fabrication and installation: more options are available for fabrication details. Welding and PWHT is same as for Class 2.

ND-5000 Examination: all butt welds are not required to be RT examined in full. NDE examination is based on material size. Construction is more economical. Personal qualifications are the same as Class 1.

Example - pressure design of piping

ASME Section III, Division 1 outlines rules for pressure design in NB-, NC-, ND-3133 [74]. The formula given to calculate the required wall thickness of straight pipes t_m is based on the following equation [74]:

$$t_m = \frac{P \cdot D_0}{2(S_m + P \cdot y)} + A \qquad (1)$$

where P is internal design pressure, D_0 is outside diameter of pipe, S_m is stress intensity, y is parameter to adjust the Boardman equation to the Lamé equation and A is additional thickness. The value of A is used to represent an additional thickness to account for material erosion and corrosion and to provide resistance against mechanical damage. The stress intensity S_m must be replaced by S_m in the equation for Class 2 and 3 components. This equation is called the Boardman equation when y = 0.4 [74].

6.4.1.2 RCC-M technical code

The RCC-M technical code for mechanical equipment was initially based on the US ASME code. When the French government decided to launch a large nuclear program, in the context of the first petroleum crises in the 1970's, the decision was taken to follow American ASME rules for design and to meet French and European standards for procurement, manufacturing and examination. Initially it combines ASME Section III code, Westinghouse PWR specifications and French construction practice. RCC-M [53] lays down design and construction rules for pressure vessels, reactor internals and nuclear island pipework and equipment supports. It codifies French industrial practice and benefits from experience from manufacture, inspection and operation of French units.

After AFCEN was created in 1980, the RCC-M code rules were developed jointly by EDF and Framatome in 1981, with successive editions, which were applied to EPR projects.

Annually, the modification sheets (MS) are gathered in an Addendum, which is published [57]. Modification sheets are sent periodically to the French safety authority. There is no more formal regulatory approval of RCC-M code or its modifications. The 2012 edition (with three addenda in 2013, 2014 and 2015) incorporated initial feedback from EPR projects. The Probationary Phase Rules (RPP) were added in 2013. This is a way of providing an alternative set of rules in cases where industry feedback has not been sufficiently consolidated for permanent inclusion in the code. The 2018 edition [57] introduced NF EN ISO 9001:2015 standard, which replaced NF EN ISO 9001:2008 standard. It also complies to European pressure equipment directive [58]. It should be also noted that there are referenced several ISO or EN standards.

The comparison between RCC-M edition 2018 and ASME code structure is shown in Table 7. The rules for Class 1 components construction are applied to the components of reactor coolant pressure boundary of light water reactors (LWR) and are therefore not applicable to the sCO2-4-NPP components. Candidates are rules for Class 2 or 3, depending on the code classification by the owner. Rules for Class 2 and 3 components cover the same general provisions as Class 1 components [53]. The difference is that approach for damage prevention is not explicitly addressed. The design by rules may be used in RCC-M code in the same way for Class 2 and 3 components as in ASME code [53].

Table 7: General RCC-M structure compared to ASME BPV code (adapted per [55] and [56])

RCC-M Section	Title	ASME code section
SECTION I	NUCLEAR ISLAND COMPONENTS	SECTION III
Subsection A	General	Subsection NCA
Subsection B	Class 1 components	Subsection NB
Subsection C	Class 2 components	Subsection NC
Subsection D	Class 3 components	Subsection ND
Subsection E	Small components	none
Subsection G	Core support structures	Subsection NG
Subsection H	Supports	Subsection NF
Subsection J	Storage tanks	NC/ND 3800-3900
Subsection P	Containment penetrations	Introduction of Subsection NE
Subsection Z	Technical appendices	Appendices
SECTION II	MATERIALS	SECTION II
SECTION III	EXAMINATION METHODS	SECTION V
SECTION IV	WELDING	SECTION IX, part QW
SECTION V	FABRICATION	Various chapters
SECTION VI	PROBATIONARY PHASE RULES	In some ways analogous to ASME "Code Cases"

The structure of subsections of the RCC-M [55] is:

1000 chapters

- Scope (it relates to Subsection A)
- Documentation (relates to Subsection Z)
- Identification (relates to Sections III Examination, Section IV Welding and Section V Fabrication)

2000 chapters

- Prevention of corrosion
- Applicable procurement specification (relates to Section II Materials)

3000 chapters

- Sizing
- Analysis (relates to Subsection Z)

4000 chapters

- Manufacturing and examination (relates to Sections III Examination, Section IV Welding and Section V Fabrication)

5000 chapters

- Hydrostatic tests

6000 chapters

- Overpressure protection

The differences between RCC-M and ASME codes are discussed in document on the codes and standards used in the EPR design [53]. In document [53] it is stated that for Class 2 equipment the regulations were less developed and rules are closer to ASME provisions, except for some changes in structure. An example is integration in RCC-M C.3200 of rules for design by analysis covered in ASME III Appendices XIII and XIV. The document [53] summarised that RCC-M is an adaptation of the ASME approach to the French and European standardisation context, with organisational aspects excluded to permit its adaptation to international projects. The document [53] concludes that although the RCC-M and ASME codes may contain different sets of requirements, they result in components of an equivalent level of quality. For details refer to [53].

Finally, during nuclear standards and codes harmonization effort it was concluded that maximum of international standards (ISO) have to be considered. For example [54], in RCC-M 2012 there were referenced around 240 standards, 170 were International ISO or EN (the rest were NF, ASTM, ANSI, AWS, IAEA and ANSI/MSS SP-43 standards).

6.4.1.3 KTA 3211 safety standard

This is German nuclear design standard. There are four "*Pressure and Activity Retaining Components of Systems Outside the Primary Circuit*" parts of KTA 3211 standard. KTA 3211.1 [75] details materials, KTA 3211.2 [76] outlines design and analysis, KTA 3211.3 [77] contains rules for manufacturing and KTA 3211.4 [78] covers in-service inspection and operational monitoring. In Germany, the most current standard editions of KTA must be used for design work in existing power plants.

KTA 3211.2 [76] for design and analysis specifies the detailed requirements to be met by:

a) the classification into test groups, load case classes and level loadings,

b) the design and analysis of components,

c) the calculation procedures and design principles for obtaining and maintaining the required quality of the components,

d) the documents for the certificates and demonstrations to be submitted.

KTA 3211.2 safety standard [76] states that it applies to the manufacturing of pressure retaining walls of pressure and activity-retaining systems and components of light water reactors important to safety which are not part of the reactor coolant pressure boundary and are operated up to design temperatures of 673 K (400 °C). This is the case, if the plant component is needed to cope with incident regarding direct heat removal system. KTA 3211.2 safety standard [76] includes corresponding KTA design standards to ASME Section III, Div. 1 Subsection NC (Class 2 components) and ND (Class 3 components).

The components shall be classified regarding test groups and materials. The components in the scope of application of the KTA 3211.1 [75] safety standard shall be classified in test groups A1, A2 or A3 depending on design data and dimensions, with consideration of the planned materials and stresses. It should be noted that the allocation to the test groups is made by the licence holder by agreement with the authorized inspector. KTA 3211.2 [76] defines the criteria for classification of a component (see Table 2-1 of [76] entitled Test Groups: Classification criteria and allocation of materials). The design stress intensity values are separately fixed for test groups A1, A2 and A3 in Table 6.6-1 of KTA 3211.2 [76]. The design stress intensity value in test group A1 is S_m (correspond to ASME, Section III, Division 1, Class 1). In test groups A2 and A3 the design stress intensity value is S (correspond to ASME, Section III, Division 1, Classes 2 and 3). The allowable stresses used for the dimensioning of pressure-retaining walls are given in Table 6.7-1 of KTA 3211.2 [76]. They are determined in dependence of the test group, service level loading and stress category. Design loading level is level 0. The load case data comprise the design pressure, the design temperature and additional design mechanical loads. The loadings for the various service limits shall be determined and limited within the analysis of the mechanical behaviour in which case the respective actual loadings and temperatures may be used. There are four loading levels A to D covering operation. Level D is for postulated accidents (design basis accidents).

Section 4 of KTA 3211.2 [76] specifies mechanical and thermal loadings as well as fluid effects that shall be taken into account in the design and calculation. Mechanical and thermal loadings are the loadings caused by the fluid, loadings caused by the component itself, loadings imposed by adjacent components and ambient loadings imposed e.g. by anchor displacement, vibrations due to earthquake.

Components shall be designed in accordance with the rules of Section 5 "Design" of KTA 3211.2 [76]. The general requirements for components and welds shall:

- meet the functional requirements,
- not lead to an increase of loadings/stresses,
- meet the specific requirements of the materials,
- meet fabrication and inspection and testing requirements,
- be amenable to maintenance.

The above mentioned general requirements explained in Section 5.2 of KTA 3211.2 [76] are correlated and shall be harmonized with respect to the component-specific requirements of Section 5.3 of KTA 3211.2 [76]. Component-specific requirements are given for pressure vessels, pump casings, valve bodies, piping systems and component support structures.

Section 6 of KTA 3211.2 [76] deals with dimensioning of welds, claddings, wall thickness allowances, wall thickness, design stress intensities, allowable stresses for dimensioning and nominal operating stress (service loading level A). Dimensioning shall be effected on the basis of the design loading level (Level 0). The components for which pertinent design rules are available in Annex A (parts of pressure retaining wall, pumps, valves and piping systems) of KTA 3211.2 [76] shall be dimensioned to these design rules.

Section 6 of KTA 3211.2 [76] also deals with the general analysis of the mechanical behaviour. It shall be demonstrated by means of the analysis of the mechanical behaviour that the components are capable of withstanding all loadings in accordance with the loading levels. The extent of verification depends on the test group. The loadings result from load cases (Level D in case of sCO2-4-NPP components). For test groups A2 and A3 dimensioning in accordance with Section 6, or if required with Section 8, and depending on the individual case, a simplified analysis of mechanical behaviour according to Section 7 or 8 shall be performed. Mechanical analysis of Section 7 includes loadings; stress/strain loadings; resulting deformations; determination, evaluation and limitation of mechanical forces and moments; mechanical system analysis; stress analysis; fatigue analysis; strain analysis (only if specified strain limits are to be adhered to for functional reasons); structural analysis (where under the effect of loading a sudden deformation without considerable increase in load may be expected); stress, strain and fatigue analyses for flanged joints and avoidance of thermal stress ratcheting for components of test group A1 (not applicable to sCO2-4-NPP components).

In Section 8 component-specific analyses are specified for vessels, pumps, valve bodies, piping systems and integral areas of component support structures.

In the following some examples are given for dimensioning of wall thickness, design by rule and materials.

Example - dimensioning of wall thickness

The KTA formula for pressure design of straight pipe is derived [76]. It is assumed that the material behavior is elastic-ideal plastic with yield stress S_m (design stress intensity for components of test group A1). The series expansion of the logarithmic equation is then as follows [74]:

$$p = S_m . \ln\left(\frac{d_a}{d_i}\right) = S_m \cdot \frac{2 \cdot s_0}{d_a - s_0} \cdot \left(1 + \frac{1}{3} \cdot \frac{s_0^2}{(d_a - s_0)}\right) + \cdots \tag{2}$$

where the variables and p, d_a, d_i, and s_0 are design pressure, outside diameter, inside diameter and wall thickness, respectively.

The KTA formula (KTA 3211.2, A 2.2-1 [76]) is the antecedent of the series expansion:

$$s_0 = \frac{d_a \cdot p}{2 \cdot S_m + p} \tag{3}$$

The calculation method hereinafter applies to cylindrical shells under internal pressure, where the ratio $d_a/d_i < 1.7$. Diameter ratios $d_a/d_i < 2$ are permitted if the wall thickness s_{0n} does not exceed 80 mm, where s_{0n} is nominal wall thickness of the shell excluding allowances according to Section 6.5 of KTA 3211.2 [76]. The discrepancy between the exact solution and the approximation of KTA is less than 1% for ratios $d_a/d_i < 1.4$ [74].

As in ASME, Section III, Division 1 [63], the stress intensity S_m must be replaced by S in the equation for test group 2 and 3 (ASME Class 2 and 3 components).

The nominal wall thickness s_n shall satisfy the following condition in consideration of the allowances c_1 and c_2:

$$s_n \geq s_0 + c_1 + c_2 \tag{4}$$

where s_0 is the calculated wall thickness, c_1 is the absolute value of the minus tolerance, which is based on the fabrication tolerance and c_2 is a value that accounts for wall thickness reduction due to wear (for details see Section 6.5 of KTA 3211.2 [76]). Paper [74] shows that for a specific example ($T_{design} = 700$ °F, $P_{design} = 2000\ psig$, $D_0 = 16"$ and $S_m = 15.1\ kSi$) wall thickness of ASME is 1 % higher than for KTA.

Example - design by rule

The component specific design criteria of KTA 3211.2 [74] are nearly identical to the methods of ASME nuclear design codes for Class 1, 2 and 3 [63]. Design criteria for vessels, pumps, valves and piping system are presented in KTA section 8.2 to 8.5, respectively. However, there are some exceptions. For ASME Class 2 and 3 (KTA test groups A2 and A3) piping the limitation of the primary stress intensity can be performed based on identical equations in KTA 3211.2 [74] and ASME, Section III or an additional equation in KTA, Section 8, which contains stress intensification factor (an i-value) instead of stress index (B-index). The range of resultant moments and amplitudes of longitudinal forces resulting from anchor motions due to reversing type dynamic level D loadings are not limited in KTA (whereas ASME provides limits). In KTA 3211.2 [74], dynamic anchor motions have to be evaluated with equations (8.5-91) and (8.5-93) of KTA 3211.2 [74] for the integrity proofs of service levels A and B.

Example - materials

KTA 3211.1 [75] does not have a general section dedicated to materials comparable to ASME Section II [67], which include information for the design analysis regarding material properties. The regulations are non-uniform in the different KTA standards. For KTA 3211.2 [76], the materials which are permitted are listed in KTA 3211.1 [75]. The stress intensities S and S_m are calculated with Table 6.6-1 of KTA 3211.2 [76] based on the material properties, which are provided in Annex A of KTA 3211.1 [75].

6.4.2 Codes and standards for electrical equipment

6.4.2.1 RCC-E technical code

The RCC-E code (technical code for electrical equipment) [53] is a unique example of a code which gathers all requirements applicable to the electrical components (I&C included) of a nuclear power plant. It makes references to international standards (IEC standards) and French standards (AFNOR standards). RCC-E is published by AFCEN. The RCC-E comprises a set of technical rules to be applied and implemented by a contractor, manufacturer or supplier in the design and construction of electrical equipment.

The RCC-E is divided in six volumes as follows [53]:

- Volume A: General and quality requirements
- Volume B: Qualification
- Volume C : Functional system design
- Volume D: Installation
- Volume E : Constituent parts of equipment
- Volume MC: Verification and testing methods

It is believed that there is no equivalent code to RCC-E in the US. The US uses IEEE standards, which are separated documents.

6.4.2.2 IEEE standards

IEEE Std. 603

There are only two standards referenced in rulemaking 10CFR50.55a, IEEE Std. 603 [105] for safety systems (including protection system) and IEEE Std. 279 [106] for protection systems, which is not applicable to the scope of sCO2-4-NPP.

IEEE Std. 603 safety criteria are important for power, control, and instrumentation systems.

The content is the following:

- 1. Overview
- 2. Normative references
- 3. Definitions
- 4. Safety system design basis
- 5. Safety system criteria
- 6. Sense and command features – functional and design requirements
- 7. Execute features – functional and design requirements
- 8. Power source requirements

Minimum functional and design criteria for the power, instrumentation, and control portions of nuclear power generating station safety systems are established. The criteria are to be applied to those systems required to protect the public health and safety by functioning to mitigate the consequences of design basis events. The intent is to promote appropriate practices for design and evaluation of safety system performance and reliability. Although the standard is limited to safety systems, many of the principles may have applicability to equipment provided for safe shutdown, post-accident monitoring display instrumentation, interlock features, or any other systems, structures, or equipment related to safety.

6.4.2.3 IEC standards

IEC 61226

IEC 61226:2020 [100] establishes, for nuclear power plants, a method of assignment of the functions specified for the plant into categories according to their importance to safety. Subsequent classification of the I&C and electrical power systems performing or supporting these functions, based on the assigned category, then determines relevant design criteria.

The design criteria, when applied, ensure the achievement of each function in accordance to its importance to safety. In this document, the criteria are those of functionality, reliability, performance, environmental qualification (e.g. seismic) and quality assurance (QA).

This edition includes the following significant technical changes with respect to the previous edition:

- to align on IAEA requirements, recommendations and terminology, particularly to take into account the replacement of NS-R-1 by SSR 2/1 and publication of SSG 30;
- to extend the scope to electrical power systems;
- to move the detailed requirements applying to functions and I&C systems to a normative annex, which will be removed after updating.

As the title of IEC 61226:2020 [100] indicates, categorization of functions and classification of systems deal with instrumentation, control and electrical power systems important to safety. It is applicable for the sCO2-4-NPP project, e.g. for conceptual development of I&C.

6.4.2.4 EN standard

EN 60780-323:2017

Work to harmonize separate IEC 60780, “Nuclear power plants - Electrical equipment of the safety system – Qualification” and IEEE 323, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations” standards into a single, globally accepted standard began in 2010. It was jointly developed by Institute of Electrical and Electronics Engineers (IEEE) and the International Electrotechnical Commission (IEC) and released in 2016 as IEC/IEEE 60780-323-2016 standard [101]. The qualification requirements in IEC/IEEE 60780-323 are intended, when met, to demonstrate and document the safety of electrical equipment under applicable service conditions and to reduce the risk of environmentally induced, common-cause equipment failure. In 2017 IEC/IEEE 60780-323-2016 [101] was adopted as European Norm EN 60780-323 “Nuclear facilities - Electrical equipment important to safety - Qualification - IEC/IEEE 60780-323:2016” [102]. EN 60780-323 describes the basic requirements for qualifying electrical equipment important to safety and interfaces (electrical and mechanical) that are to be used in nuclear facilities. The principles, methods, and procedures described are intended to be used for qualifying equipment, maintaining and extending qualification, and updating qualification, as required, if the equipment is modified. The qualification requirements in this standard, when met, demonstrate and document the ability of equipment to perform safety function(s) under applicable service conditions, including design basis events and certain design extension conditions, and reduce the risk of environmentally induced common-cause equipment failure.

6.4.3 Codes and standards for civil structures

WENRA RL Q1.1 requires that licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant’s ability to be operated safely. This means that installation of sCO2-4-NPP system should not degrade other functions performed by civil structures, e.g. containment functions. Available nuclear codes and standards are e.g. ASME BPVC Section III, Division 2 [66], AFCEN RRC-CW [47] and KTA 2201.3 [79].

RL Q2.2 requires that the process of modification shall include design and safety assessment. This means that criteria of codes and standards for civil structures should remain fulfilled.

6.4.3.1 ASME Boiler and Pressure Vessel Code

ASME BPVC, Section III, Division 2 [66] contains requirements for the material, design, construction, fabrication, testing, examination, and overpressure protection of concrete containment structures, pre-stressed or reinforced. These requirements are applicable only to those components that are designed to provide a pressure retaining or containing barrier. They are not applicable to other support structures, except as they directly affect the components of the systems. This Section contains appendices, both mandatory and nonmandatory, for Division 2 construction.

6.4.3.2 RCC-CW technical code

RCC-CW [47] describes the rules for designing, building and testing civil engineering works in PWR reactors. The code covers the following areas relating to the design and construction of civil engineering works that play an important safety role:

- local cases and combinations,
- geotechnical aspects,
- reinforced concrete structures and galleries,
- prestressed containments with metal liner,
- metal containment and pool liners,
- metal frames,
- anchors,
- concrete cylinder pipes,
- paints and coatings,
- containment leak tests.

It explains the principles and requirements for the safety, serviceability and durability of concrete and metal frame structures, based on Eurocode design principles (European standards for the structural design of construction works) combined with specific measures for safety-class buildings.

Contents of the 2016 edition of the RCC-CW Code:

- Part G - General: scope, standards, notations, quality management, general principles;
- Part D - Design: actions and combinations of actions, geotechnical aspects, pre-stressed or reinforced concrete structures, metal containment liners, metal pool liners, metal frames, anchors;
- Part C - Construction: geotechnical aspects, concrete, surface finish and formwork, reinforcement for reinforced concrete, pre-stressing processes, prefabricated concrete elements, metal containment liners, metal pool liners, metal frames, anchors, embedded pipelines, joint sealing, survey networks and tolerances;
- Part M - Maintenance and monitoring: containment integrity and rate tests.

6.4.3.3 KTA 2201.3 safety standard

KTA 2201 Part 3 [79] safety standard applies to civil structures of nuclear power plants with light water reactors in order to achieve the protective goals specified in safety standard KTA 2201 Part 1. It specifies the requirements for civil structures that must be met for the verification of their load-bearing capacity (stability) in case of a seismic event. Additionally, requirements are specified pertaining to the verification of the serviceability of civil structures as far as necessary for maintaining their safety-related function in case of a seismic event. The term civil structures as used in KTA 2201 Part 3 safety standard shall comprise buildings and structural members made of reinforced concrete, pre-stressed concrete, steel, as well as steel composite structures and brickwork. Among others, these include the containment, crane runways, platforms, anchor constructions and canals.

6.4.4 Operations and maintenance code

As stated in subsection 6.1.2.5 for WENRA Issue K, SSCs important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability. The inservice testing demonstrates the operational readiness of components. The readiness is ensured by periodic testing.

6.4.4.1 ASME Operations and maintenance code of nuclear power plants

ASME Operations and maintenance code (OM) of nuclear power plants is overviewed in [104]. Hereafter brief summary is given. Initially, in the late 1960’s through the late 1980’s, many of the Inservice Testing Requirements (IST) were listed in the ASME Section XI Code. Then in the early 1980’s, it was determined that the “pump and valve” requirements listed in the ASME Section XI would be more readily developed and updated by being separated from ASME Section XI, which primarily deals with weld/component examinations, instead of functional testing of the components. Based on this and other considerations, the ASME determined to separate the ASME Section XI Examination Requirements from the functional testing requirements and determined to develop another Standard Committee (OM) to support the newly developed testing Standard, which was the beginning of the ASME OM Code [65]. The major reason was the different specialty areas and divergent areas associated with functional testing of the components compared to the more standardized examination requirements and qualifications associated with Inservice Inspection (ISI) as compared to Inservice Testing (IST).

The ASME OM Code is divided primarily into six (6) major Subsections, Mandatory Appendices and Nonmandatory Appendices.

ASME Operations and maintenance code of nuclear power plants [65] establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness in the light-water reactor power plants. It identifies the components subject to test or examination, responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating the results, corrective action, personnel qualification, and record keeping. These requirements apply to:

(a) pumps and valves that are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident;

(b) pressure relief devices that protect systems or portions of systems that perform one or more of these three functions; and

(c) dynamic restraints (snubbers) used in systems that perform one or more of these three functions.

OM Code, Division I is divided into 6 major Subsections (for more details refer to [68] and [104]):

- Subsection ISTA – General Requirements
- Subsection ISTB – Inservice Testing of Pumps in Light Water Reactor Power Plants – Pre-2000 Plants
- Subsection ISTC – Inservice Testing of Valves in Light Water Reactor Nuclear Power Plants
- Subsection ISTD – Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light Water Reactor Nuclear Power Plants
- Subsection ISTF – Inservice Testing of Pumps in Light Water Reactor Nuclear Power Plants – Post-2000 Plants

Each subsection follows the typical organization as enumerated in the following [104]:

- 1000, Introduction (note: or Scope)
- 2000, Definitions
- 3000, General Requirements
- 4000, Instrumentation and Test Equipment
- 5000, Specific Test Requirements
- 6000, Monitoring, Analysis and Evaluation
- 7000, not used
- 8000, not used
- 9000, Records and Reports

Division II are OM standards and Division III are OM guides.

ASME

6.5 Level V rules - Conventional codes and standards

There are several codes and standards referenced in regulatory guides, including industry codes and standards. In the scope of this section is the description of representative conventional pressure vessel codes and standards for design. Conventional operation and maintenance codes and standards are out of the scope of this report as they are not needed for design of sCO2-4-NPP components. Also, it is expected that for design purposes of sCO2-4-NPP components conventional codes and standards will be used only, if adequate nuclear codes and standards will not be available and in such cases it should be demonstrated that safety requirements are satisfied.

6.5.1 Conventional pressure vessel codes and standards

In order to make some contribution to current efforts to harmonize international design codes and standards, in paper [116] a review of fatigue analysis methods for a number of selected nuclear and non-nuclear design codes and standards has been carried out. The non-nuclear design codes considered in [116] were ASME BPVC Section VIII Division 2, EN 12952, EN 13445, EN 13480 and PD 5500 (besides nuclear ones: ASME BPVC Section III, Subsection NB, RCC-M, RCC-MRx (mechanical components for experimental reactors), JSME, PNAEG and R5). In the following, Pressure Equipment Directive (PED) 2014/68/EU [113] is described first (it is not standard, but outlines them), followed by ASME B31.1 [81], ASME Section VIII [67] and each of the above non-nuclear standards, which are also briefly described.

6.5.1.1 2014/68/EU Pressure Equipment Directive (PED)

The Pressure Equipment Directive (PED) 2014/68/EU [113] (formerly 97/23/EC) outlines the standards to which pressure equipment within the EU must conform. This Directive should be limited to the expression of the essential safety requirements. In order to facilitate conformity assessment with those requirements, it is necessary to provide for a presumption of conformity for pressure equipment or assemblies which are in conformity with harmonised standards that are adopted in accordance with Regulation (EU) No 1025/2012 of the European Parliament and of the Council [114] for the purpose of expressing detailed technical specifications of those requirements, especially with regard to the design, manufacture and testing of pressure equipment or assemblies. European standards are adopted by the European standardisation organisations, namely European Committee for Standardisation (CEN), European Committee for Electrotechnical Standardisation (Cenelec) and European Telecommunications Standards Institute (ETSI), where 'harmonised standard' means a European standard adopted on the basis of a request made by the Commission for the application of Union harmonisation legislation. Where a harmonised standard satisfies the requirements which it aims to cover and which are set out in the corresponding Union harmonisation legislation, the Commission shall publish a reference of such harmonised standard without delay in the Official Journal of the European Union or by other means in accordance with the conditions laid down in the corresponding act of Union harmonisation legislation (e.g. commission implementing decision 2019/1616 [115]).

PED 2014/68/EU [113] applies to the design, manufacture and conformity assessment of pressure equipment and assemblies with a maximum allowable pressure greater than 0.5 bar. 'Pressure equipment' means vessels, piping, safety accessories and pressure accessories, including, where applicable, elements attached to pressurised parts, such as flanges, nozzles, couplings, supports, lifting lugs.

The purpose of the Pressure Equipment Directive is to ensure the free flow of stationary pressure equipment within the European Union. At the same time it is to ensure that all pressure equipment passes a high degree of safety. European Standard EN 13445 [109] is harmonized with PED 2014/68/EU [113] (see Section 6.5.1.5). The type of products covered under the PED 2014/68/EU [113] include also heat exchangers.

6.5.1.2 ASME B31.1 for piping

ASME B31.1 [81] prescribes minimum requirements for the design, materials, fabrication, erection, test, inspection, operation, and maintenance of piping systems typically found in electric power generating stations, industrial and institutional plants, geothermal heating systems, and central and district heating and cooling systems.

ASME B31.1, Section 1 of B31 structure is the following [103]:

- Scope and Definitions
- Design
- Materials
- Dimensional Requirements
- Fabrication, Assembly, and Erection
- Inspection, Examination, and Testing
- Mandatory Appendices
- Non-mandatory Appendices
- Technical / Code Inquiries
- Code Cases

Chapter II (Design) has six parts. Part 1 cover design conditions (pressure, temperature, ambient influences, dynamic effects, weight effects, thermal contraction and contraction loads) and criteria (P-T ratings for piping components, allowable stress values and other stress limits and allowances). Part 5 deals with expansion, flexibility, and pipe supporting element.

6.5.1.3 ASME Code, Section VIII

ASME Section VIII, Division 1 provides requirements applicable to the design, fabrication, inspection, testing, and certification of pressure vessels operating at either internal or external pressures exceeding 1.03 bar gauge (15 psig). Such vessels may be fired or unfired. This pressure may be obtained from an external source or by the application of heat from a direct or indirect source, or any combination thereof [67].

Specific requirements apply to several classes of material used in pressure vessel construction, and also to fabrication methods such as welding, forging and brazing.

Division 1 contains mandatory and nonmandatory appendices detailing supplementary design criteria, nondestructive examination and inspection acceptance standards. Pressure vessels are designed for pressures above 1.03 bar gauge (15 psig) and not exceeding 206.9 bar gauge (3000 psig), and having inside diameter above 0.1524 m (6 in). The pressure vessels within scope are unfired steam boilers, evaporators and heat exchangers. Division 2 requirements on materials, design, and nondestructive examination are more rigorous than in Division 1; however, higher design stress intensity values are permitted. Division 3 requirements are applicable to pressure vessels operating at either internal or external pressures generally above 68.9 MPa (10,000 psi). It does not establish maximum pressure limits for either Section VIII, Divisions 1 or 2, nor minimum pressure limits for this Division.

Within the designations of Section III and Section VIII there are subcategories with their specific regions of applicability [107]. Each of these subcategories has evolved their own unique features with respect to design rules and their implementation. Section VIII does likewise but in a different format comparing to Section III covers all phases of construction; materials, design, fabrication and installation, examination, testing, over pressure protection, and name plates, stamping and reports.

There are three divisions of Section VIII, Div 1, Div 2 and Div 3. Section VIII, Div 1 covers all aspects of construction but only applies to vessels [107]. Div 2 covers only vessels. Section VIII, Div 3 is specifically for very high pressure service. A key feature of Div 3 is that the allowable stress is limited to 2/3 of the yield strength without any limit on the tensile strength.

6.5.1.4 European Standard EN 12952

This European Standard EN 12952 [108] specifies the requirements for the design and calculation of water-tube boilers as defined in EN 12952-1. The purpose of this European Standard is to ensure that the hazards associated with water-tube boilers are reduced to a minimum by the proper application of the design according to this part of EN 12952. It applies to water-tube boilers with volumes in excess of two litres for the generation of steam, and/or hot water at an allowable pressure greater than 0.5 bar and with a temperature in excess of 110 °C as well as auxiliary installations (other plant equipment).

6.5.1.5 European Standard EN 13445

This European Standard specifies requirements for the design, materials, manufacturing and testing of pressure vessels and pressure vessel parts intended for use with a maximum allowable pressure, equal or less than 100 bar and shell wall thicknesses not exceeding 60 mm, which are constructed of ferritic or austenitic spheroidal graphite cast iron [109]. It provides one means of conforming to essential safety requirements of the Pressure Equipment Directive 2014/68/CE [113]. EN 13445 is divided into parts which cover the following items [109]:

- General (EN 13445-1)
- Materials (EN 13445-2)
- Design (EN 13445-3)
- Fabrication (EN 13445-4)
- Inspection and testing (EN 13445-5)
- Requirements for the design and fabrication of pressure vessels and pressure parts constructed from spheroidal graphite cast iron (EN 13445-6)
- Additional requirements for pressure vessels of aluminium and aluminium alloys (EN 13445-8)
- Additional requirements for pressure vessels of nickel and nickel alloys (EN 13445-10)

With regard to PED 2014/68/EU [113] commission implementing decision 2019/1616 [115] has been taken, having regard to Regulation (EU) No 1025/2012 of the European Parliament and of the Council [114]. Specifically, CEN amended standards EN 13445-2:2014, EN 13445-3:2014, EN 13445-5:2014 and EN 13445-6:2014 for unfired pressure vessels in 2018 and 2019 (part 3 for design).

6.5.1.6 European Standard EN 13480

This European Standard [110] specifies the requirements for industrial piping systems and supports, including safety systems, made of metallic materials with a view to ensure safe operation. This European Standard is applicable to metallic piping above ground, ducted or buried, irrespective of pressure. This European Standard is not applicable to items specifically designed for nuclear use, failure of which may cause an emission of radioactivity. EN 13480 is divided into parts which cover the following items:

- General (EN 13480-1)
- Materials (EN 13480-2)
- Design and calculation (EN 13480-3)
- Fabrication and installation (EN 13480-4)
- Inspection and testing (EN 13480-5)
- Additional requirements for buried piping (EN 13480-6)
- Additional requirements for aluminium and aluminium alloy piping (EN 13480-8)

CEN amended standards EN 13480-2:2017 and EN 13480-5:2017 for metallic industrial piping in 2018 and 2019, respectively (see Commission implementing decision 2019/1616 [115]).

6.5.1.7 Published Document PD 5500

PD 5500 [111] is the UK's unfired pressure vessels code. It specifies requirements for the design, construction, inspection and testing of unfired pressure vessels made from carbon, ferritic alloy and austenitic steels, aluminium, copper, nickel and titanium making it an invaluable reference tool for the design and assessment of pressure vessels.

PD 5500 was formerly a widely used British Standard known as BS 5500, but was withdrawn from the list of British Standards because it was not harmonized with the European Pressure Equipment Directive (97/23/EC). In the United Kingdom it was replaced by EN 13445 [109]. It is currently published as a "Published Document" (PD) by the British Standards Institution (BSI). BS 5500 was first published as PD 5500 in 2000.

6.5.1.8 R5 procedures

The R5 procedures have been developed within the UK power generation industry to assess the integrity of nuclear and conventional plant operating at high temperatures. EDF Energy's R5 Assessment Procedure for the High Temperature Response of Structures [112] is an established methodology, which is frequently used in safety cases for structural integrity assessments of components in the UK's Advanced Gas cooled Reactors (AGRs), which typically operate at temperatures in the range 470-650 °C.

6.5.2 Conventional civil structure code

6.5.2.1 European Standard EN 1992

European Standard EN 1992 Eurocode 2 [117], Design of concrete structures, applies to the design of buildings and other civil engineering works in plain, reinforced and prestressed concrete. It complies with the principles and requirements for the safety and serviceability of structures, the basis of their design and verification that are given in EN 1990: Basis of structural design. EN Eurocode 2 is concerned with the requirements for resistance, serviceability, durability and fire resistance of concrete structures.

It consists of four parts [117]:

- EN 1992-1-1:2004 Eurocode 2: Design of concrete structures - Part 1-1: General rules and rules for buildings
- EN 1992-1-2:2004 Eurocode 2: Design of concrete structures - Part 1-2: General rules - Structural fire design

- EN 1992-2:2005 Eurocode 2: Design of concrete structures - Part 2: Concrete bridges - Design and detailing rules
- EN 1992-3:2006 Eurocode 2: Design of concrete structures - Part 3: Liquid retaining and containment structures

Part 1-1 gives a general basis for the design of structures in plain, reinforced and prestressed concrete, while Part 1-2 deals with the design of concrete structures for the accidental situation of fire exposure. Part 2 gives a general basis for the design and detailing of bridges in reinforced and prestressed concrete. Finally, Part 3 covers additional rules for the design of concrete structures for the containment of liquids or granular solids and other liquid retaining structures.

EN Eurocode 2 is intended to be used in conjunction with EN 1990: Eurocode - Basis of structural design; EN 1991: Eurocode 1 - Actions on structures; EN 1997: Eurocode 7 - Geotechnical design; EN 1998: Eurocode 8 - Design of structures for earthquake resistance, when concrete structures are built in seismic regions and some other.

7 Conclusion

A first level of guidance on design requirements for the passive decay heat removal system is specified in the WENRA documents ([4], [6], [1] (Level I), IAEA [11] (Level II), and in the national legislations. The Level III are nuclear process oriented documents consisting of quality assurance (management system) documents (guidance and standards) and design and operation documents (WENRA guidance document, IAEA specific safety guides and technical documents and U.S. NRC regulatory guides). Level IV are nuclear component oriented documents (pressure boundary codes and standards for mechanical components, codes and standards for electrical equipment and operations and maintenance code). Finally, Level V presents conventional pressure vessel codes and standards codes and standards.

Level I high level requirements of Western European Nuclear Regulators Association (WENRA) for existing reactors [4] are harmonized European requirements. Level II rules requirement for nuclear power plant design of International Atomic Energy Agency (IAEA) [11] is the equivalent level compared to WENRA (both are substitutes for specific country legislation) with the difference that IAEA presents internationally established standards for design of nuclear power plants.

7.1 Country legislation

When using country legislation for design of systems and components, the Level I WENRA design requirements should be also taken into account. The design area includes Issue E, which provides design basis for existing reactors; Issue F for design extension conditions of existing reactors; Issue G, which sets requirements for safety classification of SSC and Issue T for natural hazards. During design, provisions for maintenance, testing and inspection should also be considered (in Issue K). The design shall take into account the quality of the products and services that contribute to safety, therefore management system including nuclear specific requirements must be implemented.

7.2 Nuclear process oriented documents

7.2.1 Quality assurance

Acceptable are IAEA management system [37], ASME NQA-1-2019 standard [48], KTA-1401 standard [49], ISO 19443:2018 [39] and supplementation of ISO 9001:2015 [38] with nuclear requirements.

7.2.2 Design and operation

From the point of guidance documents WENRA does not provide a complete set of guidance documents. International IAEA specific safety guides and technical documents or the use of national guides if they are complete, are an acceptable approach.

These may be German RSK [51] and KTA documents [50], DIN Nuclear Standards Committee (NKe) documents (most standards approved by KTA are also published as DIN standards [41]), French RFS [14] (the RFS are intended to be gradually replaced by ASN guides) and ASN guides [15], Spanish CSN guides [16], etc. In the report, an example of relevant U.S. regulatory guides is also given. These documents typically endorse the nuclear codes and standards, and conventional industry codes and standards when nuclear are not available.

The national guides and documents should at minimum cover guidance on design requirements of a passive safety system; on classification of structures, systems and components; on reliability assessment and on deterministic safety analysis. For example, document IAEA TECDOC-1787 [26] gives examples of well-established codes defining design and manufacturing requirements for pressure retaining equipment:

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

For example [54], in RCC-M 2012 there were referenced around 240 standards, 170 were International ISO or EN (the rest were NF, ASTM, ANSI, AWS, IAEA and ANSI/MSS SP-43 standards). Also KTA 3211.2 referred to few tens of DIN and other standards and documents.

7.3 Nuclear component oriented documents

Nuclear component oriented documents primarily comprise nuclear codes and standards for mechanical and electrical equipment. This level is covered neither by WENRA nor IAEA documents, as documents have not been developed. IAEA SSG-56 [23] for design just refers that codes and standards have been developed by various national and international organizations, covering areas such as materials; manufacturing (e.g. welding) and construction; civil structures; pressure vessels and pipes; instrumentation and control; environmental and seismic qualification; pre‑service and in‑service inspection and testing; the management system and fire protection. Nevertheless, IAEA recognizes (see also Table 3) as an acceptable approach for mechanical equipment to use ASME Boiler and Pressure Vessel Code, Section III, AFCEN's RCC-M (also KTA 3211 standard for pressure and activity retaining components of systems outside the primary circuit).

For electrical equipment according to IAEA (see also Table 4) AFCEN's RCC-E or IEEE Std. 603 [105] for safety systems is an acceptable approach. For electrical equipment qualification European harmonized EN 60780-323:2017 is acceptable (published also as national adoptions by DIN in Germany, NF in France or SIST in Slovenia).

For civil structures ASME Boiler and Pressure Vessel Code, Section III, Divison2, AFCEN's RCC-CW and KTA 2201 safety standard may be used.

7.4 Conventional codes and standards

It is expected that industry designers of components are familiar with conventional codes and standards. The Pressure Equipment Directive (PED) 2014/68/EU [113] is expected to be followed when nuclear codes and standards are not available for component design.

Examples of conventional codes and standards are ASME B31.1 for piping [81], ASME Section VIII [67] for pressure vessels, EN 12952 [108] for water tube boiler and auxiliary installations, EN 13445 [109] for unfired pressure vessels, EN 13480 [110] for metallic industrial piping or PD 5500 [111] for unfired fusion welded pressure vessels.

Finally, following Table 3 the conventional codes and standards acceptable by IAEA for components providing Category 3 functions (unless specific codes and requirements are applied for specific reasons) are European Pressure Directive 97/23/EC (note: since 2014 new PED 2014/68/EU [113]), ASME Code, Section VIII, Division 1 for pressure vessels and ANSI B31.1 for piping.

8 References

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