Strål
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Swedish Radiation Safety Authority

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Design Guide for Nuclear Civil Structures (DNB)

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SSM perspective

Background

Until 2010, the National Swedish Building Code (BKR, BBK) has been applied for the design and analysis of civil structures. In 2011, the National Swedish Building Code was replaced by the European Building Code, Eurocodes.

The Swedish Radiation Safety Authority (SSM) and the Swedish licensees have previously in a jointly funded research project developed a design guide for civil structures at Swedish nuclear facilities to be based on Eurocodes, DNB. The report was published in January 2014 as SSM Report 2014:06.

To further improve DNB and to ensure that the fundamentals of the recommendations will be applied correctly, has SSM commissioned Scanscot Technology AB to further develop and clarify certain parts of the report.

Objectives

The aim of the project is to further improve and clarify certain parts of DNB in order to extend the jurisdiction of DNB and to ensure that the fundamentals of the recommendations will be applied correctly.

Results

Updates of DNB are referring to at least the following areas.

- Clarification regarding evaluation of safety-related leak-tightness in the ultimate limit state
- Revised partial factors for some process-related actions
- Modification of recommended increase of calculated pressures in the containment for severe accidents
- Clarification of ”cliff edge”-effects
- Concrete strength values to be used in ASME Sect III Div 2
- Revised guidance with respect to design provisions regarding leak-tightness of safety-related pool structures
- Update of guidelines for seismic design
- Introducing a new annex regarding material testing methodology

All corrections and revisions to the current edition of DNB are compiled in a background report.

Need for further research

More research is needed in this area. DNB needs to be updated with detailed recommendations for the design of civil structures against various types of postulated fire events as well as detailed design criteria for impact- and missile loads.

Project information

Contact person SSM: Kostas Xanthopoulos

Reference: SSM2014-2777

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Abstract

The statute documents of the Swedish Radiation Safety Authority (SSM) do not include specific requirements and adequate guidance on how concrete structures at nuclear power plants and other nuclear facilities shall be structurally verified in analyses of existing structures as well as in the case of design of new buildings.

Therefore, the Swedish Radiation Safety Authority has together with the Swedish licensees commissioned Scanscot Technology AB (SCTE) to compose the present Design Guide for Nuclear Civil Structures (DNB). This Design Guide 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. The present report is the 2nd edition of the DNB. This second edition replaces the first edition that was issued by the Swedish Radiation Safety Authority in January 2014 (Report No. 2014:06).

The main aim of DNB is to complement the regulations given in *Boverkets föreskrifter och allmänna råd om tillämpning av europeiska konstruktionsstandarder (eurokoder)* (BFS 2011:10 – EKS 8)[1] for application at nuclear power plants and other nuclear facilities in Sweden. Thus, DNB is based on the partial factor method and the principles of design in limit states, as specified in the Eurocodes including the Nationally Determined Parameters chosen by Swedish Authorities.

The report is written by a project group[2] at Scanscot Technology AB with Ola Jovall as the main responsible author. Prof. em. Sven Thelandersson as well as a steering committee appointed by the Swedish Radiation Safety Authority and the Swedish licensees have independently reviewed the first edition of the report. The first edition has has also been distributed to selected stakeholders for their opinion. The second edition has been reviewed by Prof. em. Sven Thelandersson and the Swedish Radiation Safety Authority. It has also been distributed to the Swedish licensees for comments.

The original DNB report is published in Swedish. DNB has been translated into English by Björn Lundin (SCTE) and Albin Larsson (SCTE), with the assistance of the original authors. Prof. em. Sven Thelandersson and SSM have reviewed the English translation.

DISCLAIMER

This report constitute an English translation of the report "Dimensionering av nukleära byggnadskonstruktioner (DNB)[3]" (in Swedish). The Swedish edition of Design Guide for Nuclear Civil Structures (DNB) is the valid version. In the case of any discrepencies between this English translation and the Swedish edition, the Swedish edition governs.

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[1] English translation of document title: "Mandatory provisions and general recommendations on the application of European design standards (Eurocodes) (BFS 2011:10 – EKS 8)"

[2] Patrick Anderson: Section 6; Ola Jovall: Section 1, 2, 3, 5 och 8 and co-author of section 4 and 6; Johan Kölfors: Section 4; Jan-Anders Larsson: Section 7; Sven Thelandersson: Co-author of section 4.

[3] English translation of document title: "Design Guide for Nuclear Civil Structures (DNB)"

1. Introduction

1.1 General

Design Guide for Nuclear Civil Structures (DNB) contains provisions and application rules regarding the design and analysis of concrete structures at Swedish nuclear power plants and other nuclear facilities[4,5]. Regarding nuclear power plants, DNB can be applied for light-water reactors of type boiling water reactor (BWR) and pressurized water reactor (PWR).

The purpose of DNB is to complement the regulations in *Boverkets föreskrifter och allmänna råd om tillämpning av europeiska konstruktionsstandarder (eurokoder)* (BFS 2011:10 – EKS 8) [8] for application at nuclear power plants and other nuclear facilities. Thus, DNB is based on the partial factor method and the principles of design in limit states, as specified in SS-EN 1990 [20], SS-EN 1991 and SS-EN 1992-1-1 [29] as well as associated parts of BFS 2011:10 – EKS 8 [8].

The provisions given in DNB are valid when a deterministic design or verification of structures or structural members is to be implemented. In certain situations, especially for highly improbable events (event class H5)[6], other approaches may be applicable or necessary.

1.2 Outline of the design provisions

1.2.1 General set of regulations

A nuclear power plant is a facility for production of electricity for which the safety requirements are extraordinarily high. When designing such a facility and other nuclear facilities it should be demonstrated that it meets both the general requirements for conventional buildings and production facilities as well as the safety requirements of nuclear facilities as stated by the Swedish Radiation Safety Authority (SSM).

Requirements for structures of conventional buildings with respect to safety, serviceability and durability as well as the basics for the design and verification are reported in the EKS and the Eurocodes. Thus, the reactor containment and other buildings should meet the requirements of the EKS/Eurocodes.

In addition to the conventional requirements, 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. In addition, modifications and amendments to EKS and the Eurocodes have to be introduced.

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[4] The general parts of DNB, i.e. Section 1 to Section 4 as well as Section 7 and 8, may also be seen as guidelines when designing structural members of other building materials than concrete. Any adjustments of DNB, and any further modifications and/or amendments which then may be needed, shall be determined from case to case.

[5] DNB can also be applied for other facilities that during an accident may have significant radiological impact on the surroundings.

[6] Event classes are explained in section 3.4

In the safety analysis reports (SAR) of the nuclear power plants and other nuclear facilities, the licensee's interpretation of the requirements and regulations in force is presented as well as the specific requirements for each unit.

The present design provisions are based on EKS and the Eurocodes with necessary modifications and amendments for application at nuclear power plants and other nuclear facilities. For certain structural members (such as the reactor containment), and for certain verifications (e.g. design with respect to earthquake), nuclear facility specific regulations are referred to as a supplement to the Eurocodes.

1.2.2 Regulations referred to

The Eurocodes shall be applied to the design of all building structures covered in DNB. Hence, actions and combinations of actions as well as limit states and design situations according to the principles of Eurocode shall be applied to both the reactor containment and other buildings. 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, additional requirements are provided for the reactor containment based on ASME Sect III Div 2 [6].

When combinations of actions for the ultimate limit state are affecting the reactor containment, supplementary requirements regarding the containment load-carrying capacity are referred to. ASME Sect III Div 2 [6] applies to persistent, transient and accidental design situations. Regarding highly improbable design situations, unique requirements based on the Eurocodes have been established since ASME Sect III Div 2 [6] does not cover this type of events.

Since the Eurocodes do not cover safety-related leak-tightness requirements relevant for nuclear power plants, such requirements are referred to from ASME Sect III Div 2 [6] for all event classes up to improbable events, corresponding to accidental design situations. For highly improbable events, additional provisions have been introduced since ASME Sect III Div 2 [6], as already mentioned above, does not cover this type of events.

For other buildings except the reactor containment, the Eurocodes with specified modifications and amendments in this report are considered sufficient. No additional regulations have had to be referred to, except regarding design for earthquake resistance.

The earthquake section of the Eurocodes (SS-EN 1998 [33]) is not applicable for nuclear power plants or other nuclear facilities. Therefore, new provisions have been introduced for design with respect to seismic actions, primarily based on ASCE 4-98 [4]. These provisions replace SS-EN 1998 [33].

Since the Eurocodes are the basis for the design of all building structures, it is assumed that materials and products wherever possible also meets the requirements of the Eurocodes with associated standards.

A schematic figure of the arrangement of the design provisions is given in Figure 1.1.

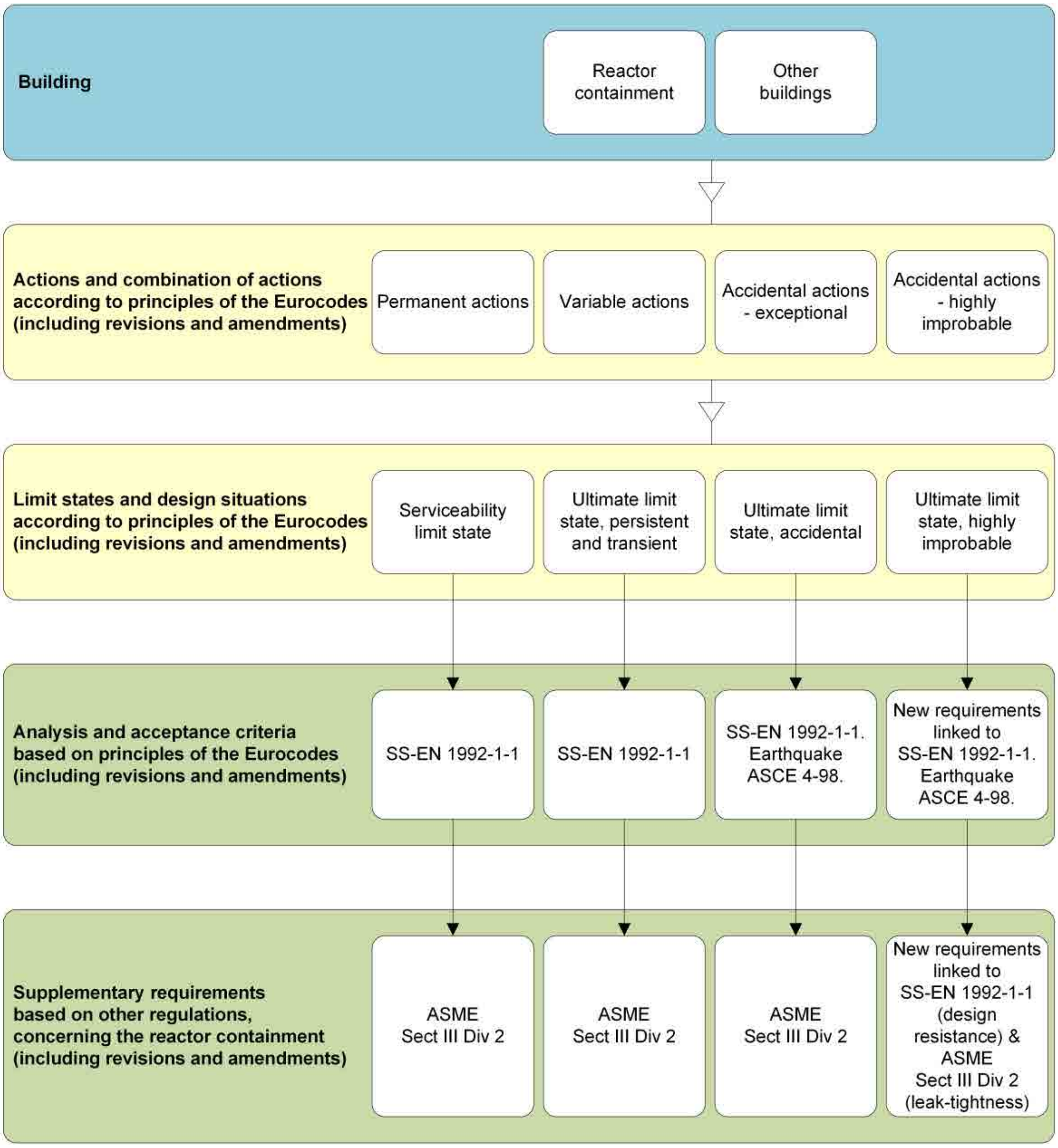

Figure 1.1 – Schematic figure of the arrangement of the design provisions

1.3 Outline of report

DNB is divided into sections, each of which connects to a specific part of Eurocode or to other regulations referred to in Table 1.1.

Table 1.1 – Connection between sections in DNB and specific parts of regulations.

DNB	Eurocode or other regulation
Section 1 Introduction	-
Section 2 General conditions	SS-EN 1990 Basis of structural design (except for annex A1)
Section 3 Basic principles of structural design	
Section 4 Actions and combinations of actions	SS-EN 1990 annex A1 and parts of SS-EN 1991 Actions on structures
Section 5 Design of the reactor containment	SS-EN 1992-1-1 Design of concrete structures and ASME Sect III Div 2 Code for Concrete Containments CC-3000 Design
Section 6 Design of other buildings	SS-EN 1992-1-1 Design of concrete structures
Section 7 Seismic design	SS-EN 1992-1-1 Design of concrete structures and ASME Sect III Div 2 Code for Concrete Containments CC-3000 Design and ASCE 4-98 Seismic Analysis of Safety-Related Nuclear Structures and Commentary and SKI Technical Report 92.3 Characterization of seismic ground motions for probabilistic safety analyses of nuclear facilities in Sweden
Section 8 Design related to the construction phase	SS-EN 1991-1-6 General actions – Actions during execution and ASME Sect III Div 2 Code for Concrete Containments CC-3000 Design

2. General conditions

2.1 General

SS-EN 1990 [20] and BFS 2011:10 – EKS 8 [8] are generally referred to with the modifications and amendments presented within this section.

2.2 Jurisdiction

The plant owners operating license is based on a safety analysis report (SAR), which constitutes the overall site-specific requirements for the nuclear facility. The SAR describes the full requirement hierarchy for the plant, including Swedish legislation, Swedish provisions and conditions issued by SSM, SAR and other regulations (normative documents, guides, codes and standards). Thus, the overall requirement specification for buildings is given in the SAR and its related references. Before a facility may be constructed and before any major reconstructions or modifications to an existing facility occur, a preliminary safety analysis report shall be compiled according to provisions of the Swedish Radiation Safety Authority.

DNB takes effect by a reference from the SAR, or via a reference from site specific or project specific documents. Specific building requirements and conditions that must be considered in design and analysis are governed by the requirements given in the SAR and associated detailed requirements specified in the design specifications of the current building (KFB) as well as in project specific documents.

DNB applies to the design of new concrete structures, to rebuildings and extensions and to the verification of existing concrete structures at nuclear facilities.

In some cases a risk assessment, based on probability theory as well as material parameters and calculation methods according to SS-EN 1990 [20] annex C, can be an appropriate or necessary supplement. The expected operational life time of the facility shall be taken into consideration for this type of analysis. Such analyses are not covered in this report.

2.3 Scope and limitations

SS-EN 1990 [20] section 1.1 is omitted.

The design provisions given in DNB cover concrete structures at Swedish nuclear power plants and other nuclear facilities. For other types of building structures as well as mechanical components that are permanently installed in the buildings, other regulations or standards may apply. In such cases, the documentation should clearly specify the boundaries of jurisdiction for each standard.

DNB contains general design provisons for the design of concrete structures and structural members at nuclear power plants and other nuclear facilities. For special conditions, specific design provisions, methods and expert investigations may be required. DNB does not cover the structure's execution, quality assurance, audit, inspection, testing or maintenance.

Furthermore, DNB gives design provisions regarding the reactor containment leak-tightness for those parts of the steel liner which are backed by the load-bearing concrete structure. An example of this distinction is presented in Figure 2.1.

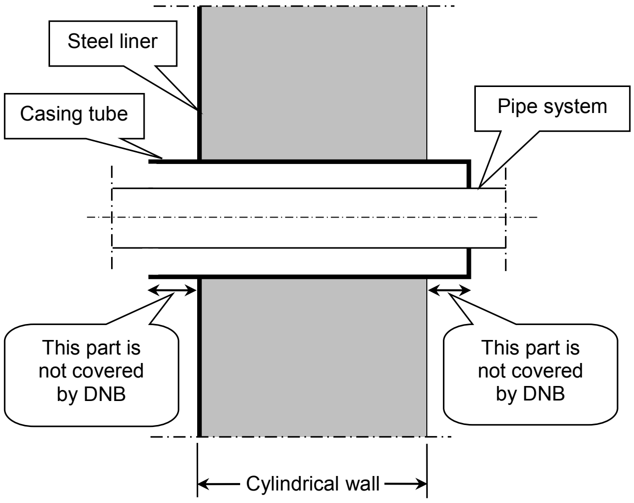

Figure 2.1 – Example of distinction regarding the steel liner of the containment, defining what is covered by DNB.

Different design provisions are provided for the reactor containment (Section 5) and for other buildings (Section 6). Section 5 is applicable to the reactor containment as well as the pressure retaining structural components that separate the primary and secondary compartment in order to maintain the pressure suppression function in BWR plants. Other parts of the building, including structural concrete members inside the containment are designed in accordance with Section 6. The distinction between each section's validity shall be determined on a case-by-case basis in case the containment is structurally integrated with the surrounding structure or with the load-bearing concrete structure inside the containment.

Note that actions on, for example the reactor containment, also can lead to effects of actions in other buildings and vice versa. These effects of actions must be considered irrespective of the distinction between the validity of the different design section. This is facilitated by the fact that the combinations of actions in Section 4 are applicable to both the containment structure and other buildings. Regarding a building that interacts with other structural members, for which less conservative design rules applies, and where this structural member significantly contributes to the building's ability to meet current requirements, it is recommended that the design is governed by the most conservative regulation.

2.4 References to codes and standards

SS-EN 1990 [20] Section 1.2 is omitted.

SS-EN 1990 [20], SS-EN 1991 and SS-EN 1992-1-1 [29] as well as associated parts of BFS 2011: 10 - EKS 8 [8] generally apply with the modifications and amendments specified in this report. SS-EN 1997 [32] and SS-EN 1998 [33] do only apply when specifically referred to. In case of conflicts, formulations stated within this document apply, if this means stricter requirements, unfavourable loading conditions, etc., compared to what is stated in SS-EN and EKS [8].

The Swedish national amendments according to Boverket (the Swedish National Board of Housing, Building and Planning) apply, see corresponding national annex (NA) in each SS-EN. Note that these annexes in turn refer to the document BFS 2011: 10 - EKS 8 [8] with modifications. This means that a reference to a certain part of the Eurocode automatically also implies a reference to EKS [8].

In some cases, DNB refer to ASME Sect III Div 2 [6], ACI 349 [2], ASCE 4-98 [4], ASCE 43-05 [4], ETC-C [9], IAEA Safety Guides and YVL Guides. A short description of these regulations is given below.

ASME Sect III Div 2 (ASME) [6] is an internationally accepted regulation for the design of concrete reactor containments. The Eurocodes and ASME [6] are based on different basic principles for design. The Eurocodes are based on the partial coefficient method and the principle of limit states, while design and verification according to ASME [6] is based on allowable stresses. Hence, ASME [6] has in DNB only been integrated within the "nomenclature" used for design according to the Eurocodes. Hence, it is ensured that the verifications resulting from applying ASME [6] according to Section 5 of DNB basically fulfill the design provisions given in ASME, with the exceptions given in section 5.2.

ACI 349 [2] is an American regulation for the design of safety-related concrete structures at nuclear facilities and is referred to as follows:

- When using the Eurocodes: The Eurocodes and ACI 349 [2] are based on the same basic principles for design, but since they relate to different fields of applications there are some differences. Therefore, when using ACI 349 [2] it is primarily the design principles for safety-related building at nuclear facilities that are utilized, rather than the introduction of precise numerical values etc.
- When using ASME Sect III Div 2 [6]: ACI 349 [2] is referred to for a few cases when ASME Sect III Div 2 [6] lacks detailed design provisions. ACI 349 [2] is consistent with ASME Sect III Div 2 [6], both regulations are based on ACI 318 [1].

ASCE 4-98 [4] is an internationally accepted regulation for seismic analysis of safety-related buildings at nuclear facilities and therefore provides more stringent design provisions than equivalent analysis requirements in conventional building codes. ASCE 43-05 [4] is only used to a limited extent as a supplement to ASCE 4-98 [4] when obtaining earthquake-related analysis practices.

ETC-C [9] is a supplier-specific regulation for design of nuclear facilities including design provisions for the reactor containment. ETC-C [9] is based on the same code format as DNB, i.e. the Eurocodes. Therefore, the ETC-C [9] is occasionally referenced, to justify introduced nuclear-related supplementary requirements in DNB.

IAEA Safety Guides are code-independent and internationally accepted guidelines regarding, among other things, nuclear power plant safety.

The YVL Guides are published by the Radiation and Nuclear Safety Authority in Finland (STUK). The authority requires leak-tightness and load-carrying resistance of the containment. In design, compliance of these requirements is verified based on applicable regulations. YVL E.6 states that the reactor containment concrete members may be designed according to EC2, and that ASME Sect III Div 2 [6] stipulates the minimum requirements. It is also stated that the leak-tightness requirements of the containment, as given in ASME Sect III Div 2 [6] applies. Since DNB is mainly based on the same design principles as YVL E.6, the YVL Guides are occasionally referenced, to justify introduced nuclear-related supplementary requirements in DNB.

DNB incorporates provisions from other publications by dated references as listed above. These normative references are given in the text where they apply. The references apply in the specified edition. Later published amendments, modifications or revised editions may be applied only when they have been incorporated in this document by amendment, modification or revision.

2.5 Assumptions

In addition to what is stated in SS-EN 1990 [20] Section 1.3, the site-specific conditions specified in SAR and KFB with associated references as well as in project-specific documents apply.

2.6 Difference between principles and application rules

A distinction is made between principles and application rules in the Eurocodes according to SS-EN 1990 [20] Section 1.4. According to Eurocode, the principles shall be followed, i.e. they are requirements, while the application rules represent generally recognised rules which comply with the principles and satisfy their requirements.

Unlike the Eurocodes, DNB contains provisions and application rules, usually in the form of modifications and amendments to apply to the Eurocodes in nuclear power plants and other nuclear facilities. When introducing modifications and amendments, it is assumed that both principles and application rules given in the Eurocodes are followed, unless otherwise stated.

2.7 Terms and definitions

Terms and definitions are presented in relevant parts of SS-EN 1990 [20], SS-EN 1991 and SS-EN 1992-1-1 [29]. Terms and definitions given in SS-EN 1998 [33] are not applicable.

Annex 4 presents terms that are not defined in the Eurocodes.

2.8 Notations

When specifying limit states and design situation (for ULS) and type of combination of actions (for SLS) in abbreviated form, the following conventions of notations are used in this report:

$\mathrm{XXX}_{\mathrm{YYY}}$-zzz

where

XXX = limit state (section 3.10.4.1 and 3.10.5.1)

YYY = type of limit state (section 3.10.5.1). May be omitted if limit state in general is referred to.

zzz = design situation for ULS (section 3.7.2), type of combination of action for SLS (section 3.10.4.3)

Example:

$\mathrm{ULS}_{\mathrm{STR}}$-exc denotes ultimate limit state (ULS), strength (STR), accidental design situation (exc).

SLS-qp denotes serviceability limit state (SLS), quasi-permanent combination of action (qp).

Notations are given in relevant parts of SS-EN 1990 [20], SS-EN 1991, SS-EN 1992-1-1 [29] and SS-EN 1998 [33].

Annex 5 provides notations that are not presented in the Eurocodes.

3. Basic principles of structural design

3.1 General

SS-EN 1990 [20] and BFS 2011:10 - EKS 8 [8] are generally referred to with the modifications and amendments reported within this section.

This section describes the basic design principles related to BFS 2011: 10 - EKS 8 [8] including referenced regulations (the Eurocodes). Also basic principles that govern the design of nuclear power plants are presented, such as safety classification, classification of events and safety-related functional requirements as well as modifications and amendments to the requirements in 2011: 10 - EKS 8 [8] including therein referenced regulations.

3.2 Classification of structures, systems and components

Structures, systems and components at nuclear power plants are divided into different classes, primarily with respect to its impact on the radiological environmental safety. This general classification normally contains the following classification categories:

- Safety class (with respect to radiological environmental safety)
- Mechanical quality class
- Leak-tightness class
- Seismic category
- Electric functional class
- High energy and low energy systems
- Areas with increased risk for explosion

Corresponding classification also normally exists for other nuclear facilities.

Safety class, leak-tightness class and seismic category have a direct impact on the requirements imposed on the building structure. The classification regarding high energy and low energy systems and areas with increased risk for explosion indicates what type of actions that need to be considered.

Furthermore, operating situations, internal events and external events are generally divided into event classes, see section 3.4. This classification also has a direct impact on the requirements imposed on the building structure.

In addition to the safety classification regarding radiological environmental safety as outlined above, a separate division into safety classes are applied to building structures, corresponding to that for conventional structures according to BFS 2011: 10 - EKS 8 [8], see section 3.3

The classification according to above, with associated requirement specification, is presented in the SAR for each plant.

3.3 Safety classes for buildings according to BFS 2011:10-EKS 8

Based on an assessment of the extent of the damage that can be expected if the requirements are not met, buildings and structural members shall according to BFS 2011: 10 - EKS 8 [8] be assigned to one of the following safety classes:

- Safety class B1: Minor risk of severe damage
- Safety class B2: Some risk of severe damage
- Safety class B3: Major risk of severe damage

Unlike BFS 2011: 10 - EKS 8 [8], the safety classes for buildings and structural members at nuclear facilities are denominated as B1, B2 and B3 respectively. This is to distinguish them from the functional classification of structures, systems and components in safety classes with respect to the importance of the radiological environmental safety.

The classification of safety classes in BFS 2011: 10 - EKS 8 [8] is primarily governed by the risk of personal injury. This also applies to nuclear power plants and other nuclear facilities, but in addition, economical damage should also be considered such as operation standstill, requirements on maintaining functions etc.

Safety class B3 is generally applicable for the buildings, unless otherwise stated in the design specifications for each building (KFB). Buildings and structural members containing or in any other way may affect equipment associated to safety class 1, 2 or 3 with respect to radiological environmental safety shall however, in order to conform with the safety demands in BFS 2011: 10 - EKS 8 [8], always conform to safety class B3.

3.4 Event classes according to SSMFS 2008:17

Possible operating situations and events shall according to the provisions of the Swedish Radiation Safety Authority be accounted for during construction and operation of a nuclear power plant and other nuclear facilities. These range from various operational situations during normal operation to highly improbable events. However, the different operational situations and different initial events that may occur have very different probabilities of occurrence.

In order to achieve a balanced risk profile, different operational situations, events and event sequences at nuclear power plants are divided into so called event classes. Each event class includes events within a given frequency range. The classification of events applied in the present report follows SSMFS 2008: 17 [39], § 2 and is presented in Table 3.1. Corresponding division should also be applied for other nuclear facilities.

More detailed descriptions regarding event classes with associated requirement specifications are given in the safety analysis report (SAR) of the plants.

Table 3.2 summarises the connection between event class, classification of actions (see Section 4), design situation (see section 3.7) and limit state (see section 3.7).

Table 3.1 – Event classes in accordance with SSMFS 2008:17 § 2.

Event class		Description	Frequency range[1)]
H1	Normal operation	Includes disturbances successfully managed by regular operations and control systems without interrupted operation.	Normal operational situations
H2	Anticipated events	Events that can be expected to occur during the lifetime of a nuclear power reactor.	Frequency $\geq 10^{-2}$
H3	Unanticipated events	Events that are not expected to occur during the lifetime of a nuclear power reactor, but which can be expected to occur if several reactors are taken into account.	Frequency (F) $10^{-2} > F \geq 10^{-4}$
H4	Improbable events	Events that are not expected to occur; this also includes a number of postulated events that are analysed to verify reactor robustness independently of the event frequency. These events are often called 'design basis events'.	Frequency (F) $10^{-4} > F \geq 10^{-6}$
H5	Highly Improbable events	Events that are not expected to occur; if the event should nevertheless occur, it can result in major core damage. These events are the basis of the nuclear power reactor's mitigating systems for severe accidents.	-
-	Extremely Improbable events	Events that are so improbable that they do not need to be taken into account as initiating events in connection with safety analysis.	Residual risks

[1)] Expected probability of an event occurring during one year.

Table 3.2 – Connection between event classes, classification of actions, design situations and limit states.

Event class	Leading action	Design situation	Limit state
H1, normal operation	Permanent, Variable	Persistent, Transient	SLS, ULS
H2, anticipated events	Permanent, Variable	Persistent, Transient	SLS, ULS
H3, unanticipated events	Accidental action	Accidental Accidental, seismic	ULS
H4, improbable events	Accidental action	Accidental Accidental, seismic	ULS
H5, highly improbable events	Accidental action	Highly improbable Highly improbable, seismic	ULS

3.5 Requirements according to SS-EN and BFS 2011:10-EKS 8

3.5.1 General

Both normal operational requirements for buildings and structural members according to SS-EN and BFS 2011: 10 - EKS 8 [8] as well as safety-related functional requirements shall be met for building structures at nuclear facilities. This section presents normal operational requirements, while section 3.6 presents safety-related functional requirements.

3.5.2 Basic requirements

In addition to what is stated in SS-EN 1990 [20] section 2.1, the site-specific requirements specified in the SAR and KFB with associated references as well as project-specific documents apply.

Structures and structural members shall be shown to withstand at least postulated accidental actions to the extent given in the SAR. However, under certain circumstances local damage of the structural members may be accepted, see section 3.7.3.

3.5.3 Reliability management

In addition to the requirements specified in SS-EN 1990 [20] section 2.2, the site-specific requirements specified in SAR and KFB with associated references as well as project-specific documents apply.

In accordance with EKS, the SS-EN 1990 [20] Annex B shall not be applied. The differentiation of the reliability of structures is instead based on safety classes according to BFS 2011: 10 - EKS 8 [8] Section B, see section 3.3.

3.5.4 Intended operational life time

DNB applies to all new and reconstructed buildings that have an intended operational life time in accordance with durability requirements according to the regulations and standards applied. Operational life time category 5 according to SS-EN 1990 [20] Section 2.3 should be applied in design unless otherwise stated in the SAR. Structural members that are not accessible for inspection and maintenance shall according to the Eurocodes conform to operational life time category 5.

3.5.5 Durability

In addition to the requirements specified in SS-EN 1990 [20] section 2.4, the site-specific requirements specified in SAR and KFB with associated references as well as project-specific documents apply.

3.5.6 Quality management

In addition to what is stated in SS-EN 1990 [20] section 2.5, the site-specific requirements specified in SAR and KFB with associated references as well as project-specific documents apply.

3.6 Safety-related functional requirements according to SAR

3.6.1 General

In addition to the functional requirements imposed on structures or structural members at nuclear facilities during normal operation, see Section 3.5, there are also safety-related requirements to protect against radiological accidents. These requirements are specified in the provisions established by the Swedish Radiation Safety Authority as well as in the SAR for each plant. The requirements are different for different units, buildings and structural members.

Safety-related functional requirements to protect against radiological accidents can generally be divided into the following classes: Containment function, leak-tightness, integrity, physical security, deformations and vibrations, environmental resistance, limitation of the spread of fire as well as radiation safety.

Table 3.3 shows the overall classification of safety-related functional requirements together with information of which limit state an evaluation is based on. A detailed description of the various safety-related functional requirements is given in the following sections.

Table 3.3 – Safety-related functional requirements.

Safety-related functional requirements	Abbreviation	See section	Evaluation is carried out in following limit state
Containment function	cont (containment)	3.6.2	Separate limit state: ULS_{CONT}
Leak-tightness	leak (leak-tightness)	3.6.3	Separate limit state: ULS_{LEAK}
Integrity	int (integrity)	3.6.4	Evaluation in limit state ULS_{STR}
Physical security	sec (physical security)	3.6.5	Requirements related to structural resistance are evaluated in limit state ULS_{STR} and requirements related to leak-tightness are evaluated in limit state ULS_{LEAK}
Deformations and vibrations	vib (vibrations)	3.6.6	Separate limit state: ULS_{VIB}
Environmental resistance	env (environmental)	3.6.7	Evaluation is not connected to any specific limit state
Limitation of the spread of fire	Not covered in DNB.		
Radiation safety	Not covered in DNB.		

3.6.2 Containment function

To achieve the required level of protection in accordance with the provisions of the Swedish Radiation Safety Authority, a nuclear power plant shall be equipped with barriers whose purpose is to contain radioactive substances. The reactor containment is such a barrier. Therefore, it shall be designed so that the allowable leakage is not exceeded for event classes up to improbable events (H4). Furthermore, according to the authority's provisions, the reactor containment design shall account for phenomena and actions that may occur during events up to the level of highly improbable events (H5), to the extent necessary to limit the discharge of radioactive substances into the environment.

The authority’s leak-tightness requirements comprise e.g.

- leak-tightness of the containment steel liner, including steel liner at inside of pools if present,
- leak-tightness of containment cover (BWR),
- leak-tightness of hatches and other service openings through the containment and
- leak-tightness of casing tubes at pipe, electricity and service penetrations through the containment.

To protect the reactor containment from damage caused by large overpressure in case of severe accidents in event class H5, a controlled safety pressure relief of the containment shall according to government's decision be possible. Hence, the pressure in the containment shall with sufficient margin be limited to the collapse pressure so that the leak-tightness is not compromised. Similarly, the temperature shall be proven to be limited.

The containment function evaluation is made in a separate limit state, ULS_{CONT}, see Table 3.3.

3.6.3 Leak-tightness

Leak-tightness requirements shall ensure that sufficient safety is maintained with respect to water and gas leakage through structural members for which such leakage is not acceptable.

Leak-tightness requirements apply to e.g. the following parts:

- In some facilities there is an outer shell outside the entire containment or parts of it, the so-called secondary containment. The authority's provisions sets leak-tightness requirements for this secondary containment up to and including improbable events (H4), in order to limit radioactive discharges in (or on the surface of) soil to acceptable levels.
- Structural members within the containment for which the leak-tightness is crucial for the maintenance of important safety functions, such as the leak-tightness between the primary and secondary compartments in order to maintain the pressure suppression function (BWR).
- Leak-tightness of steel liner in the fuel handling and fuel storage pools.
- Leak-tightness of structural members for protection against leakage from tanks in waste buildings (radioactive waste)
- Leak-tightness in culverts with respect to leakage from surrounded pipe systems containing liquid radioactive waste.
- For nuclear power plants, the main control room.

For buildings or structural members with safety functions containing steam- or water pipes, requirements regarding the internal or external leak-tightness to adjacent compartments applies to following cases:

- where redundant or diversified equipment with safety function is present
- where structural members of adjacent compartments are not designed for the pressure and temperature conditions that can occur if the leak-tightness is not maintained.

So-called pressure flow paths and water discharge channels may then need to be established in order to control and limit the influence of gas overpressure and water pressure.

Evaluation of safety-related leak-tightness of structural members is made in a separate limit state, ULS_{LEAK}, see Table 3.3.

Leak-tightness requirements for the reactor containment are presented in section 3.6.2 (containment function).

Leak-tightness requirements for the reactor containment and other buildings are verified by testing and calculation. When testing, the test procedure and acceptable leakage is specified in a specific test program.

3.6.4 Integrity

Critical structures important to safety whose function is crucial for maintaining barriers and safety functions constitute in themselves a type of safety function.

According to provisions of the Swedish Radiation Safety Authority, the safety functions that are taken into account after the initial event shall not fail due to subsequent faults. For buildings, this subsequent fault is usually collapsing structural members, but can also be the loss of gas or water leak-tightness. Buildings containing and supporting equipment conforming to radiological safety class 1-3 should maintain load-carrying functions, and remain leak-tight to the extent required. Furthermore, parts of systems, components and building structures, which primarily are not needed in the initial event, are not allowed to jeopardize the function of safety equipment that is taken into account.

Building structures may have the purpose of protecting the safety functions from both the internal and the external events postulated for the facility. In this context, the building structures can either protect against direct effects of actions, or be part of the physical separation of redundant safety systems and so-called subbing of parts of buildings. The safety philosophy of this division into "subs" is to manage fire, pressure flow paths or flooding in a robust manner (the effects are limited to one sub).

For nuclear power plants, the main control room and the surrounding building (the control building) shall in accordance with the provisions of the Swedish Radiation Safety Authority be designed in such a way that falling objects or damaged structural members cannot compromise the safety of the operators in the control room. In addition, the provisions state that there shall also be a back-up surveillance spot, connected to the main control room through a protected transport route for the operators.

The integrity of structural members are evaluated in the limit state ULS_{STR}, see Table 3.3.

3.6.5 Physical security

The physical security consists of the actual measures intended to protect a nuclear power plant or other nuclear facility against unauthorised intrusion and sabotage or other similar action that could result in a radiological accident.

Structural members can be a part of the physical security, for example to prevent unauthorised entry into the facility, i.e. the members can be part of the area security, protective shell and protection against intrusion in the main control room. Such structural members should have sufficient strength (resistance) to resist attempts of unauthorised access.

Furthermore, structural members can protect the facility from the impact and effects of actions that may occur in case of sabotage or other similar action that may cause a radiological accident. Malevolent acts in accordance with the postulated threat scenarios described below should not lead to greater consequences than what can be expected of malfunctioning equipment, failure of equipment, misconduct, events or natural phenomena. This means that war and design basis threats can be evaluated in event class H4.

According to SSMFS 2011: 3 [34] § 11 Section 2, the design related to the physical security of the plant shall be based on analyses in accordance with the National Design Basis Threat specification and be documented in a plan that states the design of the physical security, organisation, management and staffing. The National Design Basis Threat is specified in documentation established by SSM. This type of information is usually confidential.

Requirements regarding physical security related to strength of structural members are evaluated in the limit state ULS_{STR} while requirements regarding leak-tightness are evaluated in ULS_{LEAK}, see Table 3.3.

3.6.6 Deformations and vibrations

In addition to requirements with respect to limitations of deformations and vibrations at normal operation for the serviceability limit state event class H1 and H2, additional requirements of limiting of deformations and vibrations can apply to the ultimate limit state regarding safety-related events in the event class H1 to H5.

Some examples are given below

- structural dilatation joints are not allowed to close due to the static and dynamic deformations that occur in structural members.
- installed systems and components whose function or integrity must be maintained during and after the studied event, may not be jeopardized due to vibrations and temporary deformations in the building structures. However, the requirement is generally transferred to the installed component which is verified with respect to resulting induced vibrations and deformations.

Safety-related structural deformations and induced vibrations are evaluated in a separate limit state, ULS_{VIB}, see Table 3.3.

3.6.7 Environmental resistance

It is stated in SSMFS 2008: 17 [39], § 17 that a nuclear power plant's barriers and equipment belonging to the reactor's safety systems shall be designed to withstand the environmental conditions to which they may be exposed, in situations where their function is taken into account in the safety analysis of the reactor.

The SSM application rules to the paragraph above state that the requirement for environmental resistance implies that structural members, systems, components and devices which are part of safety systems should be environmentally qualified. Corresponding is stated in the SSM application rules to SSMFS 2008:01 [38] for nuclear facilities.

In order to comply with the provisions and application rules of the authority, all structural members affecting the safety functions of the facility have to be designed, constructed, manufactured and assembled as well as tested to the necessary extent for the actual environment in the facility and the expected environment in which the members are intended to fulfil their safety functions.

The evaluation of environmental resistance of structural members is not linked to any specific limit state, see Table 3.3.

3.7 Principles of limit states design

3.7.1 General

See SS-EN 1990 [20] Section 3.1.

There are additional design situations, see section 3.7.2.

3.7.2 Design situations

See SS-EN 1990 [20] Section 3.2.

Design situations are used in the Eurocodes. The following design situations are specified:

- persistent
- transient
- accidental
- seismic

In addition to the above stated design situations, the "highly improbable design situation" is introduced in DNB, with the special case "highly improbable, seismic".

Regarding seismic action, the design basis earthquake (DBE) falls within the accidental design situation, seismic. In order to ensure the robustness beyond DBE, a design extension earthquake (DEE) may for some buildings or structural members need to be verified.

Table 3.4 lists the design situations that apply in DNB.

Table 3.4 – Design situations that apply in DNB.

Design situation	Abbreviation	Explanation
Persistent	per	persistent
Transient	tran	transient
Accidental Special case: Accidental, seismic	exc exc,s	exceptional exceptional, seismic
Highly improbable Special case: Highly improbable, seismic	dec dec,s	design extension condition design extension condition, seismic

Events in event class H1 and H2 are attributed to persistent and transient design situations while events in event class H3 and H4 are assigned to accidental design situations. Finally, events in event class H5 are attributed to the design situation highly improbable. See Table 3.2.

3.7.3 Serviceability limit state

See SS-EN 1990 [20] Section 3.4.

According to the Eurocodes, the serviceability limit state concerns

- the functioning of the structure or structural member during normal use
- the comfort of people and
- the appearance of the building.

The following amendments are made in DNB:

- The definition of normal use in DNB is presented in Annex 4.
- The serviceability limit state should ensure that the future containment function is not compromised in the event of an accident, or that its operational life time is not significantly shortened, due to events during normal use.

3.7.4 Ultimate limit state

See SS-EN 1990 [20] Section 3.3.

According to the Eurocodes, the ultimate limit state concerns

- the safety of people and
- the safety of the structure.

The following amendments are made in DNB:

- The ultimate limit state also concerns the safety-related function of the structure or structural members in normal operation, anticipated events, unanticipated events, improbable events and highly improbable events[7].

Requirements for resistance shall ensure that the safety with respect to failure in the load-bearing structure is sufficient. These requirements, which apply to all event classes, may be expressed differently in different design situations.

Requirements in the ULS shall ensure that the load-carrying structures have sufficient safety with respect to material failure and instability. The requirements shall also ensure that buildings and structural members have adequate safety with respect to overturning, uplift and sliding.

In ULS-exc and ULS-dec, local damage may be acceptable provided that the damage is limited and does not lead to progressive collapse and/or collapse of any other part of the building structure. Local damage is however not accepted if it may present a risk of personal injury. Also, local damages are not accepted for nuclear facilities, if they lead to non-compliance with the safety-related leak-tightness requirements. Further on, local damages are not accepted if they directly or indirectly may jeopardize safety functions that are taken into account during the current event.

3.7.5 Limit state design

See SS-EN 1990 [20] Section 3.5.

3.8 Basic variables

3.8.1 Actions and environmental influences

3.8.1.1 Classification of actions

See SS-EN 1990 [20] Section 4.1.1

In the Eurocodes, actions are classified with respect to their variation in time, accordingly:

- Permanent actions (G)
- Variable actions (Q)
- Accidental actions (A)

According to the Eurocodes, actions shall also be classified

- by their origin, as direct or indirect
- by their spatial variation, as fixed or free
- by their nature and/or the structural response, as static or dynamic

[7] SS-EN 1990 [20] Section 3.3 states that "in some circumstances, the limit states that concern the protection of the contents should be classified as ultimate limit states". It shall be noted that such a requirement is formally introduced by agreement between the authority (Radiation Safety Authority) and the building proprietor of the project. The contact of the Radiation Safety Authority, however, is the license holder for the nuclear facility.

3.8.1.2 Characteristic values of actions

See SS-EN 1990 [20] Section 4.1.2.

Actions applicable for nuclear power plants and other nuclear facilities are presented in Section 4.

3.8.1.3 Other representative values of variable actions

See SS-EN 1990 [20] Section 4.1.3.

3.8.1.4 Representation of fatigue actions

See SS-EN 1990 [20] Section 4.1.4.

3.8.1.5 Representation of dynamic actions

In addition to what is stated in SS-EN 1990 [20] section 4.1.5, section 7.5 presents design provisions for earthquake resistance. Section 7.5 may also, in extent of applicability, be used as guidance for other global vibrational actions.

3.8.1.6 Geotechnical actions

See SS-EN 1990 [20] Section 4.1.6.

3.8.1.7 Environmental influences

See SS-EN 1990 [20] Section 4.1.7.

In addition to the Eurocodes, DNB reports requirements on environmental resistance, see Section 3.6.7.

3.8.2 Material and product properties

See SS-EN 1990 [20] Section 4.2.

This section also applies to highly improbable design situations.

3.8.3 Geometrical data

See SS-EN 1990 [20] Section 4.3.

3.9 Structural analysis and design assisted by testing

3.9.1 Structural analysis

3.9.1.1 Structural modelling

See SS-EN 1990 [20] Section 5.1.1.

3.9.1.2 Static actions

See SS-EN 1990 [20] Section 5.1.2.

When determining actions and effects of actions, concrete creep as well as the stiffness reduction caused by concrete cracking shall be considered, if these effects have an unfavourable impact. The effects may also be taken into account for when they have a favourable effect, provided it can be shown to be acceptable.

Note that load factors and load reduction factors for restraining forces such as temperature, settlement and shrinkage as reported in Section 4 do not include the above mentioned effects.

3.9.1.3 Dynamic actions

See SS-EN 1990 [20] Section 5.1.3.

Analysis methods to be applied when designing for earthquake resistance are reported in section 7.5. These methods can also be used as guidance when designing with respect to other global vibrational actions.

Effects of actions should in general be calculated based on dynamic theory for cases where the actions are of dynamic character. The approach to increase static actions with a dynamic increase factor should only be used in exceptional cases, and when it with certainty is verified to be conservative.

3.9.1.4 Fire design

See SS-EN 1990 [20] Section 5.1.4.

At nuclear power plants, other fire scenarios than those treated in the Eurocodes may occur. See the SAR of each plant.

3.9.2 Design assisted by testing

See SS-EN 1990 [20] Section 5.2.

3.10 Verification by the partial factor method

3.10.1 General

See SS-EN 1990 [20] Section 6.1.

Actions and combinations of actions are chosen according to Section 4.

The method to determine design values directly is not applied in DNB.

3.10.2 Limitations

See SS-EN 1990 [20] Section 6.2.

3.10.3 Design values

3.10.3.1 Design values of actions

See SS-EN 1990 [20] Section 6.3.1.

For design for earthquake resistance, see Section 7.

3.10.3.2 Design values of the effects of actions

See SS-EN 1990 [20] Section 6.3.2.

3.10.3.3 Design values of material or product properties

See SS-EN 1990 [20] Section 6.3.3.

3.10.3.4 Design values of geometrical data

See SS-EN 1990 [20] Section 6.3.4.

3.10.3.5 Design resistance

See SS-EN 1990 [20] Section 6.3.5.

3.10.4 Serviceability limit state

3.10.4.1 Verifications

See SS-EN 1990 [20] Section 6.5.1. The abbreviation SLS is used for the serviceability limit state (Serviceability Limit State).

3.10.4.2 Serviceability criteria

In addition to what is stated in SS-EN 1990 [20] Section 6.5.2, the serviceability criteria stated in the SAR and KFB with associated references, and project-specific documents also apply.

Amendments to the serviceability criteria as stated in the Eurocodes are needed in accordance with Section 3.7.3.

3.10.4.3 Combinations of actions

SS-EN 1990 [20] Section 6.5.3 applies, unless otherwise stated in Section 4.

Three types of combinations of actions for the serviceability limit state are specified in the Eurocodes:

- Characteristic combination (abbreviation ch (characteristic))
- Frequent combination (abbreviation freq (frequent))
- Quasi-permanent combination (abbreviation qp (quasi-permanent))

The characteristic combination is normally applied to irreversible limit states, while the frequent combination applies to reversible limit states. The quasi-permanent combination applies to long-term effects as well as effects on the structural appearance.

3.10.4.4 Partial factors for materials

See SS-EN 1990 [20] Section 6.5.4.

3.10.5 Ultimate limit state

3.10.5.1 General

SS-EN 1990 [20] Section 6.4.1 presents the ultimate limit states that generally shall be verified. The cases denominated GEO, HYD and UPL are not included in DNB.

Amendments in DNB:

Some safety-related functional requirements are classified due to their severity, as ultimate limit state, see section 3.6. Some of these safety-related functional requirements have in turn been assigned its own ultimate limit state.

Table 3.5 presents the ultimate limit states that are taken into account in DNB.

Table 3.5 – Ultimate limit states that are taken into account in DNB.

Limit state	**Abbreviation**	**Explanation**
Ultimate limit state according to SS-EN 1990 Section 6.4.1	ULS	Ultimate Limit State
	ULS$_{\mathrm{EQU}}$	Equilibrium
	ULS$_{\mathrm{STR}}$	Strength
Ultimate limit states with respect to safety-related functional requirements	ULS$_{\mathrm{CONT}}$	Containment (see section 3.6.2)
	ULS$_{\mathrm{LEAK}}$	Leak-tightness (see section 3.6.3)
	ULS$_{\mathrm{VIB}}$	Vibrations (deformations and vibrations, see section 3.6.6)

3.10.5.2 Verification of static equilibrium and resistance

See SS-EN 1990 [20] Section 6.4.2.

3.10.5.3 Combinations of actions (fatigue verifications excluded)

SS-EN 1990 [20] Section 6.4.3.1, 6.4.3.2, 6.4.3.3 and 6.4.3.4 applies unless otherwise stated in Section 4, 5 and 7. SS-EN [20] Section 6.4.3.3 can also be applied to highly improbable design situations, unless otherwise stated in Section 4.

3.10.5.4 Partial factors for actions and combinations of actions

SS-EN 1990 [20] Section 6.4.4 applies with the modifications stated in BFS 2011:10 – EKS 8 [8] section B, unless otherwise stated in Section 4.

3.10.5.5 Partial factors for materials and products

SS-EN 1990 [20] Section 6.4.5 applies, unless otherwise stated in Section 5, 6 or 7.

4. Actions and combinations of actions

4.1 General

In section 4.2, permanent and variable actions as well as accidental actions, which have been identified to be relevant for design of nuclear power plants and other nuclear facilities, are described. Other types of actions shall be considered in case they do not have a negligible effect. Actions applicable for a specific facility are reported in the SAR and KFB[8] of the plant.

Section 4.3 shows combinations of actions and load factors that according to the Eurocodes shall be applied in design in serviceability limit state and ultimate limit state as well as additional combinations of actions specific to nuclear power plants and other nuclear facilities. The additional combinations of actions to be applied are described in the SAR and KFB of the plant. For the ultimate limit state, persistent and transient design situations (conditions at normal operation), accidental design situations as well as highly improbable design situations, are covered.

4.2 Actions

4.2.1 Permanent actions

The characteristic value of a permanent action G can be determined as follows:

- if the variability of G can be assumed to be small, a single value G_k can be applied
- if the variability cannot be considered small, two values shall be applied: an upper value $G_{k,sup}$ and a lower value $G_{k,inf}$.

The following actions belong to the category permanent actions:

D	Self-weight
H_{gw}	Water pressure at normal water level
H_{ge}	Earth pressure and geotechnical action
P_{p}	Prestressing force
$\varepsilon_{\mathrm{cs}}$	Shrinkage
δ_{s}	Settlement

[8] For actions resulting from mechanical systems for which it has been established design specifications (KFM), these can provide guidance and reference for the establishment of KFB in terms of (compare section 3.8.1):

- variation of the actions in time (permanent, variable or accidental actions),
- their origin (direct or indirect), spatial variation (fixed or free) and nature (static or dynamic),
- their location, distribution and magnitude, as wall as
- what actions to be combined.

Furthermore, the KFM can provide information about what event class each action belongs to.

In accordance with section 3.2.2 of EN 1991-3 [26], static load from operation of machinery (imposed action, action induced by machinery) may in some cases be categorized as a permanent action. Furthermore, parts of a process-related action may under certain circumstances be considered a permanent action. This is described under the heading “Process-related actions – general characteristics” in section 4.2.2.

D **Self-weight**

Self-weight of structural members and fixed installations in the building, shall be assumed to be a permanent action and shall be calculated based on the nominal dimensions and characteristic values for densities according to SS-EN 1991-1-1 [21].

$\boldsymbol{H_{gw}}$ **Water pressure at normal water level**

The water pressure at the mean groundwater level, including the possible effect of continuously acting active pumping and drainage systems, or the water pressure at the mean sea water level, as well as the hydrostatic water pressure in pools corresponding to normal water level during operation shall be defined as permanent actions.

$\boldsymbol{H_{ge}}$ **Geotechnical action and earth pressure**

Geotechnical action and earth pressure caused by the self-weight of the soil or by permanent actions on the ground, shall be assumed to be permanent actions and shall be determined in accordance with SS-EN 1997-1 [32].

$\boldsymbol{P_p}$ **Prestressing force**

Prestressing force due to prestressing steel is assumed to be a permanent action. The value of the prestressing force is calculated with consideration to the resulting losses of prestress at considered time according to SS-EN 1992-1-1 [29].

a) The characteristic value at a given time, upper value $P_{\mathrm{pk,sup}}(\mathrm{t})$ and lower value $P_{\mathrm{pk,inf}}(\mathrm{t})$.

b) Mean value at a given time $P_{\mathrm{m}}(\mathrm{t})$.

$\boldsymbol{\varepsilon_{cs}}$ **Shrinkage**

Expected shrinkage of concrete is determined according to SS-EN 1992-1-1 [29] unless another assumption is shown to be more correct.

$\boldsymbol{\delta_s}$ **Settlement**

Differential settlements are determined based on assessment of the geotechnical conditions.

4.2.2 Variable actions

The following actions belong to the category variable actions:

L	Imposed actions
M_{n}	Process-related actions during normal operation and shutdown
M_{d}	Process-related actions during anticipated operational occurence
M_{t}	Process-related actions during testing of the facility
H_{qw}	Water pressure variation relative to normal water pressure

H_{qe} Earth pressure caused by variable surface action

S Snow load

W_q Wind load

ΔT Climate-related temperature difference and temperature changes

L **Imposed actions**

The specified types of imposed action below shall be assumed to act simultaneously when justified by the current design situation. The term imposed action includes the following different action types:

a) Loads from equipment, furniture, people, bulk cargo and general cargo

SS-EN 1991-1-1 [21] specifies load values that can be used as guidelines for those cases where these are not specifically mentioned in the applicable load specification.

b) Action induced by lifting equipment

Actions shall be based on SS-EN 1991-3 [26] where applicable. Cranes and overhead travelling cranes cause vertical and horizontal loads. The magnitude of the loads is determined from the nominal values specified by the crane supplier, unless other values can be shown to be more correct. These nominal values are considered as characteristic load values, unless otherwise stated. When determining the characteristic load values according to SS-EN 1991-3 [26], a lifting class that is one class higher than the current lift device shall be applied in accordance with ETC-C [12] Section 1.3.3.5.2.

c) Action induced by machinery

Actions are based on SS-EN 1991-3 [26] where applicable. Actions of easily moveable machinery are considered as free, variable action. Action from permanently installed machinery with well-defined self-weight is considered to be a permanent action and included in *D* (self-weight), as defined above. Dynamic actions due to eccentricity of rotating machinery and forces caused by switching on and off or other temporary effects are considered as variable actions.

d) Actions caused by the placement of dismantled units

When heavy units, such as parts of the reactor vessel or radiation protection blocks of concrete, are dismantled and placed on existing structures, actions are generated. These actions shall be considered in the design situations during shutdown. The magnitude and the location of the actions shall be based on existing instructions regarding the placement.

e) Heavy transports

The nature of the transport and other required information shall be specified in each particular case. This refers to both transports inside the building as well as traffic load and variable surface load on surrounding ground. In SS-EN 1991-1-1 [21], guidelines of applicable load values of vehicles are specified for the cases where load values are not specified in the current load specification.

f) Exchange of heavy components

Actions that occur during replacement of heavy components in the facility.

Process-related actions – general characteristics

The concept of process-related actions includes process-generated actions during normal operation and shutdown M_n, during anticipated operational occurrence M_d, and during testing of the facility M_t.

M_d includes actions during considered anticipated operational occurrence as well as other simultaneously acting process-related actions. M_t includes actions during considered testing as well as other simultaneously acting process-related actions.

Process-related actions can generally not be regarded as time-varying in the same way as actions of snow and wind, which are random by nature, and their variability can be estimated based on historical data. The extent to which the process actions are uncertain depends on the process and what plant condition that is considered. This can for obvious reasons not be prescribed in general.

Process-related actions shall in general be defined in the form of specified maximum values that for safety reasons can be equated with characteristic load values. Specified maximum values in this case, refers to unfavourable absolute values regardless of sign. Specified maximum values shall be determined based on the plant condition considered, e.g. normal operation and shutdown, anticipated operational occurrence, testing or exceptional impact.

As guidance for determination of these values, the probability that the maximum value is exceeded, shall be so small that it can be equated to e.g. climate-related actions, which are defined so that they can on average be expected to be exceeded only once in 50 years.

For certain cases there might be a physical upper limit for the action considered, for example regarding the water level in a pool or in a tank with safety relief valves that are released at a conservatively determined pressure. This upper limit can then be interpreted as a characteristic value.

If none of the above alternatives are possible to use and if the action is of great importance for a certain design situation, a specific maximum value needs to be determined based on a special investigation considering the uncertainty of the action.

Some types of process-related actions or impacts can have a permanent character, due to the fact that they during normal operation are relatively constant through time. Some examples of such actions are pressure differences, temperature differences between different parts of a construction and water pressure in pools. Also, regarding water pressure, section 4.2.1 states a permanent water pressure corresponding to normal water level during operation. However, the water level variation around this value is regarded as a time-varying process-related action. For some design situations it can be justified that other process-related actions such as pressure- and temperature differences are described with a permanent part corresponding to a mean value during normal operation in combination with a variable part, where the latter describes the difference between time-variable impact and the permanent value.

If a process-related time-variable action is favorable in a combination of actions, it shall normally be set to zero. Nevertheless, there can be scenarios and design situations where it might be justified to consider a favorable acting process-related action separated from zero. In such cases, a conservatively specified minimum value shall be used. A presumption is that the action for certain is active in the studied design situation.

Process-related actions during normal operation and shutdown M_n, during anticipated operational occurrence M_d, and during testing of the facility M_t

The specified types of process-related actions below shall be assumed to act simultaneously when justified by the current design situation. The concept of process-related actions includes following different types of actions:

a) Process-related actions from piping and processing systems: $M_{n,R}$, $M_{d,R}$ and $M_{t,R}$

Process-related actions from piping and processing systems refer to e.g. reaction forces from the piping system acting on the building during different plant conditions and during testing. Self-weight and mass forces of the medium inside the pipes are included, but not the self-weight of the permanently installed mechanical system components which are considered as permanent actions. During testing, reaction forces from lifting devices acting on the building are included.

b) Process-related overpressure or underpressure: $M_{n,P}$, $M_{d,P}$ and $M_{t,P}$

This action type refers to the differential pressure caused by differences between pressure inside and outside compartment of the building, such as outside and inside of the reactor containment or between different regions within the containment during different plant conditions and during testing. This action refers to e.g. the pressure difference arising during pressure test and periodic leak-tightness tests of the reactor containment. The pressure test load shall be set to $1.15P_{aL}$[9]. P_{aL} is defined in section 4.2.3. Specified maximum values in this case, refer to unfavourable absolute values regardless if it is overpressure or underpressure. If favourable values are used for this action, these are treated analogously but as conservative minimum values.

c) Process-related temperature differences and temperature changes: $M_{n,\Delta T}$, $M_{d,\Delta T}$ and $M_{t,\Delta T}$

This action refers to both the temperature distribution within and between different parts of the structure and the change in temperature over time during different plant conditions and during testing. Specified maximum values in this case, refer to unfavourable absolute values regardless of the temperature difference sign. If favourable values are used for this action, these are treated analogously but as conservative minimum values.

For structures that are exposed to outdoor climate, the maximum and minimum values for outdoor temperature are determined according to SS-EN 1991-1-5 [25], see load ΔT – “Climate-related temperature difference and temperature changes”.

Reference temperatures for the determination of temperature changes over time are estimated from case to case.

d) Process-related water pressure variations: $M_{n,Hqw}$, $M_{d,Hqw}$ and $M_{t,Hqw}$

Time-variable water level for buildings with pools is determined by water level changes that occur during different plant conditions and during testing. The variable load component of water pressure is considered as the difference between the water pressure at the time-variable water level and the water pressure at normal water level. Specified maximum values in this case, refer to unfavourable absolute values. If favourable values are used for this action, these are treated analogously but as conservative minimum values.

Actions due to surge are also included.

e) Safety relief valve blow-down or other pressure relief of high energy device: $M_{n,SRV}$, $M_{d,SRV}$ and $M_{t,SRV}$

[9] In accordance with ASME Sect III Div 2 [6]. The pressure testing is carried out with this increased value with the primary purpose to demonstrate that the containment is properly constructed and to verify an acceptable behaviour of the included parts. The magnification factor accounts for that the pressure test is conducted at room temperature, whilst in case of an accident an increased temperature act simultaneously with the increased differential pressure.

This action refers to resulting actions on the building structure due to the safety relief valve blow-down or other high-energy pressure relief device during different plant conditions and during testing.

H_{qw} Water pressure variation relative to normal water pressure

This action refers to water pressure conditions related to external climatic effects and not to process-related water pressure conditions such as water pressures in tanks and pools. The variable load component of water pressure is considered as the difference between the water pressure at the time-variable water level and the mean water pressure. Time-variable water level of ground water is based on the highest high water level, HHW and the lowest low water level, LLW, unless otherwise stated. In cases where the ground water surface is regulated continuously by active pumping and drainage systems, the groundwater pressure variation is determined according to the principles applicable for the process-related conditions during normal operation, M_n or during anticipated operational occurence M_d.

H_{qe} Earth pressure caused by variable surface action

Variable action, from e.g. vehicles, on the ground surface causes a horizontal or near horizontal earth pressure. This earth pressure is considered as a free variable action and can be determined according to SS-EN 1997-1 [32].

S Snow load

Snow load shall be expressed as weight per horizontal area. The snow load is determined in accordance with SS-EN 1991-1-3 [23] based on a prescribed characteristic value for each snow zone, and a form factor that depends on the shape of the roof surface and the risk of snow accumulation.

W_q Wind action

Wind action is determined in accordance with SS-EN 1991-1-4 [24].

ΔT Climate-related temperature difference and temperature changes

This action refers to climate-related temperature changes in the outdoor air, in watercourses and in lake and sea water. Maximum and minimum values of the outdoor temperature are determined according to SS-EN 1991-1-5 [25]. Reference temperature for the determination of temperature changes over time are estimated from case to case.

Note that this action does not concern process-related temperature loads (see $M_{n,\Delta T}$, $M_{d,\Delta T}$ and $M_{t,\Delta T}$).

4.2.3 Accidental actions - exceptional

Accidental actions refer to incidents of accidental nature and are described below for each event:

Event: Pipe rupture

The actions described below shall be considered in the event of a pipe rupture. Regarding nuclear power plants this primarily refers to pipe failure inside the containment but where applicable the impact caused by pipe rupture outside the containment shall also be considered. In design and analysis, both the local and global effects shall be taken into account, as well as the time-dependent and dynamic effects of the actions.

P_a Transient overpressure and underpressure during pipe rupture

ΔT_a Temperature differences and temperature changes associated with P_a

P_{aL} Specified pressures

ΔT_{aL} Temperature differences and temperature changes associated with P_{aL}

R Direct loads caused by pipe rupture

F Pool dynamic load

H_{if} Action due to exceptional internal water pressure at pipe rupture

Event: Other pressure differences

P_g Pressure difference not included in the pipe rupture event

Event: Safety relief valve blow-down

F_{SRVe} Pool dynamic load due to extreme safety relief valve blow-down

Event: Exceptional internal water pressure

H_{if} Action due to exceptional internal water pressure

Event: Exceptional external flooding

H_{ef} Action due to exceptional external flooding

Event: Transportation accident

Y Action due to transportation accident

Event: Earthquake

E_{DBE} Action caused by design basis earthquake (DBE)

Event: Explosions

X_e Action due to explosions

Event: Missiles

X_m Missile generated loads

Event: Air plane crash

X_{APC} Actions related to air plane crash (APC)

Event: War action and actions related to design basis threats

X_{DBT} War action and actions related to design basis threats (DBT)

Event: Other exceptional impact

X Action due to other exceptional impact

Event: Extreme climate impact

W_a Action due to extreme climate impact

Event: Fire

B Action of fire

The accidental actions included in the above identified events are described in detail below.

P_a Transient overpressure and underpressure during pipe rupture

P_a represents the transient differential pressure caused by differences between pressure inside and outside the reactor containment, between different compartments within the containment and between different areas in other parts of the facility in the event of a pipe rupture.

ΔT_a Temperature differences and temperature changes associated with P_a

This action refers to both the temperature differences between different parts of the structure and the change in temperature over time, which occurs at the same time as the pressure P_a. It also includes the reaction forces from pipe supports caused by temperature change in the pipe systems. For structures that are also exposed to outdoor climate, the maximum and minimum values for outdoor temperature are determined according to SS-EN 1991-1-5 [25]. Reference temperature for the determination of temperature changes over time are estimated from case to case.

P_{aL} Specified pressures

For a nuclear power plant, P_{aL} is the specified values of the differential pressure P_a, denoted specified pressures. This action is typically used for global verification of the reactor containment. For existing plants, the specified pressure is given in the safety analysis report (SAR).

For the design of new facilities, the specified pressure is determined from containment analyses, where the accident scenario that gives the largest differential pressure shall be used. For the determination of P_{aL}, a safety margin is added to the calculated largest differential pressure. The safety margin should be at least 10% of the associated absolute pressure, which is added to the calculated largest differential pressure[10]. This is done in order to compensate for the uncertainties associated with the calculation methods and calculation cases.

ΔT_{aL} Temperature differences and temperature changes associated with P_{aL}

ΔT_{aL} refers to temperatures which occurs in the structure when the pressure P_a reaches the value P_{aL}. Also, see the description of $\Delta T_{a.}$

R Direct actions caused by pipe rupture

Direct actions during pipe rupture are assumed to be specified in the form of representative accidental load values. The specified types of actions below shall be assumed to act simultaneously when justified by the current design situation.

a) Jet load due to pipe rupture, R_{rj}

Reaction forces caused by emanating steam or water during pipe rupture.

b) Missile load due to pipe rupture, R_{rm}

Forces caused by missiles generated at pipe rupture.

c) Pipe support reaction forces due to pipe rupture, R_{rr}

Pipe support reaction forces caused by pipe rupture.

F Pool dynamic load

[10] In accordance with YVL B.6 [36]

Loads caused by pool dynamic effects due to pipe rupture shall be considered during design (BWR plants only). The specified types of loads below shall be assumed to act simultaneously when justified by the current design situation.

a) Drag and impact loads caused by level-swell in the condensation pool, F_{ps}

Loads caused by pool dynamic effects during the initial stage of the blow-down to the condensation pool during a pipe rupture.

b) Loads due to condensation oscillations, F_{CO}

Loads due to pool dynamic effects during the intermediate stage of the blow-down to the condensation pool during a pipe rupture.

c) Loads caused by chugging, F_{CH}

Loads due to pool dynamic effects during the final stage of blow-down to the condensation pool during a pipe rupture.

d) Loads caused by safety relief valve blow-down during a pipe rupture, F_{SRVa}

Loads caused by dynamic effects during safety relief valve blow-down to the condensation pool or other pressure relief of high-energy system during pipe rupture.

P_g Pressure not included in the pipe rupture event

Reactor containments and other buildings are designed for the overpressure or underpressure that can occur as a result of other accidents than pipe ruptures.

F_{SRVe} Pool dynamic load due to extreme safety relief valve blow-down

Loads caused by pool dynamic effects at extreme safety relief valve blow-down to the condensation pool in the containment or other extreme pressure relief of high-energy system.

H_{if} Action due to exceptional internal water pressure

Reactor containments and other buildings shall be designed for exceptional water pressure resulting from the rupture of pipes or as a result of an accident or failure of a process system, such as major leaks from piping systems, tanks or pools, pumpstop (surge) and increasing water levels in pools and tanks.

H_{ef} Action due to exceptional external flooding

Reactor containments and other buildings shall be designed for actions due to exceptional external flooding including the effects of high water level and high sea water waves.

Furthermore, external flooding due to malfunctioning of active pumping and drainage system shall be considered.

Y Action due to transportation accident

Actions caused by different types of transportation accidents shall be considered in the design. Intended actions are for example actions related to lifting devices and lifting, such as unexpected braking, buffer and skewing forces, impact from swinging load in the overhead travelling crane or from heavy objects falling. This action also includes actions due to collisions with vehicles.

E_{DBE} Action caused by design basis earthquake (DBE)

The reactor containment and other buildings which require consideration of actions caused by the earthquake, shall be designed for earthquake - DBE in accordance with section 7.4.1.

X_e Action due to explosions

This action refers to different types of explosions, both inside (hydrogen explosion, etc.) and outside the facility (other activities, transportation accident, ruptured pipeline, etc.).

The following impact on the building structure may need to be considered:

a) Shock wave (in air, soil or water)

b) Heat radiation

c) Missiles

d) Influence of short-term fireball

e) Effect of fire action with longer duration

The above types of effects of actions shall be assumed to act simultaneously when justified by the current design situation.

X_m Missile generated loads

Missiles can be generated either by rotating components (detached parts) or by ruptured high energy systems.

Missiles may also occur as an effect of pipe rupture, explosion, air plane crash and tornado. The missiles are then belonging to these events. Falling objects are covered in the transportation accident event.

X_{APC} Actions related to air plane crash

A number of different types of actions related to air plane crash can occur in accordance with IAEA Safety Guide NS-G-1.5 [18]:

a) Direct hit of fuselage[11]

b) Missile load caused by broken off engines

c) Shock wave (in air, soil or water)

d) Heat radiation

e) Missiles generated by the shock wave

f) Effect of short-term fireball

g) Effect of fire action with longer duration

Actions c) - g) can occur as a result of the released air plane fuel.

The above types of effects of actions shall be assumed to act simultaneously when justified by the current design situation.

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[11] This load is generally specified as a time-history load.

X_{DBT} War action and actions related to design basis threats (DBT)

Reactor containments and other buildings shall be designed for actions caused by war or as a result of design basis threats to the extent specified by the Swedish Radiation Safety Authority (SSM). Note that this information usually is confidential.

X Action due to other exceptional impact

The buildings shall be designed for actions resulting from other exceptional effects caused by internal or external events according to the safety analysis report (SAR).

Examples of other exceptional impact are:

a) Increased temperature conditions

This impact refers to actions that are not caused by pipe rupture, which is discussed separately, see above.

b) Accidental action from malfunctioning machines

c) Short-circuit in switchgear

d) Failure of the tanks associated with high-pressure systems

e) Landslide

W_a Extreme climate impact

Buildings must be designed for actions caused by extreme wind actions, outdoor temperatures and precipitation according to the safety analysis report (SAR). In extreme wind actions tornadoes and tornado generated missiles are included. Precipitation includes effects of rain, snow, hail and ice.

B Action of fire

This action refers to different types of fires originating inside or outside the facility. For some events, such as explosions and air plane crash, heat radiation and fire are included as a part of actions on buildings. In such cases the action of fire belongs to these events.

In case of fire, required load capacity as well as limitation of fire spread shall be verified to the extent required. Fire gives rise to temperature-related load effects and a strength reduction of the building structure.

In SS-EN 1991-1-2 [22] guidelines and load values are provided, that can be used when these are not specifically mentioned in the applicable load specification of the facility. Sometimes project and room specific investigations have to be carried out.

4.2.4 Accidental actions - highly improbable events

Highly improbable events to be postulated for existing plants and design of new plants, are established by the responsible authority, and are stated in the safety analysis report (SAR) for each plant. See also section 1.1.

Accidental actions - highly improbable events are linked to incidents of accidental nature and are reported below for each event:

Event: Severe accidents

Z_{SA} Effects of actions related to severe accidents

Event: Highly improbable earthquake

E_{DEE} Action caused by design extension earthquake (DEE)

Event: Highly improbable external flooding

Z_{Hef} Action caused by highly improbable external flooding

Event: Air plane crash with large commercial aircraft

Z_{APC} Effects of actions related to air plane crash with large commercial aircraft

Event: Other highly improbable impact

Z Action due to other highly improbable impact

The accidental actions included in the above identified events are described in detail below.

Z_{SA} Effects of actions related to severe accidents

According to regulations issued by the Swedish Radiation Safety Authority (SSM), analyses must be performed to estimate the possible pressure and temperature conditions in the containment related to severe accidents.

Actions related to severe accidents are assumed to be specified in the form of representative accident load values. The types of actions specified below shall be assumed to act simultaneously when justified by the current design situation.

a) Overpressure and underpressure due to severe accidents, $Z_{SA,P}$

For existing plants, the maximum differential pressure $Z_{SA,P}$ for which the leak-tightness of the containment shall be verified, is specified in the safety analysis report (SAR). Specified internal overpressure level varies from unit to unit, but is usually somewhere between 1.5-2.0 times P_{aL}. P_{aL} is described in section 4.2.3.

Verification of the leak-tightness of the containment in new designs shall be based on the maximum calculated differential pressure. When determining the $Z_{SA,P}$, a margin shall be added to the calculated largest differential pressure[12]. This is done in order to compensate for the uncertainties associated with the calculation methods when determining pressure levels on the inside of the containment during severe accidents, and to compensate for the fact that only a selection of analysis cases are studied. The margin shall not, however, compensate for uncertainties related to the capacity of the containment. The size of the margin is determined from case to case depending on type of facility, considering the uncertainties specified above. The internal design differential pressure should however not be set lower than

- 2.5 times P_{aL} for verification of the structural capacity,
- 2.0 times P_{aL} for verification of the leak-tightness,

regardless of calculated pressure level[13].

b) Temperature differences and temperature changes due to severe accidents, $Z_{SA,\Delta T}$

c) Exceptional internal water pressure due to severe accidents, $Z_{SA,Hif}$

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[12] In accordance with YVL B.6 [36].

[13] These minimum requirements ensure that a sufficient robustness of the containment function is achieved. Note that the Radiation Safety Authority may state additional requirements.

This also applies to hydrostatic pressure caused by controlled flooding of the containment.

E_{DEE} Action caused by design extension earthquake (DEE)

Buildings and structural members that constitute, protect or support especially important safety functions shall to the extent specified by the responsible authority be verified for design extension earthquake - DEE according to section 7.4.1 in order to ensure robustness beyond the design basis earthquake E_{DBE}.

Z_{Hef} Action caused by highly improbable external flooding

This action includes loads orginating from highly improbable external flooding, including effects of high water level and high sea water waves.

Z_{APC} Actions related to air plane crash with large commercial aircraft

Responsible authority specifies the design load conditions. Note that this information is usually confidential.

A number of different types of actions related to air plane crash with large commercial passenger aircraft may occur, see the description of the action X_{APC}.

Z **Action due to other highly improbable impact**

Actions due to other postulated highly improbable impact shall be considered to the extent postulated.

4.3 Combinations of actions

4.3.1 General

Combinations of actions in the serviceability limit state (SLS), can, according to SS-EN 1990 [20] section 6.5.3 be of three types: Characteristic, frequent or quasi-permanent. See section 3.10.4.3.

In the ultimate limit state combinations of actions are specified for each design situation, see SS-EN 1990 [20] section 6.4.3. For persistent and transient design situations, in SS-EN 1990 [20], Annex A1, with national modifications in the EKS [8], a set of design values are given for actions depending on the ultimate limit state studied. The present report applies method 2 according to SS-EN 1990 [20] Appendix A1, section A1.3.1. This means that set A and set B is used. Current design situations are listed in section 3.10.5.1.

Table 4.1 shows a summary of the principal combinations of actions that according to the Eurocodes shall be taken into consideration (except for fatigue). Actions that cannot occur simultaneously, depending on the physical or functional causes, should not be considered simultaneously in combinations of actions.

Table 4.1 – Schematic summary of the combinations of actions to be considered (except for fatigue).

Limit state	**Design situation/ comb. type** 1)	**Combination of actions**
SLS	ch	Characteristic (eq. 6.14 in SS-EN 1990)
	freq	Frequent (eq. 6.15 in SS-EN 1990)
	qp	Quasi-permanent (eq. 6.16 in SS-EN 1990)
ULS_{EQU}	per tran	Set A (table B-2 in EKS)
	exc exc,s	Table A1.3 in SS-EN 1990, modification according to EKS
	dec dec,s	
Other ULS (except ULS_{FAT})	per tran	Set B (table B-3 in EKS)
	exc exc,s	Table A1.3 in SS-EN 1990, modification according to EKS
	dec dec,s	

1) DNB does not specify any specific design situations in SLS. Instead, combinations of action types according to SS-EN 1990 are used.

Values of load reduction factors ψ_0, ψ_1 and ψ_2 are given for certain imposed actions, as well as for snow loads, wind actions and climatic related temperature difference in SS-EN 1990 [20], Annex A1, with national choices in EKS [8] (see Table 4.2). For other variable actions, $\psi_0 = 1$ unless another value is shown to be more correct.

Values of ψ_1 and ψ_2 for the latter type of variable actions Q are determined from case to case if necessary. Thereby, the following applies:

- Frequent load value $\psi_1 Q_k$ is the action that is exceeded 1% of the time
- Quasi-permanent load value $\psi_2 Q_k$ is the action that is exceeded 50% of the time, alternatively the average time value of the action.
- For accidental events, the above definitions of ψ_1 and ψ_2 should only be seen as indicative. The choice of load values shall be made in accordance with the initial plant conditions to be assumed for different types of accidental events.

The above principles, together with the modifications and amendments introduced in Section 3, form the basis of the combination of actions tables presented in the following section.

Each combination of actions can contain several alternative sets regarding magnitude and position of the including actions.

Table 4.2 – Load reduction factors for certain actions included in the Eurocodes.

Reference	Action		ψ_0	ψ_1	ψ_2
SS-EN 1990, Annex A1, with national choices in EKS	Imposed actions in buildings:	L			
	Cat. A: Domestic, residential areas		0.7	0.5	0.3
	Cat. B: Office areas		0.7	0.5	0.3
	Cat. C: Congregation areas		0.7	0.7	0.6
	Cat. D: Shopping areas		0.7	0.7	0.6
	Cat. E: Storage areas		1.0	0.9	0.8
	Cat. F: Traffic area, vehicle weight < 30 kN		0.7	0.7	0.6
	Cat. G Traffic area, vehicle weight >30 kN		0.7	0.5	0.3
	Cat. H: Roofs		0	0	0
	Snow loads > 3 kN/m^2	S	0.8	0.6	0.2
	Snow loads 2-3 kN/m^2	S	0.7	0.4	0.2
	Snow loads 1-2 kN/m^2	S	0.6	0.3	0.1
	Wind actions	W_q	0.3	0.2	0
	Temperature in buildings	-	0.6	0.5	0

4.3.2 Combinations of actions in the serviceability limit state

Combinations of actions that according to the Eurocodes shall be applied in the serviceability limit state are listed in Table 4.3. Numerical values and expressions in the table specify the load factor each characteristic action shall be multiplied with in the current combination of actions.

Specified combinations of actions apply to both reactor containment and other buildings where applicable.

Table 4.3 – Combinations of actions in the serviceability limit state.

Action	applies to	Combination of action: normal operation or shutdown, irreversible limit state	normal operation or shutdown, reversible limit state	normal operation or shutdown, long-term effects	anticipated operational occurence	testing
	Number	1	2	3	4	5
Permanent actions						
Self weight[1)] -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0
Water pressure H_{gw}		1.0	1.0	1.0	1.0	1.0
Earth pressure H_{ge}		1.0	1.0	1.0	1.0	1.0
Prestressing -unfavourable $P_{pk,sup}$ -favourable $P_{pk,inf}$		 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0
Shrinkage ε_{cs}[2)]		1.0	1.0	1.0	1.0	1.0
Settlement δ_s[2)]		1.0	1.0	1.0	1.0	1.0
Variable actions[5)]						
Imposed action L		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Snow load S		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Wind action W_q		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Climate-related temperature diff. ΔT		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Water level var. H_{qw}		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Earth pressure H_{qe}		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$	$1.0\psi_0$[3)]	$1.0\psi_0$[3)]
Process action M_n[6)]		$1.0\psi_0$[3)]	$1.0\psi_2$[4)]	$1.0\psi_2$		
Anticipated operational occurence M_d[6)]					$1.0\psi_0$[3)]	
Testing M_t[6)]						$1.0\psi_0$[3)]
Event class		H1, H2	H1, H2	H1	H2	H2
Combination type		Characteristic	Frequent	Quasi-permanent	Characteristic/ Frequent	Characteristic/ Frequent

1) Regarding upper and lower values, see section 4.2.1

2) If the action is favourable, the load factor shall be set to 0.

3) If one of these actions is the leading action, 1.0 shall be used instead of 1.0 ψ_0 for this action.

4) If one of these actions is the leading action, ψ_2 shall be replaced by ψ_1 for this action.

5) Variable actions that are favourable shall be set to 0.

6) If several different process actions are correlated so that they can be expected to act simultaneously, they shall be considered as one action in the combination of actions. Regarding uncorrelated process actions, only one of them shall be the leading action.

4.3.3 Combinations of actions in ultimate limit state - persistent & transient design situations

Differentiated safety levels in the ultimate limit state are described by the factor γ_d with values as shown in Table 4.4, which depends on the safety class, see section 3.3. For structures in nuclear power plants, safety class B3 shall normally be used. The same normally also applies for safety-related buildings at other nuclear facilities.

Table 4.4 – Safety classes in ultimate limit state design.

Safety class	**Consequences**	γ_d
B3	High risk of severe damage	1.0
B2	Some risk of severe damage	0.91
B1	Small risk of severe damage	0.83

Combinations of actions that according to the Eurocodes shall be applied in the ultimate limit state are listed in Table 4.5. Numerical values and expressions in the table specify the load factor each characteristic action shall be multiplied with in the current combination of actions.

Specified combinations of actions apply to both reactor containment and other buildings where applicable.

Table 4.5 –Combinations of actions in the ultimate limit state – persistent & transient.

Action	applies to	Combination of actions: normal operation or shutdown (permanent action dominating)	normal operation or shutdown	anticipated operational occurence	testing
Number		6	7	8	9
Permanent actions[6)7)]					
Self weight[1)] -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 $\gamma_d \cdot 1.35$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0
Water pressure H_{gw} -unfavourable -favourable		 $\gamma_d \cdot 1.35$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0
Earth pressure H_{ge} -unfavourable -favourable		 $\gamma_d \cdot 1.35$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0	 $\gamma_d \cdot 1.2$ 1.0
Prestressing P_{pm} -unfavourable -favourable		 $\gamma_d \cdot \gamma_{p,unfav}$ 5) 1.0	 $\gamma_d \cdot \gamma_{p,unfav}$ 5) 1.0	 $\gamma_d \cdot \gamma_{p,unfav}$ 5) 1.0	 $\gamma_d \cdot \gamma_{p,unfav}$ 5) 1.0
Shrinkage ε_{cs} 2)		$\gamma_d \cdot 1.35$	$\gamma_d \cdot 1.2$	$\gamma_d \cdot 1.2$	$\gamma_d \cdot 1.2$
Settlement δ_s 2)		$\gamma_d \cdot 1.35$	$\gamma_d \cdot 1.2$	$\gamma_d \cdot 1.2$	$\gamma_d \cdot 1.2$
Variable actions[4)]					
Imposed action L		$\gamma_d \cdot 1.5\psi_0$	$\gamma_d \cdot 1.5\psi_0$ 3)10)	$\gamma_d \cdot 1.5\psi_0$ 3)10)	$\gamma_d \cdot 1.5\psi_0$ 3)
Snow load S		$\gamma_d \cdot 1.5\psi_0$	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)
Wind action W_q		$\gamma_d \cdot 1.5\psi_0$	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)
Climate-related temperature diff. ΔT		$\gamma_d \cdot 1.5\psi_0$	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)	$\gamma_d \cdot 1.5\psi_0$ 3)
Water level var. H_{qw} -unfavourable		 $\gamma_d \cdot 1.5\psi_0$	 $\gamma_d \cdot 1.5\psi_0$ 3)	 $\gamma_d \cdot 1.5\psi_0$ 3)	 $\gamma_d \cdot 1.5\psi_0$ 3)
Earth pressure H_{qe} -unfavourable		 $\gamma_d \cdot 1.5\psi_0$	 $\gamma_d \cdot 1.5\psi_0$ 3)	 $\gamma_d \cdot 1.5\psi_0$ 3)	 $\gamma_d \cdot 1.5\psi_0$ 3)
Process action M_n 8) -unfavourable		 $\gamma_d \cdot 1.5\psi_0$	 $\gamma_d \cdot 1.5\psi_0$ 3)		
Anticipated operational occurence M_d 8) -unfavourable				 $\gamma_d \cdot 1.5\psi_0$ 3)	
Testing M_t 8) -unfavourable					 $\gamma_d \cdot 1.5\psi_0$ 3)9)10)
Event class		H1, H2	H1, H2	H2	H2
Design situation		per	per	tran	tran

1) Regarding upper and lower values, see section 4.2.1.

2) If the action is favourable, the load factor shall be set to 0.

3) If one of these actions is the leading action, ψ_0 shall be replaced by 1.0 for this action.

4) Variable actions that are favourable shall be set to 0.

5) $\gamma_{p,unfav}$ is set to 1.2 for verification of local effects and to 1.3 at risk for instability with external prestress, see SS-EN 1992-1-1 Section 2.4.2.2. For other cases $\gamma_{p,unfav}$ is set to 1.0.

6) For EQU-verification the unfavorable factor is set to $\gamma_d \cdot 1.1$.

[7] For EQU-verification the favorable factor is set to 0.9.

[8] The load factor can be applied to specified maximum values for the loads, see section 4.2.2. If several different process actions are correlated so that they can be expected to act simultaneously, they shall be considered as one action in the combination of actions. Regarding uncorrelated process actions, only one of them shall be the leading action.

[9] For pressure differences during pressure test according to ASME Sect III Div 2 [6] CC 6000 or equivalent pressure test programs the load factor can be reduced from 1.5 to 1.35. This is because the magnitude of the load is well-defined, and that both the magnitude of the load and the response of the structure are verified during the pressure test.

[10] According to EN 1991-3 [26] Appendix A the load factor for loads from crane on crane rail beam can be reduced from 1.5 to 1.35.

4.3.4 Combinations of actions in the ultimate limit state - accidental design situations

Combinations of actions for accidental design situations (ultimate limit state), as applicable according to the Eurocodes, are listed in Table 4.6. Numerical values and expressions in the table specify the load factor each characteristic action shall be multiplied with in the current combination of actions.

Each of the combinations 10-25 is related to an event of accidental nature. Two or more of such events are not considered to occur simultaneously.

For the accidental actions P_a, ΔT_a, R, F and H_{if} in combination 12, the actions that may act simultaneously during an assumed event are included in each design case. In such cases, the time sequences of the included load components shall be considered. Combinations of actions of the simultaneous effect of the different action can be taken into account, e.g. according to the principles given for the corresponding load case in the design specification for mechanical systems (KFM) installed in the facility.

Specified combinations of actions apply to both reactor containment and other buildings where applicable.

Table 4.6 – Combinations of actions in the ultimate limit state – accidental.

Action		Combination of actions							
	applies to	accident due to pipe rupture[1)]	accident + large water level variations	pipe rupture	differential pressure not related to pipe rupture	safety relief valve blow-down	exceptional internal water pressure	exceptional external flooding	transportation accident
Number		10	11	12	13	14	15	16	17
Permanent actions									
Self weight[2)] -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0
Water pressure H_{gw}		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Earth pressure H_{ge}		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Prestressing P_{pm}		$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$	$\gamma_{p,unfav}^{6)}$
Shrinkage $\varepsilon_{cs}^{3)}$		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Settlement $\delta_s^{3)}$		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Variable actions [4)]									
Imposed action L		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Snow load S		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Wind action W_q		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Climate-related temperature diff. ΔT		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Water level var. H_{qw}		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Earth pressure H_{qe}		$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Process action $M_n^{7)}$					$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$	$1.0\psi_2^{5)}$
Anticipated operational occurence $M_d^{7)}$				$1.0\psi_2^{5)8)}$					
Accidental actions									
P_a				1.0					
ΔT_a				1.0					
P_{aL}		1.5	1.0						
ΔT_{aL}		1.0	1.0						
R				1.0					
F				1.0					
H_{if}			1.0	1.0			1.0		
P_g					1.0				
F_{SRVe}						1.0			
H_{ef}								1.0	
Y									1.0
E_{DBE}									
X_e									
X_m									
X_{APC}									
X_{DBT}									
X									
W_a									
B									
Event class		H4	H4	H3, H4	H3, H4	H3, H4	H3, H4	H3, H4	H3, H4
Design situation		exc	exc	exc	exc	exc	exc	exc	exc

Action		Combination of actions							
	applies to	Design Basis Earthquake (DBE)	explosions	missiles	air plane crash	war action and actions related to design basis threats	other exceptional impact	extreme climate impact	fire
Number		18	19	20	21	22	23	24	25
Permanent actions									
Self weight 2) -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0
Water pressure H_{gw}		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Earth pressure H_{ge}		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Prestressing P_{pm}		$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)	$\gamma_{p,unfav}$ 6)
Shrinkage ε_{cs} 3)		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Settlement δ_s 3)		1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Variable actions 4)									
Imposed action L		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Snow load S		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Wind action W_q		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Climate-related temperature diff. ΔT		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Water level var. H_{qw}		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Earth pressure H_{qe}		$1.0\psi_2$	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Process action M_n 7)			$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)	$1.0\psi_2$ 5)
Anticipated operational occurence M_d 7)		$1.0\psi_2$ 8)							
Accidental actions									
P_a									
ΔT_a									
P_{aL}									
ΔT_{aL}									
R									
F									
H_{if}									
P_g									
F_{SRVe}									
H_{ef}									
Y									
E_{DBE}		1.0							
X_e			1.0						
X_m				1.0					
X_{APC}					1.0				
X_{DBT}						1.0			
X							1.0		
W_a								1.0	
B									1.0
Event class		H3, H4	H3, H4	H3, H4	H3, H4	H4	H3, H4	H3, H4	H3, H4
Design situation		exc,s	exc	exc	exc	exc	exc	exc	exc

[1] This combination of actions is used for primary verification of the containment function during postulated failure.

[2] Regarding upper and lower values, see section 4.2.1.

[3] If the action is favourable, the load factor shall be set to 0.

[4] Variable actions that are favourable shall be set to 0.

[5] For the dominant of these actions ψ_2 shall be replaced with ψ_1.

[6] $\gamma_{p,unfav}$ is set to 1.2 for verification of local effects and to 1.3 at risk for instability with external prestress, see SS-EN 1992-1-1 Section 2.4.2.2. For other cases $\gamma_{p,unfav}$ is set to 1.0.

[7] If several different process actions are correlated so that they can be expected to act simultaneously, they shall be considered as one action in the combination of actions. Regarding uncorrelated process actions, only one of them shall be the dominating action.

[8] For BWR plants $1.0\psi_2$ should be replaced by 1.0 for the action $M_{d,SRV}$.

4.3.5 Combinations of actions in the ultimate limit state - highly improbable design situations

Combinations of actions that when applied to the principles of the Eurocodes for accidental actions shall be applied to the postulated highly improbable design situations (ultimate limit state), are listed in Table 4.7. Numerical values and expressions in the table specify the load factor each characteristic action shall be multiplied with in the current combination of actions.

Each of the combinations 26-30 is related to an event of accidental nature. Two or more of such events are not considered to occur simultaneously.

Specified combinations of actions apply to both reactor containment and other buildings where applicable.

Table 4.7 – Combinations of actions in the ultimate limit state – highly improbable.

Action		Combination of actions				
	applies to	severe accidents	design extension earthquake (DEE)	highly improbable external flooding	air plane crash with large commercial air-craft	other highly improba-ble impact
Number		26	27	28	29	30
Permanent actions						
Self weight[1)] -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0	 1.0 1.0
Water pressure H_{gw}		1.0	1.0	1.0	1.0	1.0
Earth pressure H_{ge}		1.0	1.0	1.0	1.0	1.0
Prestressing P_{pm}		1.0	1.0	1.0	1.0	1.0
Shrinkage ε_{cs}[2) 4)]		1.0	1.0	1.0	1.0	1.0
Settlement δ_s[2) 4)]		1.0	1.0	1.0	1.0	1.0
Variable actions[3)]						
Imposed action L		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Snow load S		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Wind action W_q		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Climate-related temperature diff. ΔT[4)]		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Water level var. H_{qw}		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Earth pressure H_{qe}		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Process action M_n		$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$	$1.0\psi_2$
Accidental actions						
Z_{SA}		1.0				
E_{DEE}			1.0			
Z_{Hef}				1.0		
Z_{APC}					1.0	
Z						1.0
Event class		H5	H5	H5	H5	H5
Design situation		dec	dec,s	dec	dec	dec

[1)] Regarding upper and lower values, see section 4.2.1.

[2)] If the action is favourable, the load factor shall be set to 0.

[3)] Variable actions that are favourable shall be set to 0.

[4)] Influence of shrinkage, settlement and climate-related temperature differences need for highly improbable design situations only be considered if it is of significance, e.g. for the leak-tightness of structures or for stability cases where second order effects are significant. In other cases it doesn't need to be considered, provided that the ductility and rotation capacity of the structural members are sufficient.

5. Design of the reactor containment

5.1 General

This section focuses on the design and analysis of the pressure retaining boundary of the reactor containment consisting primarily of a base mat, a cylindrical wall, and a roof slab and/or a dome. Also included are structural members forming the pressure barrier between the primary and secondary compartment in BWR units for maintaining the pressure suppression function, provided that the members consist of a load-carrying concrete structure, with or without a non-structural steel liner. The design provisions refer to reactor containments of reinforced concrete with or without prestressed reinforcement, and with an internal steel liner. For prestressed structures, only post-tensioning reinforcement with tendons are covered.

In this section, the leak-tightness of the containment is covered, for the parts of the containment liner which are backed by the structural concrete member, see the example in Figure 2.1. Steel structures that constitute the containment function but are unsupported by concrete are not covered by DNB.

Design of the reactor containment follows the layout with limit states and design situations in accordance with the Eurocodes. This provides a consistent system of actions, load factors, load reduction factors and combinations of actions in accordance with the Eurocodes for the containment as well as other buildings.

The reactor containment shall according to the conditions described in Section 1, meet the requirements of conventional buildings in accordance with the Eurocodes for the serviceability limit state as well as the ultimate limit state. In the serviceability limit state, additional requirements are introduced to assure that the future containment function during an accidental event is not compromised, or that its operational life time is not shortened, because of loads during normal use, such as for example, the initial pressure test, and periodic leak-tightness tests. These additional requirements are based on ASME Sect III Div 2 [6].

For combinations of actions in the ultimate limit state, supplementary requirements regarding the resistance capacity of the containment are referred to. The supplementary requirements for persistent, transient and accidental design situations are based on the ASME Sect III Div 2 [6], while requirements based on the Eurocodes are used for highly improbable design situations. This is due to ASME Sect III Div 2 [6] which does not cover very improbable design situations.

The design rules for the concrete structure of the containment above imply that verification for effects of actions or other effects in many cases must be carried out according to both the Eurocodes and ASME Sect III Div 2 [6]. In this situation the regulation that prescribes the most conservative design solution is used.

Leak-tightness requirements for the containment follow, for all limit states, the provisions given in ASME Sect III Div 2 [6]. The Eurocodes have no applicable provisions for this requirement. For highly improbable design situations additional acceptance criteria have been introduced.

The detailing follows the provisions in the Eurocodes. Furthermore, it shall be ensured that the requirements of ASME Sect III Div 2 [6] are met. Materials are chosen according to the Eurocodes with some additional requirements according to ASME Sect III Div 2 [6].

Modifications and amendments have been introduced in both the Eurocodes and ASME Sect III Div 2 [6].

Section 5.3 provides general provisions regarding the use of SS-EN 1992-1-1 [29] and ASME Sect III Div 2 [6] for the design of the containment, while sections 5.8 and 5.9 provide detailed design provisions for each of these two regulations. The modifications and amendments that have been introduced are specified here.

Requirements to comply with the design provisions in the serviceability limit state and the different ultimate limit states given in section 5.1, are presented in sections 5.4 – 5.7.

Detailing and materials are discussed in sections 5.10 and 5.11 respectively.

General rules according to SS-EN 1990 [20] with the modifications and amendments described in Section 3 shall be met, in order for the provisions in this section to be applicable. Furthermore, actions, combinations of actions and load factors listed in Section 4 shall be applied in design and analysis.

Actions and analyses regarding earthquake are covered in Section 7.

A comparison of requirements specifications are listed in section 5.2.

SS-EN 1992-1-1 [29] Section 11 and 12 are omitted. These sections discuss lightweight aggregate concrete structures as well as plain and lightly reinforced concrete structures. These types of structures should not be used for reactor containment pressure vessels in nuclear power plants. Also Section 10 in SS-EN 1992-1-1 [29] is omitted. Precast concrete elements are not covered by DNB.

Requirements for fire resistance are not considered.

A schematic summary of the design provisions is given in Figure 5.1.

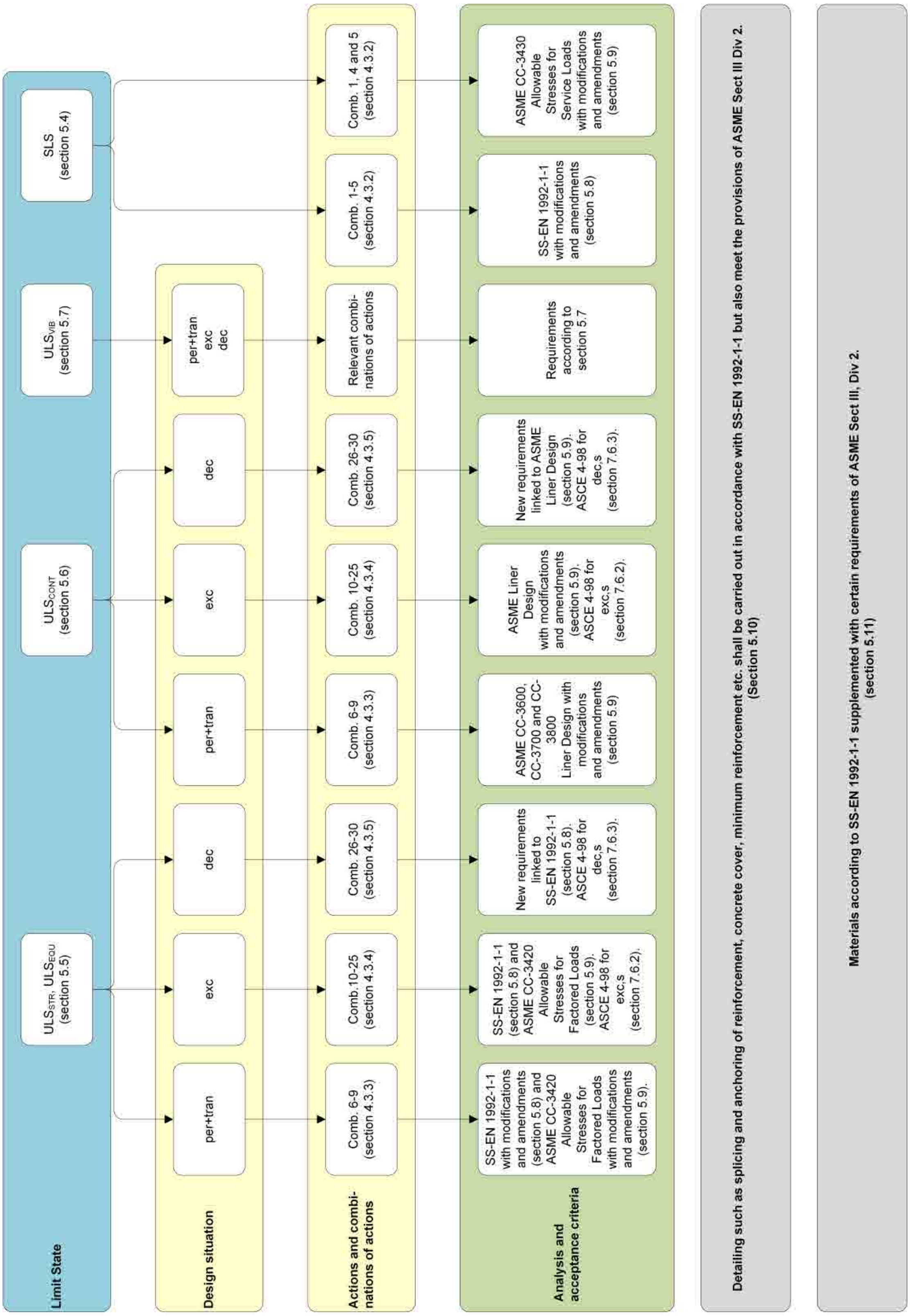

Figure 5.1 – Summary of design provisions for the containment.

5.2 Comparison of the requirements specification

The design provisions presented in DNB comply with the structural requirements of ASME Sect III Div 2 [6] Section CC-3000 Design, apart from some exceptions. These exceptions are presented in Table 5.1.

Table 5.1 - Identified deviations where the structural requirements of ASME Sect III Div 2 Section CC-3000 are not met.

Nbr	Description
1	The following combinations of actions in ASME Sect III Div 2 table CC 3230-1 are not considered: - All combinations of actions containing E_o (Operating Basis Earthquake) - Combination of actions Abnormal with $1.25R_a$ (pipe support reaction forces caused by thermal accidental load) - Combination of actions Abnormal with simultaneously acting $1.25G$ (safety relief valve blow-down) and 1.25 P_a (design-basis accident pressure) - Combination of actions Abnormal/Severe environmental with simultaneously acting $1.25W$ (designing wind load) and 1.25 P_a (design-basis accident pressure) - Combination of actions Abnormal/Extreme environmental with simultaneously acting design-basis accident pressure and SSE-earthquake - Due to the fact that the load reduction factors according to the Eurocodes for certain loads are determined case by case, for these imposed loads (L, live loads, in ASME Sect III Div 2) the combined load factor can be less than 1.3 which is prescribed in ASME Sect III Div 2.
2	Materials for concrete, reinforcement and prestressing tendon, as well as details of reinforcement and prestressing tendon are chosen primarily according to the Eurocodes, with some additional requirements according to ASME Sect III Div 2. This can result in deviations from the structural requirements of ASME Sect III Div 2, e.g. for ductility.
3	Losses of prestress force are calculated according to the provisions in the Eurocodes. The estimated losses may differ from those obtained by calculation according to ASME Sect III Div 2.

If a full compliance with article CC-3000 Design of ASME Sect III Div 2 [6] shall be demonstrated, additional requirements beyond those presented in DNB must be introduced.

Traditionally, three general design requirements are set for prestressed reactor containments in Sweden, see DRB: 2001 [19]:

1. $1.0P_{aL}$: It shall be shown that no resultant axial tension over the cross sections arise. Only local flexural tension can be accepted. Furthermore only local cracking of the concrete containment in section transitions and around penetrations is accepted. The cracks shall in this connection be small and shallow.

2. $1.5P_{aL}$: It shall be shown that the stresses in the steel liner do not reach the yield stress. Furthermore, the structure shall behave elastically, which means that tensions in the tendons and reinforcement shall be in the elastic range.
3. $2.0P_{aL}$: It shall be shown that uncontrolled leakage does not occur (was only applied to some of the existing containments).

The design provisions given in DNB imply some deviations from the traditional requirements. Obviously it is not possible to satisfy the requirement under paragraph 1 above (no resulting axial tension) for containments without tendons. Furthermore, it is not possible to meet the requirement in paragraph 2, that the stress levels in the steel liner shall be less than the yield stress, in the following cases: For a steel liner which is in direct contact with the containment atmosphere, the transient temperature increase due to accidental actions normally causes the compressive stresses in the liner to exceed the yield stress due to restraint forces.

5.3 General design provisions

The main principle for the design of the reinforced concrete structure of the containment is, according to ASME Sect III Div 2 [6], elastic behaviour ("basically elastic") for combination of actions in the serviceability limit state, and that the tensile reinforcement does not reach general yielding for primary loads in combination of actions defined in the ultimate limit state. The requirements specified in the following sections ensure that the above general principle is fulfilled. For highly improbable events, except for seismic loading (see Section 7), yielding of the reinforcement can be accepted provided that the applicable leak-tightness requirements are fulfilled.

In sections 5.4 – 5.7, parts of SS-EN 1992-1-1 [29] and ASME Sect III Div 2 [6] are referred to as requirements in both the serviceability limit state and the ultimate limit state.

In cases where requirements refer to SS-EN 1992-1-1 [29], the modifications and amendments given in section 5.8 shall be considered.

In cases where requirements refer to sections of ASME Sect III Div 2 [6], both the referred sections as well as the sections of the ASME Sect III Div 2 [6] defined in section 5.9 shall be considered with the given modifications and amendments.

5.4 Requirements in the serviceability limit state

For combinations of actions in section 4.3.2 (SLS-ch, SLS-freq and SLS-qp) requirements in the serviceability limit state according to SS-EN 1992-1-1 [29] with modifications and amendments according to section 5.8 shall be met. Furthermore, for the same combinations of actions subsubarticle CC-3430 Allowable Stresses for Service Loads in ASME Sect III Div 2 [6], with modifications and amendments according to section 5.9 shall be met. Finally, the serviceability criteria established according to Section 3.7.3 and 3.10.4 shall be met.

5.5 Requirements in the ultimate limit state, resistance and stability

5.5.1 Persistent and transient design situations

For verification of resistance regarding persistent and transient design situations (ULS_{STR}-per, ULS_{STR}-tran, ULS_{EQU}-per and ULS_{EQU}-tran) with combinations of actions according to section 4.3.3, both SS-EN 1992-1-1 [29], with modifications and amendments according to section

5.8, and subsubarticle CC-3420 Allowable Stresses for Factored Loads of ASME Sect III Div 2 [6], with modifications and amendments according to section 5.9, shall be met.

5.5.2 Accidental design situations

For verification of resistance regarding accidental design situations (ULS_{STR}-exc and ULS_{EQU}-exc) with combinations of actions according to section 4.3.4, SS-EN 1992-1-1 [29] with modifications and amendments according to section 5.8 as well as subsubarticle CC 3430 Allowable Stresses for Factored Loads of ASME Sect III Div 2 [6], with modifications and amendments according to section 5.9, shall be met. For seismic design situations (ULS_{STR}-exc,s and ULS_{EQU}-exc,s) additional requirements according to Section 7 shall be considered.

5.5.3 Highly improbable design situations

For verification of resistance regarding highly improbable design situations (ULS_{STR}-dec, and ULS_{EQU}-dec) with relevant combinations of actions according to section 4.3.5, it is sufficient if SS-EN 1992-1-1 [29] with modifications and amendments according to section 5.8 is met. Other approaches may be applicable. For seismic design situations (ULS_{STR}-dec,s and ULS_{EQU}-dec,s) additional requirements according to Section 7 shall be considered. In addition, it should be shown that sufficient margins are present to avoid “cliff edge”-effects that could lead to unacceptable consequences for the plant as a whole.

Also, in addition to the requirements above, a nonlinear finite element analysis shall be performed to determine the ultimate pressure capacity of the containment, as demanded in USNRC Reg. Guide 1.136 [45] and 1.216 [47]. In this case the effect of the increased temperatures expected during severe accidents shall be considered for cases where the effect cannot be shown to be negligible. Note that the strain criteria defined in USNRC Reg. Guide 1.136 [45] and 1.216 [47] relates entirely to an undisturbed area ("free field"). If more detailed analysis models and evaluation methods are applied other failure criteria may be formulated. Furthermore, other failure modes may need to be considered. The acceptance criteria used shall be consistent with the analytical model and the evaluation methods used, as well as the result that are used. All possible failure modes should be identified and evaluated.

5.6 Requirements in the ultimate limit state, containment function

5.6.1 Persistent and transient design situations

For verification of the leak-tightness requirements regarding persistent and transient design situations (ULS_{CONT}-per and ULS_{CONT}-tran) with combinations of actions according to section 4.3.3, subsubarticles CC-3600 Liner Design Analysis Procedures, CC-3700 Liner Design and CC-3800 Liner Design Details in ASME Sect III Div 2 [6], with modifications and amendments according to section 5.9 shall be met.

5.6.2 Accidental design situations

For verification of the leak-tightness requirements regarding accidental design situations (ULS_{CONT}-exc) with combinations of actions according to section 4.3.4, subsubarticles CC-3600 Liner Design Analysis Procedures, CC-3700 Liner Design and CC-3800 Liner Design Details in ASME Sect III Div 2 [6] with modifications and amendments according to section 5.9, shall be met. Furthermore, the additional requirements for the concrete structure as shown below shall be met.

For prestressed containments, it shall be shown that no resultant axial tension occurs over the cross sections for the number 12 combination of actions. Only local flexural tension can be accepted. Furthermore only local cracking of the concrete containment in section transitions and around penetrations is accepted. The cracks shall in this connection be small and shallow. ΔT_a does not need to be considered when implementing the verifications above.[14]

For seismic design situations (ULS_{CONT}-exc,s) the requirements of Section 7 are added.

5.6.3 Highly improbable design situations

For verification of the leak-tightness requirements regarding highly improbable design situations (ULS_{CONT}-dec) with relevant combinations of actions according to section 4.3.5, it is sufficient if subsubarticles CC-3600 Liner Design Analysis Procedures, CC-3700 Liner Design and CC-3800 Liner Design Details of ASME Sect III div 2 [6] with modifications and amendments according to section 5.9 are met. Other approaches may be applicable. For seismic design situations (ULS_{CONT}-dec,s) additional requirements according to Section 7 shall be considered. In addition, it should be shown that sufficient margins are present to avoid "cliff edge"-effects that could lead to unacceptable consequences for the plant as a whole.

A controlled safety pressure relief of the containment shall be possible as stated in section 3.6.2.

In the same way as for the ultimate pressure capacity (see section 5.5.3), the pressure and temperature conditions shall be established at the maximum leak-tightness capacity of the containment.

5.7 Requirements in the ultimate limit state, deformations and vibrations

It shall be shown that dilatation joints whose function is considered in the design of the building structures are not closed due to the combined effect of deformations and vibrations, see subsubarticle CC-3550 of ASME Sect III Div 2 [6].

Components installed in the building can be sensitive to building deformations, including relative deformations between different anchorage locations. Requirements for limitations of building deformations are specified in site-specific documents.

Verification of installed components in the building due to resulting vibrations is carried out to the extent which is necessary according to the provisions in the SAR. This evaluation is not covered by the DNB. However, Section 7 gives provisions regarding dynamic analysis of earthquake. Section 7 can also provide guidance for the dynamic analysis of other global vibrational loads.

[14] In section 5.2, the three main design requirements, traditionally imposed for prestressed reactor containments in Sweden, are presented in a numbered list. Requirements equivalent to paragraphs 2 and 3 are available through the introduction of the additional requirements of DNB to meet ASME Sect III Div 2 [6] and by the additional requirements of DNB in section **Fel! Hittar inte referenskälla.** regarding the proof of leak-tightness at 2 times P_{aL}. To ensure that new prestressed reactor containments do not have lower level of safety against leakage than the existing containments in Sweden, paragraph 1 of section 5.2 has been introduced as an additional requirement, since it is not covered by other DNB requirements.

5.8 Design based on SS-EN 1992-1-1

When SS-EN 1992-1-1 [29] is referred to in the design provisions, it shall be fully applied with the modifications and amendments described in section 6.6.

5.9 Design based on ASME Sect III Div 2

5.9.1 Introduction

When ASME Sect III Div 2 [6] is referred to in the design provisions, Article CC-3000 Design shall be fully applied with modifications and amendments as described below.

The following sections first provide a brief summary of the current subarticle in ASME Sect III Div 2 [6]. Then, for the concerned subsubarticles, introduced modifications and amendments are presented.

5.9.2 General Design (CC-3100)

General design requirements for the concrete structure as well as for the steel liner are presented in this section. For the steel liner, parts of the pressure retaining boundary backed by the concrete structure is included, the other parts of the steel structure which also serves as a pressure retaining (load carrying) structure are not covered by CC-3000 Design.

Significant terms are presented. The classification of the loads in primary and secondary actions/effects of actions should be noted. This classification, which is compiled in Table CC-3136.6-1, decides which acceptance criteria shall be met.

It shall be ensured that the construction tolerances are considered also in the design.

Subarticle CC-3100 of ASME Sect III Div 2 [6] shall apply with the following modifications and amendments.

5.9.2.1 Definition of terms

Subsubarticle CC-3130, ASME Sect III Div 2 [6]

"Service Load Category" and "Factored Load Category" is replaced with the load classification presented in Section 4.

5.9.3 Load Criteria (CC-3200)

Actions and combinations of actions are presented in this section. Generally the provisions in ASME Sect III Div 2 [6] regarding actions and combinations of actions are replaced with Section 4. The treatment of actions is therefore in accordance with the arrangement that applies to the Eurocodes.

A description of the load categories static and seismic loads, impulse loads and impact effects is given.

Subarticle CC-3200 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.3.1 General

Subsubarticle CC-3210, ASME Sect III Div 2 [6]

This section is fully replaced with the applicable sections of Section 4.

5.9.3.2 Load Categories

Subsubarticle CC-3220, ASME Sect III Div 2 [6]

This section is fully replaced with the applicable sections of Section 4.

5.9.3.3 Load Combinations

Subsubarticle CC-3230, ASME Sect III Div 2 [6]

This section is fully replaced with the applicable sections of Section 4.

5.9.4 Containment Design Analysis Procedures (CC-3300)

Analysis methods applicable to the reactor containment pressure vessel are reported in this section.

Subarticle CC-3300 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.4.1 Shells

Subsubarticle CC-3320, ASME Sect III Div 2 [6]

It is not allowed to replace design calculations of the containment with testing (“model tests”). However, testing might be a possible or necessary complement to calculations.

5.9.5 Concrete Containment Structural Design Allowables (CC-3400)

Acceptance criteria for the load categories used in ASME Sect III Div 2 [6] ("factored loads" and "service loads") are presented in this section. Which combinations of actions according to Section 4 to be classified as "factored" and "service" respectively is given in Section 5.4 – 5.6.

The main principle for the design of the reactor containment pressure vessel is basically elastic behaviour for "service loads", and that general yielding does not occur in the tensile reinforcement for combinations of actions associated with primary "factored loads."[15].

Acceptance criteria for stresses in the concrete, reinforcement and prestressing tendons are specified.

The sections below present how the relevant characteristic strength values according to the Eurocodes translate to strength values to be used in the design equations given in ASME Sect III Div 2 [6].

ASME Sect III Div 2 [6] limits the allowable yield strength of reinforcement to 400 MPa. Via a code case, see below, ASME has introduced rebar grades with higher strength. This code case is applied in the DNB.

Furthermore, the requirements of USNRC Reg. Guide 1.136 [45] have been introduced with regards to tangential shear capacity (shear stress in the shell plane).

Subarticle CC-3400 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

[15] “General yield state”, see subsubarticle CC-3110 of ASME Sect III Div 2 [6].

5.9.5.1 General

Subsubarticle CC-3410, ASME Sect III Div 2 [6]

This section shall be fully applied, except for the reference to actions and combinations of actions, compare revision of CC-3200 in section 5.9.3 above.

5.9.5.2 Allowable stress for factored loads

Subsubarticle CC-3420, ASME Sect III Div 2 [6]

f'_c (Specified Compressive Strength of Concrete according to ASME Sect III Div 2, [6]) can be set equal to f_{ck} (characteristic compressive cylindrical strength of concrete at 28 days) in accordance with SS-EN 1992-1-1 [29] provided that test results show that the concrete also meets the strength requirements in [6][16].

f_y (Specified Tensile Yield Strength of Reinforcing Steel according to ASME Sect III Div 2, [6]) can be set equal to f_{yk} (characteristic yield strength of reinforcement) according to SS-EN 1992-1-1 [29].

f_{py} (Specified Tensile Yield Strength of Prestressing Steel according to ASME Sect III Div 2, [6]) can be set equal to $f_{p0.1k}$ (characteristic 0.1% proof-stress of prestressing steel) according to SS-EN 1992-1-1 [29].

In deviation from what is stated in subsubarticle CC-3422.1, reinforcement with a yield strength up to 500 MPa is allowed in the design for membrane- and bending stresses.[17,18]

The tangential shear strength of the reinforcement with regards to shear stresses in the shell plane shall be limited to not exceed

$0.833\sqrt{f_{ck}}$ MPa.[19]

For prestressed containments the principal tensile stress of the concrete shall not exceed

$0.333\sqrt{f_{ck}}$ MPa.[20]

5.9.5.3 Allowable stresses for service loads

Subsubarticle CC-3430, ASME Sect III Div 2 [6]

See section 5.9.5.2.

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[16] If test results, showing that the strength requirements of ASME Sect III Div 2 [6] are fulfilled, are not available, a cautious approach is to put $f'_c = 0.85 f_{ck}$.

[17] According to Code Case N-807 [7]

[18] Further investigation is required regarding the ductile behaviour of the rebar before tempered reinforcement with a yield strength of 500 MPa can be utilized. This type of reinforcement is very common in Scandinavia (rebar labeled T for thermally produced steel). The ductility shall be demonstrated to be at least as large as for reinforcement permitted by ASME Sect III Div 2 [6].

[19] In accordance with USNRC Reg. Guide 1.136 [45]. This document specifies, among other things, certain additional requirements to ASME Sect III Div 2 [6] for construction of nuclear reactors in the United States. The requirements have been considered applicable also to Swedish conditions.

[20] See previous footnote.

5.9.6 Containment Design Details (CC-3500)

Design methods for normal force, bending moment and shear force are presented in this section for both "service loads" and "factored loads". There are design provisions for anchorage and splicing of reinforcement and prestressing steel, as well as how the loss of prestress is to be calculated. It also covers concrete cover, spacing between reinforcement units and limitation of crack widths.

Finally, provisions regarding requirements for separation of structures, foundation requirements and the handling of the influence of attachments mounted on the outside of the containment are given.

When calculating the loss of prestress, the provisions given in the Eurocodes shall be applied.

Requirements for radial reinforcement, also for prestressed single curved surfaces (i.e. containment cylindrical wall), have been introduced to minimize the adverse effects of any tendency of delamination of the concrete, especially during phases when the prestressing tendons are tensioned or relaxed.

Subarticle CC-3500 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.6.1 Reinforcing steel requirements

Subsubarticle CC-3530, ASME Sect III Div 2 [6]

Development and splice lengths specified in ASME Sect III Div 2 [6] shall be increased by 20 % for reinforcement with yield strength above 420 MPa. [21]

5.9.6.2 Loss of Prestress

Subsubarticle CC-3542, ASME Sect III Div 2 [6]

When calculating the loss of prestress the equations in ASME shall be replaced with corresponding equations in SS-EN 1992-1-1 [29], with the modifications and amendments given in section 6.6.

5.9.6.3 Radial tension reinforcement

Subsubarticle CC-3545, ASME Sect III Div 2 [6]

For portions of a prestressed containment with curvature, radial tension reinforcement (radial ties) shall be provided to resist radial tensile forces from curved tendons. Note that either tensioning or relaxation of the tendons can be the governing load situation.

The distance between the shear reinforcement (radial ties) shall not exceed the smaller of half the shell thickness or 600 mm.

[21] According to Code Case N-807 [7]. The reason for the increase in length is primarily to ensure a corresponding ductility of the structure as for a design with reinforcement with a yield strength less than 420 MPa.

5.9.7 Liner Design Analysis Procedures (CC-3600)

Analysis procedures for the design of the steel liner and its anchorages are presented in this section. Clarifications regarding the analysis procedures have been introduced in this and the next sections.

Moreover, a simplified method to determine an upper limit of the unbalanced forces that can act on the liner anchors has been introduced.

Subarticle CC-3600 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.7.1 Liner

Subsubarticle CC-3620, ASME Sect III Div 2 [6]

The imperfections of the liner as well as its distortion during the construction phase shall be considered in the design of both the steel liner and its anchors.

5.9.7.2 Liner anchors

Subsubarticle CC-3630, ASME Sect III Div 2 [6]

As an alternative to the implementation of biaxial tests to establish an upper limit of unbalanced forces acting on the liner anchors, it is possible to start from the uniaxial yield strength as follows: On the condition that it can be ensured that over-strength steel[22] is not delivered, an equivalent yield strength can be determined as 1.25 times f_{yk} where f_{yk} is the upper characteristic yield strength value (95-percent fractile) according to SS-EN 1993-1-1 [31].

5.9.7.3 Brackets and attachments

Subsubarticle CC-3650, ASME Sect III Div 2 [6]

This section is omitted since it refers to the design of attached steel structures that are not part of the steel liner structure.

5.9.8 Liner Design (CC-3700)

Acceptance criteria for design of the steel liner and its anchorages are presented in this section. A link between the combinations of actions according to Section 4 and the load categories listed in ASME Sect III Div 2 [6] has been introduced.

It is also shown how the strength values to be used in the design equations given in ASME Sect III Div 2 [6] can be determined. Acceptance criteria for highly improbable events have been introduced.

Note that the load factors for design of the steel liner and its anchorages differ from what is used for the concrete structure.

Subarticle CC-3700 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

[22] With the specified factor of 1.25, over-strength steel is defined as steel which has a measured upper yield strength that is no more than 5% higher than the specified upper characteristic yield strength value.

5.9.8.1 Liner

Subsubarticle CC-3720, ASME Sect III Div 2 [6]

The category ”Service” in Table CC-3720-1 corresponds to the serviceability limit state and the ultimate limit state - persistent and transient. The category “Factored” corresponds to the ultimate limit state - accidental.

Footnote (1) in table CC 3720-1 means that the steel liner can be assumed to be stress and strain free before load application in the design of categories Service and Factored, but that the imperfections of the liner as well as its distortion during the construction phase shall be considered as initial imperfections where this is unfavorable. Bending deformations due to application of service and factored loads shall be considered.

f_{py} (specified tensile yield strength of liner steel) is set equal to f_y (yield strength) according to SS-EN 1993-1-1 [31].

Combinations of actions according to Section 4 shall apply. Load factors shall hereby be set to 1.0.

The following capacities can be applied in the ultimate limit state - higly improbable events:

Allowable strain, membrane[23]: $\varepsilon_{sc} = 0.007$; $\varepsilon_{st} = 0.004$
Allowable strain, combined membrane and bending[24]: $\varepsilon_{sc} = 0.018$; $\varepsilon_{st} = 0.012$

5.9.8.2 Liner anchors

Subsubarticle CC-3730, ASME Sect III Div 2 [6]

The category “Test, normal, severe environmental, extreme environmental” in table CC-3730-1 corresponds to combinations of actions 6-9, 18 and 24 in Section 4. The category “Abnormal, abnormal/severe environmental, abnormal/extreme environmental” corresponds to combinations of actions in ultimate limit state - accidental except for combination of actions 18 and 24.

The values of F_y (liner anchor yield force capacity), F_u (liner anchor ultimate force capacity) and δ_u (ultimate displacement capacity for liner anchors) that are included in the determination of the strength of the liner anchors correspond to characteristic strength values according to definition in SS-EN 1990 [20] Section 4.2. Tests may be required in order to determine the values of F_y, F_u and δ_u.

Combinations of actions according to Section 4 shall apply. Load factors shall hereby be set to 1.0.

The capacity regarding normal force and shear force must be checked individually. Furthermore, the combined failure modes for simultaneously acting normal and shear force shall be checked.

The following strength values can be applied to ultimate limit state - highly improbable:

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[23] The values of the allowable acceptance criteria specified in ETC-C [12] for events up to and including improbable events are basically identical with those given in ASME Sect III Div 2 [6]. Further, similar load factors are applied in the two regulation. Therefore the specified values from Table 1.5.1-3 ETC-C [12] are used. The purpose is to ensure the leak-tightness of the containment structure in the case the leak-tightness has to be maintained at highly improbable events.

[24] See footnote 23.

Mechanical loads [25]: $F_a = \min.\{1.0F_y;\ 0.8F_u\}$
Displacement limited loads [26]: $\delta_a = 0.6\delta_u$

5.9.8.3 Penetration assemblies

Subsubarticle CC-3740, ASME Sect III Div 2 [6]

Design of the steel liner adjacent to penetrations shall for mechanical loads follow the design provisions for steel structures at the Swedish nuclear power plants.

For concrete anchors affected by mechanical loads ACI 349 [2][27] shall be applied.

5.9.8.4 Brackets and attachments

Subsubarticle CC-3750, ASME Sect III Div 2 [6]

Design of the steel liner adjacent to brackets and attachments shall for mechanical loads follow the design provisions for steel structures at the Swedish nuclear power plants.

For concrete anchors affected by mechanical loads ACI 349 [2][28] shall be applied.

5.9.9 Liner Design Details (CC-3800)

Details regarding the design of the steel liner and its anchorages are presented in this section.

Subarticle CC-3800 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.9.1 Liner anchors

Subsubarticle CC-3810, ASME Sect III Div 2 [6]

Regarding the requirement of biaxial testing, see section 5.9.7.2.

5.9.10 Design Criteria for Impulse Loadings and Missile Impact (CC-3900)

How to handle impact and impulse actions is presented in this section. ACI 349 [2] has been introduced as a valid reference since ASME Sect III Div 2 [6] does not provide detailed provisions.

Subarticle CC-3900 of ASME Sect III Div 2 [6] applies with the following modifications and amendments.

5.9.10.1 Penetration formulas and impulse and impactive effects

Subsubarticle CC-3923, ASME Sect III Div 2 [6]

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[25] See footnote 23.

[26] See footnote 23.

[27] ASME Sect III Div 2 [6] has no detailed design provisions in this case. Design provisions from ACI 349 [2] has been introduced since this regulation is consistent with ASME Sect III Div 2 [6].

[28] See footnote 27.

To establish the ductility determined at failure, that is used to calculate allowables, testing may be necessary to carry out.

Subsubarticle CC-3931, ASME Sect III Div 2 [6]

When selecting equations for determination of penetration depth, the provisions in ACI 349 [2] Appendix F may be applied[29].

5.10 Detailing

5.10.1 Introduction

The detailing of reinforced concrete structures shall generally be carried out in accordance with SS-EN 1992-1-1 [29], but also meet the provisions of ASME Sect III Div 2 [6]. See further below. For detailing of the steel liner, its anchorages, its adjacency and connection to penetrations, hatches and locks etc. the provisions in ASME Sect III Div 2 [6] shall be met.

5.10.2 Concrete cover and rebar spacing

Concrete cover shall meet the requirements of SS-EN 1992-1-1 [29], but also the provisions given in ASME Sect III Div 2 [6].

Minimum spacing between rebars or tendons shall meet the requirements of SS-EN 1992-1-1 [29] as well as the provisions given in ASME Sect III Div 2 [6].

5.10.3 Minimum reinforcement content and crack control

Minimum reinforcement content shall meet the requirements of SS-EN 1992-1-1 [29] section 7.3.2 and the provisions given in ASME Sect III Div 2 [6].

In SS-EN 1992-1-1 [29], provisions are given regarding maximum allowable crack widths based on durability and appearance requirements. The crack width should however be limited for cracks that occur in the reactor containment pressure vessel and other concrete structures that are designed according to the present section, although no formal requirements are given in the Eurocodes. Maximum allowed crack widths may then be determined from case to case. See also section 6.6.7.2.

5.10.4 Anchorage and splicing of reinforcement bars

Anchorage by bond and lap splices shall meet the requirements of SS-EN 1992-1-1 [29] as well as the provisions given in ASME Sect III Div 2 [6] with the modifications and amendments given in section 5.9.

Mechanical anchorage shall meet the requirements of SS-EN 1992-1-1 [29] as well as the requirements specified in ASME Sect III Div 2 [6].

Mechanical couplers shall meet the requirements of SS-EN 1992-1-1 [29] and the requirements specified in ASME Sect III Div 2 [6]. In addition, the following requirements must be met[30]: In

[29] See footnote 27.

[30] In accordance with US NRC Reg. Guide 1.136 [45]. This document specifies, among other things, certain additional requirements to ASME Sect III Div 2 [6] for construction of nuclear reactors in the United States. The requirements have been considered also applicable to Swedish conditions.

areas where the maximum calculated tensile stress produces a tension force that is greater than or equal to $0.5F_y$ the mechanical couplers in the adjacent reinforcement bars shall be displaced in relation to each other.

Welded splices shall meet the requirements of SS-EN 1992-1-1 [29] and the requirements specified in ASME Sect III Div 2 [6].

5.10.5 Anchorage and lapping of bundles of bars

Anchorage and lapping shall meet the requirements of SS-EN 1992-1-1 [29] as well as the guidance given in ASME Sect III Div 2 [6].

5.10.6 Anchorage and couplers of prestressing tendons

Anchorage and couplers shall meet the requirements of SS-EN 1992-1-1 [29] as well as the guidance given in ASME Sect III Div 2 [6].

5.11 Material qualities and products

Concrete quality, reinforcement and prestressing steel grades as well as reinforcement and prestressing steel details shall meet the requirements of SS-EN 1992-1-1 [29] and BFS 2011: 10 - EKS 8 [8], with the modifications and amendments given in section 5.8. For some materials and products it can be necessary to implement an approval process before they can be accepted for use. Such an approval process can imply that testing must be conducted and evaluated by an accredited testing center. Materials and products that do not meet the above requirements may if necessary be allowed only after special investigation and approval. Furthermore, the requirements of ASME Sect III Div 2 [6] shall be proved to be satisfied to the extent described in previous parts of this section.

Material grades for the steel liner and its anchorages shall comply with the provisions of ASME Sect III Div 2 [6].

6. Design of other buildings

6.1 General

This section refers to the design and analysis of concrete structures in nuclear power plants and other nuclear facilities. Exceptions are made for the reactor containment pressure vessel and the structural elements that constitute the pressure barrier between the primary and secondary compartment in BWR plants. These structural members are covered in Section 5.

The design and analysis of concrete structures shall according to the conditions presented in Section 1, meet regulations and provisions in accordance with SS-EN 1992-1-1 [29] with modifications and amendments specified in this section.

Section 6.6 lists the sections of SS-EN 1992-1-1 [29] referred to together with any modifications and amendments.

General rules according to SS-EN 1990 [20] with the modifications and amendments described in Section 3 shall be met, in order for the provisions in this section to be applicable. Furthermore, loads, combinations of actions and partial factors specified in Section 4 shall be applied in design and analysis.

Actions and analyses regarding earthquake are covered in Section 7.

SS-EN 1992-1-1 [29] Section 11 and 12 are omitted. These sections discuss lightweight aggregate concrete structures as well as plain and lightly reinforced concrete structures. These types of structures should not be used for buildings at nuclear power plants or safety-related buildings at other nuclear facilities[31]. Also Section 10 in SS-EN 1992-1-1 [29] is omitted. Precast concrete elements are not covered by DNB.

Requirements for fire resistance are not treated.

A schematic summary of the design provisions is given in Figure 6.1.

[31] Existing radiation protection, which also has a structural function, may formally be classified as lightly reinforced concrete if the dimensions are determined by the radiation protection requirements and not of resistance requirements.

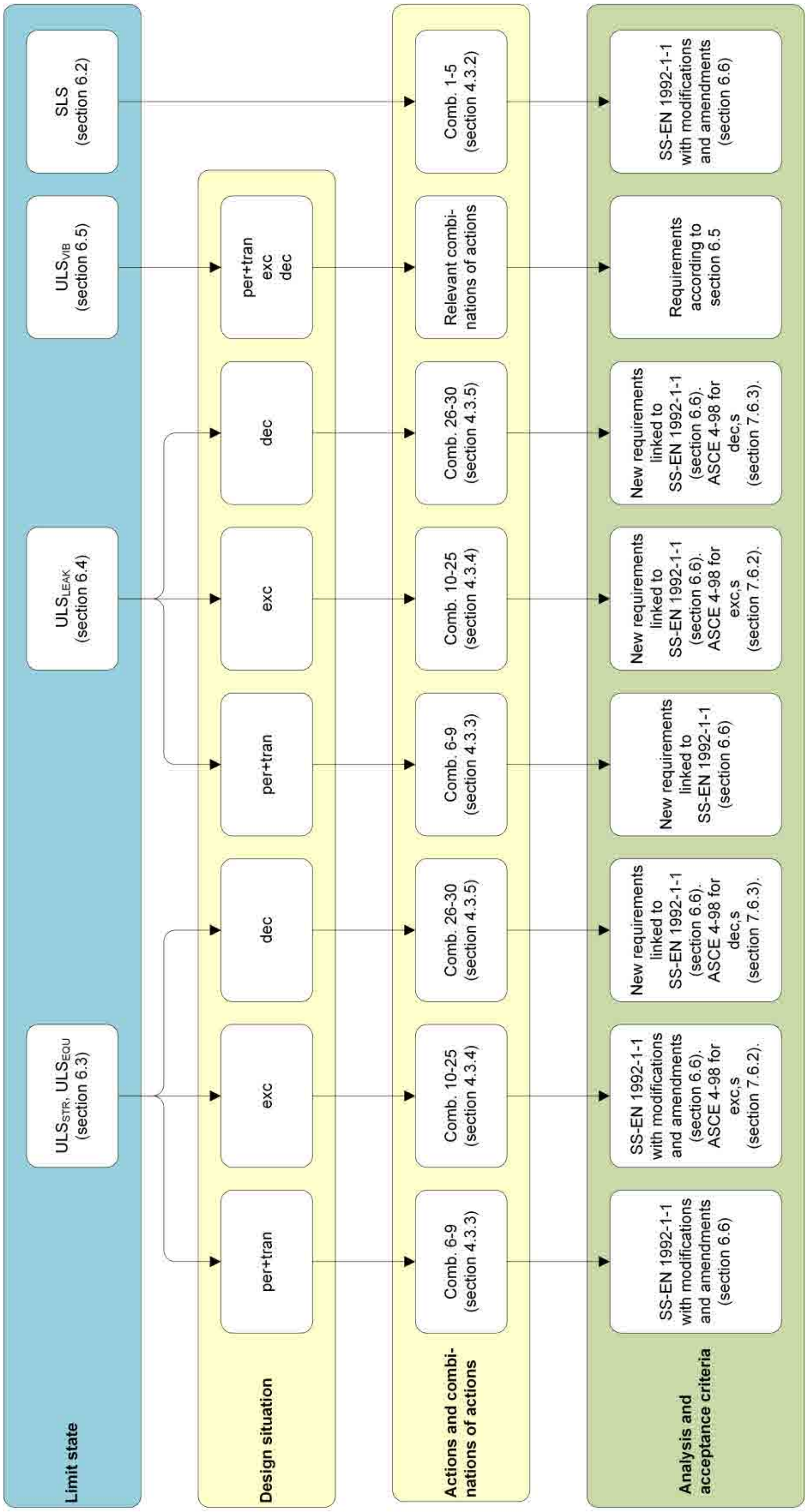

Figure 6.1 – Summary of design provisions for other buildings.

6.2 Requirements in the serviceability limit state

For combinations of actions according to section 4.3.2 (SLS-ch, SLS-freq and SLS-qp) requirements in the serviceability limit state according to SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.6 shall be met. Furthermore, the functional requirements and serviceability criteria established in line with section 3.7.3 and 3.10.4 shall be met, for example regarding requirements of leak-tightness.

6.3 Requirements in the ultimate limit state, resistance and stability

6.3.1 Persistent and transient design situations

For verification of resistance regarding persistent and transient design situations (ULS_{STR}-per, ULS_{STR}-tran, ULS_{EQU}-per and ULS_{EQU}-tran) with combinations of actions according to section 4.3.3, requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.6 shall be met.

6.3.2 Accidental design situations

For verification of resistance regarding accidental design situations (ULS_{STR}-exc and ULS_{EQU}-exc) with combinations of actions according to section 4.3.4, requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.6 shall be met. For seismic design situations (ULS_{STR}-exc,s and ULS_{EQU}-exc,s) additional requirements according to Section 7 shall be considered.

6.3.3 Highly improbable design situations

Actions and effects of actions due to postulated highly improbable events shall, if so specified in the Safety Analysis Report (SAR), be considered for building structures that are forming, protecting or supporting especially important safety functions and mitigating systems.

For verification of resistance regarding highly improbable design situations (ULS_{STR}-dec, and ULS_{EQU}-dec) with relevant combinations of actions according to section 4.3.5, it is sufficient if requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.6 are met. Other approaches may be applicable. For seismic design situations (ULS_{STR}-dec,s and ULS_{EQU}-dec,s) additional requirements according to Section 7 shall be considered. In addition, it should be shown that sufficient margins are present to avoid "cliff edge"-effects that could lead to unacceptable consequences for the plant as a whole.

6.4 Requirements in the ultimate limit state, leak-tightness function

6.4.1 General

For non safety-related structural members SS-EN 1992-3 [30] can be applied. The evaluation is then done in the serviceability limit state, see section 6.2. For safety-related structural members, leak-tightness requirements and acceptance criteria are determined case by case. For safety-related pools with steel liner, the liner should be made of stainless steel and welds in the liner should be provided with a system for drainage and indication of leakage[32]. When designing pool steel liners and its anchors, sections 5.9.7 to 5.9.9 regarding the containment steel liner may be used as guidance. For pools with steel liner, also the underlying concrete structure should be verified to be leak-tight in normal operation[33] (verification of serviceability in accordance with section 6.2). This is to prevent any unallowable leakage from the pool if the steel liner would leak.

6.4.2 Persistent and transient design situations

For verification of leak-tightness regarding persistent and transient design situations (ULS_{LEAK}-per and ULS_{LEAK}-tran) with relevant combinations of actions according to section 4.3.3, requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.4.1 and section 6.6 shall be met.

6.4.3 Accidental design situations

For verification of leak-tightness regarding accidental design situations (ULS_{LEAK}-exc) with relevant combinations of actions according to section 4.3.4, requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.4.1 and section 6.6 shall be met. For seismic design situations (ULS_{LEAK}-exc,s) additional requirements according to Section 7 shall be considered.

6.4.4 Highly improbable design situations

For verification of leak-tightness regarding highly improbable design situations (ULS_{LEAK}-dec) with relevant combinations of actions according to section 4.3.5, it is sufficient if requirements in SS-EN 1992-1-1 [29] with modifications and amendments according to section 6.4.1 and section 6.6 are met. Other approaches may be applicable. For seismic design situations (ULS_{LEAK}-dec,s) additional requirements according to Section 7 shall be considered. In addition, it should be shown that sufficient margins are present to avoid “cliff edge”-effects that could lead to unacceptable consequences for the plant as a whole.

6.5 Requirements in the ultimate limit state, deformations and vibrations

It shall be verified that dilatation joints whose function is considered in the design of the building structures are not closed due to the combined effect of deformations and vibrations.

Components installed in the building can be sensitive to building deformations, including relative deformations between different anchorage locations. Requirements for limitations of building deformations are specified in site-specific documents.

Verification of installed components in the building due to resulting vibrations is carried out to the extent which is necessary according to the provisions given in the Safety Analysis Report (SAR). This evaluation is not covered by DNB. However, Section 7 gives provisions regarding dynamic analysis of earthquake, which can also provide guidance for dynamic analysis of other global vibrational loads.

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[32] See e. g. YVL E.6 [37]

[33] As guidance, the requirements specified in SS-EN 1992-3 [30] for leak-tightness class 1 can be used.

6.6 Design based on SS-EN 1992-1-1

6.6.1 General

When SS-EN 1992-1-1 [29] is referred to in the design provisions, it shall be fully applied with the modifications and amendments described below.

Note that this section, which presents the design based on SS-EN 1992-1-1 [29], is also referred to in the design of the reactor containment. Modifications and amendments that concern the reactor containment only are specially marked.

6.6.2 General

Section 1 of SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.2.1 Scope

Section 1.1, SS-EN 1992-1-1 [29]

Modified according to sections 2.2 and 2.3.

Plain concrete structures are not included.

SS-EN 1990 [20], SS-EN 1991, SS-EN 1997 [32] and SS-EN 1998 [33] shall be applied to the extent specified in this report.

Lightweight aggregate concrete structures are omitted.

6.6.2.2 Normative references

Section 1.2, SS-EN 1992-1-1 [29]

Modified according to section 2.4.

For reinforcement SS 212540 [40] is referred to. For prestressing steel, SS 212551 [13], SS 212552 [14], SS 212553 [15] and SS 212554 [16] are planned to be referred to when valid editions are published.

6.6.2.3 Assumptions

Section 1.3, SS-EN 1992-1-1 [29]

Modified according to section 2.5.

6.6.2.4 Distinction between principles and application rules

Section 1.4, SS-EN 1992-1-1 [29]

Modified according to section 2.6.

6.6.3 Basis of design

Section 2 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.3.1 Requirements

Section 2.1, SS-EN 1992-1-1 [29]

In addition to the references made to SS-EN 1990 [20], the site-specific requirements of the references listed in section 3.5 are valid. Actions and combinations of actions are referred to Section 4.

6.6.3.2 Principles of limit state design

Section 2.2, SS-EN 1992-1-1 [29]

Modified according to section 3.7.

6.6.3.3 Basic variables

Section 2.3, SS-EN 1992-1-1 [29]

Section 2.3.1: For actions, combinations of actions and associated partial factors, Section 4 is generally referred to.

Section 2.3.1: SS-EN 1991-1-1 [21], SS-EN 1991-1-2 [22], SS-EN 1991-1-3 [23], SS-EN 1991-1-4 [24], SS-EN 1991-1-5 [25], SS-EN 1991-3 [26] and SS-EN 1997 [32] shall be applied to the extent specified in this report.

Section 2.3.1: Temperature effects, the influence of the settlement as well as the effect of shrinkage and creep shall for safety-related buildings, in addition to the serviceability limit state, also be considered in the ultimate limit state of persistent, transient and accidental design situations. For highly improbable design situations and for non safety-related buildings, the influence of the above effects only need to be considered if they are significant, e.g. for leak-tightness or stability of structures where second order effects are of importance. In other cases, the effects need not be considered, provided that ductility and rotation capacity of the structural members are sufficient.

Sections 2.3.1.2 (2), 2.3.1.3 (3) and 2.3.2.2 (2): Utilization of plastic ductility should be limited for buildings, limit states and design situations where primary elastic structural behaviour is assumed. See section 6.6.6.1.

6.6.3.4 Verification by the partial factor method

Section 2.4, SS-EN 1992-1-1 [29]

Modified according to section 3.10.

Section 2.4.2.4 (1): For highly improbable design situations, see section 3.7.2, the partial factor for material is set to the same values as for accidental design situations (i.e., 1.2 for concrete and 1.0 for reinforcing and prestressing steel).

Section 2.4.2: Load factors for shrinkage and prestress (prestressing force) is chosen according to Section 4.

Section 2.4.2: Lower value of γ_c and γ_s shall not be used.

Section 2.4.3: Actions and combinations of actions are selected according to Section 4.

6.6.3.5 Supplementary requirements for foundations

Section 2.6, SS-EN 1992-1-1 [29]

Note 2 is omitted for verification of the reactor containment because the paragraph is contrary to what is stated in the corresponding section in ASME Sect III Div 2 [6].

6.6.3.6 Requirements for fastenings

Section 2.7, SS-EN 1992-1-1 [29]

For fasteners in concrete CEN/TS 1992-4.1 [9], CEN/TS 1992-4.2 [10] and CEN/TS 1992-4.4 [11] are applicable, with the modifications and amendments specified in Annex 6. Bonded anchors and concrete screws are not allowed for safety-related concrete structures at nuclear facilities[34].

6.6.4 Materials

Section 3 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.4.1 Concrete

Section 3.1, SS-EN 1992-1-1 [29]

Section 3.1.1 (2): Lightweight aggregate concrete is omitted, see section 6.1.

Section 3.1.2: Minimum permitted strength class for other buildings should not be chosen less than C25/30. For the reactor containment the permitted strength class should be at least C40/50, but not above C70/85.

Section 3.1.2 (6): Utilization of higher compressive strength later than 28 days will primarily be determined from tests conducted at the age in question. Guidance is given in annex 7. Equation 3.1 shall first be applied after special consideration, where the influential environmental aspects and uncertainty in the time-dependent extrapolation is considered.

Section 3.1.2 (9): Utilization of increased tensile strength later than 28 days will primarily be determined from tests conducted at the age in question. Guidance is given in annex 7. Equation 3.4 shall first be applied after special consideration, where the influential environmental aspects and uncertainty in the time-dependent extrapolation is considered.

Section 3.1.4: Section regarding creep and shrinkage is applied unless other assumption is shown to be more correct.

6.6.4.2 Reinforcing steel

Section 3.2, SS-EN 1992-1-1 [29]

For reinforcement SS 212540 [40] is referred to.

Section 3.2.2 (3): Maximum allowed yield strength is f_{yk} = 500 MPa.

Section 3.2.4: The ductility of the reinforcing steel shall at least meet Class B with minimum elongation at failure and ductility ratio $(f_t/f_y)_k$ as defined in Annex C. For structural members which are designed for dynamic actions other than seismic effects, where a ductile structural behaviour is taken into account, reinforcement with higher ductility may have to be utilized. Impact and impulse actions of significant magnitude is an example of when the use of rein-

[34] In accordance with ACI 349 [2].

forcement with high ductility should be investigated. For reactor containments reinforcement of Class C should be used.

The rebar diameter should be limited to a maximum of 40 mm. In case of introduction of rebars with diameter greater than 40 mm, special investigations and tests should be carried out including studies of the bond between rebar and concrete, the concrete cracking with associated crack widths, reinforcement splicing as well as anchoring of the reinforcement.

Section 3.2.7: The design value of the strain limit ε_{ud} is limited to the lesser of ε_{uk} -0.02 and $0.9\varepsilon_{uk}$ for hot rolled steel. For highly improbable design situation (dec) the highest of $\varepsilon_{ud} = 0.9\varepsilon_{uk}$ and ε_{uk} -0.02 is allowed.

6.6.4.3 Prestressing steel

Section 3.3, SS-EN 1992-1-1 [29]

For prestressing steel, SS 212551 [13], SS 212552 [14], SS 212553 [15] and SS 212554 [16] are planned to be referred to when valid editions are published.

Section 3.3.2 (4): Wire or strand with low relaxation (class 2) shall be used.

Section 3.3.2 (6): For the prestressing tendons of the reactor containment, it is required according to Subarticle CC-2424 of ASME Sect III Div 2 [6] that the relaxation properties are determined by testing.

Section 3.3.2 (9): The temperature has a large proven effect on the relaxation losses. If the average temperature of the steel over time is expected to exceed 35°C, the relaxation losses should be particularly investigated.

6.6.5 Durability and cover to reinforcement

Section 4 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.5.1 General

Section 4.1, SS-EN 1992-1-1 [29]

In addition to what is described in section 4.1 (5) of SS-EN 1992-1-1 [29], there may be additional site-specific requirements when anchors are to be made of corrosion resistant material.

6.6.5.2 Concrete cover

Section 4.4.1, SS-EN 1992-1-1 [29]

Concrete cover for prestressing ducts shall for the containment be increased to c_{nom} = 100 mm.

6.6.6 Structural analysis

Section 5 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.6.1 General

Section 5.1, SS-EN 1992-1-1 [29]

Section 5.1.1 (7): For safety-related building structures, an essentially elastic structural behaviour is desirable in the serviceability limit state as well as the ultimate limit state for the design situations persistent/transient and accidental, for loading that do not include impact or impulse actions. Only limited plastic redistribution in determining the section force distribution can therefore be accepted, if nothing else is specifically shown to be acceptable. Note however that such redistribution is not allowed for the reactor containment.

For all seismic design situations only linear elastic idealisation of the structure should be applied. For pools, tanks etc. with safety-related leak-tightness requirements (ULS_{LEAK}) where the leak-tightness is primarily maintained by a steel liner on the inside, a linear elastic idealisation is recommended for the design situations persistent/transient and accidental. If limited plastic redistribution is still taken into account, it should be ensured that the reinforcement closest to the steel liner does not yield for cases where the steel liner is loaded in tension. Corresponding recommendations as for pools with steel liner above can also be used when limitation of the crack width is critical to demonstrate that the leak-tightness requirements of the structure is fulfilled.

However, for temperature loads and other types of restraint forces, concrete cracking may need to be taken into consideration, see section 3.9.1.2.

Recommendations in DNB regarding structural analysis are summarized in Annex 8.

Section 5.1.3: Actions and combinations of actions are referred to Section 4.

6.6.6.2 Linear elastic analysis with limited redistribution

Section 5.5, SS-EN 1992-1-1 [29]

Some restrictions have been introduced for the application of this section, see section 6.6.6.1.

6.6.6.3 Plastic analysis

Section 5.6, SS-EN 1992-1-1 [29]

Some restrictions have been introduced for the application of this section, see Section 6.6.6.1.

6.6.6.4 Prestressed members and structures

Section 5.10, SS-EN 1992-1-1 [29]

Standard value for $\Delta\sigma_{p,ULS}$ is not applicable.

6.6.7 Serviceability limit states (SLS)

Section 6 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.7.1 Stress limitation

Section 7.2, SS-EN 1992-1-1 [29]

Stresses in reinforcement and prestressing steel shall not exceed the recommended values.

6.6.7.2 Crack control

Section 7.3, SS-EN 1992-1-1 [29]

For the reactor containment the width of the concrete cracks should be limited, regardless if current exposure class requires it or not. This is to prevent that unacceptable strain levels occur locally in the steel liner, and to secure the assumed capacity of the liner anchors (strength capacity and deformation). In this connection acceptable crack widths closest to the steel liner should be determined for the current steel liner configuration. Verifying that maximum allowable crack widths stated above are not exceeded, can be done for AOC 5 with M_t corresponding to pressure and temperature levels for the initial pressure test.

6.6.7.3 Deflection control

Section 7.4, SS-EN 1992-1-1 [29]

7.4.1 (3): Site-specific requirements for deformations shall be applied where appropriate, see Safety Analysis Report (SAR) and Design Specifications for Building (KFB) as well as project specific documents.

6.6.8 Ultimate limit state (ULS)

Section 6 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.8.1 Partially loaded areas

Section 6.7, SS-EN 1992-1-1 [29]

For design of reactor containments and other building structures with important safety-functions, F_{Rdu} is limited to $2.0f_{\mathrm{cd}}A_{\mathrm{c0}}$[35].

6.6.9 Detailing of reinforcement and prestressing tendons - general

Section 8 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

6.6.9.1 General

Section 8.1, SS-EN 1992-1-1 [29]

Section 8.1 (1): According to Section 6.6.6.1, an essentially elastic structural behaviour is required during seismic design situations. Rules specified in Section 8 of SS-EN 1992-1-1 [29] are therefore considered appropriate even for seismic effects.

For other types of dynamic actions, the rules reflected in Section 8 of SS-EN 1992-1-1 [29] may be insufficient. For a structural member which is designed for impact or impulse actions, where ductile structural behaviour is accounted for, layout and splicing arrangement as well as anchorage of reinforcement be especially investigated. It is further recommended that the member, if possible, is double reinforced with the same amount of reinforcement on both sides and that the reinforcement is not curtailed.

Section 8.1 (3): Lightweight aggregate concrete is not applicable, see section 6.1.

6.6.10 Detailing of members and particular rules

Section 9 in SS-EN 1992-1-1 [29] is applied with the following modifications and amendments.

[35] In accordance with ACI 349 [2] and ASME Sect III Div 2 [6].

6.6.10.1 Foundation

Section 9.8, SS-EN 1992-1-1 [29]

Section 9.8.2.1 (2): Plain concrete is not applicable, see section 6.1. Circular footings are fully reinforced, see provisions in SS-EN 1992-1-1 [29] Section 8 and 9.3.

6.6.11 Additional rules for precast concrete elements and structures

Section 10 in SS-EN 1992-1-1 [29]. Precast concrete elements are not included in DNB.

6.6.12 Lightweight aggregate concrete structures

Section 11 in SS-EN 1992-1-1 [29]. Not applicable.

6.6.13 Plain and lightly reinforced concrete structures

Section 12 in SS-EN 1992-1-1 [29]. Not applicable.

6.6.14 Annexes in SS-EN 1992-1-1

6.6.14.1 Modification of partial factors for materials

Annex A in SS-EN 1992-1-1 [29], not applicable.

6.6.14.2 Creep and shrinkage strain

Annex B in SS-EN 1992-1-1 [29], applicable.

6.6.14.3 Reinforcement properties

Annex C in SS-EN 1992-1-1 [29], fully applied.

6.6.14.4 Detailed calculation method for prestressing steel relaxation losses

Annex D in SS-EN 1992-1-1 [29], applicable.

6.6.14.5 Indicative Strength Classes for durability

Annex E in SS-EN 1992-1-1 [29], not applicable according to EKS [8].

6.6.14.6 Reinforcement expressions for in-plane stress conditions

Annex F in SS-EN 1992-1-1 [29], applicable.

6.6.14.7 Soil structure interaction

Annex G in SS-EN 1992-1-1 [29], applicable.

6.6.14.8 Global second order effects in structures

Annex H in SS-EN 1992-1-1 [29], applicable.

6.6.14.9 Analysis of flat slabs and shear walls

Annex I in SS-EN 1992-1-1 [29], applicable.

6.6.14.10 Examples of regions with discontinuity in geometry or action

Annex J in SS-EN 1992-1-1 [29], applicable.

7. Seismic design

7.1 General

A schematic summary of the design provisions as regards earthquake resistance is given in **Figure 7.1**.

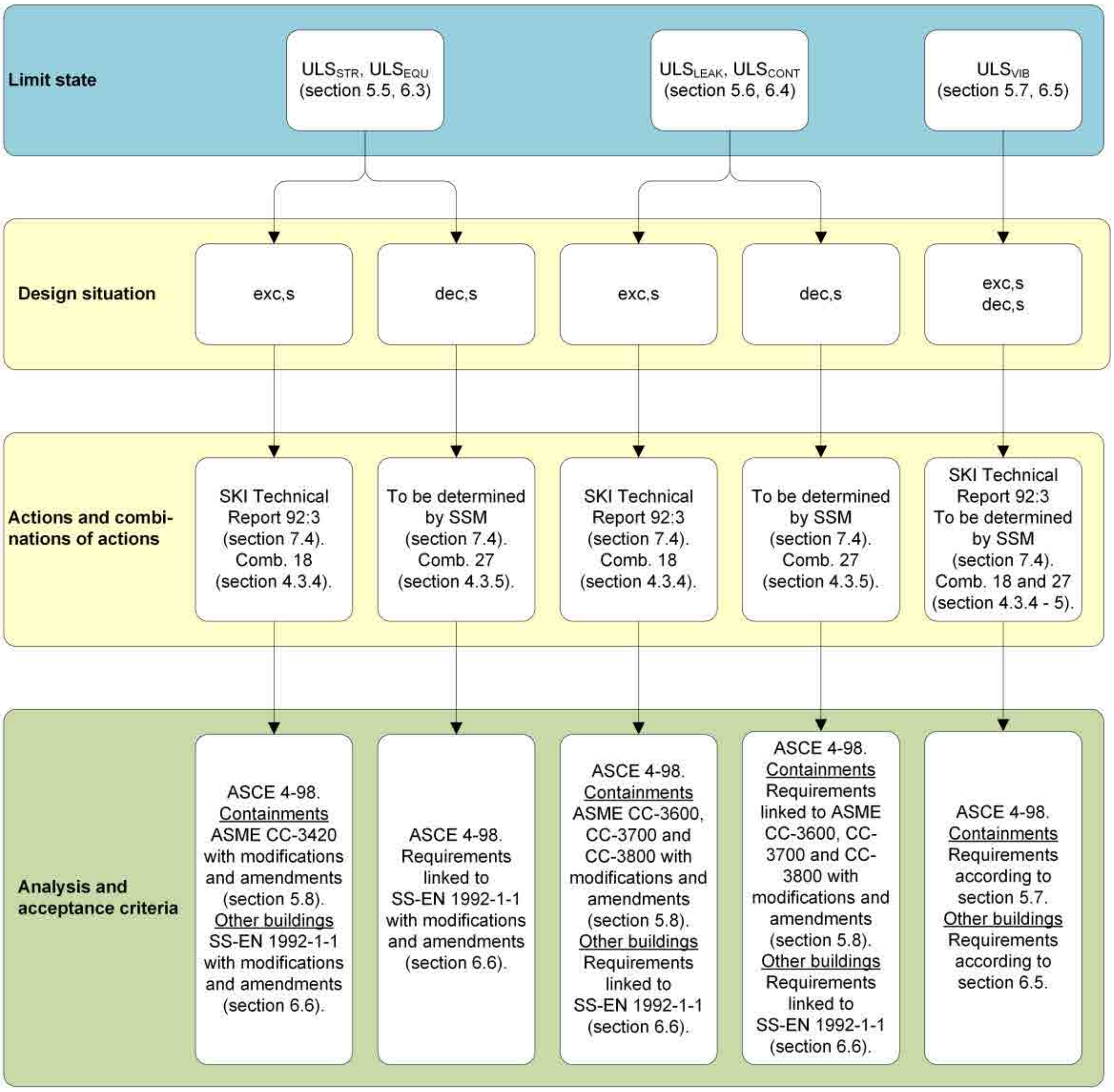

Figure 7.1 – Schematic summary of the design provisions regarding earthquake resistance.

Overall design principles and general requirements regarding seismic design are reported in section 7.2 and 7.3 respectively. Requirements for seismic input are reported in section 7.4 and for seismic analysis in section 7.5.

The approach for seismic safety evaluation is described in section 7.6, while the detailed design provisions for reactor containments and other buildings follow what is stated in Section 5 and Section 6 respectively.

7.2 General design principles

7.2.1 Introduction

The seismic activity in Scandinavia is low. Historically there are only a few registered events, which might have caused damage to an industrial facility. Thus, earthquake effects on buildings in Sweden have been regarded as negligible compared to other actions to be expected during the lifetime of a building. Accordingly, the design criteria for the oldest Swedish nuclear power facilities in the beginning of the 1970s did not include any requirements on structural integrity or maintaining safety functions due to earthquake ground motions.

Concurrently with a growing safety consciousness in the nuclear power industry, there was an increased understanding that seismic effects must be regarded for the Swedish nuclear facilities. Thus, the design criteria for the latest Swedish NPPs, Forsmark unit 3 and Oskarshamn unit 3, designed at the end of the 1970s included consideration of seismic actions. Due to lack of statistical data regarding larger earthquakes in Scandinavia, design response spectra anchored at 0.15 g PGA horizontally and 0.10 g vertically according to USNRC RG 1.60 [42] were applied.

With the purpose of deriving ground motions to be used in the safety analysis of the Swedish nuclear power facilities, a joint project was initiated in the mid 1980s between the then Swedish nuclear safety authority (SKI) and the Swedish nuclear power licensees. The project outcome is presented in SKI Technical Report 92:3 [35]. In this report, envelope ground response spectra corresponding to a certain annual probability of exceedance (10^{-5}, 10^{-6} and 10^{-7}) are defined for typical Swedish hard rock sites.

In SKIFS 2004:2 (later updated to SSMFS 2008:17 [39]) which came into force 2005, earthquake was mentioned as one of several natural events that the Swedish nuclear power plants must verify resistance to.

Within the framework of the executed modernization programs at the Swedish nuclear facilities during recent years, extensive modifications have been carried out in order to maintain necessary safety functions against the effects of an earthquake. Thereby, the facilities have been analyzed for a Safe Shutdown Earthquake corresponding to an annual exceedance probability of once in 100 000 years (10^{-5}) according to SKI Technical Report 92:3 [35].

7.2.2 Applicable standards for seismic design

7.2.2.1 Eurocode 8

SS-EN 1998 [33] applies to design and construction of buildings and civil engineering works in seismic regions in Europe. The main purpose with this standard is according to SS-EN 1998 [33], Section 1.1.1 to protect human lives, to limit damages and to secure that structures important for civil protection remain operational.

It must be observed that SS-EN 1998 [33] only includes complementary requirements in addition to the requirements of other relevant Eurocodes, to be applied for the design of structures in seismic regions. In this respect, SS-EN 1998 [33] is a complement to the other Eurocodes.

An important limitation with SS-EN 1998 [33] is that it, as for other parts of the Eurocodes, formally does not apply to nuclear power plants, offshore structures and large dam structures.

According to the Swedish National Annex to SS-EN 1998 [33], the Swedish National Board of Housing, Building and Planning (Boverket) has not found it necessary to issue any regulations or recommendations with regard to seismic actions, since SS-EN 1998 [33] would only be used in very specific cases where special expertise is required. The Swedish Transport Administration (Trafikverket) specifies as well its position in the Swedish National Annex to SS-EN 1998 [33]. Trafikverket states that seismic actions do not need to be considered in Sweden, since the other parts of the Eurocodes normally ensure the strength and durability of the structure for those earthquake hazard levels that could arise in Sweden.

According to SS-EN 1998 [33], structures in seismic regions are recommended to be constructed to withstand a design seismic action associated with a reference probability of exceedance of 10% in 50 years and with a reference return period of 475 years. With such a short return period, the design ground acceleration, $a_{g,}$ for Swedish conditions should probably result in Sweden being defined as a low seismicity case or a very low seismicity case according to the nomenclature in SS-EN-1998 [33]. Accordingly, the effects of the seismic actions should be covered by the conventional actions as for instance the wind action. Thereof, the recommendations from Boverket and Trafikverket as above, i.e. that SS-EN 1998 [33] normally does not need to be applied since other parts of the Eurocodes, are sufficient to ensure the capacity of the structures.

However, the safety conditions for safety-related structures at NPP sites in Sweden differ from the conditions for bridges, conventional structures and industrial facilities. Safety-related structures at nuclear facilities house important safety systems, which in case of failure could result in severe and unacceptable consequences for the personnel, the off-site public or the environment. Hence, safety-related structures at nuclear facilities shall be designed against external and internal hazards with much lower annual exceedance probability than conventional structures and facilities. SSM has also in SSMFS 2008:17 [39] specifically mentioned earthquake as one of several natural events that the Swedish nuclear power plants must verify resistance to.

The basis of the design philosophy in SS-EN-1998 [33] is to ensure adequate ductility to dissipate energy during the dynamic non-linear material response. This ductility is ensured by reinforcement detailing requirements. However, SS-EN-1998 [33], Section 5.2.1(2) recommends concrete buildings to be designed for low dissipation capacity and low ductility in cases of low seismicity regions (as in Sweden). This means that building structures can be designed for the seismic design situations in principle with the same methods as for other accidental loading situations according to SS-EN 1992-1-1 [29], without any specific ductile reinforcement arrangement.

To summarize, it can be concluded that SS-EN-1998 [33] is not formally mandatory in Sweden and there are not any Nationally Determined Parameters for establishing design ground response spectra. In addition, SS-EN-1998 [33] is insufficient for verification of the specific safety-related structural requirements at nuclear facilities.

7.2.2.2 ACI 318 and ACI 349

ACI 318 [1] prescribes minimum requirements for all types of ordinary concrete buildings in the U.S. In general, the structural form consists of moment resisting frames designed for an essentially elastic response for all actions and combinations of actions except those associated with strong earthquake motions, then non-linear analysis is accepted. In order to secure that the structural elements can exhibit inelastic behavior during the translational earthquake motions, ACI 318 [1], chapter 21 provides minimum requirements on the reinforcing steel detailing. However, in ACI 318 [1] Section 21.1.1.1 it is explicitly stated that the requirements in Section 21 only need to be fulfilled if the design seismic actions have been determined on the basis of energy dissipation in the non-linear range of response. For regions with low seismic hazard, the requirements in chapter 21 do not apply and the ordinary requirements in the other sections of ACI 318 [1] are considered to provide sufficient strength to the structures.

ACI 349 [2] provides requirements for design of safety-related nuclear concrete structures. The predominant structural form is shear wall and slab construction of general heavy proportions. The structural elements are designed for an elastic structural behavior for all actions (except impact or impulse actions) and combination of actions including those associated with the DBE. The main reason to the choice of structural form and the elastic design principle is of course to ensure a robust design with large safety margins.

Even though ACI 349 [2] requires safety-related nuclear structures to be designed essentially elastic to seismic actions, it provides minimum requirements for reinforcing steel detailing according to the requirements of chapter 21 in ACI 318 [1]. Besides maintaining the maximum possible compatibility between ACI 318 [1] and ACI 349 [2], the main reason for this approach is to provide additional assurance that structural integrity is maintained in the unlikely event of an earthquake beyond the design basis event DBE.

7.2.2.3 ASCE 4-98

There are a number of different handbooks covering various aspects regarding modeling and analysis of structural dynamic systems. ASCE 4-98 [4] is a standard which provides minimum requirements and acceptable methods for seismic analysis of safety-related nuclear structures. This standard provides a comprehensive survey of the seismic analysis process, also addressing requirements on input for subsystem seismic analysis. ASCE 4-98 [4] covers in principle all applicable requirements in Regulatory Guides and Standard Review Plans issued by USNRC before 1998, for instance RG 1.61 [43], RG 1.92 [44], SRP 3.7.1 [48] and SRP 3.7.2 [49] and provide more extensive background information to the intentions behind the requirements compared to the official USNRC documents.

7.2.3 The seismic design process

The seismic design process can in general be accomplished in three basic steps:

1. Establish the design earthquake level.
2. For this earthquake level identify those safety functions which need to be maintained.
3. Verify that these safety functions can be maintained during and after the earthquake.

The main principles for these steps are described in following sections.

7.2.3.1 Design Basis Earthquake

The overall safety principle to consider seismic action effects at nuclear power plants is to ensure that those SSCs essential for a safe shutdown of the reactor and to maintain it in a safe condition shall withstand a design earthquake, a so called Safe Shutdown Earthquake (SSE). In order to consider the defense in depth[36], also the reactor containment function and the systems which mitigate the consequences of a severe accident shall fulfil its safety functions in case of an SSE.

With the purpose of including other structures which although not ensuring a safe shutdown of the reactor, do maintain other essential safety functions during an earthquake, the more general definition DBE is used instead of SSE in this document.

[36] ”Defense in depth”, see Section 1 in SSMFS 2008:1 [38].

In Section 2 of SKI Technical Report 92:3 [35], it is stated that certain especially essential safety functions shall be verified against an earthquake with a magnitude beyond the design DBE, i.e. a Design Extension Earthquake (DEE).

7.2.3.2 Identification of required safety functions

Those SSCs which are identified to possess safety functions necessary to be maintained during and after an earthquake, and even SSCs which do not themselves maintain safety functions but for which loss of function could jeopardize the function of a safety-related equipment, shall be designated a Seismic Category. There are three Seismic Categories (1, P and N) depending on type of safety function, as shown in Table 7.1.

Table 7.1 – Seismic design classification for SSCs

Seismic Category	Structures	Piping system	Pump/valve	Electrical components
1	Leak-tightness	Passive function [1)]	Active function	Active function
P	Load-bearing function	Mechanical integrity	Mechanical integrity	-
N	No demand [2)]	No demand [2)]	No demand [2)]	No demand [2)]

[1)] Refers for instance to ensure free flow of water or steam.

[2)] No demand as regards leak-tightness, load-bearing function or mechanical integrity. But SSCs in Seismic Category N shall not jeopardize any safety function of SSCs in Seismic Category 1 or P.

Examples of typical requirements for building elements in respective Seismic Category can be found in Table 7.2.

Table 7.2 - Example of different types of building structures

Seismic Category	Safety function	Requirement (examples)
1	Leak-tightness	Leak-tightness over the steel liner in the containment vessel.
		Leak-tightness over the steel containment lid (BWR).
		Leak-tightness over equipment hatches and openings in the containment vessel.
		Leak-tightness over casing tubes around penetrations in the containment vessel.
		Leak-tightness between the primary and secondary compartment in BWRs.
		Leak-tightness over the steel liner in Spent Fuel Pools (SFP)
		Leak-tightness over building elements for protection against leakage from vessels in waste buildings.
		Leak-tightess of culverts, against leakage from enclosed piping containing radiactive waste in fluid phase.
P	Load-bearing function	Structural integrity of the load-bearing system.
		To provide support and to shield safety systems and components attached to the structural system.
N	No demand	No demand as regards leak-tightness, load-bearing function or mechanical integrity. But SSCs in Seismic Category N shall not jeopardize any safety function of SSCs in Seismic Category 1 or P.

7.2.3.3 Safety verification

Verification of the resistance of SSCs at a nuclear facility against seismic actions can be executed by means of one of following methods or a combination of them:

- Experience based methods
- Testing
- Numerical simulations (dynamic analysis)

The experience-based methods consist mainly of assessments of existing facilities' resistance against actual strong motion earthquakes. These methods can be used on facilities not designed against seismic actions or facilities designed for a certain earthquake hazard level, but where the site needs to be re-assessed for a more severe hazard level. The best known methods are SMA and SPSA.

Testing of components is carried out on shaking tables according to specified routines and for equipment that is difficult to evaluate by other methods. Most commonly, testing is done on electrical instrumentation and control components and devices.

The predominant method for seismic safety verification of building structures is numerical simulations by means of dynamic analyses. In section 7.5, requirements and conditions for this type of verification methods are dealt with.

7.3 Basic requirements

7.3.1 General

Actions on structures arising as a consequence of seismic ground motions are categorized as accidental actions (E_{DBE} and E_{DEE} respectively) in section 4.2.3 and 4.2.4. Design combination of actions with regard to seismic action in combination with other concurrent actions are addressed in section 4.3.4 and 4.3.5.

There are two seismic design situations:

- Accidental seismic design situation (earthquake DBE)
- Highly improbable seismic design situation (earthquake DEE)

These design situations can be categorized as shown in Table 7.3.

Table 7.3 - Categorization of the design situations.

Event	Design situation	Event class	Limit state
Earthquake - DBE	Accidental, seismic	H4	ULS-exc,s
Earthquake - DEE	Highly improbable, seismic	H5	ULS-dec,s

7.3.2 Fundamental requirements

According to what is stated in Section 7.2.2.2, ACI 349 [2] can be applied when designing safety-related buildings at nuclear facilities. One fundamental design principle in ACI 349 [2] is to ensure an elastic response for all combination of actions, including earthquake combination of actions. Chapter 21 in ACI 349 [2] is in principle equal to the corresponding chapter in ACI 318 [1], i.e. providing requirements on reinforcement detailing.

The requirements in chapter 21 in ACI 318 [1] and ACI 349 [2] have common purpose as corresponding requirements in Section 5 of SS-EN-1998 [33], that is to ensure ductility in the reinforcement detailing. However, SS-EN-1998 [33] recommends so-called non-dissipative structures for regions with low seismicity, as for instance Sweden. This means that structures then are designed against seismic actions in the same way as for other loads according to SS-EN 1992-1-1 [29] and that ductility with required complicated reinforcement detailing are not utilized.

In light of what has been described above, it is reasonable to apply a design strategy for buildings at Swedish nuclear facilities as follows.

For reactor containments, the design rules according to ASME Section III, Div 2 [6] apply according to what is stated in Section 5. Thereby, an elastic design of the reactor containment is ensured for seismic actions as well as ductile detailing, which ensure robustness against a severe earthquake beyond the design basis.

Other safety-related structures in Seismic Category 1 and P shall demonstrate an essentially elastic behavior against the DBE. The seismic actions shall be managed conventionally according to the design principles for accidental design situations as defined in SS-EN 1992-1-1 [29] and in Section 6. No ductile reinforcement detailing according to the principles in ACI 349 [2] or SS-EN-1998 [33] are required on condition that linear elastic analysis can verify that no "cliff-edge" effects arise for a severe earthquake beyond the DBE, in accordance with Section 2.39 in IAEA Safety Guide NS-G-1.6 [17].

SS-EN-1998 [33] is not used and is only applied when it is explicitly referenced.

7.4 Seismic input

7.4.1 Design ground response spectra

The Design Basis Earthquake (DBE) for the Swedish nuclear facilities in Forsmark, Oskarshamn and Ringhals is defined as an earthquake with an annual exceedance frequency of 10^{-5}, with ground response spectra for a typical "hard rock site" according to SKI Technical Report 92:3 [35], Appendice 1.

The earthquake magnitude to be applied to structures or structural elements, for which robustness shall be demonstrated for a severe earthquake beyond the design basis (DEE) is determined by the Swedish Radiation Safety Authority (SSM).

In certain cases it is appropriate to generate acceleration-time histories which target the design ground response spectra. Requirements on such artificial time histories are specified in ASCE 4-98 [4], section 2.3 and 2.4.

7.5 Requirements for seismic analysis methods

7.5.1 Requirements for structural modeling

7.5.1.1 Introduction

In contradiction to static loadcases when action values are determined independent of the mathematic model of the structure, the magnitudes of the seismic actions are dependent on the dynamic properties of the applied structural system. This means that the requirements on the structural model and analysis must be more rigorous when dealing with seismic analysis compared to conventional static analysis.

ASCE 4-98 [4] provides much more stringent and robust requirements for structural analysis methods, reflecting the stricter demands for nuclear facilities, compared to what is common practice in standards for conventional buildings, as for instance in SS-EN 1998 [33]. Hence, ASCE 4-98 [4] is in the following used as the main reference as regards requirements on structural modelling and analysis.

7.5.1.2 General requirements

ASCE 4-98 [4], section 3.1.1 provides some general basic requirements on modeling of structures.

7.5.1.3 Material properties

In linear elastic analysis of concrete structures, for calculation of eigenfrequencies as well as for determining sectional forces and moments in the structural elements, mean value of the modulus of elasticity (E_{cm}), according to the principles in SS-EN 1992-1 [29], Section 5.4 can be used. The value of E_{cm} is then calculated according to SS-EN 1992-1 [29], Table 3.1.

Recommended value of the Poisson's ratio (υ) is 0.2 for uncracked concrete and 0 for cracked concrete according to SS-EN 1992-1 [29], section 3.1.3.

For eventual non-linear calculations, the general stress-strain relation according to SS-EN 1992-1 [29] section 3.1.5 can be used.

7.5.1.4 Modeling of stiffness of concrete elements

For determination of the seismic action effects, linear elastic analysis may be carried out under assumption of uncracked concrete cross-sections and mean value of the modulus of elasticity (E_{cm}). That is, the structural model can be based on the nominal geometrical properties of the concrete elements.

However, if a linear elastic analysis indicates extensive cracking in concrete elements, the reduced stiffness must be considered. Qualified engineering assessments are needed to address the stiffness reduction in an updated linear elastic calculation, whereby ASCE 4-98 [4], Section 3.1.3 can provide guidance. An acceptable approach to consider cracked concrete properties can be to reduce the stiffness of the uncracked members by a reduction factor as described in ASCE 43-05 [5], Section 3.4.1.

7.5.1.5 Modeling of mass distribution

The inertial mass properties in the load-bearing structure can be defined directly in the structural model through the geometrical properties of the structural elements and the density of the material. In addition to the structural mass, mass equivalent to a distributed floor load of 250 kg/m^2 could be included, to represent miscellaneous dead weights such as minor equipment, piping and raceways, according to SRP 3.7.2 [49]. The mass of major permanent installed equipment should be distributed over a representative floor area or included as concentrated lumped masses at the equipment locations.

The structural model used for determining the seismic response shall also include the mass of the quasipermanent part of the imposed action ($\psi_2 Q_k$). Guidelines for applicable values on ψ_2 for different variable actions can be found in SS-EN 1990 [20] + EKS [8], Table A1.1 also shown in Table 4.2. Participating part of the mass of the imposed action at floor slabs in nuclear facilities should be determined on a best estimate basis, but not less than 25% ($\psi_2 \geq 0.25$) of the specified design imposed action, in accordance with ASCE 43-05 [5], section 3.4.2.

7.5.1.6 Modeling of damping

Damping represents the structural ability to absorb energy when responding to dynamic loading. Damping is dependent on various factors such as type of connections between the structural elements, type of material and the stress levels during loading.

Seismic action effects are usually calculated by means of modal dynamic analysis metods or direct integration of the dynamic system. Applicable damping values to be used can be found in ASCE 4-98 [4], Table 3.1-1 for various types of material. In ASCE 4-98 [4], section 3.1.2.2, the principles for determining damping values for design and structural evaluation of structures are described, as well as for determining in-structure response spectra to be used for subsystem seismic analysis.

In this connection, it must be observed that USNRC in their latest version of RG 1.61 [43] from March 2007 has revised the damping values applicable when generating in-structure response spectra to safety equipment at low stress levels. Hence, damping values for different structural types can be determined according to Table 7.4. The principles for determining stress level 1 and 2 respectively as described in ASCE 4-98 [4], section 3.1.2.2 are in all essentials compatible with corresponding principles in RG 1.61 [43]. In practice, Stress Level 2 damping values may always be used in seismic design of structures, while Stress Level 1 is most often used in development of in-structure response spectra inside the building.

Table 7.4 - Modal damping ratios according to RG 1.61 [43], with stress level definitions according to ASCE 4-98 [4].

Structure type	Stress level 1	Stress level 2
Reinforced concrete	4 %	7 %
Prestressed concrete	3 %	5 %

7.5.1.7 Modeling of hydrodynamic effects

Hydrodynamic effects of large volumes of water in for instance fuel- and service pools and condensation pools can be considered in accordance with ASCE 4-98 [4], Section 3.5.4. The effects on the dynamical properties (eigenfrequencies) as well as the resulting effects of actions in the walls and floors of the pools and eventual separating walls between different pools shall be considered.

ASCE 4-98 [4], Section 3.1.6 provides acceptable methods for modeling hydrodynamic effects of water in pools. ASCE 4-98 [4], Section 3.1.6.3 includes examples on acceptable methods for determining convective and impulsive effects of water.

7.5.2 Requirements for structural analysis

7.5.2.1 General requirements

The following methods are acceptable to use when performing a seismic response analysis of safety-related structures at nuclear facilities:

1. The time history method
2. The response spectrum method
3. The equivalent static method

Minimum requirements for each method are described in the following.

7.5.2.2 Time history method

Time history analysis can be carried out using linear or non-linear analysis methods.

Modal dynamic time history analysis is the most common linear analysis method. The earthquake is then described in the form of acceleration-time histories. Requirements for the method are described in ASCE 4-98 [4], section 3.2.2.2.1. It must be observed that USNRC does not support ASCE 4-98 [4], section 3.2.2.2.1(f) regarding how many modes that need to be included in the modal superposition. USNRC states that ASCE 4-98 [4], section 3.2.2.2.1(f) is non-conservative and recommend instead to apply RG 1.92 [44] for modal superposition and addressing that part of the mass not excited within the total modal mass (“missing mass”). Hence, ASCE 4-98 [4], section 3.2.2.2.1(f) should to be used with care and if not all mass is included in the analysis, it should to be demonstrated that the effect of missing mass can be considered negligible.

As an alternative to the modal dynamic analysis, the direct integration method can be used, see ASCE 4-98 [4], section 3.2.2.2.2.

In case where geometrical non-linearities, for instance gaps between structural elements, have a significant impact on the response or where material non-linearities as for instance plasticity or friction occur, non-linear time-history methods can be applied. Requirements on these methods are described in ASCE 4-98 [4], section 3.2.2.3.

7.5.2.3 Response spectrum method

The response spectrum method enables calculation of the maximum response in the structure when excited by an earthquake defined in the form of a ground response spectrum. The calculation of maximum values is carried out by combining the maximum responses for the participating modes. In ASCE 4-98 [4], section 3.2.3, requirements on how to apply the response spectrum method are described. As regards the application of ASCE 4-98 [4], section 3.2.2.2.1(f), see Section 7.5.2.2.

7.5.2.4 Equivalent static method

Equivalent static methods for determining seismic action effects in structures are allowed in national standards for simple structures with symmetric and uniform geometry and mass distribution. However, the method is inappropriate for structures with irregular shapes, likewise there are restrictions for the method to be used at nuclear facilities. In general, the primary usage of equivalent static methods is for simple estimates and feasibility assessments of results from more rigorous dynamic analysis for building structures and for component and distribution systems. The requirements on the use of equivalent static methods at nuclear facilities are presented in ASCE 4-98 [4], section 3.2.5.

7.5.2.5 Multiply-support systems

For structures or safety systems supported on different structures or different structural elements within a building, the effect of different input signals must be considered according to ASCE 4-98 [4], section 3.2.6.

7.5.2.6 Combination of modal and component responses

Requirements for how modes and excitation directions shall be combined when the response spectrum method is applied and how the excitation directions shall be considered when the time history method is used are described in ASCE 4-98 [4], section 3.2.7. In this connection, it is important to emphasize the requirement that the three directional components of earthquake motion in a time-history analysis must be statistically independent in order for the three excitation directions to be applied simultaneously to the numerical model in one analysis. If the directions have a statistical dependency, each excitation direction has to be applied separately and the structural response shall be adequately combined as described in ASCE 4-98 [4], section 3.2.7.2.

7.5.2.7 Soil-structure interaction

In contrast to other dynamic actions, the seismic action can be characterized in terms of the ground motion rather than an applied external action. The effective action on the building shall therefore be described in terms of this ground motion. Ground response spectra or alternatively synthetically developed time histories for Swedish nuclear facilities in SKI Technical Report 92:3 [35] describe the ground motion in the free field without any influence from the structure.

Depending on the characteristics of the earthquake, the foundation conditions and the dynamic properties of the structure, the actual motion of the foundation will deviate from the ground motion in the free field. For a light building with a flexible foundation slab founded on rock or on soil with high stiffness, the deviation will be negligible, since the building transfer only a small amount of energy to the environment through the foundation. On the contrary, a heavy building with a relatively stiffer foundation slab founded on softer soil conditions has a greater ability to radiate energy to the environment, causing the ground motion in the foundation slab to differ significantly from the motion in free field.

In case a significant difference can be expected between the motion in free field and the motion under influence from the structure, ASCE 4-98 [4], section 3.3 requires analysis to be performed by considering the interaction between soil and structure, i.e. Soil-Structure Interaction (SSI).

In ASCE 4-98 [4], section 3.3.1, it is required that SSI shall be considered for all structures not founded on rock or rock-like soil foundation material. A fixed-base support may generally be assumed when the structure is supported on rock or rock-like conditions, which approximately correspond to shear wave velocities > 1100 m/s. However, it should be verified that the interaction frequency for a model with a completely stiff structure in combination with discrete springs according to ASCE 4-98 [4], Table 3.3-1 for a circular slab and Table 3.3-3 respectively for a rectangular slab, is at least twice the fixed-base frequency in a model with a flexible structure. If the shear wave velocity > 2400 m/s, a fixed base assumption is accepted without any further verification, according to SRP 3.7.2 [49].

It shall be observed that ASCE 4-98 [4], section 3.3.1.10 regarding reduction of ground response spectra with respect to wave incoherence is not accepted by USNRC.

7.5.3 Requirements for input for subsystem seismic analysis

7.5.3.1 General requirerments

The scope of seismic design of conventional buildings includes primarily calculation of effects of actions and verification of sufficient capacity of the load-bearing structural elements. In addition for safety-related structures at nuclear facilities, the licensees also need to provide input for seismic analysis of safety equipment in the building in the form of in-structure response spectra or in-structure time histories at certain positions in the structure, normally at least at each floor level. In general, in order to provide in-structure response spectra with sufficient accuracy, the numerical model need to have higher geometrical resolution and more dense mesh to catch higher local eigenfrequencies. In ASCE 4-98 [4], section 3.4, requirements for acceptable procedures for generating in-structure response spectra and time history motions are provided.

7.6 Seismic safety verification

7.6.1 General

Basic requirements for seismic design are reported in section 7.3.

Requirements for seismic input in the form of design response spectra are shown in section 7.4.

Requirements for structural modeling and analysis are shown in section 7.5.

Safety verification shall be demonstrated for accidental, seismic and possibly for highly improbable, seismic design situation in relevant limit states according to Table 7.3. Verification for reactor containments is carried out according to ASME Sect III Div 2 [6] with amendments and modifications according to section 5 and for other buildings according to Eurocode 2 [29] with amendments and modifications according to section 6.

Conditions and acceptance criteria for respective design situation are described in section 7.6.2 and 7.6.3.

7.6.2 Accidental, seismic design situation (Earthquake-DBE)

Design of safety-related structures in Seismic Category 1 and P for the earthquake-DBE is carried out in the ULS-exc,s limit state in accordance with the combination of action for accidental seismic design situation in section 4.3.4. The design should ensure elastic structural behavior in accordance with applicable parts of Section 5 for reactor containments and Section 6 for other buildings. In order to meet the "essentially elastic structural behavior" criteria, the idealization in the structural analysis is limited to linear elastic behavior, according to Section 6.6.6.1. Possible significant concrete cracking is considered linear elastically in accordance with the principles for stiffness reduction, as described in section 7.5.1.4.

Considering the absence of ductile reinforcement detailing in the existing Swedish nuclear facilities, it is reasonable to apply the "essential elastic structural behavior" criteria for the existing structures as for new structures. It shall however be emphasized that the "essential structural elastic behavior" as stated in RG 1.208 [46] can be somewhat mitigated for Seismic Category P structures in the sense that localized inelasticity are accepted at stress concentration points, but the overall seismic response shall be essentially elastic.

In addition in SRP 3.7.2 [49], it is stated that for certain special cases (e.g., evaluation of as-built structures), reliance on limited inelastic/nonlinear behavior is acceptable when appropriate.

Safety-related structures in Seismic Category N need not comply with any formal seismic safety requirements. But structures or structural members in Seismic Category N shall not jeopardize SSCs in Seismic Category 1 or P.

7.6.3 Highly improbable seismic design situation (Earthquake-DEE)

For structures and structural members in Seismic Category 1 and P, for which resistance shall be demonstrated for an earthquake beyond the DBE according to section 7.4.1, a verification shall be performed for a DEE in the ULS-dec,s limit state in accordance with the combination of action for highly improbable seismic design situation in section 4.3.5. The design should ensure an elastic structural behavior in accordance with applicable parts of Section 5 for reactor containments and Section 6 for other buildings. For the reactor containment, sufficient resistance is indirectly ensured for an earthquake beyond the DBE, by the design performed in accordance with ASME Sect III Div 2 [6] according to what is stated in section 7.3.2. Thus, verification for an Earthquake-DEE is not necessary, as long as the seismic margin need not to be quantified.

For existing structures in Seismic categtory P for which seismic loads were not considered in the original design, localized inelasticity may be accepted as described in Section 7.6.2.

8. Design related to the construction phase

8.1 General

This section concerns design and analysis of concrete structures at nuclear power plants and other nuclear facilities related to the construction phase.

8.2 Actions and combinations of actions

Actions during execution, Q_c, shall be considered. These may include e.g. actions related to material storage, personnel and equipment, cranes, lifting and transporting, horizontal construction loads, reaction forces from machinery, loads arisen from assembly and fitting structural members, construction waste and casting pressure. Actions during execution are determined according to SS-EN 1991-1-6 [28]. Since the plant is not in operation during the construction phase and by definition thus in outage mode, seismic actions need not be considered. Load reduction factors are determined according to SS-EN 1991-1-6 [28] Annex A1, Section A1.1 (ultimate limit state) and A1.2 (serviceabilty limit state). The concept of actions during execution includes the following actions according to SS-EN 1991-1-6 [28]:

- Q_{ca} Personnel, staff, visitors and hand tools
- Q_{cb} Storage of movable items
- Q_{cc} Non permanent equipment
- Q_{cd} Moveable heavy machinery and equipment
- Q_{ce} Accumulation of waste material
- Q_{cf} Loads from parts of a structure in temporary states

In addition to these actions, horizontal loads according to SS-EN 1991-1-6 [28] Annex A1, Section A1.3 shall be considered.

Combinations of actions for actions that occur during execution are imposed in accordance with the provisions of SS-EN 1991-1-6 [28]. Normally, the following combinations of actions and design situations are considered:

- Serviceability limit state, characteristic combinations of actions
- Serviceability limit state, the quasi-permanent combinations of actions
- Ultimate limit state, transient design situations
- Ultimate limit state, accidental design situations

Combinations of actions to be applied during execution are reported in Table 8.1.

Characteristic values of climate actions such as wind and snow can be determined specifically for the execution phase based on the construction phase duration, see Table 3.1 of SS-EN 1991-1-6 [28].

Table 8.1 – Combinations of actions in the construction phase.

Action		Combinations of actions			
	applies to	serviceability limit state, characteristic	serviceability limit state, quasi-permanent	ultimate limit state, transient design situ-aions	ultimate limit state, accidental design situ-ations
Number		C1	C2	C3	C4
Permanent actions					
Self weight[1)] -unfavourable $D_{k,sup}$ -favourable $D_{k,inf}$		 1.0 1.0	 1.0 1.0	 $\gamma_d \cdot 1.2$ 1.0	 1.0 1.0
Soil movement		1.0	1.0	$\gamma_d \cdot 1.2$	1.0
Earth pressure - unfavourable - favourable		 1.0 1.0	 1.0 1.0	 $\gamma_d \cdot 1.2$ 1.0	 1.0 1.0
Prestressing - unfavourable - favourable		 $1.0\ P_{pk,sup}$ $1.0\ P_{pk,inf}$	 $1.0\ P_{pk,sup}$ $1.0\ P_{pk,inf}$	 $\gamma_d \cdot \gamma_{p,unfav}$ [8)] P_{pm} $1.0\ P_{pm}$	 $\gamma_{p,unfav}$ [8)] P_{pm} $1.0\ P_{pm}$
Pre-deformations[2)]		1.0	1.0	$\gamma_d \cdot 1.2$	1.0
Shrinkage/hydration effects[2)]		1.0	1.0	$\gamma_d \cdot 1.2$	1.0
Actions due to water - unfavourable - favourable		 1.0 1.0	 1.0 1.0	 $\gamma_d \cdot 1.2$ 1.0	 1.0 1.0
Variable actions[3)]					
Construction loads Q_c[4)]		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Earth pressure		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Prestressing		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Pre-deformations		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Shrinkage/hydration effects		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Actions caused by water		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Wind actions		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Snow loads		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Atmospheric icing		$1.0\psi_0$ [7)]	$1.0\psi_2$	$\gamma_d \cdot 1.5\psi_0$ [7)]	$1.0\psi_2$ [5)]
Accidental actions					
Wind actions					1.0[6)]
Snow loads					1.0[6)]
Actions caused by water					1.0[6)]
Acccidents					1.0[6)]
Limit state		SLS	SLS	ULS	ULS
Combination of action /Design situation		Characteristic	Quasi-perm.	Transient	Accidental

[1)] Regarding upper and lower values, see section 4.2.1.

[2)] If the action is favourable, it shall be set to 0.

[3)] Variable loads that are favourable shall be set to 0.

[4)] The concept of construction loads include a variety of actions.

[5] If one of these actions is dominant, ψ_2 shall be replaced by ψ_1 for this action.

[6] Only one of these actions is included at a time.

[7] If one of these actions is the leading action then ψ_0 shall be replaced with 1.0 for this action.

[8] $\gamma_{p,unfav}$ is set to 1.2 for the control of local effects and to 1.3 at risk of instability with external prestress, see SS-EN 1992-1-1, section 2.4.2.2. For the other cases $\gamma_{p,unfav}$ is set to 1.0.

8.3 Requirements during construction phase

During construction phase, combinations of actions in accordance with section 8.2, SS-EN 1991-1-6 [28] shall be met.

Furthermore, it shall be verified that subsubarticle CC-3430 Allowable Stresses for Service Loads in ASME Sect III Div 2 [6] and the requirements applicable to the construction phase in CC-3600 Liner Design Analysis Procedures, CC-3700 Liner Design and CC-3800 Liner Design Details in ASME Sect III Div 2 [6] are met.

In case of structural alteration or reconstruction, the remaining parts of DNB shall, in addition to the above, be met for the parts of the building structure that do not directly represent reconstruction area, but which may be affected by the actions during execution.

9. References

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Annex 1: List of figures

Annex 2: List of tables

Annex 3: Abbreviations

ACI 349	American Concrete Institute, Code Requirements for Nuclear Safety-Related Structural Concrete Structures (ACI 349-06) with Commentary
ACI 318	American Concrete Institute, Building Code Requirements for Structural Concrete (ACI 318-11) with Commentary
ASCE 4-98	American Society of Civil Engineers, ASCE 4-98 Seismic Analysis of Safety-Related Nuclear Structures and Commentary
ASCE 43-05	American Society of Civil Engineers, ASCE/SEI 43-05 Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities
ASME Sect III Div 2	American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code Section III Division 2 Code for Concrete Containments
cont	Containment
DRB DRB:2001	Scanscot Technology AB, Dimensioneringsregler för byggnader (DRB:2001)
BFS 2011:10 – EKS 8	Boverkets föreskrifter och allmänna råd om tillämpning av europeiska konstruktionsstandarder (eurokoder)
BWR	Boiling Water Reactor
ch	characteristic combination of actions in SLS
dec dec,s	Design Extension Condition Design Extension Condition, seismic
DBE	Design Basis Earthquake
DEE	Design Extension Earthquake
EKS	See BFS 2011:10 – EKS 8
env	Environmental
ETC-C	French Association for Design, Construction, and In-Service Inspection Rules for Nuclear Island Components, afcen, ETC-C EPR Technical Code for Civil Works
exc exc,s	Exceptional Exceptional, seismic
FKA	Forsmark Kraft AB
freq	frequent combination of actions in SLS
FSAR	Final Safety Analysis Report
DNB	Dimensionering av nukleära byggnadskonstruktioner (present report)
IAEA	International Atomic Energy Agency
int	Integrity
KFB	Design specification for buildings (in Swedish ”Konstruktionsförutsätt-

	ningar för byggnader”)
KFM	Design specification for mechanical systems (in Swedish ”Konstrukt-ionsförutsättningar för mekaniska system”)
leak	Leak-tightness
OKG	OKG Aktiebolag
per	Persistant
PS	Pressure Suppression
PSA	Probability Safety Analysis
PSAR	Preliminary Safety Analysis Report
PWR	Pressure Water Reactor
qp	quasi-permanent combination of actions in SLS
RAB	Ringhals AB
SAR	Safety Analysis Report
SCTE	Scanscot Technology AB
sec	Physical Security
SKB	Svensk Kärnbränslehantering
SKI	Statens Kärnkraftsinspektion (now SSM)
SKIFS	Statute book (in Swedish ”Författningssamling”) published by SKI
SKI Technical Report 92.3	Statens kärnkraftsinspektion, SKI Technical Report 92.3 Characterization of seismic ground motions for probabilistic safety analyses of nuclear facilities in Sweden
SLS	Servicability Limit State
SMA	Seismic Margin Assessment
SPSA	Seismic Probabilistic Safety Assessment
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SS-EN	Swedish Eurocodes in general
SS-EN 199x	Swedish Eurocode x
SS-EN 199x-1-2	Specific part of Swedish Eurocode x
SSI	Soil – Structure Interaction
SSM	The Swedish Radiation Safety Authority (in Swedish ”Strålsäker-hetsmyndigheten”)
SSMFS yyyy:nr	Regulations (in Swedish ”Föreskrift”) from SSM year:nbr
STR	Strength
tran	Transient

ULS	Ultimate Limit State
USNRC	United States Nuclear Regulatory Commission
vib	Vibrations
YVL Guides	Regulatory guides on nuclear safety by Radiation and Nuclear Safety Authority (STUK) in Finland

Annex 4: Terms and definitions

”Cliff edge” effect	The "cliff edge" effect means that a small change of a parameter in design gives rise to an abrupt worsening condition of the facility.
Design Basis Threats	A general description of the attributes of potential adversaries who might attempt to commit radiological sabotage or theft or diversion against which licensee's physical protection systems must defend with high assurance.
Elastic behaviour	Elastic behaviour means that the reinforcement or prestressing steel does not reach the yield stress. The concrete may however crack.
Elastic structural behaviour	An elastic structural behaviour means that the section force distribution in the structure is based on a linear elastic structural analysis.
Existing plant	The term existing plant refers to plants built before the establishment of the DNB, i.e. plants built before year 2013.
Explosion	Explosion refers to a process that gives rise to a shock wave.
Missile	Missile refers to an object that due to an initial event is propelled through the air.
Normal usage	The term "normal usage" covers all states and events in normal operation, shutdown, and anticipated operational occurence. Also included are all events associated with event classes H1 (normal operation) and H2 (anticipated events).
Nuclear facility	Formally nuclear power plant and other nuclear facility according to the Swedish Radiation Safety Authoritys statue book. However, a broader definition applies in DNB that also includes other facilities which during an accident may have significant radiological impact on the surroundings.
Regulation	A set of rules, see the word rule.

Rule	Formally a fixed provision for how a certain action shall be performed. In this sense, rules are mandatory. Here, however, a broader definition applies as follows. Rules have different status and application. Some rules are binding regulations, set by authorities; others describe technical solutions approved but not mandatory. Other rules have been established by industry associations. Many rules become regulatory as they are referred to in the contract documents. Some rules are neither set by an authority, nor intended to be included in the contract documents. This category includes, among other things, handbooks and other publications of advisory or recommendatory nature.
Safety analysis report	The safety analysis report shall show how the overall safety of the plant is provided to protect human health and the environment from radiological accidents. The report shall reflect the plant as it is built, analysed and verified, as well as show how the requirements on its structure, function, organization and operation are fulfilled. Current requirements are stated in the applicable regulations and license conditions as well as the rules, such as industry standards, which the license holder in addition applies to the plant. The term “safety analysis report” follows the terminology of IAEA. For different types of nuclear power plants in Sweden the safety analysis report is denoted either SAR or FSAR, where F stands for "Final". In the present report the term SAR is used. Before a plant may be constructed and before any major rebuilds or modifications to an existing plant is implemented, a preliminary safety report shall be compiled, usually then denoted PSAR, where P stands for “Preliminary”.
Safety-related building	Building in safety class 1-3 regarding radiological environmental safety, or building that can adversely affect other structures, systems or components in safety class 1-3.
Subs	Subs means that the building structure is divided into zones that are in some way separated from each other.

Annex 5: Symbols

δ_s	Settlement
δ_u	Ultimate displacement capacity for liner anchors according to ASME Sect III Div 2
ε_{cs}	Shrinkage
ε_{sc}	Liner strain allowable, compression according to ASME Sect III Div 2
ε_{st}	Liner strain allowable, tension according to ASME Sect III Div 2
$\varnothing$	Diameter of a reinforcing bar or of a prestressing duct
$\varnothing_n$	Equivalent diameter of a bundle of reinforcing bars
B	Action of fire
D	Self-weight
d_g	Largest nominal aggregate size
E_{DEE}	Action caused by design extension earthquake (DEE)
E_{DBE}	Action caused by design basis earthquake (DBE)
F	Pool dynamic load
f'_c	Specified compressive strength of concrete according to ASME Sect III Div 2
F_{CH}	Actions due to chugging
F_{CO}	Actions due to condensation oscillations
F_{ps}	Drag forces and impact loads caused by level-swell in the condensation pool
f_{py}	Specified tensile yield strength of prestressing steel according to ASME Sect III Div 2
f_{py}	Specified tensile yield strength of liner steel according to ASME Sect III Div 2
F_{SRVa}	Safety relief valve blow-down during a pipe rupture
F_{SRVe}	Pool dynamic load due to extreme safety relief valve blow-down
F_u	Liner anchor ultimate force capacity according to ASME Sect III Div 2
f_y	Specified tensile yield strength of reinforcing steel according to ASME Sect III Div 2
F_y	Liner anchor yield force capacity according to ASME Sect III Div 2
H_{ef}	Action due to exceptional external flooding
H_{ge}	Earth pressure and geotechnical load
H_{gw}	Water pressure at normal water level
H_{if}	Action due to exceptional internal water pressure

H_{qe}	Earth pressure caused by variable surface action
H_{qw}	Water pressure variation
L	Imposed action
l_d	Minimum length of lap for tension splices according to ASME Sect III Div 2
M_d	Process-related actions during anticipated operational occurence
$M_{d,\Delta T}$	Temperature differences and temperature changes during anticipated operational occurence
$M_{d,Hqw}$	Water pressure variation relative permanent water pressure during anticipated operational occurence
$M_{d,P}$	Overpressure or underpressure during anticipated operational occurence
$M_{d,R}$	Actions from piping and processing systems during anticipated operational oc-curence
$M_{d,SRV}$	Safety relief valve blow-down or other pressure relief of high energy device in the event class H2
M_n	Process-related actions during normal operation and shutdown
$M_{n,\Delta T}$	Process-related temperature differences and temperature changes during normal operation and shutdown
$M_{n,Hqw}$	Process-related water pressure variations
$M_{n,P}$	Process-related overpressure or underpressure during normal operation and shut-down
$M_{n,R}$	Process-related actions from piping and processing systems
$M_{n,SRV}$	Safety relief valve blow-down or other pressure relief of high energy device in the event class H1
M_t	Process-related actions during testing of the facility
P_a	Transient overpressure and underpressure during pipe rupture
P_{aL}	Specified pressures
P_g	Pressure difference related to other events than pipe ruptures
P_p	Prestressing force
$P_{pk,inf}(t)$	Corresponds to $P_{k,inf}(t)$ in SS-EN 1992-1-1
$P_{pk,sup}(t)$	Corresponds to $P_{k,sup}(t)$ in SS-EN 1992-1-1
$P_{pm}(t)$	Corresponds to $P_m(t)$ in SS-EN 1992-1-1
Q_c	Construction loads
Q_{ca}	Personnel, staff, visitors and hand tools
Q_{cb}	Storage of movable items
Q_{cc}	Non permanent equipment
Q_{cd}	Moveable heavy machinery and equipment

Q_{ce}	Accumulation of waste materials
Q_{cf}	Loads from parts of a structure in temporary states
R	Direct loads caused by pipe rupture
R_{rj}	Jet load due to pipe rupture
R_{rm}	Missile load due to pipe rupture
R_{rr}	Pipe support reaction forces due to pipe rupture
S	Snow load
ΔT	Climate-related temperature difference and temperature changes
ΔT_a	Temperature differences and temperature changes associated with P_a
ΔT_{aL}	Temperature differences and temperature changes associated with P_{aL}
W_a	Action due to extreme climate impact
W_q	Wind action
X	Action due to other exceptional impact
X_{APC}	Actions related to air plane crash
X_{DBT}	War actions and actions related to design basis threats
X_e	Action due to explosions
X_m	Missile generated loads
X_{nom}	Nominal value
Y	Action due to transportation accident
Z_{APC}	Actions related to air plane crash with large commercial aircraft
Z_{Hef}	Action caused by highly improbable external flooding
Z_{SA}	Effects of actions related to severe accidents
$Z_{SA,\Delta T}$	Temperature differences and temperature changes due to severe accidents
$Z_{SA,Hif}$	Exceptional internal water pressure due to severe accidents
$Z_{SA,P}$	Overpressure and underpressure due to severe accidents
Z	Action due to other highly improbable impact

Annex 6: Anchoring to concrete

This Annex gives provisions on how the Eurocodes together with the CEN/TS 1992-4-1 [9], CEN/TS 1992-4-2 [10] and CEN/TS 1992-4-4 [11] should be applied in the design of anchoring to concrete at nuclear power plants and other nuclear facilities. Modifications and amendments to [9], [10] and [11] with respect to anchors in safety class 1-4, are presented below.

The following sections follows the arrangement outlined in the CEN/TS documents. The sections are named and numbered in the same way as in [9], [10] and [11]. First, the considered section of the CEN/TS document is given, followed by possible modifications and amendments.

The introduced modifications and amendments are numbered from 1-20.

Scope

Section 1 in CEN/TS 1992-4-1 [9] applies, with following modifications and amendments:

1. The provisions in DNB Annex 6 applies to anchors in safety class 1-4 (radiological environmental safety)
2. Anchor channels are not covered in DNB Annex 6.
3. Concrete screws and bonded anchors shall not be used according to Section 6.6.3.6 in DNB.
4. Load-carrying fixtures should be designed with at least 2 anchors.
5. It is under certain circumstances allowed to design anchor plates that deviates from those presented in Figure 1 in [9]. It should then be ensured that:
    - the structural behavior of the anchor, including its stiffness and deformability as well as its ductility, does not deviate from what is assumed in [9].
    - actions on the anchors can be determined correctly with the methods provided in [9].
6. Requirements on minimum diameters for different types of anchors and fasteners may be specified in site-specific documents.

Section 1 in CEN/TS 1992-4-2 [10] applies with corresponding modifications and amendments as stated above.

Section 1 in CEN/TS 1992-4-4 [11] applies with corresponding modifications and amendments as stated above.

Normative references

Section 2 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

7. The Eurocodes shall be applied as specified in DNB.

Section 2 in CEN/TS 1992-4-2 [10] applies with corresponding modifications and amendments as stated above.

Section 2 in CEN/TS 1992-4-4 [11] applies with corresponding modifications and amendments as stated above.

Definitions and symbols

Section 3 in CEN/TS 1992-4-1 [9], CEN/TS 1992-4-2 [10] and CEN/TS 1992-4-4 [11] applies without modifications and amendments.

Basis of design

Section 4 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

8. The resistance of the anchor shall be determined for different types of accidental actions, to the extent specified in site-specific documents.
9. The provisions given in CEN/TS [9] are based on an anchor life time of at least 50 years. Life time expectancy for different types of anchors is usually stated in their Technical Approval ("typgodkännande" in Swedish, note that CEN/TS [9] specifies what can be considered a Technical Approval). If an anchor is utilised for a longer period than the operational life time stated in its Technical Approval, a special investigation should be conducted.
10. If the joint is subjected to dynamic loads, anchors conforming to this type of load should be used. Dynamic loads include both global and local vibration loads. Usually, anchors that are approved (tested) for seismic actions are considered to be applicable to both global and local vibration loads, provided that fatigue is not decisive.
11. Any additional requirements with respect to limitations of acceptable deformations of the anchors are specified in site-specific documents.
12. The CEN/TS [9] does not present any partial factors for resistance for various types of accidental actions. A conservative approach is to use values for transient design situations. However, regarding concrete failure, values for accidental design situations as reported in SS-EN 1992-1-1 [29] are accepted, unless otherwise stated in the Technical Approval.
13. The partial factors for resistance reported in CEN/TS [9] can be reduced for structural capacity related to severe accident. Regarding reinforcement or concrete failure, values for accidental actions as reported in EN 1992-1-1 [29] are accepted, unless otherwise stated in the Technical Approval.
14. In addition to the requirements for installation of the anchors as stated in CEN/TS [9], additional requirements may be given in site-specific documents.

Section 4 in CEN/TS 1992-4-2 [10] applies, with corresponding modifications and amendments as stated above.

Section 4 in CEN/TS 1992-4-4 [11] applies, with corresponding modifications and amendments as stated above.

Determination of concrete condition and effects

Section 5 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

15. Plastic design should not be applied to persistent or transient design situations.
16. Permissible concrete compression is determined according to Section 6.6.8.1 in DNB.

Section 5 in CEN/TS 1992-4-2 [10] applies, with corresponding modifications and amendments as stated above.

Section 5 in CEN/TS 1992-4-4 [11] applies, with corresponding modifications and amendments as stated above.

Verification of ultimate limit state

Section 6 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

17. Brittle failure of the anchors should be avoided. If this is not possible, following rules should apply:
    - For global vibration loads, section 8.4.3 in CEN/TS 1992-4-1 [9] applies.
    - The brittle failure capacity is reduced by a factor of 0.75 for safety class 1-3.

Section 6 in CEN/TS 1992-4-2 [10] applies, with corresponding modifications and amendments as stated above.

Section 6 in CEN/TS 1992-4-4 [11] applies, with corresponding modifications and amendments as stated above.

Verification of fatigue limit state

Section 7 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

18. Failure due to fatigue is not covered in DNB.

Section 7 in CEN/TS 1992-4-2 [10] applies, with corresponding modifications and amendments as stated above.

Section 7 in CEN/TS 1992-4-4 [11] applies, with corresponding modifications and amendments as stated above.

Verification for seismic actions

Section 8 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

19. The load factor for seismic action E_d, given as 2.5 can be reduced to 1.25 for cases where a linear elastic analysis approach has been applied to all steps when determining the effects of seismic actions without utilising the so-called "behavior"-factor.

Section 8 in CEN/TS 1992-4-2 [10] applies, with corresponding modifications and amendments as stated above.

Section 8 in CEN/TS 1992-4-4 [11] applies, with corresponding modifications and amendments as stated above.

Verification of serviceability limit state

Section 9 in CEN/TS 1992-4-1 [9] applies with following modifications and amendments:

20. See number 11.

Annex 7: Assessment of concrete strength based on in-situ testing at nuclear facilities

This Annex have not been translated into English.

Annex 8: Structural analysis - a summary

This annex summarizes some of the recommendations given in DNB regarding structural analysis. The summary covers cases regarding strength capacity and leak-tightness in the ultimate limit state of the reactor containment and for other safety-related building structures. The summary below shall be seen as a general approach, each specific situation must be assessed case by case. The presentation is in tabular form. The following notations are used in the table:

Structure

- Reactor containment = RI

- Other safety-related building structures = SRB

- Non safety-related building structures (not covered in Annex 8)

Limit states (with associated design situation)

- SLS (not covered in Annex 8)

- ULS_{EQU} and ULS_{VIB} (not covered in Annex 8)

- ULS_{STR}-per; ULS_{STR}-tran	ULS_{LEAK}-per; ULS_{LEAK}-tran	ULS_{CONT}-per; ULS_{CONT}-tran
- ULS_{STR}-exc	ULS_{LEAK}-exc	ULS_{CONT}-exc
- ULS_{STR}-exc,s	ULS_{LEAK}-exc,s	ULS_{CONT}-exc,s
- ULS_{STR}-dec	ULS_{LEAK}-dec	ULS_{CONT}-dec
- ULS_{STR}-dec,s	ULS_{LEAK}-dec,s	ULS_{CONT}-dec,s

Type of action

- Primary actions (external actions) = Primary

- Secondary actions (temperature effects, effect of settlement as well as effects of shrinkage and creep) = Secondary

Structural analysis

- Elastic analysis = Elastic

- Limited plastic redistribution allowed = VPO

- Plastic analysis = Plastic

Case		**Type of action**		**Structural analysis**			**Footnote**
Struct.	**Limit state**	**Primary**	**Second.**	**Elastic**	**VPO**	**Plastic**	
RI	ULS_{STR} -per/tran and -exc	X		X			
		X	X	X			
		X	(X)			X	1)
	ULS_{STR}-dec	X	(X)			X	
	ULS_{STR} -exc,s and -dec,s	X	(X)	X			
	ULS_{CONT} -per/tran and -exc	X		X			3)
		X	X	X			3)
		X	(X)			X	1)+3)
	ULS_{CONT}-dec	X	(X)			X	3)
	ULS_{CONT} -exc,s and -dec,s	X	(X)	X			3)
SRB	ULS_{STR} -per/tran and -exc	X			X	(X)	2)
		X	X		X	(X)	2)
		X	(X)			X	1)
	ULS_{STR}-dec	X	(X)			X	
	ULS_{STR} -exc,s and -dec,s	X	(X)	X			
	ULS_{LEAK} -per/tran and -exc	X			X		3)
		X	X		X		3)
		X	(X)			X	1)+3)
	ULS_{LEAK} -per/tran and -exc	X		X	(X)		3)+4)+5)
		X	X	X	(X)		3)+4)+5)
	ULS_{LEAK}-dec	X	(X)			X	3)
	ULS_{LEAK} -exc,s and -dec,s	X	(X)	X			3)

X = Action included / Acceptable analysis method
(X) = Action possibly included / acceptable analysis method under certain conditions (see footnote for details).
1), 2), …, 5) = Footnote according to below.
+ = When there are plus signs between footnotes, all footnotes must be met.
Note that for all cases where secondary type of actions are included, concrete cracking need to be taken into consideration, unless otherwise stated
It is allowed to use the less favourable analysis methods than those indicated in the table (elastic analysis or VPO, instead of plastic analysis, if plastic analysis is indicated, etc.).

1) This case can only be applied when the impulse or impact action is the main load.
2) The utilization of plastic analysis is only allowed if specifically shown to be acceptable.
3) This case can be applied provided that the leak-tightness requirements can be met.
4) This case applies to tanks, pools etc. with steel liner. This case can also be used when limitation of the crack width is essential to show that the leak-tightness requirements of the structure are fulfilled.
5) If limited plastic redistribution is taken into account, it should be ensured that the reinforcement closest to the steel liner does not yield for cases where the steel liner is loaded in tension

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The Swedish Radiation Safety Authority has a comprehensive responsibility to ensure that society is safe from the effects of radiation. The Authority works to achieve radiation safety in a number of areas: nuclear power, medical care as well as commercial products and services. The Authority also works to achieve protection from natural radiation and to increase the level of radiation safety internationally.

The Swedish Radiation Safety Authority works proactively and preventively to protect people and the environment from the harmful effects of radiation, now and in the future. The Authority issues regulations and supervises compliance, while also supporting research, providing training and information, and issuing advice. Often, activities involving radiation require licences issued by the Authority. The Swedish Radiation Safety Authority maintains emergency preparedness around the clock with the aim of limiting the aftermath of radiation accidents and the unintentional spreading of radioactive substances. The Authority participates in international co-operation in order to promote radiation safety and finances projects aiming to raise the level of radiation safety in certain Eastern European countries.

The Authority reports to the Ministry of the Environment and has around 300 employees with competencies in the fields of engineering, natural and behavioural sciences, law, economics and communications. We have received quality, environmental and working environment certification.

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