Autoriteit Nucleaire Veiligheid en Stralingsbescherming

Handreiking VOBK

Veilig ontwerp en bedrijfsvoering van kerninstallaties

Guideline for the safe design and operation of nuclear installations

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Document Number	ANVS-2025/7894
Address	Authority for Nuclear Safety and Radiation Protection
	Koningskade 4 | 2596 AA | Den Haag

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Introduction

1.1 Scope and Goal

The guideline for the safe design and operation of nuclear installations, in Dutch the 'Handreiking Veilig Ontwerp en Bedrijfsvoering van Kerninstallaties' (VOBK), describes the ANVS review criteria for the assessment of nuclear safety for all nuclear installations. It is based on the latest developments in nuclear safety, for which the IAEA Safety Standards are considered to be leading. It is used by the ANVS as an assessment framework for nuclear safety in a license application to build a nuclear installation, implement major revisions in existing nuclear installations, and can be used by licensees as part of the assessment framework for the periodic safety review. This Guideline replaces the 2023 version of the *Handreiking VOBK* and *Dutch Safety Requirements*.

The *Handreiking VOBK* is applicable to **all nuclear installations**, as licensed under article 15, section b, of the Nuclear Energy Act (*Kernenergiewet* – Kew). Specific requirements per type of installation are noted where applicable.

The *Handreiking VOBK* is the assessment framework for the ANVS for the license application to **construct** and the license application to **operate** a new nuclear installation.

The *Handreiking VOBK* can be used as an assessment framework for the ANVS for the license application for **major modifications** in existing nuclear installations.

The *Handreiking VOBK* can also be used by licensees of existing nuclear installations as a frame of reference to identify reasonably achievable improvements for their **periodic safety review**, as required by the Ministerial Decree on Nuclear Safety of Nuclear Installations (*Regeling Nucleaire Veiligheid Kerninstallaties* - Rnvk).

1.2 Status

'Handreikingen' or Guidelines are informative documents that the ANVS publishes for stakeholders such as license holders or license applicants, that set the ANVS expectations on a given topic. This means that the *Handreiking VOBK* is not legally binding. However, it does provide guidance on how to fulfill legally binding criteria.

The Dutch legal framework is illustrated in figure 1:

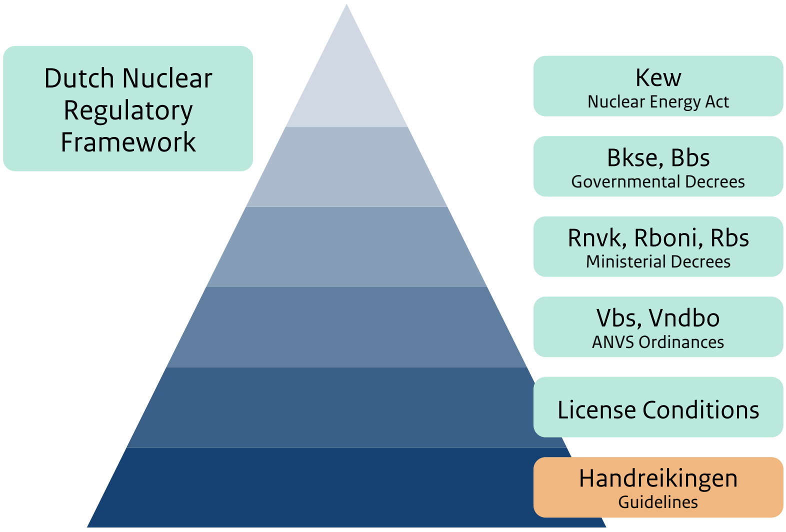

Fig. 1: Dutch nuclear regulatory framework. Laws and Decrees are referred to by their Dutch names, and are further explained in the text.

In the assessment of **license applications** for nuclear installations, the *Handreiking VOBK* provides guidance on what the ANVS must assess in order to issue a license. These legal criteria are goal-oriented and can be found in the different levels of legally binding regulations:

- The grounds for refusal of a license application in the interest of environmental protection, safety, security, and international obligations (article 15b, section 1, Kew)[1];
- The ground for refusal of a license application due to an outdated technical design (article 15b, section 2 Kew);
- The obligation to submit, among others, a safety report in a license application for a nuclear installation (articles 6 to 11 *Besluit kerninstallaties, splijtstoffen en ertsen* - Bkse). The safety report contains a description of the measures that

1 These include all grounds for refusal of a license with the exception of the insurance for liability. This is assured through the *Wet aansprakelijkheid kernongevallen* (Wako) and license conditions (as required in article 39 Bkse).

will be taken to ensure the prevention of damage, or limit the chance of damage, including measures to prevent damage outside of the site, during normal operations, and the prevention of damage coming from postulated initiating events, as well as a risk analysis of the damage outside of the site as a consequence of these events and beyond design basis accidents[2];
- The requirements for radiation protection, justification, optimalisation, dose limits, expertise, as described in the Decree on basic safety standards for radiation protection (*Besluit basisveiligheidsnormen stralingsbescherming* – Bbs) (article 18, section 1, Bkse);
- The radiological acceptance criteria for normal operation and design basis accidents (article 18, section 2, Bkse)
- Individual and group risk limits for reactors (article 18, section 3, Bkse).
- The safety security interface (article 22 Bkse, section 5)
- The requirements on radioactive waste and decommissioning (article 30 Bkse)
- Avoiding large and early releases (article 6 *Regeling nucleaire veiligheid kerninstallaties* - Rnvk)
- Defence in depth (article 7 Rnvk)
- Effective nuclear safety culture (article 8 Rnvk)
- Management system (article 9 Rnvk)

These criteria, among others, aim to ensure that a license is only issued when the nuclear safety of a nuclear installation has been demonstrated. The *Handreiking VOBK* provides explanation as to what the ANVS considers to be sufficient to fulfill these criteria.

However, an applicant may use a different approach to that in the *Handreiking VOBK* to fulfill the legally binding criteria, as long as a licensee or license applicant can demonstrate that the goals are achieved to an **equivalent** level of safety.

If there are any **contradictions** between the requirements in the *Handreiking VOBK*, directly or through reference, and the legal framework, the legally binding criteria will always take precedence.

The use of the *Handreiking VOBK* for **periodic safety review** is based on the legal obligation under article 11, paragraph 4, sub d, Rnvk. It describes that a license holder shall take into account the relevant developments and insights in the area of nuclear safety in the review. The ANVS considers the *Handreiking VOBK* to be the reference framework in that context.

1.3 Background

The first version of the VOBK was published in 2015 with minor editorial revisions in 2023. It consisted of an introduction and the 'Dutch Safety Requirements' (DSR). The DSR was applicable to light water-cooled nuclear power plants, complimented by an annex on how to apply the requirements with a graded approach for research reactors.

A decade on, the IAEA Safety Standards Series have been substantially improved and expanded, following the lessons learned from the Fukushima Daiichi accident. Also, there is increased interest in newbuild large scale and small scale reactors in the Netherlands. In 2025, the ANVS has conducted an extensive revision of the VOBK in order to align it with the latest developments in nuclear safety. The revision aimed to:
- improve harmonization with international standards;
- apply the latest developments in nuclear safety;
- expand the scope to all nuclear installations and be more technology-inclusive;
- further clarify the *Handreiking*'s relation to Dutch legislation.

This has resulted in a new *Handreiking VOBK* that maintains the exact same goals of setting the standard for the latest developments in terms of nuclear safety, but adopts a novel approach. It sets the IAEA standards as the base requirements, and adds specification or elaboration only:
- as necessary in the Dutch licensing context;
- where IAEA standards leave room for national interpretation;
- as required through the WENRA Safety Reference Levels for Existing Reactors [1] and WENRA Safety Objectives for New Reactors [2].

2 The risk analysis relating to the damage outside of the site as a consequence of beyond design basis accidents is mentioned as a separate sub-article but in practice is usually included in the safety report.

A graded approach is implemented into the new *Handreiking VOBK* by virtue of this new structure. The IAEA-requirements are applied per installation type, and complemented by goal oriented ANVS-specific requirements. This approach effects a graded approach implicitly as demonstrating compliancy with goal-oriented requirements is simpler for installations with lower complexity and risk profiles.

The implementation of the WENRA SRLs, as relevant to nuclear safety in the design and operation of a nuclear installation, have been incorporated into the *Handreiking VOBK* such that by following its guidance these do not need to be assessed separately.

1.4 Structure of the Guideline

This first chapter of the *Handreiking VOBK* describes the scope, goal, status and background of the VOBK.
Chapter 2 provides the requirements on the application of the IAEA Safety Standards in the Dutch licensing context.
Chapter 3 provides the ANVS specification of certain design requirements and on safety demonstration and documentation. Explanatory text is written in black text. Requirements are written in *orange highlighted text* and numbered in the form *x.y (z)*. Annex A gives an example of the identification of relevant postulated initiating events and specific acceptance criteria.

2

Implementation of IAEA Standards

In the assessment of a license application for a nuclear installation, the ANVS views the IAEA Safety Standards as the leading framework for nuclear safety. This chapter describes which IAEA Safety Requirements (SSRs and GSRs) and associated documents are expected to be applied in different situations. By demonstrating that the requirements as set out in this chapter are met, in addition to the specific requirements necessary in the Dutch licensing context described in the following chapter, a license applicant can show that they meet the 'state of the art' in terms of nuclear safety.

2.1 Fundamentals

The IAEA Safety Fundamentals [3] establish the fundamental safety objective, safety principles and concepts that provide the bases for the IAEA Safety Standards. These principles and concepts are all incorporated into the Dutch nuclear regulatory framework, and are considered to be fulfilled upon implementation of the regulations. The IAEA Nuclear Security Fundamentals [4] presents the objective and essential elements of a State's nuclear security regime and as such are not aimed at license applicants.

2.2 IAEA Safety Requirements & Security Recommendations

The ANVS expects the IAEA requirements to be implemented as follows:

2.2 (1) Site evaluations for nuclear installations shall be performed according to the requirements in IAEA SSR-1 [5].

2.2 (2) Design and operation of nuclear power plants shall be performed according to the requirements in SSR-2/1 (Rev. 1) [6] and SSR-2/2 (Rev. 1) [7].

2.2 (3) Design and operation of research reactors shall be performed according to IAEA SSR-3 [8].

2.2 (4) Design and operation of fuel cycle facilities shall be performed according to IAEA SSR-4 [9].

2.2 (5) Leadership and management for safety shall be performed according to GSR Part 2 [10].

2.2 (6) Safety assessments shall be performed according to GSR Part 4 (Rev. 1) [11].

2.2 (7) Security measures shall follow the recommendations directed at license holders/applicants in the IAEA Nuclear Security Recommendations 13, 14 and 15 [12, 13, 14].

2.2 (8) For small modular reactors and non-water cooled reactors, the considerations in SRS-123 [15] may be taken into account when applying the requirements above.

GSR Part 1 relates to the functioning of the regulatory body and hence is not relevant for applicants. The content of GSR Part 3 [16] is covered in Dutch law through the implementation of the basic safety standards (Bbs). Some elements of GSR Parts 5, 6, and 7 [17, 18, 19] are applicable to licensees in the design and operation of a nuclear installation. These documents are referenced in the SSRs where relevant.

2.3 IAEA Safety Guides

The IAEA Safety Guides provide guidance on how to comply with the safety requirements. The ANVS will therefore use IAEA Guides as a reference in assessing compliance with the requirements. Following the Guides is not the only way in which to fulfill the requirements, however the applicant should in that case demonstrate that the same level of safety is nonetheless achieved.

2.4 Evolving standards in response to innovative technologies and applications

Section 2.2 refers to current IAEA requirements at the moment of publishing the *Handreiking VOBK*. Changes in IAEA requirements will result in an update of the VOBK. The assessment framework for a license application will be the version of the VOBK at the moment of application. If the VOBK is being revised during pre-licensing discussions, the implications will be discussed with the applicant, such as whether the changes are relevant to the design under consideration and whether an equivalent level of safety can be demonstrated with requirements from the previous version. This is an example of the use of a different approach to demonstrating an equivalent level of safety as stated in section 1.2.

The Safety Standards that have been developed over the last decades have evolved to incorporate the latest insights in safety thanks to the experience of operating predominantly water-cooled reactors. The IAEA Safety Report Series-123, as noted in 2.2 (8), provides a detailed review of the extent to which the current safety standards can be applied to non-water cooled reactors and small modular reactors. It specifically details requirements that may not be applicable to certain innovative technologies. The gaps identified will be gradually integrated into the Safety Standards Series as and when they come up for review, and the SRS-123 serves as a valuable tool in the interim.

Additional insight can be found in publications such as IAEA TECDOCs [20, 21, 22]. These publications are not subject to the extensive consensus forming review process of the standards and guides, and are not as highly edited. However, they share experiences from Member State regulators and industry, which can serve as input for new revisions, highlighting gaps in the current standards and means to address them. The applicability of TECDOCs in the assessment of aspects of the design may be discussed during the pre-licensing stage.

3

ANVS-Specific Requirements

Chapters 3 sets out the additional requirements necessary in the Dutch licensing context. This includes:

- Interpretation on certain IAEA requirements where the IAEA leaves room for national interpretations;
- Implementation of WENRA Safety Reference Levels where these go beyond the IAEA Safety Standards;
- Specification on how the Dutch regulatory framework relates to IAEA requirements or definitions.

Chapter 3 addresses all the ANVS-specific requirements on design, safety demonstration and documentation in the Dutch licensing context. If a topic is not specifically mentioned here, the ANVS refers to the IAEA Safety Standards and has no specific further expectations. This includes topics such as filtered venting, the application of the concepts leak before break or break preclusion, and shared safety systems.

3.1 Defence in Depth

Rnvk article 7; SSR-2/1 Req 7; SSR-3 Req 10; SSR-4 Req 10

The defence in depth concept as mentioned in article 7 of the Rnvk can be interpreted in terms used in IAEA Safety Standards as follows:

Rnvk art.7 sub 2	Translation of Dutch sub-article	International term
b[3]	Prevention of abnormal operation and failures;	Level 1: Normal operation
c	Control of abnormal operation and detection of failures;	Level 2: Anticipated operational occurrences
d	Control of accidents within the design basis;	Level 3a/3: Design basis accidents
e	Control of severe conditions, prevention of accident progression; severe accident mitigation;	Level 3b/4a: Design extension conditions A[4] Level 4/4b: Design extension conditions B
f	Implementation of organizational structures as described in the installation's emergency plan.	Level 5: Mitigation of consequences of significant radiological releases

3.1 (1) The defence in depth concept used by the applicant in their safety case shall reflect all elements mentioned in Rnvk article 7 and clearly distinguish between each level.

3.2 Single failure criterion

SSR-2/1 Req. 25; SSR-3 Req. 25; SSR-4 Req. 23; WENRA SRL F4.7

In the IAEA Specific Safety Requirements, the single failure criterion is used in the analyses of anticipated operational occurrences and design basis accidents[5] and the design of SSCs. These SSRs do not explicitly require the single failure criterion to be met in design extension conditions, though this may provide significant safety gains.

3.2 (1) A quantitative analysis shall be performed to justify[6] the decision whether or not to apply the single failure criterion to SSCs for design extension conditions. Where reasonably practicable, the single failure criterion shall be applied.

Maintenance

In the case of maintenance, the time period for the unavailability of an item important to safety can influence the total reliability of its safety function. For operational reasons, this could be a reason to design certain systems with a higher degree of redundancy than would be strictly necessary to fulfill the single failure criterion.

3.2 (2) The admissible time of unavailability due to surveillance, testing or corrective maintenance (repair) shall be specified in the operational limits and conditions. The allowed outage time shall be covered by availability assumptions in the probabilistic safety analysis.

3 Article 7 sub 2 a is not a plant state but concerns the protection concept and robustness of plant design against external hazards.

4 DEC-A is sometimes considered part of level 3 as level 3b, or as part of level 4 as 4a, depending on the approach, as described in table 1 of IAEA SSG-88 [23]. The ANVS considers both approaches to be acceptable.

5 The requirements use the term 'safety group used in the design', which according to the IAEA safety and security glossary should be read as anticipated operational occurrences and design basis accidents.

6 Examples of justification are probabilistic or deterministic arguments, engineering judgement, the complexity of the SSC, robust maintenance and/or testing, or a cost-benefit analysis.

3.2 (3) No unavailability due to preventive maintenance that violates the single failure criterion shall be allowed. This means that preventive maintenance on systems that are required to fulfill the single failure criterion can only be allowed in an operational state[7] when their functioning is not required for the safe operation of the nuclear installation.

Passive systems
In the application of the single failure criterion to passive systems, the ANVS refers in the following requirement to the graded levels of passivity 'A' through 'D' as described in TECDOC-626 [24].

3.2 (4) No single failure has to be postulated in passive safety systems that fall within type A and B passive systems. In type C passive systems, the single failure criterion shall be applied to any components with moving mechanical parts. In type D systems, the single failure criterion shall also be applied to any initiating signals that are required.

3.3 Redundancy, diversity, independence and separation

SSR-2/1 Req. 24; SSR-3 Req. 26; SSR-4 Req. 23, WENRA SRL E9.4, SV5.4c
Redundancy, diversity, independence and physical separation are required in the IAEA SSRs to be considered to increase reliability and reduce the potential for common cause failures. Here, the ANVS sets specific expectations for the implementation of these concepts in line with WENRA SRLs.

3.3 (1) If a design depends on the functioning of active systems to prevent events within the design basis reaching severe accident conditions[8], the main safety functions shall, in principle, be performed by redundant, diverse, independent and separated components. Exceptions shall be justified[9].

3.3 (2) If a design depends on the functioning of passive systems to prevent events within the design basis reaching severe accident conditions, and and the single failure criterion must be applied according to 3.2 (4), the redundant components shall, in principle, be diverse, independent and separate. Exceptions shall be justified[10].

3.4 Automation

SSR-2/1 5.11, 5.12, 5.59; SSR-3 6.42, 6.43, 6.105; SSR-4 6.13, 6.14, 6.84; WENRA SRL E9.3
In the IAEA SSRs, manual initiation of safety systems, intervention or other operator actions should not be required before a "sufficiently long" period of time. Here, the ANVS sets its expectation for this time period, in line with WENRA SRLs.

3.4 (1) Activations and control of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes of the initiating event. Exceptions, i.e. any operator actions required by the design within 30 minutes of the initiating event, shall be justified[11].

3.5 Self-sufficiency

WENRA F4.5; SSG-34 7.48, 7.49; SSG-56 6.73, 6.90; SSG-39 8.19 – 8.35
The WENRA SRLs state that an NPP shall be autonomous regarding supplies supporting safety functions for a period of time. For consumable items such as fuel, lubrication oil (SSG-34 [25]) or, in some cases, feedwater (SSG-56 [26]), the IAEA Safety Standards expect that on-site sources are sufficient until no longer needed or until these items can be replenished. The IAEA Safety Standards does not specify the capacity for the power supply of specific accident instrumentation (SSG-39 [27]). Here, the ANVS sets its expectation for a self-sufficiency

7 Operational states as used in this context mean for example startup, power operation, shutdown, maintenance, testing and refueling.

8 This refers to the combination of systems in DBA and DEC-A.

9 Examples of justification are probabilistic or deterministic arguments, engineering judgement, the complexity of the SSC, robust maintenance and/or testing, or a cost-benefit analysis.

10 See footnote 9

11 See footnote 9

period for all nuclear installations, which comprises both a limit and target value. The target value is set to promote considerations of measures that could further extend the robust design of the nuclear installation.

3.5 (1) The period of time for which the site is self-sufficient regarding supplies supporting safety functions shall be no less than 72 hours with a target value[12] of 7 days.

3.5 (2) The period of time in which accident instrumentation is available even in case of station blackout (SBO) shall be no less than 10 hours.

3.6 Hazard protection concept

SSR-2/1 Req 17; SSR-3 Req 19; SSR-4 Req 16; WENRA SV5.1, TU4.2, TU5.1
The ANVS expects a hazard protection concept in line with WENRA SRLs and IAEA Safety Standards. While the IAEA Safety Standards treat the range of external hazards to be considered in depth in their guidance, no specific goal is set for the minimum exceedance frequency to be considered.

3.6 (1) For external hazards, an exceedance frequency no higher than 10^{-4} per annum shall be used to determine which events fall within the design basis. Where deterministic methodologies are used in the hazard evaluation it shall be justified that an equivalent level of safety is reached.

3.7 Safety analyses

Bkse article 18; SSR-2/1 Req 42; SSR-3 Req 41; SSR-4 Req 20, 21; GSR part 4
The basic acceptance criteria in the Dutch legal framework are formulated in Bkse article 18 as:
- radiation protection criteria for normal operation (sub article 1);
- dose limits for anticipated operational occurrences, design basis accidents and design extension conditions A[13] (sub article 2);
- risk limits for design extension conditions B[14] (sub article 3).

For nuclear power plants and research reactors, the methods of safety assessments, both deterministic (DSA) and probabilistic (PSA), are well documented in SSG-2 [28], SSG-3 [29] and SSG-4 [30], complimented by the *"ANVS Guide on Level 3 PSA"* [31]. The general concepts of the safety assessments as used for reactors (defining specific acceptance criteria, identifying enveloping postulated initiating events (PIEs) and determining event sequences) are applicable for the quantitative risk assessment (QRA) for other types of facilities as well. As an example of the level of detail that is expected by the ANVS, an example set of PIEs and acceptance criteria relevant for light water reactors are provided in Annex A.

3.7 (1) In the deterministic safety analysis (DSA) that is performed as part of the license application to operate a nuclear installation, the design basis accident analyses shall be best estimate plus uncertainty, conservative, or combined. The design extension conditions analyses may be best estimate if complemented with a sensitivity analysis[15].

3.7 (2) Loss of off-site power shall be considered in the analysis of all design basis accidents as an additional conservative assumption.

12 When reaching the 'target value' the ANVS will not require additional measures to improve this value.

13 The Dutch legal system uses the distinction between 'design basis accidents' and 'beyond design basis accidents'. These terms are not used in the rest of this publication to align as much is possible with international terminology but corresponds to the transition to severe accidents.

14 See footnote 13

15 The terms 'conservative', 'combined', 'best estimate plus uncertainty', and 'best estimate' refer to the terms as defined in SSG-2 table 1 in options 1, 2, 3 and 4 respectively.

3.7 (3) For reactor facilities, a failure to insert the control element[16] with the highest reactivity worth shall be assumed in the analysis of all design basis accidents as an additional conservative assumption, unless already assumed as single failure.

3.7 (4) For a license application to operate a reactor, the risk analyses to demonstrate compliance to the basic acceptance criteria on individual and societal risk in article 18 sub 3 of the Bkse shall be performed with a three level PSA that includes all operational modes, events and hazards, and relevant sources of radioactivity.

3.7 (5) For a license application to operate a nuclear installation that is not a reactor, these risk analyses shall be performed by a quantitative risk assessment that as far as reasonably practicable contains the same elements as a three level PSA and includes all operational modes, events and hazards, and relevant sources of radioactivity.

3.7 (6) For a license application to construct a nuclear installation, safety analyses shall be performed to justify confidence that meeting the acceptance criteria can be demonstrated in the license application to operate. These analyses shall be both deterministic and probabilistic, but can be limited in scope.

3.8 Codes and standards

The goal-oriented Dutch regulatory framework does not prescribe specific codes and standards (C&S). The ANVS expects an applicant to choose which C&S will be used for which part of the design.

3.8 (1) The ANVS expects that the applicant can substantiate that the selected C&S assure a level of quality that corresponds to the level of reliability and robustness that is assumed in the safety analyses and the safety classification of the SSC.

3.9 Large Early Releases

Rnvk article 6; SSR-2/1 Req 20; SSR-3 Req 22; SSR-4 Req 21; WENRA Safety Objective O3

The Rnvk demands the avoidance of early radioactive releases or large radioactive releases. Furthermore, the IAEA requires the practical elimination of plant events sequences or conditions that could lead to a large or early radioactive release. Both these cases are broadly defined as demonstrating that such events are physically impossible or extremely unlikely with a high degree of confidence. As stated in Rnvk article 6.1a, ‘early’ releases are releases where there is insufficient time to implement the required emergency measures outside the nuclear installation. Here, the ANVS sets quantified expectations for what it considers to be ‘large’, ‘extremely unlikely’ and ‘high degree of confidence’.

3.9 (1) Releases that, according to the Dutch intervention values[17], would require evacuation beyond 5 km of the site boundary and sheltering beyond 20 km of the site boundary shall be considered large releases.

This requirement specifies an upper limit to be used when determining which events must be practically eliminated and is distinct from the actual effect zones. A smaller risk profile can also be demonstrated. The emergency planning zones are addressed separately as described in the national emergency plan [32].

3.9 (2) The local ‘safety region’ (veiligheidsregio) shall be consulted to determine what timeframe[18] is sufficient to implement the required emergency measures in the various scenarios.

16 This is to be understood as any separately moving, reactivity-relevant part in the specific reactor design’s means of shutdown. For example, a control rod, drum or other moving element.

17 The Dutch intervention values can be found in the national emergency plan [32].

18 For reference, a timeframe of 6 hours is used for Borssele NPP in the national emergency plan [32].

3.9 (3) When a probabilistic argument is used to support the justification[19] of practical elimination, extremely unlikely shall be considered to be less than 10^{-6} /y with a target value[20] of 10^{-7} /y for the sum of the frequencies of all events leading to large or early releases.

3.9 (4) In demonstrating compliance with this frequency criterion for large or early releases, a high degree of confidence shall be achieved by the use of source terms from the PSA/QRA, and a 95 % confidence level for weather conditions.

3.10 Safety documentation

Bkse article 6 - 9; GSR Part 4 Req 20; SSR-3 Req 1; SSR-4 Req 5

The information that an applicant must provide as part of the license application is described in articles 6 - 9 of the Bkse. As the application is published by the ANVS, a version of this information that is suitable for publication (i.e. without any confidential information) must be provided at the same time as the license application. This may be in Dutch or English. Further information on the application process can be found in the ANVS-Guideline *Handreiking Vooroverleg Nieuwbouw Kernreactoren* [33]. Upon receiving a license to construct, a licensee is required to have an authorized decommissioning plan and an authorized security package, in line with the Bkse and Rboni. However, these authorization processes are separate from the process to obtain a license to construct.

3.10 (1) An applicant shall submit a Safety Analysis Report (SAR)[21] to substantiate the claims made in the license application.

3.10 (2) The SAR shall reflect the topics as defined by the IAEA[22]. If an alternative structure is used, it shall be clearly mapped against the structure of an IAEA-style SAR.

3.10 (3) The SAR shall be supplemented with the topics of nuclear security and safeguards[23], demonstrating security by design and safeguards by design and justifying confidence that the design can comply to security and safeguards requirements during commissioning and operation.

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19 Examples of justification are probabilistic or deterministic arguments, engineering judgement, the complexity of the SSC, robust maintenance and/or testing, or a cost-benefit analysis.

20 When reaching the 'target value' the ANVS will not require additional measures to improve this value.

21 To reflect different stages in the lifetime of the facility, subsequent versions such as the Preliminary Safety Analysis Report (PSAR), Pre-Operational Safety Analysis Report (POSAR) and Final Safety Analysis Report (FSAR), can be used.

22 This includes for example, a description of how the applicant plans to handle radioactive waste.

23 For example, by expanding a SAR with chapters on cybersecurity, physical security, and safeguards.

References

No.	Reference
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Annex A Example set of acceptance criteria and postulated events

This annex presents examples of specific acceptance targets and criteria and event lists for pressurized water reactors (PWR) and boiling water reactors (BWR). As of yet, a similar list cannot yet be provided for other reactor types, though inspiration can be taken from these example lists. These event lists are not comprehensive, but present the level of detail and potential structure that would be required in the safety demonstration. In this example, the defence in depth structure uses the approach with levels 2, 3a and 3b as described in the WENRA Report on Safety Objectives for New NPPs [34]. As described in section 3.1, different defence in depth structures may be implemented, as justified by the applicant.

The generic event lists of Tables 3-1, 3-2 and 3-3 include events in defence in depth levels 2 to 3b, for PWRs, BWRs and the spent fuel pool. No events were defined for defence in depth level 4. Events specific to malicious acts are not included in this generic events list.

The events are assigned to different operating states as defined in Table 1-1 for PWRs and Table 1-2 for BWRs. The specific acceptance targets and criteria assigned to the levels of defence 2 to 3b are presented in Tables 2-1, 2-2, 2-3 for the reactor and in Table 2-4 for fuel assembly storage and handling.

The events are labeled with a letter representing the relevant affected main safety function:
- control of reactivity (R)
- cooling of the fuel assemblies (K)
- confinement of radioactive material (I)

Events that could affect the radiological safety objectives are labeled (S)

The event lists are divided into event categories.

For PWRs:
- change of secondary-side heat removal
- secondary-side heat removal leakages
- change of flow rate in the primary circuit
- pressure change in the primary circuit
- increase of reactor coolant inventory
- decrease of reactor coolant inventory
- loss of residual heat removal
- change of reactivity and power distribution
- loss of coolant within the containment
- loss of coolant outside the containment
- release of radioactive material from nuclear auxiliary systems
- loss of energy supply
- internal event
- anticipated transient without scram (ATWS)
- loss of component cooling
- loss of secondary-side heat removal

For BWRs:
- main steam or feedwater side change of heat removal
- change of flow rate in the reactor coolant system
- increase of reactor coolant inventory
- decrease of reactor coolant inventory
- loss of residual heat removal
- change of reactivity and power distribution
- loss of coolant within the containment, not isolable
- loss of coolant outside the containment
- release of radioactive material from nuclear auxiliary systems
- loss of energy supply
- internal event
- anticipated transient without scram (ATWS)
- loss of component cooling

For the fuel pool:

- Reduced heat removal from the fuel pool
- loss of coolant from the fuel pool
- loss of energy supply
- reactivity changes in the fuel pool
- events during handling and storage of fuel assemblies and heavy loads

Operating states

Table 1-1 Definition of the operating states for pressurized water reactors (PWR)

Operating state	Definition	System states (normal operation)	k_{eff}[1]
A	Nuclear power and startup operation	Power state as well as hot or intermediate shutdown state with all the automatic reactor protection functions available	≥ 0,99
B	Subcritical hot	Residual heat removal system not connected	< 0,99
C	Subcritical cold Primary circuit pressure-tight	Intermediate and cold shutdown, with the residual heat removal system in operation and the primary coolant system closed	< 0,99[2]
D	Subcritical cold Primary circuit not pressure-tight	Cold shutdown with the primary coolant system open	< 0,95
E	Refuelling	Cold shutdown with the reactor cavity flooded	< 0,95
F	Fuel assembly storage	Cold shutdown with the reactor core totally unloaded Cooling of the fuel assemblies via the spent fuel pool cooling systems	< 0,95

Table 1-2 Definition of the operating states for boiling water reactors (BWR)

Operating state	Definition	System states (normal operation)	k_{eff}
A	Nuclear power and start-up operation	Power state or start-up operation (beginning of withdrawal of control elements)	≥ 0,99
B[3]	Subcritical hot	All control elements completely inserted Residual heat removal system not connected	< 0,99
C	Subcritical cold Primary circuit pressure-tight	Intermediate and cold shutdown, with the residual heat removal system in operation and the primary coolant system closed	< 0,99[4]
D	Subcritical cold Primary circuit not pressure-tight	Cold shutdown with the primary coolant system open and reactor cavity not completely flooded	< 0,99
E	Refuelling	Cold shutdown with the reactor cavity flooded Fuel elements in reactor and in spent fuel storage pool	< 0,99[5] in reactor < 0,95 in pool
F[6]	Fuel assembly storage	Cold shutdown with the reactor core totally unloaded Cooling of the fuel assemblies via the spent fuel pool cooling systems	< 0,95

1 The safety demonstration with respect to the control of events on levels of defence 2 and 3 may result in further requirements for the keff values required according to the operating procedures (margin for event sequences to be controlled).

2 With the control elements withdrawn from the reactor core.

3 Upon start-up of the BWR, there is a direct transition from operating state C to operating state A, due to the nuclear heat-up caused by the withdrawal of the control elements.

4 In zero-load inspections, only the number of control elements is withdrawn that will ensure that criticality is avoided.

5 Not during function or subcriticality tests nor during the shutdown safety test; here, however, 2 control elements at the most not inserted.

6 In a BWR, operating state F generally only occurs in special cases (e.g. pressure test of the reactor pressure vessel).

Acceptance Criteria

Table 2-1 Specific acceptance criteria of levels of defence 2 to 3b for the reactor and the main safety function "control of reactivity"

Level of defence	2	3a	3b
Main safety function	**Control of reactivity (R)**		
Acceptance targets	**Power adjustments or reactor shutdown**[7]	**Reactor shutdown**	
Acceptance criteria	See "Cooling of the fuel assemblies" (Tab. 2-2) and "Confinement of radioactive material" (Tab. 2-3)		
Acceptance target	**Ensuring sub-criticality**		
Acceptance criteria[8] **"Amount of shutdown reactivity"**	≥ 1 %	≥ 1 %	sub-criticality k_{eff} < 0,999

Table 2-2 Specific acceptance criteria of levels of defence 2 to 4 for the reactor and the main safety function "cooling of the fuel assemblies"

Level of defence	2	3a and 3b	4
Main safety function	**Cooling of the fuel assemblies (K)**		
Acceptance targets	**Unrestricted reuse of the fuel assemblies**	**Possibility of shutdown and cooling of the reactor core**	
Acceptance criteria	For anticipated operational occurrences with respect to cooling of fuel elements there shall be 95 % probability at 95 % confidence level that departure from nucleate boiling or dry-out will be avoided. No internal melting of the fuel	$T_{Cladding}$ < 1200 °C Shutdown and coolability in the short and long term	Removal of the residual heat in the long term

7 Only operating state A

8 Acceptance criteria for the effectiveness of reactor scram (only operating state A as well as, for boiling water reactors (BWRs), also temporarily in operating state E during refuelling) and shutdown in the long term (all operating states). During refuelling (operating state E), the failure of the most effective control element to insert fast need not be postulated.

Table 2-3 Safety-related acceptance targets and acceptance criteria of levels of defence 2 to 3b for the reactor and the fundamental safety function "confinement of radioactive material"

Level of defence	2	3a	3b
Main safety function	**Confinement of radioactive material (I)**		
Acceptance target	**To maintain barrier integrity**		
Acceptance criteria	• pressure increase in containment below limits of the reactor protection system • BWR: Keeping of specified temperatures in the pressure-suppression pool • pressure in the primary system below design pressure • pressure in the primary system below pressure limits for opening of safety valves • no PCI[9] • For anticipated operational occurrences with respect to cooling of fuel elements there shall be 95 % probability at 95 % confidence level that departure from nucleate boiling or dry-out will be avoided.	• pressure in the containment below design pressure of the containment • BWR: Keeping of specified temperatures in the pressure-suppression pool • pressure in the primary system below 1.1 times the design pressure • hydrogen concentration everywhere inside the containment below ignition limit • maximum cladding oxidation must remain lower than 17 % of the cladding thickness • leakage ≤ 0.1 A[10]: integrity of the fuel rods • leakage > 0.1 A: number of damaged fuel rods ≤ 10 % • less than 1 % of the total available Zirconium inventory is allowed to react with water	• pressure in the containment below design pressure of the containment • boiling water reactors (BWR): Keeping of specified temperatures in the pressure-suppression pool • pressure in the primary system below 1.3 times the design pressure

9 Only operating states A and B (PCI: Pellet Cladding Interaction)

10 In this annex "A" refers to the cross-sectional flow area of the respective tube or pipe. In this context, "2A" would correspond to a double-ended break (two full cross-sections), and "0.1A" to a leak size equal to 10% of the cross-sectional area.

Table 2-4 Safety-related acceptance targets and criteria of level of defence 2 to 3b for fuel assembly storage and handling

Level of defence	2	3a	3b
Main safety function	**Control of reactivity (R)**		
Acceptance target	Ensuring sub-criticality		
Acceptance criteria **“neutron multiplication factor keff”**	< 0,95[11]	< 0,95	< 0,999
Main safety function	**Cooling of the fuel assemblies (K)**		
Acceptance targets	Limitation of the pool water temperatures to values which limits release of volatile radioactive substances from the pool water	Limitation of the pool water temperatures to values below the design temperature of the pool	Limitation of the pool water temperatures to values which ensure pool integrity
	Sufficient water coverage for ensuring the required inlet condition for the pool pumps	Sufficient water coverage for ensuring fuel assembly cooling	Sufficient water coverage for ensuring spill or evaporation cooling (maintenance of fuel rod integrity)
Acceptance criteria	pool water temperature ≤ 45°C	pool water temperature ≤ 60°C	pool water temperature ≤ 80°C

11 A coolant density that leads to the largest neutron multiplication factor and being possible under the given circumstances shall be assumed. The demonstration of criticality safety shall be based on the assumption that the coolant is pure water.

Generic event lists

The first column of the event lists gives the number of the event. For numbering the general listing Xyx; X denotes whether it concerns: pressurized water reactors (D), boiling water reactors (S), or the fuel pool (B), y denotes the level of defence, and x denotes the consecutive number of the events on the respective level or in the respective table. This is followed by a description of the events in the next column. The following columns describe the main safety functions affected, the relevant operating states, additional explanations regarding the acceptance criteria and, if necessary, detailed information about supplementary boundary conditions or notes specific to the event.

Table 3-1 Event list power operation and low-power and shutdown operation of pressurized water reactors (PWR)

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
Level of defence 2				
Change of the secondary-side heat removal				
D2-01	Malfunction in the main steam system or in the feedwater supply system which leads to an unplanned temperature/ pressure decrease in the steam generator or primary circuit	R	A	**Note:** E.g. control fault, loss of high-pressure feedwater heater, inadvertent actuation of a main steam turbine bypass, inadvertent actuation of auxiliary steam supply.
D2-02	Malfunction in the main steam system or in the feedwater supply system which leads to an unplanned temperature/ pressure increase in the steam generator or primary circuit.	K	A-B	**Note:** E.g. turbine control faults, partially inadvertent closure of main steam isolation valves.
D2-03	Inadvertent closure of valves leading to significant changes in main steam or feedwater flow rate.	K, I	A-B	
D2-04	Turbine trip with opening of the bypass station	R, K, I	A	
D2-05	Turbine trip with delayed failure of the bypass station or without opening of the bypass station	R, K, I	A	
D2-06	Loss of main heat sink	R, K, I	A-B	
D2-07	Load rejection to auxiliary power	R, K, I	A	**Additional boundary condition:** With and without switching to off-site power supply.
D2-08	Failure of a main feedwater pump without actuation of the standby pump	R, K	A	
D2-09	Failure of all operating main feedwater pumps with and without actuation of the standby pump	R, K	A	
Change of flow rate in the primary circuit				
D2-10	Loss of a main coolant pump	R, K	A-B	
D2-11	Loss of all main coolant pumps	R, K, I	A-B	**Note:** Coastdown behaviour as per design of the reactor coolant pumps is assumed.
Pressure change in the primary circuit				
D2-12	Pressure drop due to inadvertent pressuriser spraying actuation or inadvertent valve opening	K	A-B	
D2-13	Pressure increase due to inadvertent switch-on of pressuriser heater	I	A-C	

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
Increase of reactor coolant inventory				
D2-14	Inadvertent injection or reduction of extraction rates by operational systems or safety systems	K, I	A-C	
Decrease of reactor coolant inventory				
D2-15	Inadvertent opening of a pressuriser safety valve or pressuriser relief valve for a short time	K, I	A-C	**Additional boundary condition:** • For a short time so that the rupture discs of the pressuriser relief tank remain intact. • For the pressuriser safety valve, only operating states B and C are considered.
D2-16	Malfunction in the volume control system leading to a reduction of the coolant inventory	K	A-C	
D2-17	Level drop during mid-loop operation	K	C-D	**Note:** The successful prevention of the failure of the residual heat removal pumps caused by the level drop has to be demonstrated.
D2-18	Leakages at pressuriser (in steam region)	K	A-B	**Note:** Without automatic actuation of the safety system.
Loss of residual heat removal				
D2-19	Loss of a train in operation of the residual heat removal system including cooling chain	K, I	C-E	**Additional boundary condition:** Single failure is not postulated
D2-20	Loss of all residual heat removal trains due to inadvertently triggered signals (short term)	K, I	C-E	**Additional boundary condition:** The limit values for taking the residual heat removal system into operation are not exceeded.
Change of reactivity and power distribution				
D2-21	Malfunction in the reactor power control system	R, K	A	
D2-22	Inadvertent withdrawal of the most effective control element or the most effective control element group without failure of the limitation systems	R, K	A-B	
D2-23	Inadvertent drop or insertion of one or more control elements	R, K	A	
D2-24	Inadvertent injection from a system carrying deionised water or low-borated coolant (external boron dilution; homogeneous and heterogeneous)	R	A-E	
D2-25	Most unfavourable misloading of the most reactive fuel assembly	R, K	E, A	**Additional boundary condition:** Reactor startup with misloaded fuel assembly is analysed regarding protection goal K in operating state A. Comment: • Fundamental safety function R (subcriticality) in operating state E • Fundamental safety function K in operating state A
D2-26	Non-compliance with the actuation conditions upon the start-up of a main coolant pump following 3-loop operation	R, K	A	

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D2-27	Cold water injection into the reactor coolant system from a connected system (e.g. bypass of the recuperative heat exchanger of the volume control system)	R	A-B	
Loss of energy supply				
D2-28	Loss of offsite power for less than 10 hours	R, K, I	A-E	Additional boundary condition: The restoration of the external electrical power supply has to be analysed as well.
Level of defence 3a				
Change of the secondary-side heat removal				
D3a-01	Major malfunction in the main steam system or in the feedwater supply system, leading to an unplanned temperature or pressure reduction in the steam generator or in the primary circuit	R, I, S	A-C	**Additional boundary condition:** Operationally permissible steam generator tube defects are considered. **Note:** E.g. inadvertent complete opening of main steam bypass valve, inadvertent opening of main steam safety and main steam relief valves. Relevant with regard to radiology (since no N16 detection) in state B or in state A at low power. Inadvertent opening in state B more probable than in state A due to performance of tests.
D3a-02	Major malfunction in the main steam system or in the feedwater supply system, leading to an unplanned temperature or pressure increase in the steam generator or in the primary circuit	K, I, S	A-B	**Additional boundary condition:** Operationally permissible steam generator tube damage has to be taken into account. Cases to be considered: e.g. inadvertent closing of two up to all main steam isolation valves.
D3a-03	Loss of feedwater supply	K	A-B	**Note:** This is to be understood as the loss of the main feedwater supply as well as of the installations used during startup and shutdown (startup and shutdown system or emergency feedwater system in operating mode).
D3a-04	Malfunction in the feedwater supply, leading to an impermissible increase of the coolant level in the steam generator	K	A-B	
Secondary-side heat removal – leaks				
D3a-05	Secondary-side leak or secondary-side break within the containment	R, K, I	A-C	**Additional boundary condition:** At low secondary circuit pressures, the effectiveness of the actuation due to dp/dt and / or containment pressure difference at the respective leak spectrum has to be considered.

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3a-06	Leak/break in main steam or feedwater system or other high-energy piping systems in the annulus and in the valve compartment	R, K, I, S	A-B	**Additional boundary condition:** Operationally permissible steam generator tube defects are considered for leak/break in the main steam and feedwater system. **Special consideration of:** the integrity of the containment, humidity, pressure build-up, differential pressures, temperature, jet and reaction forces, etc. with impacts affecting more than one redundancy, the integrity of safety-relevant structures of the reactor building and the valve compartment.
	Leak/break in the main steam or feedwater system downstream of the main steam isolation valve and upstream of the feedwater isolation valve	R, K, I, S	A-C	**Additional boundary condition:** Operationally permissible steam generator tube defects are considered for leak/break of the main steam line.
D3a-08	Main steam line rupture after first isolation with maximum 2A break of a steam generator tube	R, K, I, S	A-B	**Additional boundary condition:** The accidental steam generator tube rupture can be considered as a single failure in the safety analysis.
D3a-09	Inadvertent opening of a main steam safety valve with consequential 2A break of a steam generator tube	R, K, I, S	A-B	**Additional boundary condition:** The accidental steam generator tube rupture can be considered as a random failure.
Change of flow rate in the primary circuit				
D3a-10	Forced decrease of reactor coolant flow (all pumps)	R, K, I	A-B	**Note:** Fast coastdown of the main coolant pumps (see also D2-13)
D3a-11	Reactor coolant pump seizure (blocked rotor)	R, K, I	A-B	
D3a-12	Reactor coolant pump shaft break	R, K, I	A-B	
Increase of reactor coolant inventory				
D3a-13	Inadvertent injection by operational systems or safety systems in case of ineffectiveness of limitation measures provided	K, I	A-C	
Decrease of reactor coolant inventory				
D3a-14	Inadvertent level drop during mid-loop operation with consequential loss of residual-heat removal pumps	R, K, I	C-D	• Fundamental safety function R affected due to reflux condenser mode in State C. • Fundamental safety function B is relevant for operating state C (primary circuit closed)
Loss of residual heat removal				
D3a-15	Loss of a train in operation of the residual heat removal system including cooling chain	K, I	C-E	**Additional boundary condition:** In contrast to event D2-19, here with consideration of the single failure criterion.
D3a-16	Shutdown of all residual heat removal trains by inadvertently triggered signals	K, I	C-E	**Additional boundary condition:** The analysis has to take the ineffectiveness of operator actions required at short notice into account (see event D2-20)
Change of reactivity and power distribution				
D3a-17	Inadvertent withdrawal of the most effective control element or control element group with loss of limitation systems	R, K	A-B	

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3a-18	Ejection of the most effective control element	R, K	A-B	
D3a-19	Misloading of the reactor core with more than one fuel assembly	R	E	
D3a-20	Drop of a fuel assembly on the reactor core	R	E	**Additional boundary condition:** Verification of subcriticality for fuel assembly on the core
D3a-21	Inadvertent injection from a system carrying deionised water or low-borated coolant with loss of limitation systems or preceding procedures (external boron dilution; homogeneous and heterogeneous)	R, K	A-E	**Additional boundary condition:** The following is considered: • all possibilities and amount of an influx of demineralised water, • operator error or inadvertent filling of tanks, • input from connected systems via heat exchanger tubes, seals and / or valve seat leakages, and • inadvertent injection into the primary circuit. • feedwater injection during shutdown under loss of offsite power conditions after steam generator tube rupture. It shall be demonstrated that reactivity changes due to injection of ionised water into the reactor coolant system remains limited to such values where • for an initially critical reactor the safety-related acceptance target for the reactivity accident according to Table 3.1b and Table 3.1c and • for an initially subcritical reactor the amount of shutdown reactivity required according to Table 3.1a are complied with.
D3a-22	Formation of low-borated areas in the primary circuit (internal boron dilution)	R, K	A-C	**Additional boundary condition:** Potential sources of formation of low-borated areas shall be investigated. Causes may be, e.g., • reflux condenser operation after small LOCA under consideration of the inserted control elements (under consideration of "Safety requirements for Nuclear Power Plants" subsection 3.2 (6)) and the time-dependent xenon concentration, and • shutdown with three circuits and secondary-side isolated steam generator and injection of low-borated coolant after restart of natural circulation. It shall be demonstrated that reactivity changes due to injection of ionised water into the reactor coolant system remain limited to such values where for an initially subcritical reactor the amount of shutdown reactivity required according to Table 3.1a is complied with.

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3a-23	Subcooling transients due to leak or break of main steam or feedwater line	R, K	A-B	**Specification of the acceptance criteria:** • Recriticality is only permissible in the case of leaks in the main steam line with high and rapid cooldown of the primary circuit if the criteria for fuel assembly cooling are fulfilled. • The leak size leading to the highest degree of subcooling has to be identified.
Loss of coolant within the containment				
D3a-24	Small leak within the containment	R, K, I, S	A-B	**Additional boundary condition:** Reflux condenser mode shall be considered (see D3a-23). **Note:** Characteristic feature: Secondary-side heat removal necessary for the control of this postulated single initiating events
D3a-25	Medium leak within the containment (leak cross section ≤ 0.1 A)	R, K, I, S	A-B	**Note:** Characteristic feature of the medium leak: Heat removal via leak sufficient => secondary-side heat removal for control of this postulated single initiating event not generally necessary.
D3a-26	Large leak within the containment (leak cross section > 0.1 A)	R, K, I, S	A-B	**Additional boundary condition:** The double-ended break of a main coolant line (“2A break“) determines the dimensioning of the emergency core cooling and residual heat removal system, the pressure design of the containment, the design of the pump flywheels against failure due to overspeed and the failure resistance of all safety-relevant components in the containment required for the control of accidents. **Specification of the acceptance criteria:** Subcriticality in the short term without taking the control elements into account unless effectiveness of the control elements has been demonstrated, and in the long term without taking the control elements into account.
D3a-27	Leak in the pressuriser steam space without reaching the containment pressure criterion	R, K, I, S	A-B	**Note:** With automatic actuation of the safety system.
D3a-28	Leak at the connecting nozzle of the main coolant line on reactor pressure vessel	K	A-B	**Additional boundary condition:** • It shall be demonstrated that impermissible impacts on the structure of the reactor cavity and the anchoring of the reactor pressure vessel are practically eliminated. • The consequences of an event regarding sufficient coverage of sump suction lines with coolant in case of considered dead volumes of the reactor cavity shall be considered.
D3a-29	“20 cm^2” leak in reactor pressure vessel below upper edge of the core	R, K, I, S	A-B	**Additional boundary condition:** The leak size of 20 cm^2 is design-relevant for the flow-off conditions at the biological shield and the maintenance of its safety function.

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3a-30	Leak in RPV closure head area	R, K, I, S	A-B	**Additional boundary condition:** In connection with the control of this event, it also has to be demonstrated in particular that the sufficient draining of the coolant into the containment sump is ensured, also considering the routine operational processes during and after plant standstills, i.e. a sufficiently dimensioned connection between the reactor cavity and the sump in operating states A and B must be ensured.
D3a-31	Leak due to faulty maintenance or switching failures at the primary circuit	K, I, S	C-E	**Additional boundary condition:** • The leak size is determined by the largest free cross section in the lines connected with the primary circuit or its components (e.g. manholes). • The analysis shall consider that in case of an incident a fuel assembly is transported in the most unfavourable position. Here, the acceptance criterion is to maintain the cladding tube integrity. • Requirement for emergency cooling effectiveness; limited availability of safety systems (e.g. reactor protection) shall be considered.
D3a-32	Inadvertent opening and / or stuck-open of a pressuriser safety valve or pressuriser relief valve, e.g. during functional tests	K, I	A-C	**Additional boundary condition:** The limited availability of safety systems (e.g. reactor protection) is considered.
D3a-33	Failure of a steam generator tube (larger than operationally permissible leakages and up to max. 2A)	K, I, S	A-B	**Additional boundary condition:** The event shall be investigated with and without reaching the limit value of the main steam activity regarding actuation of the reactor protection system. Without actuation, e.g. at small thermal load, zero load or 3-loop operation.
D3a-34	Small leak loss of coolant accident in external systems (up to 50 mm diameter)	R, K, I, S	C-E	
D3a-35	Intermediate break and large break loss of coolant accident (up to the surge line break in states A and B)	R, K, I, S	A-B	
D3a-36	Rupture of two steam generator tubes in one steam generator	K, I, S	A-B	**Additional boundary condition:** Leak size: up to 2A of an exchanger tube.
Loss of coolant outside the containment				
D3a-37	Leak in residual heat removal system in rooms between containment and surrounding building during residual heat removal operation	K, I, S	C-E	**Additional boundary condition:** Spiking effect shall be considered.
D3a-38	Leak/break in heat exchangers carrying primary coolant in case of demand	K, I, S	A-E	**Additional boundary condition:** Leak size: up to 2A of an exchanger tube.
D3a-39	Loss of coolant from the containment via systems connected to the reactor coolant pressure boundary	K, I, S	A-C	

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3a-40	Leaks in systems with flooding potential in the rooms between containment and surrounding building	K, I, S	A-E	**Additional boundary condition:** All relevant sources from leaks and containment failure of systems and devices in the annulus, in particular the containment sump suction line, shall be considered.
Release of radioactive material from nuclear auxiliary systems				
D3a-41	Leak in the volume control system outside the containment	S	A-F	**Additional boundary condition:** Spiking effect shall be considered.
D3a-42	Rupture of a line carrying primary coolant outside of the containment (e.g. sampling line)	S	A-F	
D3a-43	Leak/break in a pipe or break of a filter in the off-gas or gas treatment system	S	A-F	
D3a-44	Leak in container with active medium	S	A-F	**Additional boundary condition:** • The container with the largest radiological hazard potential shall be identified. • Analysis also has to cover container failure due to an earthquake.
Loss of energy supply				
D3a-45	Long term loss of offsite power (> 10h)	R, K, I, S	A-C	**Additional boundary condition:** Operationally permissible steam generator tube leakages shall be considered.
Internal event				
D3a-46	Potential activity release as a result of plant-internal fires (including filter fires) or explosions	S	A-F	**Additional boundary condition:** Fires and explosions affecting components and in system areas with high activity release potential have to be considered.
D3a-47	Break of a control element nozzle with control element ejection	R, K, I, S	A-B	**Additional boundary condition:** In addition to the control of the resulting leak it shall be demonstrated that the ejection of the control element does not lead to an impermissible damage of the containment. Further, it shall be demonstrated that no consequential damages of neighbouring drives occur that impair the functional safety of other control elements. If consequential damage cannot be practically eliminated, it shall be demonstrated that the acceptance criteria are also fulfilled.
Level of defence 3b				
Anticipated transient without scram (ATWS)				
D3b-01	Loss of main heat sink, e.g. by loss of condenser vacuum or closure of the main steam isolation valve with available house load supply	R, K, I	A	
D3b-02	Loss of main heat sink with unavailable house load supply	R, K, I	A	
D3b-03	Maximum increase of steam extraction, e.g. by opening of the bypass station or of the main steam safety valves	R, K, I	A	
D3b-04	Total loss of main feedwater supply	R, K, I	A	

No.	Events PWR	Main safety function	Operating state	Additionally considered comments, boundary conditions and notes
D3b-05	Maximum reduction of the coolant flow rate	R, K, I	A	
D3b-06	Maximum reactivity insertion by withdrawal of control elements or control element groups on the basis of the operating conditions “full load” and “hot subcritical”	R, K, I	A	
D3b-07	Depressurisation due to inadvertent opening of a pressuriser safety valve	R, K, I	A	
D3b-08	Maximum reduction of the reactor inlet temperature caused by a fault in an active component of the feedwater supply	R, K, I	A	
Loss of energy supply				
D3b-09	Loss of offsite power cumulated with the failure of all emergency diesel generators	R, K, I	A-F	**Remark:** mid-loop operation in state C or D DC power supply and back-up AC power supply available
D3b-10	Loss of offsite power and all onsite AC power sources	R, K, I	A-F	**Remark:** DC power supply available
Loss of component cooling				
D3b-11	Loss of the component cooling water system	R, K, I	A-F	**Remark:** • mid-loop operation in state C or D • essential service water system cooling chains
Loss of secondary site heat removal				
D3b-12	Total loss of feedwater		A	**Remark:** loss of the main feedwater, startup and shutdown, emergency feedwater systems
Loss of coolant accidents				
D3b-13	Small break loss of coolant accident (up to 50 mm diameter) and loss of the medium head safety injection trains	R, K, I	A+C	
D3b-14	Small break loss of coolant accident (up to 50 mm diameter) and loss of the low head safety injection trains	R, K, I	A+C	
D3b-15	Small break loss of coolant accident and simultaneous loss of the component cooling water system/essential service water	R, K, I	A	
D3b-16	Rupture of several steam generator tubes	R, K, I	A	**Remark:** up to 10 tubes in one steam generator
D3b-17	Steam line break and simultaneous rupture of several steam generator tubes	R, K, I	A	**Remark:** up to 10 tubes in the affected steam generator
D3b-18	Steam generator tube rupture with a main steam relief train stuck open at the affected steam generator	R, K, I	A	**Remark:** • one steam generator tube ruptured • leak size: up to 2A of an exchanger tube.

Table 3-2 Event list power operation and low-power and shutdown operation of boiling water reactors (BWR)

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
Level of defence 2				
Main-steam or feedwater-side change of heat removal				
S2-01	Malfunctions in the main steam system or in the feedwater supply system which lead to an unplanned temperature or pressure decrease in the reactor coolant system	R, K	A-B	**Additional boundary condition:** Impact on stability of the core is considered. **Note:** E.g. control fault, loss of high-pressure preheater, inadvertent actuation of a main steam turbine bypass, inadvertent actuation of auxiliary steam supply or of S&R valves.
S2-02	Malfunctions in the main steam system or in the feedwater supply system which lead to an unplanned temperature/ pressure increase in the reactor coolant system	R, K, I	A-B	**Note:** • e.g. malfunction of turbine control, inadvertent closure of individual valves. • Relevant for pressure control, in particular of the main steam bypass
S2-03	Turbine trip with opening of the turbine bypass	R, K, I	A	
S2-04	Turbine trip with delayed loss of the bypass or without opening of the turbine bypass station	R, K, I	A	
S2-05	Loss of main heat sink	R, K, I	A-B	
S2-06	Load rejection to auxiliary power	R, K, I	A	**Additional boundary condition:** With and without switch-over to offsite power supply.
S2-07	Loss of a main feedwater pump without connection of standby pump	R, K	A-B	
S2-08	Loss of all main feedwater pumps with and without connection of standby pump	R, K	A-B	
Change of flow rate in the reactor coolant system				
S2-09	Loss of individual / several / all reactor recirculation pumps	R, K	A-B	**Additional boundary condition:** Effect on neutron-physical thermal hydraulic stability of the core has to be considered.
Increase of reactor coolant inventory				
S2-10	Malfunction in the coolant level control or removal of excess water or inadvertent injection by operational systems or safety systems	R, I	A-C	**Note:** Relevant for level limitation. Prevention of water entry into the main steam line.
S2-11	Inadvertent injection with a train of the emergency core cooling systems	- - -	D	**Additional boundary condition:** • Relevant for procedures. • Only relevant in operating state D due to overfilling of reactor pressure vessel in case of not installed reactor cavity seal liner. **Specification of the acceptance criteria:** Ensuring coolant inventory in the long term.
Decrease of reactor coolant inventory				
S2-12	Leakage from RPV bottom resulting from maintenance work	K	E	**Note:** • Relevant for procedures. • Limit: leakage can be overfed by operational systems.

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
Loss of residual heat removal				
S2-13	Loss of a train, in operation or in demand, of the residual heatremoval system	K, I	C-E	**Additional boundary condition:** Single failure is not postulated.
S2-14	Shutdown of all active residual heat removal trains due to pressure increase or coolant level decrease	K, I	C-D	
Change of reactivity and power distribution				
S2-15	Withdrawal of the most effective control element or the most effective control element group	R, K	A, C, E	
S2-16	Inadvertent fast rod insertion or inadvertent insertion of a control rod	R, K	A	
S2-17	Inadvertent insertion of all control rods at high power	R, K	A	
S2-18	Maximum reduction of the reactor inlet temperature caused by a fault in an active component of the feedwater supply or by inadvertent injection by operational systems or safety systems (subcooling transient)	R, K	A	**Additional boundary condition:** Effect on neutron-physical thermal hydraulic stability of the core has to be considered.
S2-19	Malfunction in the reactor power control	R, K	A	
S2-20	Most unfavourable misloading of the most reactive fuel assembly	R, K	E, A	**Additional boundary conditions:** Reactor startup with misloaded fuel assembly shall be analysed regarding fundamental safety function K in operating state A. **Comment:** • Fundamental safety function R (subcriticality) in operating state E • Fundamental safety function K in operating state A
S2-21	Inadvertent speed increase of the reactor recirculation pumps	R, K	A-B	**Additional boundary condition:** Increase of pump speed from minimum speed with maximum speed gradient.
Loss of energy supply				
S2-22	Loss of offsite power for 10 hours or less	R, K, I	A-E	**Additional boundary condition:** The restoration of the external power supply also has to be analysed.
Level of defence 3a				
Main-steam or feedwater-side change of heat removal				
S3a-01	Major malfunction in the main steam system or in the feedwater supply system which leads to a temperature or pressure decrease in the reactor coolant system.	R, K	A-B	**Note:** In contrast to S2-01, in this case simultaneous inadvertent opening of several valves, e.g. inadvertent complete opening of main-steam bypass station, inadvertent opening of safety and relief valves.
S3a-02	Major malfunction in the main steam system or in the feedwater supply system which leads to a temperature or pressure increase in the reactor coolant system.	R, K, I, S	A-B	**Note:** E.g. inadvertent closure of all main steam isolation valves.

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
S3a-03	Loss of all main feedwater pumps without addition of standby pump	R, K	A	**Additional boundary condition:** In contrast to event S2-08, here with consideration of the single failure criterion
Increase of reactor coolant inventory				
S3a-04	Functional failure with increase of coolant level in the reactor pressure vessel or inadvertent injection by operational systems or safety systems	R, I	A-C	**Additional boundary condition:** In contrast to event S2-10, here with consideration of the single failure criterion.
Loss of residual heat removal				
S3a-05	Loss of a train, in operation or in demand, of the residual heat removal system	K, I	C-E	**Additional boundary condition:** In contrast to event S2-13, here with consideration of the single failure criterion
S3a-06	Shutdown of all residual heat removal trains due to pressure increase or coolant level decrease	K, I	C-D	**Additional boundary condition:** In contrast to event S2-14, here with consideration of the single failure criterion
Change of reactivity and power distribution				
S3a-07	Inadvertent reactivity insertion due to loss of high-pressure preheater and unavailability of limitation systems	R, K	A	
S3a-08	Withdrawal of the most effective control element or control element group with loss of limitation systems	R, K	A, B, D	
S3a-09	Ejection of the most effective control rod	R, K	A	
S3a-10	Drop out of the most effective control rod	R, K	A	**Additional boundary condition:** Drop out over the length of a latch distance.
S3a-11	Drop of a fuel assembly into the reactor core during refueling	R, K	E	
S3a-12	Drop of a fuel assembly onto the reactor core	R	E	**Additional boundary condition:** Verification of subcriticality for fuel assembly on the core.
S3a-13	Inadvertent withdrawal of control rods during loading	R, K	E	
S3a-14	Inadvertent withdrawal of a control rod during zero-power test or shutdown safety test	R, K	C, E	
S3a-15	Misloading of the reactor core with more than one fuel assembly	R	E	
S3a-16	Nuclear-thermal hydraulic instability	R, K	A	**Additional boundary condition:** The boundary conditions of the possible initiating events have to be considered. Without consideration of limiting measures. In-phase and out-of-phase oscillations have to be analysed. The effectiveness of reactor protection actions for the timely detection of neutron flux oscillations and reactor shutdown has to be demonstrated.
S3a-17	Inadvertent speed increase of the reactor recirculation pumps	R, K	A	**Additional boundary condition:** Increase of pump speed from minimum speed with maximum speed gradient without limitations.

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
Loss of coolant within the containment, not isolable				
S3a-18	Leak/break within the containment (leak cross section ≤ 0.1 A of the respective line considered)	R, K, I, S	A-B	**Additional boundary condition:** In addition to main steam and feedwater lines, all other coolant-retaining systems shall be considered.
S3a-19	Leak/break within the containment (leak cross section > 0.1 A of the respective line considered)	R, K, I, S	A-B	**Additional boundary control:** In-addition to main steam and feedwater lines, all other coolant-retaining systems shall be considered. The double-ended break of the main-steam line (2A break) has to be analysed for the design of the pressure suppression system, the reactor pressure vessel internals necessary for cooldown and core cooling, as well as the pressure design of the containment and the accident resistance of all safety-relevant systems and components necessary for accident control.
S3a-20	80 cm2 leak in RPV bottom	R, K, I, S	A-B	
S3a-21	Leak due to faulty maintenance or switching failures at the reactor coolant system	K	C-E	**Additional boundary condition:** A maximum leak resulting from faulty maintenance or switching failures is postulated. The leak size is determined by the largest free cross section in the lines connected with the reactor coolant system The analysis considers that in case of an incident a fuel assembly is transported in the most unfavourable position. Here, the acceptance criterion is the integrity of the cladding tube. **Note:** This may result in requirements for the sump function of the containment (locks included).
S3a-22	Leak in the reactor cavity seal liner	K, S	D-E	**Additional boundary condition:** The constructively possible leak cross section in case of seal failure is postulated. **Note:** Relevant for establishment of the sump function and procedures.
S3a-23	Leak in RPV bottom due to: - inadvertent pulling of a pump shaft, or - work on control rod drives or detector assemblies	K, S	E	**Note:** Where applicable, temporary requirement for the sump function of the containment until reliable function of the isolating equipment has been verified (locks included).
S3a-24	Leak in the blow-off pipe of a safety and relief valve within the gas space of the pressure suppression pool	K, I, S	A-B	
Loss of coolant outside the containment				
S3a-25	Leak/break in the main steam or feedwater system and other high-energy piping systems between containment and first isolation possibility outside the containment	R, K, I, S	A-B	**Special consideration of:** the integrity of the containment, humidity, pressure build-up, differential pressures, temperature, jet and reaction forces, etc. with impacts affecting more than one redundancy, and the integrity of safety-relevant structures of the reactor building.

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
S3a-26	Leak/break in the main steam or feedwater system within the turbine building	R, K, I, S	A-B	
S3a-27	Leak/break in an instrumentation line carrying coolant, in the reactor building	S	A-C	**Additional boundary condition:** 2A break of an instrumentation line in the reactor building that cannot be isolated for 30 min. The most unfavourable operating state is analysed with regard to radiology (spiking effect).
S3a-28	Leak/break in the reactor water cleanup system in the reactor building	S	A-E	**Additional boundary condition:** The spiking effect shall be considered.
S3a-29	Leak/break in coolers, carrying reactor coolant, in case of demand	B, S	A-E	
S3a-30	Leakage from the wetwell	K	A-B	**Additional boundary condition:** The event is relevant for the transition to residual heat removal via RHR train from RPV and flooding of reactor building.
S3a-31	Leak/break in reactor scram system in the reactor building	R	A	**Note:** Relevant for the design of the reactor scram system.
S3a-32	Leak in residual heat removal system in the reactor building during residual heat removal operation	K, I, S	C-E	**Additional boundary condition:** The spiking effect shall be considered.
S3a-33	Loss of coolant from the containment via systems connected to the reactor coolant pressure boundary	K, I, S	A-C	
Release of radioactive material from nuclear auxiliary systems				
S3a-34	Leak/break in a pipe or break of a filter in the off-gas or gas treatment system	S	A-F	
S3a-35	Leak in container with active medium	S	A-F	**Note:** • The container with the largest radiological hazard potential shall be identified. • Analysis also has to cover container failure due to earthquake.
Loss of energy supply				
S3a-36	Longterm loss of offsite power (> 10 hours)	R, K, I, S	A-E	**Additional boundary condition:** Cooldown under emergency power conditions also has to be analysed.
Internal event				
S3a-37	Potential activity release as a result of internal fires (including filter fires) or explosions	S	A-F	**Additional boundary condition:** Fires and explosions on components and in system areas with great activity release potential have to be analysed.
S3a-38	Break of a control rod nozzle with control rod ejection.	R, K, I, S	A-B	**Additional boundary condition:** In addition to the control of the resulting leak it shall be demonstrated that the ejection of the control rod does not lead to an impermissible damage of the containment. Further, it shall be demonstrated that no consequential damages of neighbouring drives occur that impair the functional safety of other control rods. If consequential damage cannot be excluded, it shall be demonstrated that the acceptance criteria are also fulfilled.

No.	Events BWR	Main safety functions	Operating state	Additionally considered comments, boundary conditions and notes
Level of defence 3b				
Anticipated transient without scram (ATWS)				
S3b-01	Loss of main heat sink, e.g. by loss of condenser vacuum or closure of the main steam bypass valve with available house load supply.	R, K, I	A	**Note:** For ATWS it is postulated that the nut follow-up movement (if available) for the control rods is effective.
S3b-02	Loss of main heat sink with unavailable house load supply	R, K, I	A	
S3b-03	Maximum increase of steam extraction, e.g. by opening of the bypass station or of the safety and relief valves	R, K, I	A	
S3b-04	Total loss of main feedwater supply	R, K, I	A	
S3b-05	Maximum reactivity insertion by withdrawal of control rods or control element rods on the basis of the operating conditions “full load” and “hot zero power condition”	R, K, I	A	
S3b-06	Maximum decrease of the feedwater temperature.	R, K, I	A	
S3b-07	Steam line isolation with available house load supply	R, K, I	A	
S3b-08	Steam line isolation with unavailable house load supply	R, K, I	A	
S3b-09	Maximum increase of feedwater flow rate	R, K, I	A	
S3b-10	Startup of the recirculation pumps with maximum speed gradient	R, K, I	A	
Loss of energy supply				
S3b-11	Loss of offsite power cumulated with the failure of all emergency diesel generators	R, K, I	A-F	**Additional boundary condition:** DC power supply and back-up AC power supply available
S3b-12	Loss of offsite power and all onsite AC power sources	R, K, I	A-F	**Additional boundary condition:** DC power supply available
Loss of component cooling				
S3b-13	Loss of component cooling water system	R, K, I	A-F	
Loss of coolant accidents				
S3b-14	Small break loss of coolant accident and simultaneous loss of the component cooling water system/essential service water	R, K, I	A	
S3b-15	Loss-of-coolant accident with failure to shut off emergency cooling after flooding of the core and failure of steam line isolation	R, K, I	A	
Loss of residual heat removal				
S3b-16	Transient with simultaneous complete loss of emergency cooling	R, K, I	A	

Table 3-3 Event list spent fuel pool

No.	Events spent fuel pool	Fundamental safety functions	Operating phase	Additionally considered comments, boundary conditions and notes
Level of defence 2				
Reduced heat removal from the fuel pool				
B2-01	Loss of a train in operation or unplanned short-term (max. 30 min) interruption of heat removal	K	A-F	-
Loss of coolant from fuel pool				
B2-02	Leakage from the spent fuel pool or loss of water from via connecting pipes (corresponding as a maximum to a cross-sectional area of DN25)	K	A-F	-
Loss of Energy supply				
B2-03	Loss of offsite power for 10 hours or less	K	A-F	-
Reactivity changes in the fuel pool				
B2-04	Disturbances in the boron concentration	R	A-F	-
B2-05	Most unfavourable misloading of the fuel pool or transport and storage cask with a most reactive fuel assembly	R	A-F	-
Level of defence 3a				
Reduced heat removal from the fuel pool				
B3a-01	Loss of two trains of the fuel pool cooling system for a longer period (> 30 min.)	K	A-F	**Additional boundary condition:** For the safety demonstrations, grace times and repair possibilities can be taken into account.
Loss of coolant from fuel pool				
B3a-02	Loss of coolant from the spent fuel pool through leaks in the pool or via connecting pipes (corresponding to a cross-sectional area of > NB25)	K, I	A-F	**Additional boundary condition:** Maximum leak cross-sectional area: area of the largest connecting pipe.
B3a-03	Leak in the reactor cavity or the setdown pond for steam separators at opened refueling slot gate	K, I	E	**Additional boundary condition:** Effects of leaks that may occur in the reactor coolant system during refuelling also have to be considered.
B3a-05	Internal leak in heat exchangers of the fuel pool carrying coolant	K, I, S	A-F	-
B3a-06	small leak loss of coolant accident in external systems (up to 50 mm diameter)	I, S	A-F	-
Reactivity changes in the fuel pool				
B3a-07	Water/steam ingress in the spent fuel dry storage facility	R	A-F	Specification of the demonstration criteria keff < 0.98
B3a-08	Geometry changes due to earthquake (fuel pool, spent-fuel dry storage facility)	R, K, I	A-F	-
B3a-09	Drop of a fuel assembly into the fuel pool	R	A-F	**Additional boundary condition:** A dropped-down fuel assembly is lying on the storage racks or standing directly adjacent to a storage rack.
B3a-10	Misloading of the fuel pool or the transport and storage cask with more than one fuel assembly	R	A-F	-
B3a-11	Boron dilution in the fuel pool	R	A-F	-

No.	Events spent fuel pool	Fundamental safety functions	Operating phase	Additionally considered comments, boundary conditions and notes
Events during handling and storage of fuel assemblies and heavy loads				
B3a-12	Fuel assembly damage during handling	I	A-F	**Additional boundary condition:** Damage of all fuel rods at exterior side of a fuel assembly is postulated. **Note:** The analysis serves to verify that the release into the environment resulting from the release of radionuclides in the containment without loss of coolant is sufficiently limited.
Loss of energy supply				
B3a-13	Long term loss of offsite power (> 10 h)	R, K, I	A-F	-
Level of defence 3b				
B3b-01	total loss of the spent fuel pool cooling system	S	A-F	-
B3b-02	Loss of offsite power cumulated with the failure of all emergency diesel generators	R, K, I	A-F	**Additional boundary condition:** DC power supply and back-up AC power supply available
B3b-03	Loss of offsite power and all onsite AC power sources	R, K, I	A-F	**Additional boundary condition:** DC power supply available

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Koningskade 4 | 2596 AA Den Haag
Postbus 16001 | 2500 BA Den Haag
www.anvs.nl

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