Note:
This is a translation of the RSK statement entitled
"Bewertung der Umsetzung von RSK-Empfehlungen im Nachgang zu Fukushima"
In case of discrepancies between the English translation and the German original, the original shall prevail.

RSK Statement
(496th meeting of the Reactor Safety Commission (RSK) on 06 September 2017)

Assessment of the implementation of RSK recommendations in response to Fukushima

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Part A Introduction

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After the accident at Fukushima, the RSK dealt with the robustness of the safety concept of the nuclear power plants in Germany. The questions were

- whether and to what extent the safety concept was still "sustainable" even under higher loads and stresses than assumed for the design and

- with which measures robustness could be further increased.

As a result of these consultations, several RSK statements and recommendations were issued. The purpose of this RSK statement is to assess how the recommendations of the RSK have been taken up by the plant operators and whether the concepts presented are suitable to meet the RSK recommendations. It is not the subject of this statement to evaluate whether the individual recommendations have been implemented in a plant-specific manner according to the concepts presented by VGB.

For practical reasons, the assessment is carried out separately for PWRs and BWRs. Since not all recommendations apply to PWRs and BWRs in the same way, the following sections deal with the recommendations in a type-specific manner. The assessment of the implementation of RSK recommendations for the German PWR plants is presented in Part B from page 4 and for the German BWR plants in Part C from page 41.

In the following, the recommendations of the RSK are listed consecutively with a short description, divided into PWR (Part B) and BWR (Part C) and supplemented by an excerpt from the corresponding RSK statement or RSK recommendation. Afterwards, the respective concept of VGB for the implementation of the recommendations is summarised and assessed. Topics that were dealt with in more detail by the RSK within the framework of the robustness assessment (earthquakes, internal flooding, load crash, leak tightness of the reactor coolant pump (RCP) seals in case of failure of the primary heat sink) are additionally presented in more detail in separate appendices.

The recommendations discussed below are taken from the following documents:

1 RSK statement "Plant-specific safety review (RSK-SÜ) of German nuclear power plants in the light of the events in Fukushima I (Japan)" (*Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Kernkraftwerke unter Berücksichtigung der Ereignisse in Fukushima-I (Japan))*, 437th RSK meeting from 11 to 14 May 2011 [1]

2 Recommendations of the RSK on the robustness of the German nuclear power plants (*RSK-Empfehlung zur Robustheit der deutschen Kernkraftwerke*) from the 450th RSK meeting on 26/27 September 2012 [2], including the RSK statement "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" (*Mindestwert von 0,1 g (ca. 1,0 m/s²) für die maximale horizontale Bodenbeschleunigung bei Erdbeben*) from the 457th RSK meeting on 11 April 2013 [6]

3 RSK statement "Loss of the ultimate heat sink" (*Ausfall der Primären Wärmesenke*) from the 446th RSK meeting on 05 April 2012 [3]

4 RSK statement "Specification of requirements related to 10-hours self-sufficiency in the event of external man-made hazards (man-made hazard conditions)" (*Konkretisierung von Anforderungen im Zusammenhang mit der 10 h-Autarkie bei zivilisatorischen Einwirkungen von außen (Notstandsfälle)*) from the 459th RSK meeting on 20 June 2013 [4]

5 RSK statement "Assessment of the coverage of extreme weather conditions by the existing design" (*Einschätzung der Abdeckung extremer Wetterbedingungen durch die bestehende Auslegung*) from the 462nd RSK meeting on 6 November 2013 [5]

6 RSK statement "Hydrogen release from the containment" (*Wasserstofffreisetzung aus dem Sicherheitsbehälter*) from the 475th RSK meeting on 15 April 2015 [7]

Notes:

- The recommendations of the RSK statement "Plant-specific safety review (RSK-SÜ) of German nuclear power plants in the light of the events in Fukushima I (Japan)" from the 437th RSK meeting from 11 to 14 May 2011 [1] are not addressed in the following for PWRs, since they were generally further specified by the RSK recommendations or statements listed above and are dealt with here in specified form; for BWRs, they are addressed insofar as they are still relevant. Regarding the recommendation on attacks on software-based systems, it had already been stated in [1] that further treatment would take place in another procedure. For the recommendation on blast waves, it was stated in [2] that the yet outstanding point concerns only the verification of distances to potential explosion causes, which is not to be carried out generically but on a plant-specific basis.

- The investigations to assess the robustness with regard to an aircraft crash have not yet been completed and are therefore dealt with separately.

Certain individual recommendations that have already been dealt with by the RSK are explained in [8].

Part B Assessment of the implementation of RSK recommendations in response to Fukushima: PWR plants

In the following, the concepts of VGB for the implementation of the recommendations in the German PWR plants are assessed from the RSK's point of view.

B.1 Recommendations of the RSK on the robustness of the German nuclear power plants from the 450th RSK meeting on 26/27 September 2012 [2], including the RSK statement "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" from the 457th RSK meeting on 11 April 2013 [6] 5

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B.1 Recommendations of the RSK on the robustness of the German nuclear power plants from the 450th RSK meeting on 26/27 September 2012 [2], including the RSK statement "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" from the 457th RSK meeting on 11 April 2013 [6]

B.1.1 Systematic analysis of the robustness of German nuclear power plants

Recommendation of the RSK, [2], Part 1

To ensure the vital safety functions in case of beyond-design-basis external or internal hazards, a systematic analysis should be conducted to identify potential for increasing robustness appropriately, for which supplementary measures should be designed where required.

Thus, the design margins in the existing safety installations or emergency systems are to be assessed with regard to whether and when the required safety function of safety installations or emergency systems may be endangered in case of increased (beyond-design-basis) assumptions on external and internal hazards. These analyses can be performed by means of engineered judgements.

On this basis it is then to be assessed whether an increase of robustness is possible

- *either by appropriate measures to upgrade existing safety installations or emergency systems or*
- *by existing or additional accident management measures to ensure vital safety functions in case of expected failure of safety installations or emergency systems. These accident management measures must not lose their operability by those impacts that in the analyses have led to a functional failure of safety installations or emergency systems.*

With the accident management measures to ensure the vital process-based safety functions designed in this way it is then possible to derive the tasks for auxiliary functions and thus for appropriate accident management measures to compensate for possibly occurring failures in the safety-related auxiliary functions (in particular electrical energy supply and service water supply).

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 12 to 24, 62 to 65, 67 and 76,
- VGB presentation of 04 November 2014 [14],
- Brokdorf Nuclear Power Plant (KBR), Recommendations of the RSK on the robustness of the German nuclear power plants from the 450th RSK meeting on 26/27 September 2012, Statement to the Environment Ministry in Schleswig-Holstein (MELUR) of 12 August 2013, pages 6 to 8 and Appendix 1 [19], hereinafter referred to as KBR-MELUR letter of 12 August 2013.

According to the presentation of VGB, the following procedure was applied to examine the robustness and to derive robustness-increasing measures:

- Consideration of the range of relevant scenarios with the potential for cross-redundancy failures of safety equipment:

  - beyond-design-basis external and internal hazards (aircraft crash, earthquake, flood, load crash, flooding of the annulus),

  - postulated beyond-design-basis failures of safety equipment (total failure of the three-phase power supply including assumed long-term failure of the external grids due to design earthquake or design basis flood, total failure of primary-side cooling due to assumed blockage of the safety-relevant intercoolers).

- Determination of conservative assumptions relating to the boundary conditions for the relevant scenarios

  - loss of external power supply for at least 10 h, for the external hazard cases for seven days.
  - help from outside with technical equipment in the natural external hazard cases only after three days.

- Consideration of representative plant states (power operation, low-power and shutdown operation with variants that differ significantly in terms of the boundary conditions for the analysis).

- Establishment of time ranges that differ in terms of opportunities to implement manual measures or substitute measures (< 2 h, 2 to 10 h, 10 to 72 h, three to seven days)

- Determination of the "vital functions" in order to avoid serious impacts on the environment in such assumed scenarios and different plant states.

- Assessment of the functionality of existing safety and emergency systems to ensure vital functions in these scenarios.

- If the operability of the existing safety and emergency systems is in doubt in the light of the beyond-design-basis assumptions, derivation of supplementary verifications or measures (in the case of safety equipment and existing emergency measures or those to be supplemented).

Based on the results of this robustness analysis, measures with corresponding effectiveness conditions have been derived (VGB presentation of 4 November 2014 [14], slides 22 to 24), in particular:

- Mobile diesel generators for SBO conditions, storage in the emergency feed building or outside, operation outside.

- Additional SG feeding, e.g. via additional mobile pump stored in the emergency feed building/outside, operation inside/outside.

- Mobile diesel generator for the operation of an emergency residual-heat removal chain, storage and operation with feed point accessible during design floods and earthquakes.

- Optimisation of emergency feedwater pool replenishment:

    - inventory make-up of the emergency feedwater pools via connection nozzles on/in the emergency feedwater building with the help of a mobile fire-fighting pump, kept in flood-proof storage,

    - suction e.g. from demineralised-water tanks, security trench, river, cooling tower basin.

- Optimisation of continuous diesel operation:

    - increased lubricant supplies secured for more than three days available,

    - procedure to conserve fuel by switching off individual emergency diesels, supply of important consumers via cross-connections if necessary,

    - mobile pump-over station for fuel and contractually secured delivery and refuelling of the diesels in external hazard scenarios.

- Flood protection of the annulus

    - Procedure to prevent impermissible flooding by early manual measures in case of water accumulation in several annulus quadrants (reactor scram, shutdown of auxiliary cooling water pump),

    - creation of additional options for pumping out medium, for example with mobile shortened cooling chain equipment or with fire-fighting equipment and existing mobile submersible pumps, or for draining water from the annulus.

- If necessary, extended flood protection, for example flood-proof connecting options to demineralised-water and fuel oil tanks and appropriate installation/connections for mobile fire-fighting/emergency pumps as well as mobile diesel generators, temporary precautions, such as stop logs to protect safety-relevant buildings.

- Mobile shortened cooling chain (only for systems without cell coolers or without cooling via well water), suction from additional water reservoirs that may still be available, depending on the scenario, for example security trenches or cooling tower basins, with a mobile pump for perfusing the residual-heat removal coolers and the cooling loads of the pool cooling or residual-heat pumps, return to the open air.

- Procedure for resuming residual-heat removal mode during mid-loop operation or removal of the decay heat by evaporation and make-up feeding from the accumulators or the spent fuel pool.

- Procedure for the removal of the decay heat in the spent fuel pool by evaporation and make-up feeding of coolant inventory, make-up feeding from the demineralised-water tanks, the emergency feedwater tanks or the fire-fighting system, among others with mobile pumps.

- Procedure for sump suction with feeding into the spent fuel pool in the event of a loss of coolant from the flooded reactor pool that cannot be shut off and the slot gate drawn.

- Additional emergency measure for containment pressure limitation and thus limitation of the boiling temperature in the spent fuel pool at approx. 120 °C in case of spent fuel pool cooling by evaporation and feeding (for this purpose, proof of the integrity of the spent fuel pool for a temperature of at least 130 °C). [30]

Following the robustness analysis that was carried out in the plants until summer 2013 and is outlined above, further robustness-increasing measures have been derived on the basis of RSK statements adopted afterwards [4], [5], [7] and on the basis of additional discussions in meetings of the RSK Robustness WG with VGB. These include in particular:

- Option of feeding the reactor coolant system via an emergency measure with use of the volume control system in case of an assumed unavailability of the other feed functions (not for plants with cell coolers) [18], s. in the following Chap. B.1.5 "Concretisation of the recommendation on annulus flooding",

- Option for an emergency measure using existing equipment for circulating the annulus atmosphere (avoiding increased local H_2 concentrations) and controlled exchange of the atmosphere (limiting the H_2 concentration) in the event of core damage states with entry of H_2 into the annulus [18], s. in the following Chap. B.4 "RSK recommendation "Hydrogen release from the containment" from the 475th RSK meeting on 15 April 2015“,

- Administrative measures in connection with extreme weather conditions, s. in the following Chap. B.3 "RSK statement "Assessment of the coverage of extreme weather conditions by the existing design" from the 462nd RSK meeting on 6 November 2013",

- Robustness in the event of an earthquake during low-power and shutdown operation, s. in the following Chap. B.1.3 "Concretisation of the recommendations on earthquake".

Assessment by the RSK

The procedure for the robustness analysis has been presented to the RSK in the form of key points. A more detailed description of the procedure for the systematic robustness analysis is given in a report [19] prepared for one plant, which is supposed to be representative for all PWR plants with regard to the procedure. The system for deriving supplementary verifications and measures meets the requirements of the RSK.

This assessment refers to the basic approach of the robustness analysis. Measures derived from this analysis are discussed in detail below. Insofar as specific measures proposed by the RSK are not to be implemented, this is also discussed below.

B.1.2 Aiming for robustness level 1 or degree of protection 2

Recommendation of the RSK [2], Part 1

The RSK considers it appropriate that ultimately, at least robustness level 1 or at least degree of protection 2 (man-made hazards) should be aimed at.

Concept of VGB for the implementation of the recommendations

The robustness levels shown in the RSK-SÜ were not used by VGB to assess robustness.

From the point of view of VGB, however, the analyses carried out, where appropriate with consideration of additional robustness-increasing measures, have demonstrated a high degree of robustness against beyond-design-basis events.

Assessment of the RSK

Due to the systematic robustness analysis and the measures derived from it, a further increase in robustness was achieved in comparison to the 2011 safety review.

Even though VGB did not pursue the assignment to robustness levels, the RSK concludes on the basis of the available information that if the concepts presented by VGB are implemented appropriately for the specific plant and if the RSK's comments in this statement are taken into account, a level of robustness will be achieved as far as possible which is in the range of robustness level 1, and in some cases even higher. Even regarding the events or features for which there was no assignment to robustness levels in 2011 (see e.g. B.3 "RSK statement "Assessment of the coverage of extreme weather conditions by the existing design" from the 462nd RSK meeting on 6 November 2013" and B.4 "RSK recommendation "Hydrogen release from the containment" from the 475th RSK meeting on 15 April 2015"), there is a significant increase in robustness with the implementation of the concepts, i.e. even if the design requirements are exceeded, there will still be significant reserves to prevent serious effects on the environment.
Note: Degrees of protection as defined in [1] for man-made hazards are not addressed here because the corresponding man-made hazards are not addressed in this statement.

B.1.3 Concretisation of the recommendations on earthquakes

Recommendation of the RSK, [2], Part 1, and [6]

a) *For plants for which results of probabilistic seismic safety analyses are available, the robustness with respect to beyond-design-basis earthquake effects has to be assessed. The assessment has to be based on the HCLPF (High Confidence of Low Probability of Failure) values of the structures and equipment required to ensure vital safety functions.*

b) *For plants for which no results of probabilistic seismic safety analyses are available, an applicability assessment may be chosen (possibly supported by an inspection of the plant by a commission of experts)* *based on results according to a) for the assessment of robustness against beyond-design-basis earthquake impacts.*

In terms of robustness, a simultaneous occurrence of short-term operating states during low-power and shutdown operation and an earthquake is to be considered additional to the existing requirements of the rules and regulations. For the analysis of robustness, it must be shown that the design earthquake will not lead to any significant effects in the surrounding area during short-term operating states.

Particular attention should be paid to situations in which vital safety functions may be impaired due to the circumstance

- *that changed mass distributions (e.g. filled reactor cavity during reloading) in the reactor building lead to higher earthquake-induced loads on safety-relevant equipment and building structures than during power operation;*

- *that certain equipment is used exclusively (e.g. reactor cavity seal liner in a BWR) or in a specific mode of operation (e.g. refuelling machine outside the parking position) in low-power and shutdown operation for which no specific or no enveloping design verification against earthquakes is available;*

- *that equipment (e.g. fuel assembly transport casks, heavy components) and operating agents (lubricating oils and solvents) that are brought into the plant or handled there during low-power and shutdown operation lead to earthquake-related damage to safety-relevant equipment and structures;*

- *that safety-relevant measures and equipment that are required for the control of the earthquake effects are only available to a limited extent in the event of earthquakes during low-power and shutdown operation (e.g. disconnection of residual-heat removal trains, short-term manual measures).*

For plants that are permanently in low-power and shutdown operation, the robustness verification for prolonged states also has to be carried out for beyond-design-basis earthquakes according to a) and b) (see above).

With regard to a PGA (peak ground acceleration) minimum value of 0.1 g for the maximum horizontal ground acceleration, the RSK stated in [6]:

Proof of compliance with the IAEA requirement for a PGA value of 0.1 g can be provided by reassessing the seismic resistance of the affected plants using the methods of IAEA Guide NS-G-2.13. By means of the "Seismic Margin Assessment" method mentioned in NS-G-2.13 (if necessary, using data from an existing seismic PSA), it would have to be demonstrated for plants with a maximum ground acceleration of < 0.1 g that the plant is also sufficiently resistant to a ground acceleration of 0.1 g. This procedure is already included in principle in the recommendations of the RSK statement on robustness [2]. Should a PGA value < 0.1 g be determined site-specifically at an assumed intensity corresponding to robustness level 1, the RSK recommends determining the margins available in the design for an assumed PGA value of 0.1 g.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013, slides 29 to 31 and 68 [9],
- VGB presentation of 25 June 2015, 108^{th} AST meeting [20],
- VGB presentation of 17 July 2015 [23],
- KBR-MELUR letter of 12 August 2013 [19], pages 8 to 12,
- VGB, e-mail Earthquake Robustness, 02 October 2015 [29],
- VGB letter of 03 March 2016 [30],
- VGB, e-mail Split Dowel, 20 October 2015 [31].

VGB focused on the following aspects (for details, see Appendix Earthquake, from page 74):

- principles of seismic design applied to NPPs with power operation during the period of their construction,
- criteria for the assessment of robustness against earthquakes,
- assessment of design margins against earthquake loads for the plants in operation,
- Seismic Probabilistic Safety Analysis (SPSA),
- applicability of results from plants studied in more detail to other plants,
- capability of plants with a design earthquake and < 0.1 g horizontal peak ground acceleration (PGA) to withstand horizontal accelerations up to 0.1 g.

The operators conclude that for the plants in power operation, the practice of designing against seismic impacts has resulted in significant margins.

According to the operators, a complete seismic probabilistic safety analysis (SPSA) was carried out for a total of three PWR plants. For the two plants with detailed analyses, a margin of at least a factor of 2 related to the maximum horizontal peak ground acceleration (PGA) of an earthquake with an occurrence frequency of approx. 10^{-5}/a resulted. However, not all actually existing margins had been determined.

The results of the SPSAs carried out were applicable to the other PWR plants in power operation due to the similarity of the plant and design concepts, which was also verified by specific individual studies and plant inspections.

Since a factor of 2 in the horizontal peak ground acceleration (PGA) corresponds to about one order of magnitude in the occurrence frequency (see Appendix D Earthquakes), it can thus be assumed that all PWRs for power operation have design margins with which the loads from earthquakes that are at least one order of magnitude lower in occurrence frequency than for an earthquake with an occurrence frequency of approx. 10-5/a will also be shed.

According to the operators, this also means that plants for which a PGA value of 0.5 m/s² or 0.7 m/s² was originally used for the design are robust for a PGA value of 1.0 m/s² (corresponding to approx. 0.1 g).

Furthermore, a systematic assessment of the robustness with respect to the event "earthquake in low-power and shutdown operation" was carried out representatively for some PWR plants. For this purpose, design documents were reviewed with regard to special conditions of low-power and shutdown operation/FA change (including load transfer with modified mass arrangements), and plant inspections were performed. The main focus of the inspections was on the installation of components/internals/auxiliary tools and the resulting hazard potential in the event of an assumed earthquake. Taking into account some optimisation measures during refuelling outages (e.g. securing against slipping), even earthquake scenarios in low-power and shutdown operation with an occurrence frequency of the combination of 10^{-6}/a would not lead to a loss of vital functions. The results of the assessment should be implemented in all plants.

Assessment by the RSK

The assessment is limited to the questions of whether the approach presented by the operators for all PWR plants together corresponds to the recommendations of the RSK in [2] and whether the results of the corresponding investigations are basically plausible. An assessment of the plant-specific implementation is not carried out. Overall, the RSK arrives at the following assessments:

- The methodological approach chosen by the operators for the designation of margins is in line with the RSK recommendation. The procedures described by the operators for the determination of margins (by means of an SPSA) are in line with standard international practice.

- The reserves indicated by VGB for plants with SPSA (presented were GKN II, KWG, additionally SPSA available for KKP 2) with regard to the PGA values are about a factor of 2 above the values taken as a basis during construction. This corresponds approximately to an impact increased by one intensity level compared to the design impact. At the same time, according to VGB, the design earthquakes for all sites are comparable to or stronger than the earthquake that has been determined probabilistically according to the current state of the art with a frequency of occurrence of 10^{-5}/a.

  However, to the RSK's knowledge, the current design earthquake (with reference to a frequency of occurrence of 10^{-5}/a) for the KWG plant is above the values taken as a basis during construction. [41]

- Based on the available information, the RSK is of the opinion that the achievement of robustness level 1 is plausible for plants with SPSA - except for the KWG plant. Against the background of the reportable event ME 16/063 "Defective connecting bolts on ventilation duct supports" in KKP-2 in December 2016, however, the RSK is currently still conducting a generic discussion on the robustness of the SPSA results.

- From the RSK's point of view, the elements of the applicability assessment presented by VGB are suitable to fulfil the implementation of the RSK recommendation for those plants for which no seismic PSAs were prepared. Applicability was ensured by targeted individual observations and plant-specific inspections with a focus on those components that have the lowest seismic capacities on the basis of experience with the SPSAs carried out (such as electrotechnical components, diesel periphery, instrumentation and control cabinets). Since the occurrence frequencies of the design earthquakes assumed for these systems are $\leq 10^{-5}$/a, a margin of a factor of 2 results for these systems compared to an earthquake with an occurrence frequency of 10^{-5}/a.

- For three of the PWR plants still in power operation, PGA values of 0.5 m/s² and 0.7 m/s² (approx. 0.05 g and 0.07 g) had been assumed in the original design. An SPSA for the KWG plant with 0.5 m/s² resulted in a load-bearing capacity for the relevant functions for at least 1 m/s² (approx. 0.1 g). From the point of view of the RSK, the basic considerations put forward by VGB for an applicability of this result to other PWR plants in power operation with a design PGA value < 0.1 g are comprehensible.

- In the opinion of the RSK, the methodical approach for assessing the robustness of PWR plants against seismic impacts in low-power and shutdown operation and during short operating states is also suitable to fulfil the implementation of the corresponding RSK recommendation. The prerequisite is that the walkdowns required for the assessment are carried out plant-specifically and that it is ensured that the conclusions drawn from them are observed in all further LP&S states.

B.1.4 Concretisation of the recommendation on flooding

Recommendation of the RSK [2], Part 1

If a level at which a risk to vital safety functions has to be considered cannot be excluded on the basis of the site-specific conditions, the criteria from the safety review [1] for at least Level 1 have to be applied. Alternatively, site-specific justification can be provided that a postulated run-off quantity determined by extrapolation of existing probabilistic curves to an occurrence frequency of 10^{-5}/a will not lead to a loss of vital safety functions. An analogous procedure applies to tidal sites. The methodology applied here has to be explained in a comprehensible manner.

The buoyancy safety of canals and buildings must be taken into account.

Concept of VGB for the implementation of the recommendations
- VGB presentation of 11 December 2013, slide 32 [9],
- VGB presentation of 31 May 2012, 80th AST meeting,
- KBR-MELUR letter of 12 August 2013 [19], page 13 f.
- VGB letter of 02 May 2016 [37]
- VGB letter of 10 June 2016 [38]

During the review of the flood analyses for the plants, the supervisory authorities confirmed, according to VGB, that the flood analyses on which the plant designs are based are also enveloping according to the current state of knowledge. The realised "protection heights" of the plants are therefore clearly above the design basis flood with a frequency of occurrence of 10^{-4}/a to be applied today.

In order to assess the design margins for the postulated exceeding of the water level on the plant site during a design basis flood, further data were compiled [42] on the basis of which the site-specific influence on the increase in the water level can be estimated. According to this, it follows for

- four plants that the difference in level height between the hundred-year or thousand-year flood and the design basis flood (10^{-4}/a) is smaller or at least not greater than between the design basis flood and the plants' protective height for the vital functions,

- two other plants that, due to the possibility of run-off into the surrounding area, the water level can hardly rise any further and thus remains far below the protection level for the vital functions or cannot flood the plant site.

Thus, the protection height of the equipment for the vital functions covers water levels that would be expected if the design basis flood level were extrapolated to a frequency of occurrence of 10^{-5}/a.

Assessment by the RSK
The following emerges from the information provided by the operators:

- A determination of the design basis flood (frequency approx. 10^{-4}/a) from the last ten years is available for all plants.

- For the PWR plants still in operation, the Δ in the level between the design basis flood and the 100- or 1,000-year flood can be determined from the available data, or it can be justified why the design basis flood level cannot increase significantly even with an even lower frequency of occurrence.

- For all plants, this Δ is <u>not greater</u> than the difference between the design basis flood level and the protective height relevant for the vital functions, i.e. this would also cover a flood with an extrapolation to an occurrence frequency of 10^{-5}/a.

If site-specific confirmation of the operator's data is provided, the criteria for robustness level 1 specified by the RSK are fulfilled.

B.1.5 Concretisation of the recommendation on annulus flooding

Recommendation of the RSK [2], Part 1
The following should be presented or clarified:

- *Indication of the safety-relevant equipment that would fail in the event of a flooding height of 2 m at the lower annulus level. In particular, it has to be examined what effects the flooding of measuring transducers and other electrical and instrumentation and control equipment in the annulus may have on residual-heat removal and on the boration of the primary coolant. It has to be shown whether measures can be hindered, prevented or inadvertently triggered.*

- *Taking this point into account, it has to be shown in detail which measures will be available, depending on the operating phase, in order to avoid an inadmissibly long failure of vital safety* *functions under the boundary conditions of beyond-design-basis flooding of the annulus up to a flood height of 2 m. In particular, it has to be shown with which measures*

- *in the event of beyond-design-basis flooding, starting from power operation, secondary-side heat removal and in addition a shutdown to reach a cold, pressureless, subcritical state are ensured in the short run and what equipment must be credited and will be available for this purpose.*

- *fuel pool cooling can be ensured within the required time frame in the event of beyond-design-basis flooding, both in power operation and in low-power and shutdown states.*

- *in the event of beyond-design-basis flooding in low-power and shutdown states with a lowered level in the reactor coolant lines, make-up of the evaporated inventory can be achieved in the short and medium run (by proving e.g. that accumulator feeding is securely available and can be activated).*

Furthermore, it has to be shown how, in operating phases with flooded reactor pool, scenarios with water losses into the annulus from the connected system (RPV - reactor compartment - fuel pool) will be prevented under all operating conditions of the fuel pool cooling and pool cleaning systems (incl. leaks caused by erroneous actions or inadvertent excitations of reactor protection signals) and controlled in case of failure of the precautionary measures possibly provided for this purpose.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 20,
- VGB presentation of 04 November 2014 [13], slides 5 to 11,
- KBR-MELUR letter of 12 August 2013 [19], pages 14 to 18,
- oral explanations, 2nd meeting of the RSK WG on Robustness on 04 June 2014,
- VGB presentation of 19 May 2015 [18], slides 5 to 10
- oral explanations, 5th meeting of the RSK WG on Robustness on 19 May 2015 [26],
- VGB, e-mail HP transfer pumps, 24 November 2014 [32].
- VGB, oral explanations, 10th meeting of the RSK WG on Robustness on 02 February 2016 [35]

In the opinion of VGB, due to the existing and further improved precautionary measures against water entry into the annulus as well as the countermeasures in case of an assumed water entry, beyond-design-basis flooding with comprehensive failure of safety equipment in the annulus is not to be assumed. For the postulate of the RSK regarding such beyond-design-basis flooding, VGB at most sees a potential for plants with auxiliary cooling water intake from a river, but not for plants with cell coolers, i.e. with a closed auxiliary service water circuit, since beyond-design-basis flooding of the annulus is considered to be excluded there due to the limited inventory.

VGB therefore developed a generic concept for controlling the effects of a postulated beyond-design-basis flooding in the annulus only for plants with auxiliary service water intake from a river. Based on this generic concept, the respective plants have evaluated whether and to what extent this concept should be implemented plant-specifically (evaluation process not yet completed for all plants).

The following aspects of the generic concept were presented (for more details, see Appendix Flooding of the Annulus (PWR and BWR) from page 82 onwards):

- According to the operating manual, in the event of a strong rise of water levels with an uncontrollable cause, reactor scram and fast secondary-side shutdown must be triggered at an early stage and the auxiliary service water pump of the affected quadrant must be switched off.

- It has to be examined on a plant-specific basis whether an opening of the annulus towards the reactor auxiliary building may be useful in order to gain non-intervention time.

- An analysis was made of the failures of safety-relevant equipment including the measuring transducers (MT) in the event of an assumed flooding height of 2 m at the lower annulus level. Depending on the processing of the measured values in the logic, the associated reactor protection actions are either triggered or blocked in the event of flooding of the MTs (e.g. blocking of main-steam discharge).

- The countermeasures depend on the prevailing operating phase of the plant. (In any case, measures are first taken to stop the cooling water induction):

  - In <u>power operation</u>, reactor scram should first be triggered and the shutdown of the entire plant initiated in order to use the time until failures occur (e.g. also injecting borated coolant with the extra borating system as long as this is still available). In the first instance, it is essential to secure/restore heat removal via the secondary side, for which existing measures are taken to reduce pressure on the secondary side of the SG, in particular resetting the shut-off signal for the atmospheric steam dump stations and opening blow-off control valves via actuation from the control room (can be carried out within a few minutes). Injection with the emergency feedwater system, which may be triggered by the failure of the MTs, is terminated by intervention in the reactor protection system in order to prevent overfeeding of the SG.

    In the longer run, further boration of the primary circuit and compensation of the volume contraction are required for shutdown, for which first the extra borating system pumps and then the pump of the volume control system can be used after cooling has been established via emergency measures, since these will not have been flooded due to their spatially high point of installation and only cooling via the component cooling system has to be replaced by an emergency measure. With this emergency measure, it is also possible to inject into the reactor cooling system against high pressure if need be (for more details see Appendix "Flooding of the Annulus (PWR and BWR)" from page 82 onwards, on the emergency measure of using a volume control system pump see Ch. B.1.9 "Review of the accident management concept with regard to injection options for cooling the fuel assemblies and ensuring subcriticality").

    The spent fuel pool can be cooled by feeding from areas outside the annulus in combination with a discharge of the evaporating coolant from the containment, thus limiting the boiling temperature in the spent fuel pool to approx. 120 °C (regarding the emergency measure for feeding water, see Ch. B.1.11 "Greater consideration of the wet storage of fuel assemblies").

- In low-power and shutdown operation, with the reactor cavity flooded, there are long non-intervention times until measures are necessary to cool the fuel assemblies in the reactor pressure vessel (>> 10 h). The shortest non-intervention times occur in the case of mid-loop operation. However, by feeding from at least four accumulators, heat removal by evaporation is possible for a longer time (available non-intervention time > 10 h), after which coolant must be injected via the volume control system.

- For the state "All fuel assemblies in the spent fuel pool and refuelling slot gate closed", evaporation cooling and operational injection of demineralised water can be used to supplement the coolant inventory in the spent fuel pool (no active components in the annulus are required for this).

Although it has to be expected that in the assumed scenario the flooding of consumers in the annulus connected to emergency busbars (D1) will lead to short circuits, this would be controlled by the selective staggered disconnection of these consumers from the emergency busbar without de-energising the emergency busbars [34]. If, due to an additionally assumed fault, a disconnector should not open and thus an emergency busbar would be disconnected, the faulty disconnector can be located and opened manually. The emergency busbar can then be reconnected within the non-intervention time (volume control system pumps will only be required after several hours).

Assessment by the RSK

In the opinion of the RSK, the generic concept for measures and strategies to control beyond-design-basis flooding of the annulus that has been presented by the operators for plants with intake of the auxiliary cooling water from a river is suitable to fulfil the corresponding RSK recommendations, provided that the following conditions are fulfilled:

- Since beyond-design-basis annulus flooding with failure of measuring transducers can lead to complex plant states with the consequence that interventions in the reactor protection system as well as substitute information in place of failed measurements may become necessary at short notice, a special chapter in the emergency manual should be implemented in the plants, describing the system behaviour to be expected and the measures required to control such scenarios.

- The feasibility of implementing the planned measures within the non-intervention time should be verified on a plant-specific basis.

If this concept is implemented properly, the RSK considers the recommendations to be fulfilled.

In the opinion of the RSK, scenarios with the consequence of flooding of the annulus beyond the design limits cannot be excluded with sufficient certainty even in plants with cell coolers. Therefore, the emergency measures presented by the operators should also be provided in these plants.

B.1.6 Concretisation of the recommendations on earthquake

Recommendation of the RSK [2], Part 1

The following is recommended:

- *The effects of a fuel assembly transport cask dropping into the fuel pool have to be analysed with regard to a loss of pool water. The overfeeding capability of any loss of pool water that may occur has to be checked and, if necessary, specific emergency measures have to be provided.*

- *Also, the effects of loads dropping into the RPV or onto the slot gate between the RPV and the spent fuel pool during low-power and shutdown operation have to be analysed. If necessary, specific emergency measures have to be provided depending on the consequences.*

- *With regard to the handling of loads in the vicinity of necessary safety equipment, it has to be analysed whether, after a postulated drop of a load, inadmissible repercussions on the pressure boundary may occur or cross-redundancy damage may result, which may lead to "cliff-edge" conditions in the plant.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 20,
- VGB presentation of 04 November 2014 [13], slides 12 to 25,
- KBR-MELUR letter of 12 August 2013 [19], pages 19 to 25.

The aim of the deliberations of the VGB Demonstration Procedure Working Group (VGB-AK "Nachweisverfahren" - VGB-AK "NWV") was to show that a drop of heavy loads can be ruled out or that the consequences of an assumed drop will not lead to the failure of vital safety functions (for more details, see Part F Appendix "Load drop (PWR and BWR)" from page 89 onwards).

With regard to the vital safety functions, it is essentially a matter of ensuring the protection goal of fuel cooling since a significant release of radioactive substances would only be possible in the event of a meltdown of nuclear fuel on a large scale. Such a cooling failure is practically only conceivable as a result of an assumed drop of loads that, due to their weight, are only moved with the reactor building crane main hoist (approx. 30 to 40 operating hours per year).

The concept of the VGB-AK "NWV" is based

- on the one hand on minimising the frequency of occurrence of load drops that could cause a major cooling failure, and

- on the other hand on providing measures to restore impaired cooling to the required extent.

Precaution against load crash

For hoists and load attachment rigging complying with the present state of nuclear safety standards KTA 3902, 3903 and 3905, the failure frequency for hoists according to Section 4.3 of KTA 3902 is estimated to be < 10^{-6}/a (documentation document on the amendment of safety standard KTA 3902, 21st KTA meeting, p. D-6).

Moreover, since not every assumed drop leads to a risk to vital functions, it can be assumed that a load crash with the potential to impair vital functions has a frequency of occurrence that is at least one order of magnitude lower than the above-mentioned value.

Handling of FA transport casks

As planned, FA transport casks (TC) are only moved within the reactor building during power operation (concrete cover above the reactor cavity closed). Due to the design and in-service inspections of the building crane, the possibility of a cask dropping has so far been ruled out. Travel beyond critical areas is prevented by technical measures and administrative specifications (travel range interlocks of the building crane when the TC is stopped, and monitoring of the process).

For the movements of the TC on the crane hook at a low height above the reactor service floor, it can be assumed in the case of a postulated drop that the massive reinforced concrete structures of the reactor service floor can absorb the impact loads. For the assumed drop of the transport cask when it is lowered into the transport cask pool, no evidence is available to show that the resulting impact load will be transferred from the bottom of the transport cask pool without any significant cracking. However, for the actual spent fuel pool, due to the massive reinforced concrete structures between the spent fuel pool and the transport cask pool (threshold present) that are footed on a transverse wall located below, it is not to be expected that cracks with leakage could occur in the spent fuel pool that could not be overfilled for cooling (see below).

A failure of the cranes and hoists exactly in the short time period in which the TC is suspended over the aperture of the transport cask pool in such a way that it would touch down on the edge of the transport cask pool when dropping and then tip over in the direction of the spent fuel pool is probabilistically ruled out.

Handling of the RPV closure head and the upper core structure

The consideration of the RPV closure head is enveloping with regard to mechanical impacts on the flange area of the RPV. If it is assumed that the hoisting gear fails, an impact load is to be expected when the closure head hits the flange area of the RPV, which leads to plastic deformations in the suspension and thus to a certain lowering of the RPV (vessel support and support paws). However, an assessment of the German Risk Study, Phase B (DRS-B, Part F, Appendix Load Crash) has shown that a rupture or failure of the adjoining reactor coolant lines is not to be expected. In the event of assumed damage in the downstream reactor coolant lines, the level in the RPV would drop to the lower edge of the reactor coolant lines. Nevertheless, due to the significantly deeper position of the reactor core and the corresponding water inventory in the RPV, there is sufficient time (approx. 1 h) until the coolant in the RPV has to be supplemented to prevent the level from dropping into the area of the reactor core to be cooled. For coolant make-up in such a scenario, measures were added to the operating documentation for injection by means of accumulators and residual-heat removal pumps held on standby.

Handling of other loads

Other loads moved within the reactor building with hoists and cranes are covered by the cases discussed above with regard to their weight. In the building areas where these loads are moved, the physical separation of engineered safety features ensures that, even in the case of postulated consequential damage, a maximum of one redundant system train of engineered safety features would be affected.

Cooling options

Several diverse options are available for injecting borated coolant into the fuel pool or the RPV. After feeding the inventories of the residual-heat removal system, the residual-heat removal/pool-cooling pumps can be switched to sump operation and thus ensure heat removal in the long run.

Assessment by the RSK

Even if, in the opinion of the operators, a drop of heavy loads need not be postulated, the possibility of a drop of heavy loads and possibly occurring effects was investigated. For the postulated scenarios, the information provided by the operators shows that - taking into account emergency measures if necessary - the consequences remain so limited that serious effects on the environment are not postulated. The operators rule out the possibility of the FA transport cask dropping into the spent fuel pool as the cask is prevented from travelling over the spent fuel pool by interlocks and administrative regulations, and a combination of a postulated inadvertent travel movement with a simultaneous failure of the hoisting gear need not be assumed.

From the RSK's point of view, however, the drop of the fuel assembly transport cask into the spent fuel pool cannot yet be excluded with sufficient certainty with regard to the prevention of cliff-edge effects, since, in addition to technical precautionary measures, administrative precautionary measures are also required. Therefore, the RSK further recommends analysing the effects of the drop of a spent fuel transport cask into the spent fuel pool with regard to a loss of pool water. The overfeeding capability of any loss of pool water has to be checked and, if necessary, specific emergency measures have to be provided. Alternatively, it has to be described in detail by what measures a drop or the tipping of a spent fuel transport cask into the spent fuel pool is prevented so reliably that it can be excluded with regard to cliff-edge effects.

Regarding the further analyses of the operators, the RSK comes to the following conclusions:

- A crash into the transport cask storage pool was classified as very unlikely, but nevertheless postulated. With regard to the impacts, it was concluded on the basis of engineering considerations that even in the event of damage to the transport cask storage pool, any spent fuel pool leakages that may occur as a result can still be compensated. The RSK agrees with this assessment.

- For an assumed drop of the RPV closure head onto the RPV, investigations within the framework of the German Risk Study (DRS) had already shown that no leakages that cannot be compensated would occur at the pressure boundary as a result. In the DRS (main volume, p. 468ff), certain boundary conditions in the processes were assumed for the assessment; in case of any possible deviations from these processes practised today, a plant-specific assessment has to be made as to whether sufficient reactor coolant line integrity for coolant injection into the RPV is nevertheless not called into question.

- No potential for inadmissible damage to the pressure boundary or for cross-redundancy impacts has been identified for other, non-excluded load crash events. This is endorsed by the RSK.

Note: The topic "Transport cask on the hook of the building crane during earthquakes" is dealt with in Chap. B.1.3 "Concretisation of the recommendations on earthquake" and in the associated Part D Appendix "Earthquake (PWR and BWR)".

B.1.7 Achieving the safety-related objective of emergency measures even in the case of natural external hazards

Recommendation of the RSK [2], Part 2

The safety-related objective of the emergency measures is also to be achieved during or after natural external design basis hazards. In particular, the following aspects have to be taken into account during/after natural external hazards:

- *access restrictions to the power plant site and power plant buildings, if any,*
- *the operability of the emergency measures, and*
- *the availability of the remote shutdown and control station.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 35,
- KBR-MELUR letter of 12 August 2013 [19], pages 25 to 27.

In principle, the accessibility of the plant even under external hazard conditions is regulated by the disaster control plan of the authorities and can be ensured by the technical possibilities of disaster control and the Nuclear Emergency Brigade.

In the course of the robustness analysis, it was shown that the vital safety functions can be maintained solely with the personnel available at the plant and the robustly stored emergency equipment and consumables, even assuming that accessibility is impaired as a result of design basis external hazards (with regard to the power supply required for this purpose, see also Section B.1.8 "Availability of the three-phase current supply for vital safety functions").

A simultaneous unavailability of the remote shutdown and control stations during design basis events affecting the power plant need not be assumed due to the physical separation (remote shutdown and control stations are located several kilometres away from the plant site and are designed according to DIN standards to withstand earthquakes).

Assessment by the RSK

On the basis of the explanations of VGB, the RSK comes to the conclusion that the safety-related objective of the emergency measures is also achieved during and after natural external design basis events (external hazards). With regard to the availability of the remote shutdown and control station, the RSK points out the required proper implementation of the RSK/SSK general guidelines for emergency planning.

B.1.8 Availability of the three-phase current supply for vital safety functions

Recommendation of the RSK [2], Part 2

a) *It has to be shown that the three-phase power supply required for the vital safety functions is available even if no grid connection is available for up to one week.*

b) *In the event of a postulated station blackout, the necessary vital safety functions have to be maintained or restored in good time before any cliff-edge effects are reached. This includes the following:*

- *The direct current supply required for the vital safety functions has to be available even if a three-phase current supply is not available for up to 10 hours. A self-supporting charging unit for recharging relevant batteries, which is provided with protection against external hazards, may be credited if the non-intervention times for connection and use of such a charging unit are safely sufficiently long.*

- *Furthermore, it has to be shown that a three-phase power supply can be restored within a plant-specifically determined non-intervention time with backup generators. From the point of view of the RSK, this includes:*

  - *external-hazard-protected arrangement of standardised feed points outside the buildings for supplying the systems necessary for maintaining the vital safety functions. An appropriate design of the feed points is to ensure that the emergency busbars required for this purpose and the emergency busbars supplying the emergency feedwater system can be supplied without impairing the protected status of the corresponding buildings (e.g. ventilation isolation and flood protection) with respect to the relevant external hazard. The feed points have to be designed to be nonreactive.*

  - *at least one external-hazard-protected mobile emergency diesel generator with a capacity for supplying one redundant residual-heat-removal system train.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [98], slides 44 to 49,
- KBR-MELUR letter of 12 August 2013 [19], pages 27 to 29,
- VGB concept, robustness analysis to verify the effectiveness of the vital safety functions in case of beyond-design-basis external and internal hazards. [21]

The operators have explained that the fourfold redundant D1 and D2 diesel generators already provide very reliable protection in the event that all external grid connections should become unavailable. Their robustness analyses have shown that additional equipment cannot contribute significantly to risk prevention. Regardless of this, mobile diesel generators with connections were installed at the emergency feedwater building.

Here, the following measures and equipment were realised:

- For an unavailability of the grid connection of up to one week: extension of the diesel running time to up to seven days using secured fuel stocks (diesel storage tanks on the plant site or, if these are unavailable, by switching off diesels that are not needed and pumping over their fuel). Note: No credit has yet been taken here of the possible staggering of the operation of D1 and D2 diesels.

- Maintaining the vital safety functions in the event of a postulated total failure of the three-phase power supply: Implementation of mobile diesel generators and, if not yet available, an additional fire-fighting/emergency pump with the following tasks:

  - One mobile diesel generator (approx. 200 to 650 kW, depending on the model), stored inside or outside the emergency feedwater building, to supply the instrumentation and control equipment in the emergency feedwater building and which can be connected within 1 to 2 hours.

  - One further mobile diesel generator, stored physically separate from the first mobile diesel generator, for connection to a D2 busbar.

  - Capability of supplying an emergency residual-heat removal chain within less than 10 h.

  - One additional fire-fighting pump/emergency pump for SG feeding (protected inside or outside the emergency feedwater building, depending on the plant layout).

Assessment by the RSK

If the concept intended by VGB is implemented properly, the recommendation of the RSK to safeguard the vital functions in case of an assumed unavailability of the grid for up to one week as well as in case of an assumed SBO is fulfilled.

B.1.9 Review of the accident management concept with regard to injection options for cooling the fuel assemblies and ensuring subcriticality

Recommendation of the RSK [2], Part 2

Review of the accident management concept with regard to injection options for cooling the fuel assemblies and ensuring subcriticality. The following aspects must be taken into account:

- *external-hazard-protected staging of mobile pumps and other injection equipment (hoses, connectors, couplings, etc.) as well as of boron with a specification of non-intervention times for the staging including delivery*

- *ensurance of a water intake that will still be available even after external hazards*

- *options for injecting water into the steam generator, the reactor coolant system and, if necessary, the pressure suppression pool as well as the containment (also taking into account higher back pressures) without having to enter areas with a high hazard potential (dose rate, debris load) and in order to be able to compensate for local destruction (e.g. by means of permanently installed and physically separate injection paths).*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 36 and 44,
- VGB presentation of 04 November 2014 [13], slide 49,
- KBR-MELUR letter of 12 August 2013 [19], pages 29 to 32.

The VGB concept comprises the following measures:

- external-hazard-protected staging (physical separation by distance from reactor building and emergency feedwater building, flood-proof site of installation, safe against debris loads in case of earthquakes) of a fire-fighting/emergency pump for injecting into a steam generator or the fuel pool (see also Section B.1.11 "Greater consideration of the wet storage of fuel assemblies").

- Water intake available after external hazard, see below, Section B.2.3 "Measures to control the loss of the ultimate heat sink".

- Water injection in PWR:

    - into the steam generators via an additional fire-fighting/emergency pump with self-sufficient drive and connection in or at the emergency feedwater building,

    - into the reactor coolant system via the volume control system with separate cooling of the pumps (this measure is intended for plants where the failure of other injection possibilities cannot be excluded due to postulated flooding of the annulus, see Ch. B.1.5 "Concretisation of the recommendation on annulus flooding").

With regard to an additional injection option into the RCS, VGB stated that the PWR 1300 plant concept already offers a wide range of options for injecting borated coolant into the RCS via

- the volume control system,
- the emergency core cooling and residual-heat removal system with HP and LP injection pumps,
- the extra borating system,
- the emergency residual-heat removal pumps (if necessary, supplied by the mobile diesel generators).

From VGB's point of view, a common-cause failure of all injection options need not be assumed:

- In the event of a postulated complete failure of the three-phase power supply (i.e. a more far-reaching postulate compared to an SBO abroad), the supply of a D2 busbar can be established via mobile diesel generators within the non-intervention time until an injection of borated coolant is required (see Ch. B.1.8 "Availability of the three-phase current supply for vital safety functions"), so that an injection with the additional borating system or the emergency residual-heat removal pumps is possible. Alternatively, the supply of a D1 busbar can be established via the 3rd grid connection, with injection via the HP injection system or the volume control system.

- Flooding of the annulus beyond design limits is prevented by precautionary measures against flooding and countermeasures in the event of water entering the annulus. If, despite the precautions taken, the postulate of beyond-design-basis annulus flooding is applied to plants with auxiliary cooling water intake from a river, this would result in a simultaneous failure of all injection options. In this case, however, injection can be provided again via the volume control system. (see Ch. B.1.5 "Concretisation of the recommendation on annulus flooding").

From the point of view of VGB, no additional findings regarding additional primary-side injection can be derived from the systematic robustness analysis beyond these two cases.

Assessment by the RSK

The concept for the installation and connection of an additional fire-fighting/emergency pump with self-sufficient drive and the additional water intake (see Section B.2.3 "Measures to control the loss of the ultimate heat sink") as well as with additional measures for steam generator or fuel pool injection (see Section B.1.11 Greater consideration of the wet storage of fuel assemblies) corresponds to the recommendation of the RSK.

With regard to an additional <u>injection into the reactor coolant system</u>, the following can be stated:

- The injection options into the reactor coolant system for the PWR plants comprise three systems with a total of 17 pumps, whereby redundancy and diversity are realised with regard to the pump types, the injection paths, the coolant/borated water supplies used, and the electrical power supply.

- The recommendation to review the injection options for cooling the fuel assemblies and ensuring subcriticality in the reactor core had resulted, on the one hand, from the discussion on the complete failure of the three-phase power supply and on annulus flooding and, on the other hand, independent of the scenario, from the demand for additional water injection options beyond the water volume available in the reactor building.

- In the event of an assumed complete failure of the three-phase power supply, a D2 busbar (for the operation of the additional borating system and the emergency residual-heat removal pumps) can be resupplied via mobile diesel generators or a D1 busbar of the emergency power supply (for the operation of the safety injection system) via the 3rd grid connection within the given non-intervention time. This means that borated coolant can be injected into the primary circuit even at higher back pressures.

- In the case of a postulated flooding of the annulus beyond the design limits, a simultaneous unavailability of all injection options is to be expected since all pumps except for those of the volume control system are arranged on the lowest level. However, the cooling of the pumps of the volume control system depends on the function of the component cooling pumps arranged on the lowest level. With the emergency measure provided in the VGB concept for cooling the pumps of the volume control system, feed operation of these pumps is possible even in the event of a failure of the component cooling system.

- Further scenarios beyond SBO and annulus flooding that could lead to the simultaneous unavailability of the diverse injection options were not identified.

- If it is planned to use the HP pumps of the volume control system for the emergency measure of injection into the primary circuit, the RSK considers this to be compatible with its recommendation if the following boundary conditions are met:

    - cooling of the cooling loads of the HP pump even if the component cooling system is unavailable (implementation must be possible if the lower area of the annulus is not accessible),

    - power supply for the operation of the HP pump even if the auxiliary power supply and the D1 diesel are unavailable (e.g. via 3rd grid connection or emergency diesel),

    - possibility of injecting water and boron from areas outside the reactor building,

    - consideration of measures in emergency procedures to override required interlocks that would prevent injection via the HP pump,

    - feasibility of implementing the emergency measure within a few hours.

Overall, the measures provided in the VGB concept fulfil the recommendations of the RSK if they are implemented properly and if the boundary conditions mentioned with regard to the use of the HP injection pumps of the volume control system are complied with. However, with regard to a fundamental, additional possibility to inject water beyond the water volume available in the reactor building, the RSK is of the opinion that the emergency measures described for restoring primary-side injection should be realised in all PWR plants with a power operation licence (also in plants with cell coolers).

A separate option for injection into the containment was discussed with regard to emergency measures for possible external RPV cooling in BWRs. For the PWRs considered here, such an emergency measure is not considered to be promising. It is rather the problem of an increased water-melt interaction in scenarios with severe core damage and RPV failure that is seen for the PWR.

If it is possible to inject water into the RPV of a PWR even in the case of core meltdown scenarios, it may also be possible to cover the melt with water outside the RPV in this way. An additional injection option into the containment is then not required by the recommendation.

B.1.10 Filtered venting during and/or after natural external design basis hazards and in the event of a station blackout

Recommendation of the RSK [2], Part 2

The equipment for filtered venting has to be secured in such a way that the venting can also be carried out repeatedly during or after natural external design basis hazards and in the event of a station blackout. In addition, the effectiveness of the installations for hydrogen decomposition in the containment has to be ensured accordingly.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 33 and 44,

The design and layout documents of the wet and dry systems were evaluated with regard to design earthquakes. The findings on stability during an earthquake had been presented to the Land authorities by 12/2014.

Venting can be carried out repeatedly in all plants even in the event of a postulated station blackout. In some plants, the use of mobile ventilation units is planned to avoid detonable mixtures in the exhaust air section.

Regarding the formation of explosive hydrogen-air mixtures in the vent path during filtered venting, the operators have carried out studies for the wet and dry systems in use within the framework of the VGB operators' group. (see Section B.4 "RSK recommendation „Hydrogen release from the containment“ from the 475th RSK meeting on 15 April 2015")

Assessment by the RSK

With proof that the functionality of the equipment required for venting is not questioned by a design-basis earthquake and with the possibility of restoring the power supply to the required extent, the recommendation of the RSK is fulfilled.

On the third aspect, see Section B.4 "RSK recommendation „Hydrogen release from the containment“ from the 475th RSK meeting on 15 April 2015".

B.1.11 Greater consideration of the wet storage of fuel assemblies

Recommendation of the RSK [2], Part 2

B.1.11 Greater consideration of the wet storage of fuel assemblies within the framework of the emergency protection concept with consideration of the following aspects:

- *Options for injecting water into the wet fuel storage facility without having to enter areas with a high hazard potential (dose rate, debris load) and in order to be able to compensate for local destruction (e.g. through permanently installed and physically separated injection paths).*

- *To safeguard evaporation cooling: updating of the verifications for the spent fuel pool, reactor well, setdown pool, reactor cavity seal liner to boiling temperature.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 50 to 52 and 67,
- VGB presentation of 04 November 2014 [13], slides 50 to 54,
- KBR-MELUR letter of 12 August 2013 [19], page 33.
- VGB letter of 03 March 2016 [30]

The concept of VGB pursues the goal of ensuring fuel cooling even under beyond-design-basis failure assumptions in such a way that the occurrence of serious fuel damage including H_2 formation can be ruled out. Essential components of the concept are:

- Additional provisions for coolant make-up in the spent fuel pool
  - by means of sump operation in the case of a water loss from the spent fuel pool/reactor pool,
  - in case of an assumed complete failure of the pool cooling systems by feeding e.g. from demineralised-water tanks or emergency feedwater pools or the fire-fighting system i.a. with mobile pumps without having to enter the containment (connection in the annulus to permanently installed pipes leading to the spent fuel pool).

- Removal of the decay heat in the spent fuel pool by steam release into the containment and make-up feeding.
  The required feed rate to compensate for evaporation is up to 7 kg/s, depending on the pool allocation and decay period. The non-intervention time for feeding to prevent the water level from dropping into the area of the active zone of the FAs is approx. 2 d or more.

- Ensuring the integrity of the spent fuel pool (lining) by containment venting - plant-specifically via suitable flow paths from the containment (e.g. via the compressor line) - to limit the pool water temperature to approx. 120 °C and to limit the containment overpressure to approx. 1 bar (plant-specific investigations on the integrity of the pool structure for temperatures of at least 130 °C), [30].

Since no FA damage occurs with the FAs covered with water, venting can in principle be carried out unfiltered.

Assessment by the RSK
With the implementation of the described concept and due to the long non-intervention times until feeding into the spent fuel pool must be restored after an assumed failure of pool cooling as well as due to the comparatively simple measures for feeding into the spent fuel pool, a cooling failure with the consequence of serious FA damage is practically excluded from the RSK's point of view. Therefore, if the evaporated coolant is discharged from the containment via the compressor line, filtering to reduce the activity release is not necessary in this situation.

In scenarios with core damage, this venting path is not available. In this case, filtered venting can be used. A loss of the integrity of reinforced-concrete structures and thus of the spent fuel pool is not to be assumed at the temperatures that would then occur. Any leakage from the lining can be overfed.

In the opinion of the RSK, the concept presented by VGB corresponds to the recommendations of the RSK if implemented properly.

B.1.12 Early introduction of the Severe Accident Management Guidelines (SAMG)

Recommendation of the RSK [2], Part 2
Furthermore, the RSK considers it necessary that the Severe Accident Management Guidelines (SAMG) be introduced without delay.

Concept of VGB for the implementation of the recommendations
- VGB presentation of 11 December 2013 [9], slides 38, 44, 56 to 58,
- VGB presentation of 19 May 2015 [18], slides 29 to 57,
- KBR-MELUR letter of 12 August 2013 [19], page 34.

The plant-specific manuals of mitigative emergency measures (Severe Accident Management Guidelines - SAMG) have been completed. The structure and contents of the SAMG have already been presented at information events of the Länder authorities and the SAMG working group of the RSK-AST. First staff trainings and internal emergency exercises have been conducted.

Two technical discussions on the remaining issues raised by GRS took place within the framework of the SAMG working group of the RSK-AST at the end of September/beginning of October 2014. The Severe Accident Management Guidelines (SAMG) were introduced in all plants by the end of 2014. The transitions to the SAMG are regulated in the corresponding operating documents of the plants.

Detailed questions that arose in the course of the further discussion (sufficient availability of information/instrumentation for initiating and implementing sets of measures in the SAMG, procedure in the event of activity releases on the site) were clarified in further meetings.

Regarding the robustness of the instrumentation, VGB stated [18], [26] that the qualification of the instrumentation according to KTA 3502 guarantees the LOCA resistance of the measuring points. In the opinion of VGB, a systematic analysis has shown that the measuring points in the containment (SHB) function safely up to the failure of the RPV, since expected pressures, temperatures, radiation exposure and humidity are covered by the qualification as LOCA-resistant. VGB states that the qualification of the measuring points used means that at least core damage state C (core melt outside the RPV, RPV defective) will be detected for the expected loads and that a successive failure of measuring points after entry into core damage state C will be taken into account in the accident analyses.

Regarding the robustness of the power supply of the SAMG-relevant measuring points, VGB explained [18], [26] that aids exist for the crisis team that show how substitute values can be obtained for some of the measuring points that are only supplied by the D1 grid. According to VGB, this is possible for all D1-supplied measured values and, from VGB's point of view, does not justify a general upgrading of the measuring points to a D2 supply. For some measuring points, VGB argues that they are of minor importance for accident control. In summary, VGB came to the conclusion that even if the D1-supplied measured values are not available, a suitable strategy will be chosen to minimise the consequences of the accident, including any release.

Assessment by the RSK

The Severe Accident Management Guidelines (SAMG) have been introduced in the plants. Technical discussions with participation of GRS have shown that there are no open generic questions regarding the structure of the SAMG. Any suggestions for optimisation of the depth and scope of presentation resulting from the feedback of experience, e.g. during emergency exercises, can be implemented during operation, if necessary.

The RSK additionally points out the following aspects that the operators should consider in the further optimisation:

- From the RSK's point of view, the availability of the instrumentation of the parameters relevant for the SAMG is of particular importance. The availability of the measuring points referenced in the SAMG under consideration of the ambient conditions of the different scenarios and core damage states should be determined in detail for each plant.

- Appreciating the statements of VGB [18], [26], the RSK comes to the conclusion that, against the background of the already very robust power supply by the D2 network in terms of design as well as the additional robustness-increasing measures taken by the operators in the area of the D2 supply (cf. also B.1.8 Availability of the three-phase current supply for vital safety functions), the aim should be that the measuring points credited in the SAMG be supplied by the D2 network. Where this is not the case, suitable specifications for obtaining equivalent alternative measured values should be anchored in the SAMG.

- The effects of the core damage states dealt with in the SAMG on the usability of the infrastructure should be systematically analysed in the sense of a worst-case analysis. The effects of unavailabilities identified therein should be taken into account in the presentation in the SAMG.

B.2 RSK statement "Loss of the ultimate heat sink" from the 446th RSK meeting on 05 April 2012

B.2.1 Measures to review and possibly improve the reliability of the ultimate heat sink with regard to blockage of cooling water intake

Recommendation of the RSK [3]

The RSK considers it necessary to re-assess the ultimate and, if existing, the alternate heat sink site-specifically, taking into consideration the operating experience gained in Fukushima and in other plants.

Assessment by the RSK

These are plant-specific aspects in the area of design that were not considered further here.

B.2.2 Measures to strengthen the reliability of the ultimate heat sink with regard to the occurrence of rare external hazards

Recommendation of the RSK [3]

The RSK recommends checking whether the assumptions for flood events also take into account the dynamic peak loads of incoming tidal waves to be expected in the area of the cooling water intake.

In connection with the re-assessment of flood protection as well as of the design against earthquakes and other very rare events, such as aircraft crashes and their impacts in the vicinity of the plant, it has to be determined if all failure causes that may result from these events have been considered in the design of the ultimate heat sink to the extent required.

Assessment by the RSK

These are plant-specific aspects in the area of design that were not considered further here.

B.2.3 Measures to control the loss of the ultimate heat sink

Recommendation of the RSK

Residual heat removal from the plant and the spent fuel pool must be ensured in all plant operating states also in case of loss of the ultimate heat sink due to failure causes in the area of cooling water intake and cooling water return by an alternate heat sink (possibly also different heat sinks in combination). The installations required for it must at least meet the requirements for accident management measures and their effectiveness is to be demonstrated.

The safety analysis should be performed under the following conditions. It has to be demonstrated

- *that a loss of the ultimate heat sink and the emergency diesel generators cooled by it and a simultaneous loss of power supply can be controlled. In this respect, all relevant operating conditions as well as the cooling of the spent fuel pool is to be considered.*

- *that for at least 72 hours the necessary technical installations as well as supplies and materials are available at the plant and can be used effectively. The feasibility of the measures is to be demonstrated under the event-induced conditions.*

- *that the protection goals can be maintained until restoration of power supply (also offsite-power supply), for at least 7 days. After the expiry of 72 hours, reliably prepared and available external assistance measures in can be credited the verification process.*

- *that measures generally can only be credited if the required power supply and availability of the necessary supplies and materials is demonstrably ensured. Furthermore, the boundary conditions during a loss of the ultimate heat sink (e.g. failure of room and component cooling, especially the cooling of the reactor coolant pump seals in PWR plants, or of the seals of reactor recirculation pumps of BWR plants) have to be considered.*

The RSK recommends - if not already implemented – introducing an additional accident management measure so that cooling water can be supplied to the nuclear component cooling systems and be discharged again. Supply can be provided through mobile devices. The quantities supplied must be sufficient for the removal of decay heat from the reactor and spent fuel pool and of the heat loss of the components required for such a cooling operation.

The RSK recommends providing appropriate accident management measures to ensure cooling water return for plants having a CCF potential due to the cross-redundancy merging of the cooling water return lines.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 44, 53 to 55,
- VGB presentation of 04 November 2014 [13], slides 36 to 54,
- further information in connection with the VGB presentation of 19 May 2015 [18], slides 4 to 10.

According to VGB, no additional measures are planned for those plants that already have a sufficiently diverse auxiliary service water supply (for non-diverse components, measures to prevent blockage) since a high level of robustness has already been demonstrated for these plants within the framework of the RSK review.

For these plants, there are two variants:

- secured auxiliary service water (four-strand) via cell coolers, alternatively emergency residual-heat removal chains (two-strand) from receiving water,
- secured auxiliary service (four-strand) via receiving water, alternatively emergency residual-heat removal chains (two-strand) with well water.

In these plants, no mechanisms for a cross-redundancy blockage of heat exchangers could be derived from theoretical considerations or the evaluation of operating experience.

For the other plants in power operation, the concept presented by VGB for avoiding serious effects in the event of an assumed failure of the ultimate heat sink comprises the following main elements:

a) Equipment and measures for an additional mobile shortened cooling chain for plants that do not have a largely diverse auxiliary service water supply, such as cell coolers and emergency auxiliary service water from receiving waters or well water for removing the decay heat.

b) Verification that failure of the reactor coolant pump seal sections in the event of a cooling failure and thus increased leakage is not a concern that would require operation of the emergency cooling system.

c) Emergency measures for additional injection options into the reactor coolant system (RCS) and the spent fuel pool as well as for evaporation cooling of the fuel pool.

To a)

With the mobile shortened cooling chain, water is to be injected from a diverse water reservoir outside the reactor building via a nozzle into the feed line of train 10 or 40 of the component cooling system in such a way that the nuclear residual-heat removal unit and the cooler of the emergency residual heat-removal pump or pool cooling pump are supplied with cooling water. For the return flow, a pipe section on the suction side of the component cooling pump is removed and replaced by an assembly part so that the heated water can drain off from there to outside the reactor building via a hose line. This drainage is led to the outside via an escape door. (Depending on Land-specific requirements, the remaining opening may or may not be closed with a sheet metal insert). The required components (mobile pump with self-sufficient drive, elbow, NB 200 hose sections) are stored partly in the reactor building near the escape door and partly outside the reactor building protected against natural external hazards.

The pumps and hoses of the mobile shortened cooling chain are dimensioned - taking into account the distance to the diverse water reservoir - such that the cooling water flow rate is sufficient to cover the demand from approx. 10 h after shutdown from power operation.

To b)

The design of the RCP seal sections in the plants varies depending on the pump manufacturer and possibly on the year of manufacture (see detailed Appendix "Robustness of the RCP seals" from page 95). For both pump types (KSB, Andritz), detailed investigations have shown that, according to the current design status of the seal sections, even in the scenarios "Failure of the ultimate heat sink in power operation and station blackout

(SBO)", no temperature distributions occur that would lead to a failure of the seals with the consequence of an increased amount of leakage. In the case of KSB pumps, it may be necessary to either close open HP leakage drain lines after 1 to 2 h or to start shutting down the plant. Corresponding specifications are provided in the operating documents. In the case of Andritz pumps, according to the available information, no measures are required within 2 d to avoid excessively high temperatures at the seals.

To c)

Description and assessment see Sections B.1.7 "Achieving the safety-related objective of emergency measures even in the case of natural external hazards", B.1.9 "Review of the accident management concept with regard to injection options for cooling the fuel assemblies and ensuring subcriticality" and B.1.11 "Greater consideration of the wet storage of fuel assemblies.

Assessment by the RSK
In the plant-specific safety review (RSK safety review) of German nuclear power plants taking into account the events in Fukushima-I (Japan) in May 2011, the RSK had found that, with the exception of jointly used coolers, three plants had diverse heat sinks. For this reason, these plants were assigned robustness level 2 with regard to the aspect "failure of the auxiliary service water supply".

From the RSK's point of view, this assessment remains unchanged. Thus, no further measures are to be derived from the above recommendation for these plants, provided that a plant-specific assessment has shown that in general and, in particular, in connection with the event occurrence of failure of the ultimate heat sink, there is no potential for a blockage of intercoolers and that no blockage of the cooling water return across redundancies is possible (due to countermeasures that may have been taken).

For the two plants with cell coolers for the secured auxiliary service water system, no relevant potential for a cross-redundancy blockage of intercoolers can be identified. For the plant where, if necessary, the cooling water supply is to be ensured by well water, it has to be shown by specific analysis that there is no relevant potential for blockage of the intercooler. Furthermore, it has to be shown for this plant that a blockage of the cooling water return can be avoided or resolved by appropriate measures.

For these plants, the retrofitting of an emergency measure with injection of cooling water into the closed cooling water system for reactor services mentioned in the penultimate paragraph of the RSK recommendation is then not necessary.

For all other plants, a concept for an emergency measure with shortened cooling chain was presented. If the concept is implemented properly as described and the effectiveness of the measures (in particular required injection rates, heat removal capacity and year-round, long-term availability) is confirmed on a plant-specific basis, the recommendation of the RSK is fulfilled.

According to the concept presented for the shortened residual-heat removal chain, this is effective for failure causes in the area of the auxiliary service water intake as well as the auxiliary service water outfall.

The conditions to be met for verifications (provision of electrical power supply and operating and auxiliary supplies) mentioned in the recommendation must be taken into account for the credited diverse heat sink.

B.3 RSK statement "Assessment of the coverage of extreme weather conditions by the existing design" from the 462nd RSK meeting on 6 November 2013

Recommendation of the RSK [5]

At its 457th meeting held on 11 April 2013, the RSK recommended that "analyses should be conducted to demonstrate robustness against design basis weather conditions with a return frequency of 10^{-4}/a in line with international developments (ENSREG, RHWG/WENRA). As far as impacts in this frequency range cannot be determined with sufficient reliability, effective management of events and a high level of robustness should be demonstrated deterministically using engineering judgement." In addition, it was suggested with a view to robustness that impacts beyond these impacts should be taken into account by engineering estimates for the determination of safety margins. [5]

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [16],
- VGB presentation of 17 July 2015 [17],
- VGB letter of 24 August 2015 [27]
- VGB letter of 03 March 2016 [30].

According to VGB, for the control of all the impacts discussed below, it can be credited that the respective plant can be shut down sufficiently quickly and thus the demand for cooling water can be reduced to the amount required for the vital functions.

A1 – Freezing rain / ice storm / snowstorm

Essentially, it was assessed whether snow or ice accumulation could impair vital functions (e.g. supply of combustion air for diesels, circulation of cooling water). Among others, the following precautions have been implemented or measures are planned:

- physically separated air intake openings oriented differently to the wind direction, partly protected by nearby buildings,
- cooling water installations located outside the buildings at frost-proof depths in the ground,
- monitoring measurements and electrical heating of critical points in cell coolers,
- bypass operation options to avoid excessively low temperatures in cell coolers,
- instructions for inspections at critical points and, if necessary, for the removal of accumulations.

A2 – Ice floes (on the receiving water)

Essentially, it was assessed whether the supply of cooling water could be impaired by ice floes and pack ice formation. The following precautions have been implemented or measures are planned to counteract this:

- in many plants, intake from de facto stagnant waters without potential for pack ice formation,
- cooling water intakes laid very low (approx. -4 m) and thus designed to be frost-proof,
- in the case of intake from outflowing waters, intake structures are aligned with the flow in such a way that blockage due to pack ice formation is very unlikely,
- site-specific option of admixing preheated return cooling water into the inlet,
- site-specific diverse heat sink with capacity for shutdown plant.

A3 – Air temperature extremely low

Essentially, it was assessed whether liquid media might freeze or become too viscous due to temperature reduction. The following precautions, among others, have been implemented against this:

- safety-relevant systems are largely housed inside buildings in enclosed, heated and monitored rooms,
- piping of safety-relevant systems outside the buildings is located at a frost-proof depth in the ground or protected against frost effects,
- admixture of antifreeze in instrument lines of emergency diesel generators,
- preheating of emergency diesels with auxiliary equipment,
- operational reliability of diesel generator sets when used in regions with extremely low temperatures, i. a. no adverse effects known in NPPs in Russia, Canada or Finland.

A4 – Ambient temperature extremely low

Covered by combinations of other weather effects under consideration and the corresponding countermeasures.

A5 – Slush ice and floating ice

Essentially, it was assessed whether slush ice and floating ice could block the screens of the cooling water purification system and thus impair the supply of cooling water.

Under the site conditions of the plants, the occurrence of slush ice and floating ice is very unlikely. Regardless, any impacts are controlled by the same countermeasures as for A2.

A6 – Receiving water level extremely low for prolonged periods of time

Essentially, it was assessed whether there remains a sufficient supply of cooling water. To this end, the following measures, among others, have been realised:

- suction lift of the intake structures a few metres below the lowest measured water level,
- metrological monitoring (especially differential pressure, filling level) of critical points,
- reduction of the required throughput when limit values are reached,
- use of additional water supplies (wells, water reservoirs),
- in some plants, safety-related independence of the receiving water through the use of cell coolers.

A7 – External fire

Essentially, it was assessed whether an impermissible effect of fire gases and heat on safety-relevant equipment is possible:

- no large fire loads in the vicinity of the plant,
- monitoring of the field outside of the protected-area perimeter for any minor fires that may nevertheless occur, and their suppression,
- sufficient distance between safety-related plant components and the power plant fence to avoid direct impact from fires,
- in the event of the occurrence of flue gases, ventilation isolation of the air supply in the safety-relevant buildings,
- greater physical separation of the intakes of the various emergency power diesels, use of units (especially filters) that are robust in the face of smoke loads,
- proven operating performance of other diesel units under operating conditions that are much less favourable (e.g. firefighting).

A8 – Prolonged periods of heavy rain

Essentially, it was assessed whether the ingress of water into safety-relevant buildings is possible. The following precautions, among others, have been implemented against this:

- For flood-free sites: Extreme precipitation can run off the plant site even if the surface drainage systems are assumed to be unavailable, without impairing safety-relevant equipment. In particular, the drainage conditions in the area of the emergency diesel buildings were checked. The emergency diesel buildings represent a high point. The building has thresholds that are 15 cm high and prevent rainwater from entering.
- For sites with flooding on the plant site: watertight design of the buildings up to above the maximum flood level. This covers considerations regarding heavy rainfall.
- Protection of the air intake against water ingress.

The German nuclear power plant operators have generally reviewed the issue of water drainage in the area of the emergency diesel building for all plants. Due to the structural conditions, the terrain profiles in the area of the emergency diesel building and the geodetic height profiles of the terrain derived from the measurement profiles during floods, a water ingress into the building during heavy rain can be excluded due to conservative boundary conditions for the runoff of heavy rain.

A9 – Strong wind, tornado

The overpressures possible during high winds and tornadoes are covered by the design of the structures required for vital functions against aircraft crash and blast wave.

The negative pressure possible during a tornado (maximum approx. 0.2 bar) cannot affect the stability and function of these structures and their equipment, even if the opening of individual escape doors as a result of the negative pressure cannot be excluded.

A10 – Sand/dust storms

Essentially, it was assessed whether the ingress of sand or dust into safety-relevant buildings could lead to impairments of safety-relevant systems. Among other things, the following precautions have been taken or measures are planned:

- no major sand and dust reservoirs that may be mobilised by winds in the vicinity of the sites,
- staggered filtration of the supply air for the buildings (grilles against larger airborne particles, treatment sections to adjust humidity, air temperature and air purity),
- intake of the combustion air of the diesels via their own filters, physically separated intake options,
- differential pressure monitoring of the filters with signalling, filter replacement or filter cleaning is possible if required,
- operating experience shows that the filters of the emergency diesel generators in particular have large margins against dust contamination. For example, in one plant, no increase in the differential pressure across the air filters of the emergency diesel generators could be detected during operation of the emergency diesel generators within the scope of in-service inspections and with filter service lives of eight years.

VGB comes to the conclusion that, due to the design margins and the countermeasures possible in the plants, a high level of robustness of the plants is given even against the effects of extreme weather exceeding the design basis events. Probabilistic considerations were not made since there is usually no sufficient statistical basis for weather conditions in the range of a frequency of 10^{-4}/a or even rarer.

Assessment by the RSK

The approach of estimating and evaluating the safety margins in the design of the plants against the weather-related design basis events instead of relying on occurrence frequencies in the range of 10^{-4}/a or even rarer was also included as an alternative in the RSK recommendation.

The RSK considers the assessment that there is a clear robustness against these impacts due to the design concept and the design safety margins to be plausible despite the non-existing quantification of safety margins.

With regard to impact A1 (Freezing rain, ice storm, snow storm), sufficient robustness for the intake of the combustion air of the diesels is seen if the weather protection grilles can be electrically heated.

With regard to impact A6 (Receiving water level extremely low for prolonged periods of time), sufficient robustness is considered to be achieved if 10 % of the runoff volume required for full-load operation can be used for the cooling required for safety reasons one day after shutdown [39].

On the subject of "lightning effects", reference is made to the separate statement of the RSK "Lightning with parameters exceeding the standardised lightning current parameters" [51].

B.4 RSK recommendation "Hydrogen release from the containment" from the 475th RSK meeting on 15 April 2015

Recommendations of the RSK [7]

Recommendation 1:

With regard to the hydrogen release during filtered venting of the containment it has to be examined for PWR plants (see Section 3.2) on the basis of representative analyses in accordance with the approach outlined in Section 3.1 which emergency measures to prevent combustible conditions during containment venting in shared exhaust air systems, such as in the exhaust air chamber and the stack, can be provided. Alternatively, it has to be demonstrated that hydrogen combustion will not lead to safety-relevant impacts. To what extent these measures are actually realised in an appropriate manner has to be shown on a plant-specific basis.

Recommendation 2: (BWR)

Recommendation 3:

With regard to the release of hydrogen into rooms outside the containment of the PWR (see Section 3.3), a measure for preventing the formation of combustible gas mixtures is to be developed and implemented within the framework of Severe Accident Management Guidelines (SAMG) for the recirculation of the atmosphere in the annulus (break-up of stratification) and controlled ventilation (limiting the increase in H_2 concentration) in a timely manner. For the annulus air extraction required for it, it has to be assessed whether measures to reduce the release of radioactive substances into the environment can be used here (e.g. filtering, discharge via stack). Alternatively, measures for hydrogen recombination can be provided.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 19 May 2015 [18], slides 13 to 15

With regard to Recommendation 1 (avoidance of combustible conditions during containment venting in shared exhaust air systems), the relevant investigations will be carried out in plant-specific procedures and, if necessary, measures will be defined. The corresponding measures are not dealt with in this Statement.

With regard to Recommendation 3, it was explained how the annulus atmosphere can be circulated (avoiding increased local H_2 concentrations) and exchanged in a controlled manner (limiting the H_2 concentration), using existing equipment:

- With a recirculation system for the upper annulus (provided in normal operation to prevent condensation from forming when the temperature falls below dew point), air can be extracted at the top of the dome via a duct and injected into the annulus, distributed via an annular duct at about the height of the containment equator (throughput approx. 11 m³/s).

Thus, the areas in which H_2 discharge from the containment into the annulus would be possible within the scope of the design leakage in the event of an assumed severe accident are covered by the circulation.

- By opening e.g. a flap in the supply air system as well as by extraction via the annulus air extraction system (if necessary via the optional filter system in the reactor auxiliary building), an air exchange is achieved to limit/reduce the H_2 concentration (throughput approx. 4000 m³/h). Since the air is fed into the lower area of the annulus, this also allows mixing of the lower annulus atmosphere. The extracted air is always guided through filter sections.

The fan of the recirculation system for the upper annulus is located separately from the annulus atmosphere in an enclosure in the annulus, while the fans of the annulus air extraction system are located in separate rooms of the annulus. A failure of the fans in both systems due to humidity and temperatures is therefore not to be expected.

The fan of the recirculation system for the upper annulus is supplied electrically from the normal mains, but if necessary, a power supply can also be established in the switchgear building via mobile equipment. The other fans that may be required are connected to the emergency power supply.
The annulus does not have to be entered to carry out the measures.

Assessment by the RSK

The RSK considers the implementation of a measure using the above-described equipment for circulation and exchange of the annulus atmosphere as targeted if the effectiveness and operability including the required power supply under the plant-specific conditions is demonstrated for the implementation and the associated procedural strategy is included in the Severe Accident Management Guidelines.

Part C Assessment of the implementation of RSK recommendations in response to Fukushima: PWR plants

In the following, the Recommendations of the RSK with reference to Fukushima and the Concept of VGB for the implementation of the recommendations in the German BWR plants are evaluated from the point of view of the RSK.

C.1 RSK Statement "Plant-specific Safety Review (RSK-SÜ) of German Nuclear Power Plants with Consideration of the Events at Fukushima I (Japan)" from the 437th RSK meeting from 11 to 14 May 2011 42

C.1.1 Optimisation of high-pressure injection in BWRs 42

C.1.2 Limitation of releases from the fuel pool 42

C.1.3 Twin-unit plants 42

C.2 RSK recommendations on the robustness of the German nuclear power plants from the 450th RSK meeting on 26/27 September 2012, including the RSK Statement "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" from the 457th RSK meeting on 11 April 2013 45

C.2.1 Systematic analysis of the robustness of the German nuclear power plants 45

C.2.2 Targeting robustness level 1 or degree of protection 2 48

C.2.3 Concretisation of the recommendation on earthquakes 49

C.2.4 Concretisation of the recommendation on flooding 51

C.2.5 Concretisation of the recommendation on flooding of the annulus 53

C.2.6 Concretisation of the recommendation on load drop 56

C.2.7 Achieving the safety objectives of accident management measures even in the case of natural external hazard events 58

C.2.8 Availability of three-phase power supply for vital safety functions 59

C.2.9 Review of the accident management concept with regard to injection options for the cooling of fuel assemblies and for ensuring subcriticality 61

C.2.10 Filtered containment venting during or after external natural design basis hazards and in the event of a station blackout 62

C.2.11 Increased consideration of the wet storage of fuel assemblies 63

C.2.12 Short-term implementation of the Severe Accident Management Guidelines (SAMG) 64

C.3 RSK Statement "Loss of the ultimate heat sink" from the 446th RSK meeting on 05 April 2012 66

C.3.1 Measures to check and, if necessary, improve the reliability of the ultimate heat sink with regard to blockages of the cooling water intake. 66

C.3.2 Measures to strengthen the reliability of the ultimate heat sink with regard to the occurrence of rare external hazards 66

C.3.3 Measure to control the loss of the ultimate heat sink 66

C.4 RSK Statement "Assessment of the coverage of extreme weather conditions by the existing design" from the 462nd RSK meeting on 06 November 2013 68

C.5 RSK recommendation "Hydrogen release from the containment" from the 475th RSK meeting on 15 April 2015 72

C.1 RSK Statement "Plant-specific Safety Review (RSK-SÜ) of German Nuclear Power Plants with Consideration of the Events at Fukushima I (Japan)" from the 437th RSK meeting from 11 to 14 May 2011

C.1.1 Optimisation of high-pressure injection in BWRs

Recommendation of the RSK [1]

Optimisation of the BWR accident management measure of steam-driven high-pressure injection in a SBO to prevent the high-pressure path during core meltdown (maintenance of a sufficient pressure suppression capability at increased pressure suppression pool temperature)

Assessment by the RSK

This recommendation is no longer relevant as it was specific to the BWR-69 reactor type. Power operation is no longer permitted for plants of this type. The plants are in the post-operational phase. For this purpose, the provision of a high-pressure injection system is no longer necessary.

C.1.2 Limitation of releases from the fuel pool

Recommendation of the RSK [1]

Measures for the limitation of releases from the fuel pool in BWRs in the postulated event of severe fuel assembly damage, considering possible H_2 formation.

Assessment by the RSK

This recommendation has not been upheld in the context of the RSK recommendations of 2012 [2] on the systematic robustness analysis as due to additional measures to keep the FAs covered with water in the storage basin, a dry-out of the FAs and thus serious damage need no longer be postulated.

C.1.3 Twin-unit plants

Recommendation of the RSK [1]

Against the background of the experience gained from Fukushima, the RSK recommends that an analysis of this issue should be carried out as part of the supervisory procedure for the twin-unit plants concerned. Based on the postulated damage states of the neighbouring unit (i.a. fires, activity releases, core damage states, core meltdown), this analysis has to examine the consequences for the maintenance of the vital functions of the unaffected unit.

Concept of VGB for the implementation of the recommendations

- VGB Power Tech e. V., Additional generic BWR-specific information, 05 November 2014 [15], slide 15
- VGB presentation of 02 February 2016 [36], slides 2-4
- VGB letter to the RSK Secretariat of 03 March 2016 [44]
- VGB letter to the RSK Secretariat of 14 March 2017 [46]
- BStMUV, e-mail of 11 August 2017 [48]
- TÜV-Süd, e-mail of 21 June 2017 [49]

According to VGB, the topics of fire, activity releases, core damage states and core meltdown were investigated for the BWR type with regard to effects in twin-unit plants.

For fires, the investigations show that no new findings have to be taken into account compared to the Information Notice of GRS WLN 07/2008 ("Intrusion of fire gases into the control room of the Krümmel nuclear power plant during the fire of a generator transformer on 28.06.2007"). The protection of the safety-related equipment is effected by the ventilation isolation or by physical separation in case of cross-redundancy impacts.

With regard to the topic of activity release, the investigations did not reveal any special aspects for twin-unit plants. With regard to the control room supply air filtration, it was explained that by creating an overpressure of 0.2 mbar in the control room and the computer rooms, the intrusion of gases is prevented. The air volume required to maintain the overpressure is filtered in a mobile filter unit for each unit and supplied to the rooms by means of a fan.

With regard to the core damage states and core meltdown, it was shown that for BWR twin-unit plants, there is only the common aspect of common filtered venting to consider. A simultaneous core meltdown in both units, which would require simultaneous venting, is considered a very unlikely scenario due to the conditions at KRB II in view of the systems design. Nevertheless, the possibility of alternating venting was investigated, in which venting is periodically switched between the units.

According to VGB's presentation, alternating venting can be carried out without restrictions on level of defence 4b. Venting can be performed both remotely from the control room and manually by operating manual valves. Radiological protection during manual operation is provided by a wall between the handwheels and the pressure relief line. The containment isolation dampers in the raw gas lines are supplied via the uninterruptible AC grid; in addition, supply via mobile emergency diesel generators is possible. Manual operation is thus not necessary even in hypothetical scenarios, but it would also be feasible from a radiation protection point of view [48, Appendix Feedback]. According to [36], the valves are located in the reactor building, so there is no interference from the neighbouring unit.

Analysis results [36] of an hourly alternating filtered venting in Units B and C for a period of 17 hours indicate the pressure-reducing influence during venting. However, for both units, after an initial drop, a rise of the mean containment pressure by 1.5 bar within 14 hours is shown, since the heat removal capacity via the venting system is not sufficient in the investigated time range (up to 24 hours after the start of the event) to maintain the low-pressure value during alternating operation.

VGB states in [46] that the venting system is designed for a saturated-steam mass flow of 14 kg/s at a containment pressure of 7 bar_{abs}. Taking only the evaporation heat conservatively into account, this results in a heat removal of approx. 29 MW. However, according to VGB's estimate, at a containment pressure of 10 bar, a power of about 40 MW can be dissipated via the venting system.

For containment failure, the failure of the loading cover is most significant. Its failure pressure was determined as a function of the loading cover temperature using temperature-dependent material parameters. Up to a loading cover temperature of 220 °C, the failure pressure is approx. 11 bar_{abs}, at higher temperatures it drops. According to MELCOR analyses, temperatures above 200 °C would only be exceeded in the loading cover area at advanced core melting. At 400 °C, the failure pressure of the loading cover as well as that of the confinement is then 8 bar_{abs}.

In the opinion of VGB, the heat removal via the venting system is sufficient to limit containment pressure to values below the failure pressure even with alternating venting of both units on level of defence 4b.

Furthermore, radiation exposure during manual operation of the containment isolation dampers on level of defence 4c was estimated to be 35 mSv 15 hours after accident occurrence and approx. 165 mSv after 3 days under conservative assumptions [48, Appendix Feedback]. Actuation of the handwheels would thus still be possible.

In addition, the operator points out that all investigations on core meltdown scenarios carried out within the framework of the Level 2 PSA had shown that containment failure would occur within < 24 h after the start of core meltdown [48, Appendix Feedback]. With containment failure, venting is no longer possible.

Assessment by the RSK
Based on the presentation of VGB, the RSK comes to the conclusion that the joint use of parts of the venting system is the decisive aspect with regard to a mutual influence of the two units.

With regard to the maintenance of the vital function of residual-heat removal on level of defence 4b, the assumed simultaneous failure of the residual-heat removal systems in both units represents the most unfavourable boundary condition. On the basis of the data on the design of the venting system mentioned by VGB, the RSK considers it plausible that at a containment pressure of 10 bar, a power of about 40 MW can be removed via the venting system. With alternating venting, this corresponds to an output of approx. 20 MW per unit. The steam temperature at 10 bar is 40 K below the temperature specified by VGB for a loading cover failure pressure of 11 bar. If the increase in containment pressure shown in [36] is extrapolated, a value of 10 bar_{abs} would be reached more than two days after the start of the event. In the opinion of the RSK, a heat removal capacity of 20 MW per unit is then sufficient to limit the containment pressure to 10 bar_{abs}. Thus, when switching venting between the two units, the capacity of the venting system on level of defence 4b appears sufficient for maintaining the vital function of residual-heat removal. It is assumed here that the switching takes place without delay.

For a postulated "simultaneous" core meltdown accident and postulated failure of the remote operation of the containment isolation dampers of the venting system, TÜV SÜD concludes on the basis of its own estimates for the scenario "operation of the handwheels on site" that the dose exposure for the person carrying out the entire measure (i.e. dose received on the way + dose received during the measure) remains below 250 mSv. This applies to all cases, i.e. also to the repeat case and 3 d after the occurrence of the core meltdown accident. This means that in the case of alternating venting, the relevant value of the Radiation Protection Ordinance can be complied with if the personnel is changed for each individual measure. Against this background, the RSK comes to the conclusion that alternating venting can be repeatedly performed even in case of postulated core melt accidents. However, as described by the plant operator, the medium-term significance of venting in the event of a progressive core meltdown accident is to be regarded as rather low, since in this case there is a high probability of a containment failure.

In the opinion of the RSK, there is no risk of hydrogen entering the neighbouring unit via the venting system due to multiple isolating valves.

C.2 RSK recommendations on the robustness of the German nuclear power plants from the 450th RSK meeting on 26/27 September 2012, including the RSK Statement "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" from the 457th RSK meeting on 11 April 2013

C.2.1 Systematic analysis of the robustness of the German nuclear power plants

Recommendation of the RSK [2]

To ensure the vital safety functions in case of beyond-design-basis external or internal hazards, a systematic analysis should be conducted to identify potential for increasing robustness appropriately, for which supplementary measures should be designed where required (see Annex 1).

…

Thus, the design margins in the existing safety installations or emergency systems are to be assessed with regard to whether and when the required safety function of safety installations or emergency systems may be endangered in case of increased (beyond-design-basis) assumptions on external and internal hazards. These analyses can be performed by means of engineered judgements.

…

On this basis, it is then to be assessed whether an increase of robustness is possible

- *either by appropriate measures to upgrade existing safety installations or emergency systems*
- *or by existing or additional accident management measures to ensure vital safety functions in case of expected failure of safety installations or emergency systems. These accident management measures must not lose their operability by those impacts that in the analyses have led to a functional failure of safety installations or emergency systems.*

With the accident management measures to ensure the vital process-based safety functions designed in this way it is then possible to derive the tasks for auxiliary functions and thus for appropriate accident management measures to compensate for possibly occurring failures in the safety-related auxiliary functions (in particular electrical energy supply and service water supply).

Concept of VGB for the implementation of the recommendation

- VGB presentation of 11 December 2013 [9], sides 12-24, 62-65, 67, 76;
- VGB presentation of 04 November 2014, details on the systematic robustness analysis (PWR) [14], slides 7 - 24,
- VGB presentation of 05 November 2014, generic BWR-specific additional information [15], slides 7-14
- VGB concept, Robustness analysis to check the effectiveness of the vital functions in the event of beyond-design-basis external and internal hazards, BWR, March 2015 [42]

- VGB presentation of 02 February 2016 [36], slides 5-11
- VGB letter of 14 March 2017 [46], items 4 and 6

According to the presentation of VGB, the following have been considered so far for the analysis of robustness (cf. also *Table C-1*):

- the relevant spectrum of events with the potential for cross-redundancy failures of safety equipment
  - beyond-design-basis external and internal hazards
  - cross-redundancy postulated failures of safety equipment (loss of the ultimate heat sink and Station Blackout)
- the vital functions of BWRs to avoid serious effects on the surroundings in the case of such postulated events and different plant states
- the plant conditions analysed for BWRs at the beginning of the event.
- the operational capability of existing safety and emergency equipment for vital functions during these events
- the results of the robustness analysis for the beyond-design-basis earthquake for BWRs (shown as an example of the as-is condition)
  the results of the robustness analysis for the beyond-design-basis internal hazards "loss of the ultimate heat sink" and "Station Blackout (SBO)"
- insofar as the functional capability is questioned due to the beyond-design-basis assumptions, derivation of supplementary verifications or measures (for safety equipment or also for supplementary emergency measures).

Table C-1: Overview of VGB's considerations regarding robustness-enhancing measures for the BWR-72 [46]

Objective	**Robustness-enhancing measures (BWR-72)**
Power supply ► keeping S&R valves open ► I&C in the reactor building RPV feeding ► with mobile pump time range ► 2 to 10 h after start of the event	Power supply ► emergency diesel for SBO conditions ► provision outside, operation outside RPV feeding ► increase of possible feed pressure
Power supply ► consumers of the battery system, ventilation and light distribution of one of the safety sub-systems 2 or 3 Time range ► from 3 to 4 h after start of the event possible	Power supply ► supply via mobile diesel
Optimisation of make-up feeding into the RPV	Additional tank storage for diesel fuel for mobile pumps, increase of possible feed pressure
Optimisation of the continuous operation of the emergency diesel generators through the use of all internal fuel reserves and extended lubricant stocks.	Lubricant stocks available for more than seven days, fuel conservation by switching off individual emergency power diesels, pumping-over possible, new, additional fuel storage, thereby self-sufficiency of the site for more than seven days
Optimisation of flood protection in the reactor building	Flooding does not lead to a loss of vital safety functions
Optimisation of flood protection	Plant-specific review revealed considerable additional safety margins, extended flood protection (already implemented before review), for example protection of buildings with safety-critical components by means of sheet piling, acquisition of boats
Removal of the decay heat via a diverse heat sink	was already realised
Optimisation of coolant make-up into the reactor coolant system	New specification in the operating manual for early opening of the diverse pressure relief valves, increase of the possible feed pressure of mobile pumps
Optimisation of coolant make-up into the spent fuel pool	Removal of the decay heat in the spent fuel pool by evaporation and replenishment of coolant inventory, injection from the fire extinguishing system or with mobile pumps, e.g. special fire-fighting vehicles.
Sump recirculation mode in case of water loss from the spent fuel pool/reactor pool	In the event of a loss of coolant from the flooded reactor pool that cannot be shut off when the slot gate is drawn, the slot gate can be closed quickly; since it is a swivel gate, no additional equipment such as a crane and hangers are required for this; there are various options for injection into the spent fuel pool
Ensuring spent fuel pool integrity by means of containment venting	In the area of the spent fuel pool, pressures of more than 1 bar are not possible.

Assessment by the RSK

The procedure for the robustness analysis has been presented in bullet points. The system applied has been explained by means of a sequence plan for the evaluation process and the derivation of necessary improvement measures. The ensurance of vital safety functions was exemplified for the beyond-design-basis earthquake and the events "loss of the ultimate heat sink" and "station blackout (SBO)". Robustness was demonstrated for the systems and their redundant system trains available four different time ranges. The operators pointed out that the procedure of the systematic analysis was closely oriented on that of the PWR.

This assessment refers to the basic approach of the robustness analysis. Measures derived from this assessment are discussed in detail below. Insofar as specific measures proposed by the RSK are not to be implemented, these are also discussed below. (see Ch. C.2.8 "Availability of three-phase power supply for vital safety functions, concept regarding mobile diesels", Ch. C.3.3 "Measure to control the loss of the ultimate heat sink regarding additional injection into a closed cooling water system for reactor services".

C.2.2 Targeting robustness level 1 or degree of protection 2

Recommendation of the RSK [2], Part 1

The RSK considers it appropriate that ultimately, the target should be at least Robustness Level 1 or at least Degree of Protection 2 (man-made hazards).

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 75

The robustness levels indicated in the RSK-SÜ were not used to assess robustness.
From VGB's point of view, however, the analyses carried out, including additional robustness-increasing measures if necessary, have demonstrated a high level of robustness against beyond-design-basis events.

Assessment by the RSK

Due to the measures implemented, a further increase in robustness was achieved compared to the RSK-SÜ of 2011.

Even if VGB did not pursue the assignment to robustness levels, the RSK comes to the conclusion on the basis of the available information that, if the concepts presented by VGB are plant-specifically appropriately implemented if the RSK's comments in this Statement are taken into account, a level of robustness is largely achieved that is in the range of robustness level 1, in some cases even higher. Even for the events or features for which no assignment to robustness levels was made in the SÜ of 2011 (see e.g. C.4 and C.5), clear robustness is given, i.e. even if the design requirements are exceeded, there are still clear design margins to prevent serious impacts on the environment.)
Note: Degrees of protection as defined in [1] for man-made hazards are not discussed here, because the corresponding man-made hazards are not dealt with in this Statement.

C.2.3 Concretisation of the recommendation on earthquakes

Recommendation of the RSK, [2],Part 1, and [6]

a) *For plants for which results of probabilistic seismic safety analyses are available, the robustness against beyond-design earthquake impacts has to be assessed. The assessment should be based on the HCLPF (High Confidence for Low Probability of Failure) values of the buildings and installations required to ensure the vital safety functions.*

b) *For plants for which no results of probabilistic seismic safety analyses are available, there is the option to assess robustness against beyond-design earthquake impacts by means of applicability considerations (possibly supported by a plant walkdown by an expert commission) based on results according to a)).*

To improve robustness, the superposition of operating conditions during low-power and shutdown operation of short duration with an earthquake should be considered. For the analysis of robustness, it has to be demonstrated that the design earthquake will not lead to significant impacts in the environment during temporary operating conditions of short duration.

Here, particular attention has to be paid to situations where vital safety functions may be impaired in the case that

- *changes in mass distributions (e.g. filled reactor well during reloading) in the reactor building lead to higher seismic loads on safety-relevant installations and building structures than during power operation,*
- *certain installations are only operated (e.g. reactor cavity seal liner in a BWR) or are in a specific mode of operation (e.g. refuelling machine outside the parking position) during low-power and shutdown operation for which there are no specific or higher-level proofs regarding seismic loads,*
- *parts (e.g. fuel assembly transport casks, heavy components) and operating media (lubricating oils and solvents) that are introduced into the plant or handled during low-power or shutdown operation cause damage to safety-relevant installations and building structures due to an earthquake,*
- *in the event of an earthquake, safety-relevant measures and installations are only available to a limited extent during low-power and shutdown operation (e.g. isolation of residual-heat removal trains, short-term manual actions), which are required to manage the consequences of an earthquake.*

For plants that are permanently in low-power and shutdown operation, the proof of robustness has to be provided for longer-lasting states also for beyond-design earthquakes according to a) and b) (see above)).

With regard to a PGA minimum value of 0.1 g for the horizontal peak ground acceleration, the RSK stated in [6]:

Demonstration of compliance with the IAEA requirement for a PGA value of 0.1 g can be achieved by reassessing the seismic resistance of the affected installations using the methods of IAEA Guide NS-G-2.13. By means of the "Seismic Margin Assessment" method mentioned in NS-G-2.13 (if necessary, using data from an existing seismic PSA), it would have to be shown for installations with a maximum ground acceleration of < 0.1 g that the installation is also sufficiently resistant to a ground *acceleration of 0.1 g. This procedure is already included in principle in the recommendations of the RSK Statement on robustness [2]. Should a PGA value < 0.1 g be determined site-specifically at an assumed intensity corresponding to robustness level 1, the RSK recommends determining the design margins available for an assumed PGA value of 0.1 g. The RSK also recommends determining the PGA value of 0.1 g for a site-specific design.*

Concept of VGB for the implementation of the recommendations

In addition to the overarching issues addressed in Section B.1.3 „Concretisation of the recommendations on earthquake" of this statement, BWR-specific aspects are addressed in

- VGB letter of 24 August 2015 [27]
- VGB concept – Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015. [42]
- VGB presentation of 02 February 2016 [36], slides 16-18
- VGB letter of 14 March 2017 [46], items 7 and 8
- BStMUV, e-mail of 11 August 2017 [48]
- TÜV-Süd, e-mail of 21 June 2017 [49]
- BStMUV, e-mail of 05 September 2017 [50].

In [27], VGB refers to its remarks on the robustness of German nuclear power plants against seismic impacts - see Part D Appendix "Earthquake (PWR and BWR)". According to [27], the principles described for design against external hazards apply to all German nuclear power plants. The same applies to the basic statements on earthquakes in low-power and shutdown operation.

The methodology of the seismic probabilistic safety analyses (PSA) presented using the example of two PWR plants and the related results are, in the opinion of VGB, also representative for the BWR plant, which lies in the seismic hazard level presented and is thus comparable with the PWR plants mentioned as examples with regard to seismic impacts.

According to [46], a simplified procedure was carried out with regard to the seismic PSA according to the PSA methods volume. The theoretical basis is formed by the relevant EPRI and NUREG reports and the related concepts (HCLPF, GERS - Generic Equipment Ruggedness Spectrum). An assessment of the conditional failure probabilities of safety functions for earthquakes with intensities I = VII (design) and I = VIII (design +1 intensity level) was carried out. The conditional failure probability of the high-pressure injection system (no vital safety function) is the highest with approx. 2% for both cases (flat curve of the failure probability distribution in the acceleration range between I = VII and I = VIII). The failure probability of the other safety functions is again at least one order of magnitude lower, also for both cases.

Analyses of the stability of the two turbine buildings of the BWR plant KRB II during a design earthquake (intensity I = VII) were carried out. Compared to the original safe shutdown earthquake for the construction (1974), a comparison with assessment spectra from a current study (2016) to verify the KRB II design earthquake showed lower acceleration values in the lower frequency range [49], [50], so that it was to be expected that the turbine buildings would also safely withstand an earthquake of intensity I = VIII [36]. To confirm this assessment, computational analyses were commissioned [46, item 7], which have since been completed with positive results [48].

According to [46], an earthquake-related plant inspection was carried out, in the course of which the installation of components, internals and auxiliary tools and any resulting hazard potential in the event of an assumed earthquake during ongoing overall maintenance and refuelling work was assessed. According to this, cliff-edge effects due to relocation of large components during severe earthquakes are not possible. Some possible optimisations were identified and have been implemented.

Assessment by the RSK

The assessment is limited to the questions whether the procedure presented by the operators corresponds to the recommendations of the RSK in [2] and whether the results of the corresponding investigations are generally plausible. Overall, the RSK arrives at the following assessments:

- The methodological approach chosen by the operators for the designation of design margins is in line with the RSK recommendation. The procedures described by the operators for the determination of design margins (by means of an SPSA) are in line with standard international practice.

- The reserves specified by VGB for an earthquake with an impact increased by one intensity level compared to the design intensity indicate, according to VGB, a conditional failure probability of high-pressure injection of approx. 2%. According to VGB, the probability of failure of the other safety functions is at least one order of magnitude lower. Based on this information, the RSK is of the opinion that it is plausible that robustness level 1 is achieved. Against the background of the reportable event ME 16/063 "Faulty connecting bolts on brackets of ventilation ducts" in KKP-2 in December 2016, however, the RSK is currently still conducting a generic discussion on the reliability of the SPSA results.

- In the opinion of the RSK, the methodical approach for the assessment of robustness against seismic impacts in low-power and shutdown operation and during short operating states is also suitable to fulfil the implementation of the corresponding RSK recommendation. The prerequisite is that the conclusions drawn from the inspection carried out in accordance with the VGB account are adhered to in all other low-power and shutdown operating states and short operating states.

- The PGA value for the considered earthquake of intensity VIII for the BWR plant is 1.6 m/s^2 [48, Appendix 3]. This results in a PGA value of > 0.1 g for robustness level 1, so that compliance with the IAEA requirement is plausible on the basis of the criteria mentioned in [6].

C.2.4 Concretisation of the recommendation on flooding

Recommendation of the RSK [2], Part 1

If a water level that may endanger vital safety functions cannot be excluded due to site-specific conditions, the criteria specified in the safety review [1] for at least Level 1 shall be referred to. Alternatively, it may be *demonstrated on the basis of site-specific conditions that a postulated discharge quantity, which is determined by extrapolation of existing probabilistic curves to an occurrence frequency of 10^{-5}/a, will not result in the loss of vital safety functions. For sites located near tidal waters, an analogous approach is to be applied. The methodology used for it has to be explained in a comprehensible manner.*

In this respect, the uplift resistance of canals and buildings has to be considered.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [15], slide 16
- VGB letter of 02 May 2016 [37]
- VGB e-mail of 10 June 2016 [38]
- VGB concept, Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015 [42]

For the flood that according to KTA 2207 occurs with a probability of occurrence of 10^{-4}/a (10,000-yearly, HQ10,000), a discharge of 2,100 m^3/s was determined with an associated flood level of 433.33 m above sea level [42]. At this design flood level, the plant site (433.0 m above sea level) would be flooded, but with a design margin of more than 1 m before the water could affect safety-relevant systems and components.

After the event at Fukushima, the design flood level was newly determined for a discharge of 2,100 m^3/s according to the current state of the art in science and technology (two-dimensional calculations). It amounts to a maximum of 432.92 m above sea level. At this design flood level, the plant site is not flooded.

In order to further quantify the design margins against floods exceeding design limits, the water level was also determined in 2012 that would be expected in the event of a 50% increase in the discharge of the Danube (3,150 m^3/s) compared to HQ10,000. This discharge would result in a maximum water level of 433.18 m above sea level, i.e. a water level rise of only 26 cm. A discharge that would result in a water level of 434.50 m above sea level, i.e. that would exhaust the existing freeboard, can be considered impossible. Within the framework of the robustness analysis, the site was therefore assessed as safe during flooding events.

In addition, provisions have been made for the temporary installation of mobile sheet piling to improve the accessibility of access doors. Furthermore, three boats for passenger transport have been procured.

Assessment by the RSK

According to the operator, new analyses have been carried out for the flood levels at the BWR plants (for design floods with frequencies of occurrence of 10^{-4}/a and for a flood with a 50% higher assumed discharge rate). It was found that the plant site remains free of water for the design flood and that the existing freeboard is by far not exhausted even with the increased discharge rate. On this basis, robustness level 1 is fulfilled. Furthermore, due to the lead time for extreme floods, the additional measures that are still available can be taken if necessary.

C.2.5 Concretisation of the recommendation on flooding of the annulus

Recommendation of the RSK [2], Part 1

The following issues should be explained or clarified:

- *Identification of a safety-relevant installation failing in case of a flooding level of 2 m at the lower annulus level. Here, it has to be examined in particular which impacts the flooding of transducers and other electrical and I&C equipment located in the annulus may have on residual-heat removal and the boration of the primary coolant. It has to be shown whether measures may be hindered, prevented or triggered incorrectly.*

- *Taking into consideration this issue, it has to be specified what measures will be reliably available in the different operating phases under the boundary conditions of a design basis flooding of the annulus up to a flooding level of 2 m for the prevention of an impermissibly long loss of vital safety functions. In particular, it has to be shown by which measures*

    - *secondary-side heat removal and, moreover, shutdown into a cold unpressurised, subcritical state are ensured in the short term in case of beyond-design-basis flooding during power operation, and which installations have to be taken into account for it and are available.*

    - *cooling of the fuel pool can be ensured within the required time in case of beyond-design-basis flooding both during power operation and low-power and shutdown operation.*

    - *replacement of the evaporated inventory can be achieved in the short and medium term in case of beyond-design-basis flooding during low-power and shutdown operation with a lowered level in the reactor coolant lines (also demonstrating, e.g. that the accumulator injection system is reliably available and can be activated).*

Furthermore, it has to be shown how in operating phases with flooded reactor pool scenarios with water losses into the annulus from the connected system (RPV - reactor well - fuel pool) are prevented under all operating conditions of the spent fuel pool cooling and purification system (including leakage caused by human error or false triggering of reactor protection signals) and, in case of failure of the precautionary measures provided, be managed.

The recommendation formulated in [2] referred to the boundary conditions specific to PWR plants in Germany. Further considerations on the part of the RSK have shown that also in the case of BWRs, there is the possibility of beyond-design-basis flooding in the reactor building, with the consequence that vital functions might become unavailable.

VGB was therefore asked for information that would allow an assessment of whether beyond-design-basis flooding in the reactor building might lead to a failure of vital functions.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 02 February 2016 [36], slide 15

- VGB letter to the RSK Secretariat of 3 March 2016 [44]
- VGB letter to the RSK Secretariat of 14 March 2017 [46], item. 9
- BStMUV, e-mail of 11 August 2017 [48]

In [36] and [44], VGB addresses the issue of annulus flooding for the BWR. A large leak in the nuclear service water system of redundant system trains 2 and 3 is considered to be a relevant event since the heat exchangers are located in the reactor building (for more detailed information, see Part E2 „Flooding of the annulus BWR“)). In contrast, the heat exchanger of redundant system train 1 is installed in the nuclear services building, so that such an event in redundant system train 1 cannot lead to flooding in the reactor building annulus. Reference was made by VGB to the compartmentalisation of the redundant system train areas within the reactor building up to an elevation of 0 m, so that approx. 1,600 m³ of water can be absorbed by each of the redundant system trains 2 and 3 before any cross-redundancy effects occur. Two-channel sump level measurements are available in accordance with the requirements of KTA Safety Standard 3501, which lead to early alarms (safety hazard alarms) and to automatic measures with shutdown of the affected pumps of the VE, TF and TH systems. In addition, it was pointed out that the entire reactor protection system is installed at level +8.3 m and thus flooding is practically excluded. Thus, according to VGB's presentation, the isolation valves of the feedwater and main steam systems remain controllable, whereby residual-heat removal remains available either via the main steam system RA to the main heat sink or via the opening of the diverse pressure relief valves and the release of steam from the pressure suppression pool via the venting system when feeding the RPV with the feedwater system RL.

If the compartment of redundant system train 2 is flooded, the emergency measure for RPV feeding via mobile pumps is not feasible. Possible measures according to [46] would be the injection of feedwater and heat removal via the main steam lines and, as measures with which time can be gained, the injection of coolant via control rod drive pumps or cooling water injection via the seal water pumps of the coolant recirculation pumps.

Assessment by the RSK:
There is a potential for flooding of the reactor building annulus due to the auxiliary cooling water lines and heat exchangers of redundant system trains 2 and 3 in the respective compartments in the reactor building. Leakage quantities of up to 1,600 m³ do not lead to failures across redundancies. In the event of an assumed flooding of a compartment, the flooding of further compartments takes place through overflowing water.

In undisturbed power operation, the service water pumps are generally switched off and in standby mode (only brief operating states for surveillance testing or temperature maintenance of the pressure suppression pool), so that the potential for a cross-redundancy annulus flooding to occur in power operation can be considered to be low. When the unit is shut down, at least one line of the service water system is in operation.

In the event of an assumed leak in a service water line with water entering the respective compartment of redundant system trains 2 or 3, the respective service water pump is switched off via a limiting function. Flooding of the compartments when the pump is switched off is not a concern at river water levels up to the level of the current design flood due to the geodetic conditions. Any cross-redundancy annulus flooding from the service water system thus requires, in addition to the leak, the failure of this automatic measure and of manual measures, which the shift is requested to take by the safety hazard alarm generated by the flooding.

According to [48], the authorised expert assumes that in the case of the postulated flooding of compartment 2 (3), a relevant transfer of water into the other compartment 3 (2) would only occur after > 40 min. Since diverse measures are available and can be implemented in 20 min or less to stop the inflow of water, the authorised expert considers flooding of both compartments to be very unlikely.

If, independently of this, a cross-redundancy flooding of the annulus were to be assumed, then, according to VGB [44] and [46], in the event of a flooding of the compartments, no flooding of I&C equipment would occur in such a way that further event control would be impeded by false signals due to the height arrangement of the reactor protection system.

Based on this, the RSK is of the opinion that steam removal from the RPV can be ensured even in case of an assumed cross-redundancy flooding of the annulus:

- If the RPV is closed pressure-tight, the steam generated in the RPV can in any case be discharged via the motor-driven diverse pressure relief valves into the pressure suppression pool and the residual heat can be removed from the pressure suppression pool via the venting system.
- If the RPV is not closed pressure-tight, steam is released into the reactor building. The reactor building has overflow apertures to the turbine building that open at a pressure difference of approx. 60 mbar - see Section C.2.11 "Increased consideration of the wet storage of fuel assemblies".

The following options exist with regard to coolant make-up feeding:

- When the flood chamber is filled and the swivel gate is open and when the core is fully discharged, coolant can be made up via the emergency measures for the spent fuel pool - see section C.2.11 Increased consideration of the wet storage of fuel assemblies.
- In plant states with closed RPV in which the feedwater system or the main condensate system is still available, these can be used for RPV feeding.
- Feeding from the condensate storage tank by means of TE and YT pumps. According to the authorised expert, the condensate storage tank is only emptied for testing purposes; it can be replenished according to [52] from the fire extinguishing system by means of emergency measures.[1]

Overall, the RSK comes to the conclusion that there is sufficiently high robustness for the BWR with regard to potential flooding of the annulus.

[1] This measure is available independently of the previous two and is sufficient to replenish the evaporated coolant when the plant is shut down.

C.2.6 Concretisation of the recommendation on load drop

(Drop of a fuel assembly transport cask into the fuel pool, drop of loads into the RPV, inadmissible repercussions on the reactor coolant pressure boundary)

Recommendation of the RSK [2], Part 1

Therefore, the following is recommended:

- *The impacts of the drop of a fuel assembly transport cask into the fuel pool should be analysed regarding the loss of pool water. The possibility of overfeeding a loss of fuel pool water should be checked and specific accident management measures should be introduced, if required.*
- *Likewise, the impacts of the drop of loads into the RPV or onto the connection between RPV and fuel pool established during low-power and shutdown operation should be analysed. If necessary, specific accident management measures should be introduced in dependence of the consequential impacts.*
- *Regarding the handling of loads in the environment of necessary safety-related installations, it should be analysed whether a postulated drop load leads to inadmissible retroactive effects on the reactor coolant pressure boundary or damage affecting more than one redundancy that may lead to "cliff-edge" conditions in the plant.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [15], slides 4 – 6
- VGB presentation of 02 February 2016 [36], slides 16-18
- VGB presentation of 05 November 2014 [15] slides 4-6
- VGB letter of 14 March 2017, item 10 [46]
- BStMUV, e-mail of 11 August 2017 [48]

For the BWR, remarks were made on the postulated load drop of the RPV closure head onto the RPV as well as on the postulated load drop of the transport vessel (for further information, see Appendix F).

Within the framework of the BWR safety analysis, an investigation was carried out by GRS in 1991 on the load drop of the RPV closure head onto the RPV with effects on the support skirt. It was determined there that a risk to the coolability of the fuel assemblies due to the drop need not be assumed [45]. Therefore, the VGB does not see any cliff-edge effect.

For the fuel assembly transport casks, the following applies:

- Transport and handling operations are regulated in the 40-metre manual and are only carried out inside the reactor building using the reactor building crane in accordance with an approved step sequence plan over specially designated/proven areas. The highest transport height of approx. 1.30 m is reached when the cask is lifted over the railing at the fuel pool (1.10 m). Administrative regulations ensure that the transport cask is moved exclusively in the peripheral area of the reactor building between the transport shaft and the separate transport cask pool (slide 17 in [36]).
- Below the transport shaft floor (0 m) is the antechamber of the main interlock (-5 m and -3,5 m), below which there is an empty plant compartment (-8,3 m). Hence any damage of safety equipment through the drop of a CASTOR cask in the transport shaft is not possible.

- Interlocks on the reactor building crane prevent load transport in the pool area under load. Without a key switch, the crane hooks can only be moved unloaded and in the maximum position above the pool. Moving the crane hooks above the pool under load is only possible with a key switch.
- The transport cask pool and the traversing area of the FA transport casks are designed against a drop in accordance with [36]. The scenario of a drop of the transport cask into the transport cask pool was investigated during the construction of the plant with the result that the floor of the transport cask pool can withstand such a drop, so that no potential for a "cliff-edge" effect can be identified here either.
- The toppling-over of an FA transport cask into the spent fuel pool (also due to incorrect positioning on the edge of the transport cask pool and subsequent toppling-over into the spent fuel pool) is excluded by VGB since events with overlapping of incorrect travel of the KTA crane and simultaneous failure of the KTA lift rig in terms of the footnote in the SiAnf, item 2.5 (1) need not to be assumed. Furthermore, unhooking of the hoist in the event of a faulty touchdown on the edge of the transport cask pool with incipient toppling also need not be assumed due to constructive measures in the hoisting gear [48, Annex 1]. Thus, the drop of a transport cask into the spent fuel pool need not be assumed as a whole.

Assessment by the RSK

Even if, in the opinion of the operators, a drop of heavy loads need not be assumed, the possibility of a drop of heavy loads and any effects that might occur were investigated. For the assumed scenarios, the information provided by the operators shows that - if necessary, taking into account emergency measures - the consequences remain so limited that serious effects on the environment need not be assumed. The operators rule out the possibility of the FA transport cask falling into the spent fuel pool as the cask is prevented from passing over the basin by interlocks and administrative regulations, and a combination of a postulated spurious movement and a simultaneous failure of the hoisting gear need not be assumed.

Since even unhooking of the hoist in case of a faulty placement on the edge of the transport cask pool need not be assumed, the RSK agrees with this assessment.

Regarding the further analyses of the operators, the RSK arrives at the following assessments:

- A drop into the transport cask pool was classified as very unlikely but was nevertheless assumed. Due to the design of the transport cask pool, the conclusion is that no relevant damage resulting in leakage will occur.
- For an assumed drop of the RPV closure head onto the RPV, investigations within the framework of the BWR safety analysis had already shown that this would not result in any leakages that could not be overfed.
- No potential for inadmissible damage to the reactor coolant pressure boundary or for cross-redundancy impacts has been identified for other, non-excluded load drop events. This view is followed by the RSK.

Note: The topic "Transport cask on the hook of the building crane during an earthquake" is dealt with in Part D Appendix "Earthquake (PWR and BWR)", Section f "Low-power and shutdown operation, transient operating states" ([23], slides 19-60).

C.2.7 Achieving the safety objectives of accident management measures even in the case of natural external hazard events

Recommendation of the RSK [2], Part 2

The safety objectives of the accident management measures mentioned in Part 2 should also be achieved during or after natural external design basis hazards. In particular, the following aspects should be considered during/after these hazards:

- *limitations of the accessibility of the power plant area and power plant buildings that may have to be postulated,*
- *operability of the accident management measures, and*
- *availability of the remote shutdown and control station.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 35
- Action plan December 2014 [24]
- VGB presentation of 02 February 2016 [36], slide 19

In principle, the accessibility of the plant even under external hazard conditions is regulated by the disaster control plan of the authorities and can be ensured with the technical possibilities of disaster control and the Nuclear Emergency Brigade.

In the course of the robustness analysis, it was shown that the vital safety functions can be maintained solely with the personnel available on the plant and the robustly stored emergency equipment and machinery materials, even assuming that accessibility is impaired as a result of external design hazards (with regard to the power supply required for this, see also Section C.2.8 "Availability of three-phase power supply for vital safety functions").

According to [36], new flood analyses show that the site remains flood-free, so that all emergency measures remain feasible. For the design earthquake, the vital safety functions can be fulfilled even without considering emergency measures. The contribution of additional emergency measures to increasing robustness can therefore be rated as low for the design earthquake. Irrespective of this, the feasibility of all emergency measures was assessed under the conditions of a design earthquake and a design basis flood.

According to [36], emergency measures that are additionally available in the event of a design earthquake even without an electrical power supply are filtered venting, RPV feeding with a mobile pump or fire-fighting vehicle, and make-up feeding of the spent fuel pool with fire-fighting water.

A simultaneous unavailability of the remote shutdown and control stations in case of design basis events affecting the power plant need not be assumed due to the physical separation (remote shutdown and control stations several kilometres away from the plant site and designed against earthquakes according to DIN).

According to the action plan, a systematic review of the robustness of emergency measures, taking external hazards into account, has been carried out and documented. A new remote shutdown and control station has been created to increase availability.

Assessment by the RSK

In the opinion of the RSK, the accessibility of the power plant site and of power plant buildings as well as the operability of the emergency measures are given in case of design earthquakes and design basis floods. With regard to the availability of the remote shutdown and control station, the RSK refers to the required proper implementation of the RSK/SSK's general guidelines on emergency preparedness.

C.2.8 Availability of three-phase power supply for vital safety functions

Recommendation of the RSK [2], Part 2

a) *It has to be demonstrated that the supply of three-phase alternating current required for the vital safety functions is ensured even if there is no grid connection available for up to a week.*

b) *In the case of a postulated station blackout, the vital safety functions have to be maintained or re-established in time before reaching "cliff-edge" effects. This involves the following:*
   - *The direct current supply required for the vital safety functions also has to be ensured if three-phase alternating current supply is not available for up to 10 hours. An independent battery charger for recharging of relevant batteries, which is protected against external hazards and kept available, can be credited if it is ensured that there is sufficient grace time for connection and use of such a battery charger.*
   - *Furthermore, it has to be demonstrated that three-phase alternating current supply can be re-established within a plant-specifically determined non-intervention time by means of back-up units. From the point of view of the RSK, this includes:*
     - *layout of standardised hook-up points protected against external hazards outside the buildings for supplying the systems required to maintain the vital safety functions. The aim of an adequate layout of the hook-up points is to ensure supplying the emergency power busbars and, if necessary, emergency power busbars required for it without impairing the degree of protection of the respective buildings (e.g. ventilation isolation and flood protection) against the respective external hazard. The hook-up points are to be installed so that they have no retroactive effects,*
     - *at least one mobile emergency power generator protected against external hazards with sufficient capacity for supplying one redundant residual-heat removal train.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 44-49,
- VGB presentation of 05 November 2014 [15], slides 9, 13, 16 and
- Action plan December 2014 [24]
- VGB presentation of 02 February 2016 [36], slide 20

- VGB concept, Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015 [42]
- VGB letter of 14 March 2017 [46]

The following measures and installations were realised within the framework of the VGB concept:

- For unavailability of the grid connection for up to one week:
  - Extension of diesel running time up to seven days using secured fuel stocks (with shutdown of diesels that are not needed (fuel saving)), additional diesel tank, stockpiling of lubricants, construction of central tank facility.
- Maintaining vital safety functions in SBO as follows:
  - The battery-buffered power supply is available for at least 5 hours and up to 15 hours for individual busbars in the event of an assumed complete power supply failure.
  - Opening of the three diverse pressure relief valves is possible during the first 10 hours of the accident sequence. An optimisation in the operating manual for opening the three diverse pressure relief valves has been carried out.
  - Implementation of an additional mobile emergency diesel per unit with hook-up points and an additional mobile submersible pump with its own power supply.
    With the help of the emergency diesel, which can be made ready for use within 80 minutes, the direct current supply to one of the safety subsystems 2 or 3 can be ensured via a 10-kV emergency power switchgear.
  - Use of fire-fighting vehicles for RPV injection.
  - Measures to increase the feed pressure for RPV injection by blocking safety valves in the service water system.

With the mobile diesel generator mentioned above, it is not possible to additionally supply a complete residual-heat removal chain [46, item 6]. However, from the point of view of VGB, this is not necessary since the additional independent residual-heat removal system (ZUNA) already provides a diverse option for heat removal. Furthermore, due to the possibility of feeding water directly into the reactor pressure vessel (RPV), heat removal is possible entirely without operating pumps of the emergency core cooling and residual-heat removal system. For this reason, the decision was made against a diesel generator that is heavier and therefore less convenient to handle.

Assessment by the RSK

If the concept envisaged by VGB is implemented properly, the recommendation of the RSK to safeguard the vital functions in case of an assumed unavailability of the grid for up to one week and in case of an assumed SBO is fulfilled to a sufficient extent.

C.2.9 Review of the accident management concept with regard to injection options for the cooling of fuel assemblies and for ensuring subcriticality

Recommendation of the RSK [2], Part 2

Review of the accident management concept with regard to injection option for the cooling of fuel assemblies and for ensuring subcriticality. Here, the following aspects have to be taken into account:

- *Availability of mobile pumps and other injection equipment (hoses, fittings, couplings, etc.) protected against external hazards as well as of boron under consideration of the specified grace time for preparation and delivery.*
- *Water intake points whose availability is also ensured after an external hazard.*
- *Possibilities of injecting water into the steam generator, the reactor coolant system and, if required, the containment (in the latter case also taking into consideration the higher back pressures) without the need to enter areas with high risk potential (dose rate, debris load) and to be able to compensate local destruction (e.g. by installed and physically separated injection paths).*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slide 37
- VGB presentation of 05 November 2014 [15], slides 13, 16-18
- VGB Power Tech e. V., Answers to questions posed by the Ad-hoc Working Group on Robustness relating to BWR, 02 February 2016 [36]
- VGB concept, Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015 [42]
- VGB letter to the RSK secretariat of 14 March 2017 [46]
- Volker Noack, e-mail of 29 March 2017 [47]

Various injection options into the RPV were already available for KRB in the emergency manual before Fukushima (e.g. injection with TH filling pumps, control rod drive pumps, seal water pumps, passive injection with steam pressure from the feedwater tank).

The VGB concept now includes the following additional measures for water injection for the BWR:

- Injection into the RPV by submersible pump with connection to the VE2 service water system in the circulating water structure and injection via TH2, as well as by the additional option of using a special fire-fighting vehicle as mobile pump.
  According to [46], the emergency manual (NHB) recommends starting the injection process at a RPV pressure of less than 7 bar_{abs}. The emergency manual also provides for injection up to an RPV pressure of 10 bar_{abs}. The mobile pump is limited to a pressure of 13 bar_{abs} with a safety valve, which means that it pumps into the RPV at up to 13 bar_{abs}. (13 bar_{abs} is not the zero head, the injection rate at this pressure is still 18 kg/s). For this measure, the existing safety valves in the VE system were blocked after investigations of the pipes showed that no pressure limitation is required in the pipes of the VE system [46]. As a result, it is possible to inject into the RPV via mobile equipment even in postulated beyond-design-basis scenarios in which the venting system is used to remove heat from the containment.
- In addition to the already existing emergency measure (connection of a fire extinguishing hose to the existing fire extinguishing pipe of the +40 m level at the level of the spent fuel pool), a

device is permanently installed as an emergency measure for cooling the spent fuel pools so that there is no need to enter endangered spaces in case of demand - see Section C.2.11 Increased consideration of the wet storage of fuel assemblies.

The storage of emergency equipment was reviewed with regard to its availability in case of external hazards. Optimisations were made: for example, the mobile pumps are stored together with a vehicle for their transport in a heatable tent. This means that the equipment will still be available with a high degree of reliability in the event of beyond-design-basis earthquakes.

Assessment by the RSK

The VGB concept fulfils the recommendations of the RSK for the external-hazard-safe storage of mobile components and with regard to additional injection options into the reactor pressure vessel and the spent fuel pool. Due to the fact that in BWRs, boration of the coolant is not required to ensure subcriticality, the above-mentioned RSK recommendation on the provision of boron is not relevant here.

C.2.10 Filtered containment venting during or after external natural design basis hazards and in the event of a station blackout

Recommendation of the RSK [2], Part 2

The filtered containment venting system has to be designed such that pressure relief can also be repeatedly performed during or after natural external design basis hazards and in the event of a station blackout. Furthermore, the effectiveness of installations to reduce hydrogen in the containment is to be ensured accordingly.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 33, 44 and
- VGB presentation of 05 November 2014 [15]]
- VGB presentation of 02 February 2016 [36], slides 20 and 21
- VGB letter of 14 March 2017 [46]

The underlying design and layout documentation of the containment venting system under design earthquake loads was evaluated. According to [36], the analysis showed that the pressure relief system remains available during design earthquakes. However, there is no seismic qualification for the system. The main prerequisite for availability during design earthquakes was the stability of the turbine building, which could be demonstrated by means of a stability assessment (see also C.2.3 "Concretisation of the recommendation on earthquakes").

If integrity is ensured, the formation of ignitable hydrogen-air mixtures on the relief path is not a concern since the corresponding exhaust gas line is routed to the end of the stack and thus air admixtures cannot occur on the exhaust air path.

Filtered venting can also be initiated and terminated via handwheels in the absence of a power supply. Radiological shielding is provided by the wall between the two hand wheels and the pressure relief line. This means that venting can be carried out repeatedly even if the power supply is unavailable.

Assessment by the RSK

According to VGB's statement on the design of the system for filtered venting for the BWR, it has been shown that the system can withstand the loads of the design earthquake. In particular, this could be shown by a stability analysis of the turbine building. The measure can be repeated. Thus, the recommendation of the RSK is fulfilled.

C.2.11 Increased consideration of the wet storage of fuel assemblies

Recommendation of the RSK [2], Part 2

Increased consideration of wet storage of fuel assemblies in the accident management concept, taking the following aspects into account:

- *Possibilities of injecting water into the wet storage facility for fuel assemblies without the need to enter areas with high risk potential (dose rate, debris load) and to be able to compensate local destruction (e.g. by installed and physically separated injection paths).*
- *To ensure evaporation cooling: updating of the safety demonstrations for the fuel pool, reactor cavity, setdown pool, reactor cavity seal liner at boiling temperature.*

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 44, 50-52, 67,
- VGB presentation of 05 November 2014 [15], slides 13, 18
- Action plan December 2014 [24]
- VGB presentation of 02 February 2016 [36], slide 22
- VGB concept, Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015 [42]
- VGB letter to the RSK secretariat of 3 March 2016 [44]
- VGB letter of 14 March 2017 [46]

The concept of VGB pursues the goal of safeguarding fuel cooling even under beyond-design-basis failure assumptions in such a way that the occurrence of serious fuel damage including H_2 formation can be excluded. Essential components of the concept for the BWR are:

- Additional provisions for coolant make-up in the spent fuel pool.
  In the event of an assumed complete failure of the pool cooling systems, coolant make-up can be by injection via a fire-fighting vehicle or via a mobile pump. The connection to the feed line is made via a permanently installed connection point by means of a flexible hose. The connection point is located in a stairwell in the reactor building outside the confinement. Access from the outside can be via an

escape door located nearby. This means that it is not necessary to enter any potentially hazardous areas of the reactor building.

- Removal of the decay heat in the spent fuel pool by steam release into the reactor building and make-up feeding.
  The required feed rate to compensate for evaporation is a maximum of 7 kg/s. The non-intervention time for feeding to prevent the water level from dropping into the area of the FA's active zone is more than one day, even when the spent fuel pool is fully loaded. For evaporation cooling, water temperatures of 100 °C were considered. Higher temperatures than slightly above 100 °C do not have to be considered since no relevant pressure build-up occurs in the reactor building outside the containment. The reactor building has overflow openings to the turbine building that open at a pressure difference of approx. 60 mbar. The turbine building is also equipped with pressure relief dampers to the atmosphere that open temporarily at a pressure difference of more than 12 mbar. Overall, the maximum overpressure above the spent fuel pool is therefore approx. 60 mbar, so that the temperature of the pool can only slightly exceed 100 °C at the surface. Heat removal from the fuel pool is therefore ensured as long as the fuel assemblies are sufficiently covered or surrounded by water. The addition of pool water may be necessary in the long run.

- Ensuring the integrity of the spent fuel pool during evaporation cooling
  Engineering analyses were carried out according to [44], [46] to show that the BWR pool structure can withstand a pool water temperature of about 100 °C without damage. A temperature higher than about 100 °C is not a concern since no relevant pressure build-up can occur within the reactor building due to open exhaust air ducts and/or pressure relief dampers between the reactor building and the turbine building during evaporation of the pool water.

Assessment by the RSK

If the concept described above is implemented, and also thanks to the long non-intervention times until injection into the fuel pool has to be restored after a postulated failure of the pool cooling system as well as on account of the comparatively simple measures for make-up feeding into the spent fuel pool, the recommendation of the RSK is fulfilled.

C.2.12 Short-term implementation of the Severe Accident Management Guidelines (SAMG)

Recommendation of the RSK [2], Part 2

Furthermore, the RSK considers it necessary that the Severe Accident Management Guidelines (SAMG) be implemented in the short term.

Concept of VGB for the implementation of the recommendations

- VGB presentation of 11 December 2013 [9], slides 38, 44, 56-58
- VGB presentation on SAMG BWR of 02 December 2014 [12]

- VGB presentation of 05 November 2014 [15], slide 17
- VGB letter of 14 March 2017 [46], item 19

The "Severe Accident Management Guidelines - SAMG" have been implemented at the BWR plants in operation and their contents have been imparted and practiced the in context of an emergency exercise. The SAMG consist of two volumes. Volume A of the SAMG contains instructions for the mitigation of accident sequences for the plant states of power operation and low-power and shutdown states of the reactor plant and the spent fuel pool to be considered. It contains instructions for determining relevant plant data, for diagnosing the extent of core damage and for diagnosing and determining the containment condition. This is followed by the determination of the strategy relevant to the established plant condition (six different strategies are available in the SAMG), which in turn comprises various blocks of measures that are to be worked through. Volume B of the SAMG, which contains background knowledge on core meltdown accidents, justifications for the individual strategies and explanations of the tools, serves as an additional aid. This volume is not necessary for accident mitigation but serves as an additional reference work.

According to VGB [46], the decision to apply the SAMG, the strategy selection and the control of whether initiated emergency measures are successful are based on measurements which, because of their design, can still provide reliable information even in the event of damage to the core and/or the containment. For each measurement referenced in the SAMG, the measuring/indication range and use under the ambient conditions prevailing during a core meltdown event (for example, pressure, temperature, humidity, radiation) were verified. In cases where the reliability of a measurement is not beyond doubt, suitable substitute measurements or simple aids for validating the respective plant parameters are specified in the SAMG. Information that can be derived indirectly or in an unconventional way from the measured values of other measuring points is also used.

The implementation of a SAMG concept in the BWR plants was carried out after approval by the supervisory authority of the necessary modifications to the BHB/SLS[2]. A positive opinion has been issued by the supervisory authority's expert. The recommendation has been implemented.

Assessment by the RSK

SAMG have been implemented in the BWR plants. Technical discussions with the participation of GRS have shown that there are no open generic questions regarding the structure of the SAMG. From the feedback of experience, e.g. during emergency exercises, indications for optimisation of the depth and scope of representation can be implemented during operation, if necessary.

The authorised expert confirms that the relevant measurement points are supplied by the uninterruptible power supply and that long-term power supply can be ensured by the mobile diesels (see "C.2.8 Availability of three-phase power supply for vital safety functions").

The RSK additionally points out the following, which the operator should consider in the further optimisation:

[2] operating manual (Betriebshandbuch) / accident sequence diagram (Störfall-Leitschema)

- The effects of the core damage states dealt with in the SAMG on the usability of the infrastructure should be systematically analysed in the sense of a worst-case consideration. The effects of unavailabilities identified therein should be taken into account in the presentation in the SAMG.

C.3 RSK Statement "Loss of the ultimate heat sink" from the 446th RSK meeting on 05 April 2012

C.3.1 Measures to check and, if necessary, improve the reliability of the ultimate heat sink with regard to blockages of the cooling water intake.

Recommendation of the RSK [3]

The RSK considers it necessary to re-assess the ultimate and, if existing, the alternate heat sink site-specifically, taking into consideration the operating experience gained in Fukushima and in other plants.

Assessment by the RSK

These are site-specific aspects in the area of design that were not considered further here.

C.3.2 Measures to strengthen the reliability of the ultimate heat sink with regard to the occurrence of rare external hazards

Recommendation of the RSK [3]

The RSK recommends checking whether the assumptions for flood events also take into account the dynamic peak loads of incoming tidal waves to be expected in the area of the cooling water intake.

In connection with the re-assessment of flood protection as well as of the design against earthquakes and other very rare events, such as aircraft crashes and their impacts in the vicinity of the plant, it has to be determined if all failure causes that may result from these events have been considered in the design of the ultimate heat sink to the extent required.

Assessment by the RSK

These are site-specific aspects in the area of design that were not considered further here.

C.3.3 Measure to control the loss of the ultimate heat sink

Recommendation of the RSK

The RSK recommends:

Residual-heat removal from the plant and the spent fuel pool must be ensured in all plant operating states also in case of a loss of the ultimate heat sink due to failure causes in the area of cooling water intake and cooling water return by an alternate heat sink (possibly also different heat sinks in combination). The *installations required for it must at least meet the requirements for accident management measures and their effectiveness has to be demonstrated. ...*

The RSK recommends - if not already implemented – introducing an additional accident management measure so that cooling water can be supplied to the nuclear component cooling systems and be discharged again. Supply can be provided through mobile devices. The quantities supplied must be sufficient for the removal of decay heat from the reactor and spent fuel pool and of the lost heat of the components required for such a cooling operation.

The RSK recommends providing appropriate accident management measures to ensure cooling water return for plants having a CCF potential due to the redundancy-wide merging of the cooling water return lines. [3]

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [15], slide 13
- VGB presentation of 02 February 2016 [36], slides 23 to 25
- VGB concept, Robustness analysis to check the effectiveness of vital functions in the event of external and internal beyond-design-basis impacts, BWR, 27 March 2015 [42]
- VGB letter of 14 March 2017 [46]

The following are available as alternate heat sink:

- Coolant make-up and residual-heat removal by the additional independent residual heat-removal (ZUNA) system
  This allows injection into the RPV as well as the cooling of the pressure suppression pool or the reactor well via the system's own auxiliary cooling water system VE4 with the wet cell coolers. The permanent pressure relief of the RPV into the pressure suppression pool (with the RPV closed) can be achieved by opening the three motor-driven diverse pressure relief valves.
- Emergency measure of early opening and keeping-open of the three diverse pressure relief valves in combination with filtered venting and RPV injection bypassing the nuclear intercooler (information on the initiation of the measures in the operating manual, for implementation in the emergency manual).
- The spent fuel pool can be replenished via the additional permanently installed line. The heat can be removed via evaporation cooling. Since the reactor building is permanently open to the turbine building via pressure relief dampers after reaching 60 mbar overpressure, the temperature of the pool water outside the fuel channels cannot reach more than 102 °C (see C.2.11 "Increased consideration of the wet storage of fuel assemblies").

From VGB's point of view, an additional emergency measure with which cooling water can be fed into and discharged from the closed cooling water system for reactor services is not necessary since with the additional independent residual-heat removal system and the possibility of direct RPV feeding via emergency measures, there are further options for cooling even in the event of a failure of the ultimate heat sink.

Assessment by the RSK

The complete loss of the auxiliary service water supply from the receiving water can be controlled via the additional independent residual-heat removal system and, in addition, by filtered venting in combination with the injection of water into the RPV. The objective of the RSK recommendation is thus fulfilled.

C.4 RSK Statement "Assessment of the coverage of extreme weather conditions by the existing design" from the 462nd RSK meeting on 06 November 2013

Recommendation of the RSK

At its 457th meeting held on 11 April 2013, the RSK recommended "that analyses should be conducted to demonstrate robustness against design basis weather conditions with a return frequency of 10-4/a in line with international developments (ENSREG, RHWG/WENRA). As far as impacts in this frequency range cannot be determined with sufficient reliability, effective management of events and a high level of robustness should be demonstrated deterministically using engineering judgement." In addition, it was suggested with a view to robustness that impacts beyond these impacts should be taken into account by engineering estimates for the determination of safety margins. [5]

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [16],
- VGB presentation of 17 July 2015 [17],
- VGB letter of 24 August 2015 [27]
- VGB letter of 03 March 2016 [30].

According to VGB, for all of the impacts discussed below, credit can be taken for their control that the respective plant can be shut down sufficiently quickly and thus the demand for cooling water can be reduced to the amount required for vital functions.

A1 – Freezing rain / ice storm / snow storm

Essentially, it was assessed whether snow or ice deposits could impair vital functions (e.g. supply of combustion air for diesels, circulation of cooling water). Among other things, the following precautions have been implemented or measures are planned:

- physically separated air intake openings oriented differently to the wind direction, partly protected by nearby structures,
- cooling water installations located outside the buildings at frost-proof depths in the ground,
- monitoring by measurements and electrical heating of critical points in multiple-cell coolers,
- options for bypass operation to avoid excessively low temperatures in multiple-cell coolers,
- instructions for visual inspections of critical points and, if necessary, for the removal of deposits.

A2 – Ice floes (on the receiving water)

Essentially, it was assessed whether the supply of cooling water could be impaired by ice floes and pack ice formation. The following precautions have been implemented or measures are planned to counteract this:

- in many plants, intake from de facto standing water without potential for pack ice formation,
- cooling water intakes are very deep (approx. -4 m) and thus frost-proof,
- in the case of intake from outflowing waters, intake structures are aligned with the flow in such a way that clogging due to pack ice formation is very unlikely,
- depending on the location, it is possible to add preheated return cooling water to the intake,
- site-specific diverse heat sink with capacity for the shut-down plant.

A3 – Air temperature extremely low

Essentially, it was assessed whether liquid media could freeze or become too viscous due to temperature reduction. The following precautions, among others, have been implemented against this:

- safety-relevant systems are largely housed inside buildings in closed, heated and monitored rooms,
- piping of safety-relevant systems outside the buildings are located at frost-proof depths in the ground or protected against frost effects,
- admixture of antifreeze in the instrument lines of the emergency diesel generators,
- preheating of emergency diesel generators with auxiliary equipment,
- operational testing of diesel generator sets when used in regions with extremely low temperatures; among others, no adverse effects are known from NPPs in Russia, Canada or Finland.

A4 – Ambient temperature extremely low

Covered by combinations of other weather effects considered and the corresponding countermeasures.

A5 –Slush ice/floating ice

Essentially, it was assessed whether slush ice and floating ice could clog the strainers of the cooling water purification system and thus impair the supply of cooling water.

Under the site conditions of the installations, the occurrence of slush ice and floating ice is very unlikely. Regardless, impacts are controlled by the same countermeasures as for A2.

A6 – Receiving water level extremely low for prolonged periods of time

Essentially, it was assessed whether a sufficient supply of cooling water remains. For this purpose, the following measures, among others, have been realised:

- intake height of the intake structures a few metres below the lowest measured water level,
- monitoring by measurements (especially differential pressure, filling level) of critical points,

- reduction of the required flow rate when limit values are reached,
- use of additional water supplies (wells, water reservoirs),
- in some installations, safety-related independence from the receiving water by means of multiple-cell coolers.

A7 – External fire

Essentially, it was assessed whether an impermissible effect of fire gases and heat on safety-relevant equipment is possible:

- no large fire loads in the vicinity of the plant,
- monitoring outside of the protected-area perimeter for small fires that may nevertheless occur, and their suppression,
- sufficient distance between safety-relevant plant components and the perimeter fence of the plant to avoid direct exposure to fires,
- in the event of the occurrence of flue gases, ventilation isolation of the intake air in the safety-relevant buildings,
- greater physical separation of the intake openings of the various emergency power diesels, use of equipment (especially filters) that is robust against smoke loads,
- proven operating experience of other diesels under much less favourable conditions (e.g. firefighting).

A8 – Prolonged periods of heavy rain

Essentially, it was assessed whether an entry of water into buildings important for safety is possible. The following precautions, among others, have been implemented against this:

- For flood-free sites: Extreme precipitation can run off the plant site even if the surface drainage systems are assumed to be unavailable, without impairing safety-critical equipment. In particular, the drainage conditions in the area of the emergency diesel buildings were checked. The emergency diesel buildings represent a high point. There are thresholds at the building in the order of 15 cm in height that prevent rainwater from entering.
- For locations with flooding on the plant premises: watertight design of the buildings up to above the maximum flood level. This covers considerations for heavy rainfall.
- Protection of the air intake against water entry.

The German nuclear power plant operators have generally reviewed the issue of water drainage in the area of the emergency diesel building for all plants. Due to the structural conditions, the terrain profiles in the area of the emergency diesel building and the geodetic height profiles of the terrain derived from the measurement profiles during floods, an entry of water into the building during heavy rain can be ruled out due to conservative boundary conditions for the runoff of heavy rain.

A9 – Strong wind, tornado

Due to the design of the structures required for vital functions against aircraft crash and blast wave, the overpressures that are possible during high winds and tornadoes are covered.

The negative pressure possible during a tornado (maximum approx. 0.2 bar) cannot affect the stability and function of these structures and their installations, even if the opening of individual escape doors as a result of the negative pressure cannot be ruled out.

A10 – Sand storms and dust storms

Essentially, it was assessed whether the ingress of sand or dust into safety-relevant buildings can lead to impairments of safety-relevant systems. Among other things, the following precautions have been taken or measures are planned:

- no major sand and dust reservoirs that can be mobilised by winds in the vicinity of the sites,
- staggered filtration of the supply air for the buildings (screens against larger airborne particles, treatment sections to adjust the humidity, temperature and purity of the air),
- intake of the combustion air of the diesels via their own filters, physically separated intake options,
- differential pressure monitoring of the filters with signalling, filter replacement or filter cleaning is possible if required,
- operating experience shows that the filters of the emergency diesel generators in particular have large safety margins against dust contamination. In one plant, for example, no increase in differential pressure across the air filters of the emergency diesel generators was detected during operation of the emergency diesel generators within the scope of in-service inspections (ISI) and with filter service lives of eight years.

VGB comes to the conclusion that due to the design margins and the countermeasures possible in the installations, a high level of robustness of the installations is ensured, even against the effects of extreme weather conditions exceeding the design basis events. Probabilistic considerations were not carried out, as there is usually no sufficient statistical basis for weather conditions in the range of a frequency of 10^{-4}/a or even less.

Assessment by the RSK

The approach of estimating and evaluating the margins in the design of the plants in relation to the weather-related design basis events instead of relying on occurrence frequencies in the range of 10^{-4}/a or even less was also included as an alternative in the RSK recommendation.

The RSK considers the assessment that due to the design concept and the design margins there is a clear robustness against these impacts to be plausible, despite the non-existing quantification of margins.

With regard to impact A1 (Freezing rain, ice storm, snow storm), the diesel combustion air intake is considered to be sufficiently robust if the weather protection screens can be electrically heated.

With regard to impact A6 (Receiving water level extremely low for prolonged periods of time), sufficient robustness is considered to be achieved if the cooling required for safety reasons can be carried out one day after shutdown with 10 % of the discharge volume required for full-load operation [39].

On the topic of "lightning effects", reference is made to the separate RSK Statement on "Lightning with parameters above the standard lightning current parameters" [51].

C.5 RSK recommendation "Hydrogen release from the containment" from the 475th RSK meeting on 15 April 2015

Recommendation of the RSK

Recommendation 1: (PWR)

Recommendation 2:

With regard to the release of hydrogen into rooms outside the containment of the BWR-72 boiling water reactor (see Section 3.3), measures have to be introduced in the Severe Accident Management Guidelines (SAMG) to remove the air-hydrogen mixture from the rooms of the reactor building in which a combustible mixture may be formed. In this respect, the possibilities of activity retention have to be taken into account.

Recommendation 3: (PWR)

Concept of VGB for the implementation of the recommendations

- VGB presentation of 05 November 2014 [15] and
- Action plan of December 2014 [24]
- VGB letter of 14 March 2017 [46]
- BStMUV, e-mail of 11 August 2017 [48]

In the German Action Plan, under N7, it is required for spent fuel pools located inside the reactor building but outside the containment that passive safety systems be installed to prevent the accumulation of explosive hydrogen above the spent fuel pool area. These systems should also be effective for a long-term station blackout after 10 hours. For the two BWR plants, twelve passive autocatalytic recombiners were installed in the reactor building area above the spent fuel pool ("40-m" level) to limit the resulting hydrogen concentration there during an accident sequence.

In the RSK recommendation "Hydrogen release from the containment" of the 475th meeting of 15 April 2015 [7], it is recommended for the BWR with regard to the control of hydrogen outside the containment to provide measures or procedures in the "Severe Accident Management Guidelines (SAMG)" to be able to flush out the air-hydrogen mixtures from the rooms of the reactor building in which ignitable mixtures may occur.

In [46], VGB states that corresponding specifications were included in the SAMG during their preparation. Among other things, the SAMG stipulate that the accident subatmospheric pressure system should be operated at an early stage. In case the operation of the accident subatmospheric pressure system is not possible, the possibility of using the natural draught of the stack for subatmospheric pressure maintenance is pointed out. In the course of processing the RSK recommendation, a note was added to establish a certain air flow rate in rooms of the reactor building.

In addition, VGB explained that all strategies in the SAMG contain a block of measures for activity retention in the reactor building [48, annex feedback discussion]. This provides for ventilation isolation of the reactor building and ensuring that the doors to other buildings or the surrounding area are closed. Furthermore, the operation of the TL26 accident subatmospheric pressure system in the event of an accident has to be controlled and, in the event of a failure of the fans of the accident subatmospheric pressure system, has to be aligned with the natural draught of the stack. In the event of a filter failure of the TL26 accident subatmospheric pressure system, it has to be shut off and the exhaust air of the reactor building has to be guided through the TL81/84 activated-carbon filter exhaust air system.

The block of measures for containment venting serves to prevent the entry of hydrogen from the containment into the reactor building by lowering the pressure in the RPV and reducing the amount of hydrogen. The reduction of hydrogen in the reactor building is achieved by the block of measures for making the autocatalytic recombiners available on the +40-m level (reactor service floor).

The TL 26 accident subatmospheric pressure system exhausts at three points in the reactor building annulus at a level of +36.50 m. Should ventilation isolation have occurred, four overflow dampers from the auxiliary building will open automatically in order to limit the negative pressure in the reactor building. These four overflow dampers from the auxiliary building to the reactor building are distributed to different levels (+16.8 m, +18.5 m, +21.5 m and +28.5 m). Thus, in the room areas relevant for containment leakages (above +18.5 m), there is a directional flow from bottom to top during operation of the accident subatmospheric pressure system.

If challenged, the accident subatmospheric pressure system would exhaust a gas volume of 4,000 m^3/h. The room area in the containment above the +18.5 m level has a volume of 7,500 m^3. This would provide a sufficient exhaust of mixtures containing hydrogen in the area above the +18.5 m level.

Assessment by the RSK

With the installation of the 12 passive autocatalytic recombiners above the spent fuel pool of the two BWR units, requirement N7 of the German Action Plan is fulfilled. However, based on the results of the robustness analyses and the robustness of the BWR plants derived from them, VGB assesses the relevance of this measure as being low.

Under the given boundary conditions, the measures described by VGB are suitable to avoid impermissible hydrogen concentrations in the reactor building.

Part D Appendix Earthquake (PWR and BWR)

The explanations in this appendix on earthquakes supplement the statements in Chapters B.1.3/C.2.3 "Concretisation of the recommendations on earthquake" and B.1.6/C.2.6 "Concretisation of the recommendations on earthquake".

The explanations are taken from the report of AST to the RSK of 15 October 2015 [40]. Insofar as the RSK Working Group on Robustness dealt with additional questions, the text passages supplemented on the basis of information from VGB are printed in italics.

In the following, the situation with PWR plants is discussed first, followed by an explanation of the extent to which the results can be applied to BWR plants.

Reports by the operators [20], [23]

In their reports, the operators focused on the following aspects:

a) seismic design principles applied to the NPPs with power operation during the period of their construction,

b) criteria for the assessment of robustness against earthquakes,

c) assessment of the design margins against seismic loads for the plants in operation,

d) seismic probabilistic safety analysis (SPSA),

e) applicability of the results of more detailed investigations to other plants and ability of plants with a design earthquake < 0.1 g horizontal peak ground acceleration (PGA) to withstand horizontal accelerations up to 0.1 g.

f) robustness in low-power and shutdown operation and during short-term plant states.

The slide numbers given in the following refer to the VGB presentations "Investigation of the robustness of German nuclear power plants during earthquakes" of 25 June 2015 [20] and 17 July 2015 [23].

a) Principles of seismic design ([20], slides 8 - 20; [27])

The operating nuclear power plants were designed to withstand external hazards such as earthquakes, aircraft crash and blast waves, in which in particular a horizontal dynamic load acts on the structures. However, during the construction period, it was often the case that no complete dynamic verifications were provided for the entire design chain but - tending to be conservative - dynamic verifications were supplemented by quasi-static verifications of the load transfer.

The aim of the design was to ensure that - even taking into account data scatter and model uncertainties in the calculations - the stress from the impact would with sufficient certainty be less than the resistance of the affected installation.

b) Criteria for robustness ([20], slides 22 – 29)

Various seismic parameters can be used as reference values for the assessment of robustness. The occurrence frequency stands out as an adequate reference value, which is also shown by the international comparison. For the robustness assessment, an earthquake with an occurrence rate of 10^{-6}/a can be assumed, which corresponds to an occurrence frequency reduced by one order of magnitude compared to the design value (KTA 2201.1). The reference value "+ one intensity level" proposed by the RSK, on the other hand, is less suitable, since - with regard to the occurrence frequency usually used internationally for the assessment - this may mean a change of the occurrence frequency by about 1 order of magnitude or even more, depending on the site, i.e. it represents different things for different sites.

c) Design margins ([20], slides 31 – 45; [25]; [27])

In order to limit the computational effort, the overall calculation was divided into calculation steps for the design of the plants, which were decoupled from each other for simplification, i.e. assumptions were made at the interfaces of the calculation steps in such a way that, on the one hand, the possible variation of the results in the previous calculation step was covered and, on the other hand, (attenuating) repercussions on the previous calculation step were neglected.

Furthermore, to simplify the calculation, linear-elastic methods were used as a matter of principle, i.e. the attenuations and margins associated with plastic deformations were only taken into account to a limited extent. With regard to the material properties, characteristic values that were on the unfavourable side were used.

With the procedure according to a) and c), a design was achieved that had to contain clear margins. The existence - and partly also the quantification - of these margins was repeatedly confirmed by the evaluation of experience with severe earthquakes, by experimental investigations (e.g. bending and vibration tests) as well as by exemplary detailed comparative calculations.

The overall high conservatism, which is based on the engineering approaches in the design and is proven by experimental investigations and experience from destroying earthquakes (no differentiation between BWR and PWR plants), thus applies to both plant types according to [27].

Significant margins are also present in the buried auxiliary service water pipes, which were not designed and verified according to the generic specification on basic safety but according to a special specification [30].

With regard to anchorings using conical split sleeves remaining in the installations, there are sufficient load-bearing capacity margins in the beyond-design range due to margins on the impact side, the support with anchor groups, and the low number of crack opening cycles in earthquakes in Germany. [30], [31].

d) SPSA ([20], slides 47 – 98)

The procedure for a complete SPSA was presented using the examples of 2 PWR plants, one with a site of very low seismicity and sedimentary subsoil and another site with increased seismicity for German conditions and solid subsoil, which means that the selected sites are representative of the spectrum of sites in Germany. An SPSA essentially comprises the following elements:

- Determination of the site-specific frequencies of earthquake impacts. Result for the examples: The design earthquake/PGA value on which the design was based corresponded to an occurrence frequency of approx. 10^{-5}/a at one site and approx. 10^{-7}/a at the other site.

- Determination of earthquake-related failure probabilities of structures, systems and components (SSC):

  - Compilation of a comprehensive list of SSC to be considered:
    Components were compiled that may have relevance in the context of the PSA analyses with regard to the determination of the frequency of hazard and core damage states (*Gefährdungszustände / Kernschadenszustände* - GZ / KSZ).

  - Plant inspection for the selection of the SCC to be considered in more detail:
    E.g. selection of components that should be considered more closely due to the design and installation situation.

  - The quantification was carried out using so-called fragility curves and the so-called safety margin factor procedure, with which the median values (A_M) and the uncertainty parameters β_R and β_U for the fragility curves were determined. In the safety margin factor procedure, it is determined for those parameters that are included in the calculation of the impact or resistance which factor lies between the value typically used conservatively in the design and the value to be expected in reality and thus describes the margin in the design in each case. These factors thus depend on how conservatively the design was carried out during construction; they can therefore only be applied from one plant to another if the same design concept (standardisation, verification philosophy, calculation methods, ...) was used. The product of these individual factors then results in the safety margin factor valid for the respective component or structure under investigation.

  Result for the investigated plants: Related to the PGA value with an occurrence frequency of approx. 10^{-5}/a, design margins between 2 and 6 - irrespective of whether it is a vital component or not - were determined, i.e. at least one margin for a PGA value higher by a factor of 2 was confirmed (see slide 71).

Note: The SPSAs presented were not specifically prepared to determine the minimum given robustness, i.e. for several of the factors addressed above, only a part of the existing margins was credited. However, a quantification of the remaining margins would require an increased investigation effort.

- Determination of the frequencies of earthquake-related hazard and core damage states (GZ / KSZ)

  - Identification of earthquake-related initiating events:
    This involves transients caused by earthquake-related failures of structures and process engineering as well as electrical and I&C equipment, e.g. failure of the external power supply due to failure of the ceramic insulators.

  - Creation / adaptation of the plant model in the PSA.

  Quantification

  In accordance with international practice, the results can be converted into a so-called "plant level fragility" by plotting the conditional system unavailability (frequency GZ / KSZ divided by occurrence frequency of earthquakes) for each earthquake considered above the associated PGA.

  Result for the investigated plants: For the PGA value with an occurrence frequency of approx. 10^{-5}/a, a conditional probability of clearly < 10^{-3} per event was determined in the plant with the relatively lower margins for a system unavailability that could lead to a core damage state; for an earthquake with doubled PGA value, the conditional system unavailability increases by almost two orders of magnitude, but still remains below the value of 5 %, so that the HCLPF criterion is fulfilled (see slide 98). For the other plant, the conditional system unavailability also increases by about two orders of magnitude, albeit starting from a lower level of about 10^{-6} for the conditional system unavailability.

Note: In addition to the two plants on the basis of which the SPSA methodology was presented (KWG and GKN II), a complete SPSA has also been carried out for a third plant (KKP 2).

An assessment of the earthquake-induced failure probability of safety functions was carried out; the quantification was carried out for earthquake I = VII (design) and I = VIII. Earthquake-induced failure probabilities were determined for selected SSC, which can be decisive with regard to robustness against earthquakes. The conditional failure probability of the high-pressure injection (no vital safety function) is the highest with approx. 2% for both cases (I = VII and I = VIII). The failure probability of the other safety functions is again at least one order of magnitude lower, also for both cases. From the comparison of the failure probabilities with design loads, similar design margins can thus be derived as for the PWR plants.

e) Applicability and assessment 0.1 g ([20], 99 –106)

The operators assume that the results of the plants with SPSA are applicable to the other operating PWR plants for the following reasons:

- The plants with SPSA are representative of the other plants in power operation with regard to the plant components and structural installations as well as the procedure for verifying stability, functionality and integrity. This also applies to the original design practice that led to the design margins.

- With regard to the seismic parameters, the other German plants are in the range of the plants with SPSA. Upon request, VGB explained that the design earthquake for all sites used during construction is comparable to or higher than the presently probabilistically determined earthquake with an occurrence frequency of approx. 10^{-5}/a.

- In addition to the seismic design, all plants are designed to withstand aircraft crashes and blast waves in accordance with the requirements of the RSK guidelines and thus have a high (comparable) degree of robustness, especially for seismic impacts.

- For all other plants, plant inspections have been carried out; for sites with seismic intensity between I = VI and I = VII simplified SPSAs have also been performed.

- Overall, the plant inspections did not reveal any findings that would call the applicability of the results into question.

It can thus be assumed that all PWRs in power operation have design margins with which the loads can also be transferred from earthquakes that are at least one order of magnitude lower in frequency of occurrence than an earthquake with a frequency of occurrence of approx. 10^{-5}/a.

For two PWR plants in northern Germany, a PGA value of about 0.5 m/s², i.e. about 0.05 g, was used in the original design for the design earthquake. For one of these plants, it was shown with an SPSA that the HCLPF values of all functions investigated are at least 0.1 g. The other plant is known to have a PGA value of about 0.05 g. For the other plant, it is known that it has additional margins due to the conservative modelling of its pile foundation (a static equivalent load of 0.5 g horizontal was used for the design). A further plant had already been designed with a PGA value of 0.7 m/s². Since the present investigations showed a margin of at least a factor of 2 in relation to the PGA value of the design, it can be concluded that the plants are also robust for a PGA value of 0.1 g.

f) Low-power and shutdown operation, transient operating states ([23], slides 19-60)

A systematic assessment of the robustness with regard to the event "earthquake during low-power and shutdown operation" was carried out representatively for some PWR plants on the basis of a review of the available documents and plant inspections during low-power and shutdown operation (LP&S).

The boundary conditions in the PWR plants are similar or comparable with regard to systems engineering, the plant operating conditions to be considered, and the regulations for LP&S as well as for inspection procedures and activities including FA handling.

The main focus of the plant inspection and document review was on the list of components/installations/auxiliary tools and the resulting potential hazards, as well as the load transfer with modified mass arrangements in the event of an assumed earthquake.

Key findings of the plant inspections/document review were:

- Components and structures that are specifically required for conditions in LP&S (e.g. lining of the reactor pool) to safeguard vital functions are either robust against the loads of an earthquake with an occurrence frequency of 10^{-6}/a or the functions can be safeguarded within the respective non-intervention time by manual and substitute measures [29].

- *The refuelling machine has been verified to withstand design earthquakes, not only for the parking position. Should an earthquake occur while the machine is moving, the vibrations are detected by sensors and the machine will brake immediately to a standstill. Additional precautions (claws) ensure stability; slipping (guided on the rails) is still possible [30]. The frequency of occurrence for a design earthquake during operation of the loading machine is less than 10^{-6}/a.*

- Equipment for overall maintenance inspections or other masses stored in the overall maintenance inspection are - partly due to optimisations after the inspections - stored, set up or fastened in such a way that an impairment of vital functions due to slipping or overturning during earthquakes need not be assumed.

- Special load combinations, such as a transport cask suspended from the hook of the building crane at the simultaneous occurrence of a design-basis earthquake, are extremely unlikely due to the brevity of the corresponding operating conditions (a few hours per year). Irrespective of this, they will not lead to the loss of vital functions under the conditions existing in the plants. *Due to the very low natural frequency of the "pendulum" (transport cask on the crane cable) in the horizontal direction and the only short strong earthquake phase under the seismic conditions in Germany, only very little oscillations of the pendulum would be excited, so that there would be neither any relevant additional horizontal loads on the building crane nor any hitting of the cask against walls. In the vertical direction, no relevant additional loads will be imposed due to the elasticity of the crane cable and the vertical accelerations, which are lower anyway. [30].* (Note: The load combination mentioned can practically be ruled out for LP&S operation since transport casks are generally not moved during overall maintenance inspections in the reactor building).

- In the buildings in which the components for vital functions are installed (reactor building, annulus incl. valve chamber, emergency feedwater building, pump building), no deficits were identified with regard to the effects of an earthquake during LP&S operation. Storage locations and installation positions of components/equipment are safely chosen and correspond to an "earthquake-proof" situation. For example, existing scaffolding is also secured against external hazards, work equipment - such as boxes - is held horizontal, and lifting gear is available without a trolley or with a locked trolley.

Generally, optimisation potential for securing the existing equipment against possible slipping was identified in the following areas:

- area in front of the refuelling machine (mobile inspection equipment),
- reactor cover slab ledgers on the SG covers with positioned lifting beams or components, and
- area on the cover of the pressuriser.

Plant-specific assessments were carried out and plant-specific optimisation measures were derived (e.g. securing equipment on the reactor service floor against possible slipping, anti-slip devices, etc.).

The equipment and the operating materials are installed in such a way that they will not pose a hazard under seismic conditions. For example, the installation site is chosen in such a way that a hazard to relevant SSC can be excluded (EK- IIa topic) [29].

Within the scope of maintenance measures on the trains of the spent fuel pool cooling system, plugs are inserted into the piping according to the plant-specific conditions. These are usually plugs specially designed for the pipe diameter [30], [36]. Sealing is achieved by surface pressure. Rubber is used as sealing material. The surface pressure and thus sealing function (reinforced via thread, spring tension) is maintained in the event of vibrations, even during earthquakes.

g) Applicability to the BWR plants ([27], slides 19-60, [36], [42], [46]

According to [27], the principles described for design against external hazards (EVA) apply to all German nuclear power plants. The design concept of all plants corresponds to the status of the late 1970s to mid-1980s. The BWR was also designed against dynamic loads due to EVA by similar engineering verifications, according to the specifications of the nuclear rules and regulations, and by the underlying assumptions, such as the model-based representation of the impacts and the resistance.

Against this background, according to [27], the statements on the robustness of German PWRs against earthquakes can be applied to BWR plants. The presented conservativeness in the individual steps of the design chain were also applied to BWRs. Thus, the overall high degree of conservativeness, which is based on the engineering approaches in the design and is proven by experimental investigations and experience from destroying earthquakes (with no distinction being made between BWR and PWR plants), also applies to the BWR.
The methodology of the seismic probabilistic safety analyses (PSA) presented on the example of two PWR plants and the related results are, in the opinion of VGB, also representative for BWR plants that lie within the presented seismic hazard level and are thus comparable with the PWR plants mentioned as examples with regard to seismic impact.

A seismic PSA was also carried out for the BWR according to the specifications of the PSA Methods Volume. A comprehensive compilation of the systems, structures and components (SSC) to be considered for the BWR seismic PSA has been carried out. On this basis, a plant inspection with the participation of authorised experts was also carried out, the scope of which was similar to that for the PWR plants. As a result, seismic failure probabilities were determined for selected bounding SSC. From a comparison of these with design impacts, similar design margins can be derived, as for the PWR plants. Irrespective of the differences in systems engineering between PWRs and BWRs, all the information presented by the operators on the *robustness assessment of German nuclear power plants also applies to BWRs. This also applies to compliance with the IAEA requirement regarding a rigid-body acceleration of 0.1 g.*

In [36], the seismic PSA is supplemented by the fact that a simplified procedure was carried out according to the PSA Methods Volume. The theoretical basis is formed by the relevant EPRI and NUREG reports and the related concepts (HCLPF, GERS - Generic Equipment Ruggedness Spectrum). An assessment of the conditional failure probabilities of safety functions for earthquakes with intensities I=VII (design) and I=VIII was carried out. The conditional failure probability of the high-pressure injection (no vital safety function) is the highest with approx. 2% for both cases. The failure probability of the other safety functions is again at least one order of magnitude lower, also for both cases.

Analyses of the stability of the two turbine buildings of the BWR plant KRB II during a design earthquake (intensity I = VII) were carried out. A comparison with the original safe shutdown earthquake for the construction (1974) and the assessment spectra from a recent study (2016) to verify the KRB II design earthquake showed lower acceleration values in the lower frequency range [49], [50], so that it was to be expected that the turbine buildings would also safely withstand an earthquake of intensity I = VIII [36]. To confirm this assessment, computational investigations were commissioned [46, item 7], which have since been completed with a positive result [48, Appendix 3].

According to [30], the plugs for sealing main steam lines achieve their sealing function through the surface pressure of pinched rubber seals. According to VGB, this pressure remains guaranteed even in the event of vibrations caused by earthquakes.

According to [46], an earthquake-related plant inspection was carried out, in the course of which the installation of components, fixtures and auxiliary tools and the resulting potential hazard in the event of an assumed earthquake during ongoing overall maintenance inspection activities were assessed. According to this, cliff-edge effects due to relocation of large components are not possible during severe earthquakes. Some potential for optimisation has been identified and implemented.

Part E Appendix Flooding of the Annulus (PWR and BWR)

The explanations in this Appendix on flooding of the annulus complement the statements in Chapters B.1.5/C.2.5 "Concretisation of the recommendation on annulus flooding".

1. Flooding of the annulus PWR

The explanations in this Appendix on flooding of the annulus complement the statements in Chapter B.1.5 "Concretisation of the recommendation on annulus flooding".

VGB made several presentations on this topic:

- VGB presentation of 11 December 2013 [9], slide 20;
- Presentation of 04 June 2014 [10],
- VGB presentation of 04 November 2014, Generic additional information specific to PWRs [13], sides 5-11,
- VGB presentation of 19 May 2015 [18], slides 5-10
- Oral explanations, 5th meeting of the Robustness WG on 19 May 2015 [26]
- VGB, e-mail HP transfer pumps, 24 November 2014 [32]
- VGB, oral explanations, 10th meeting of the RSK's Robustness WG on 24 February 2016 [35]

1.1 General

In general, from the point of view of VGB, a distinction has to be made between plants with auxiliary service water supply from a river or from a closed circuit (multiple-cell cooler). For plants with multiple-cell coolers, the probability of flooding of the annulus with the consequence that safety-relevant equipment will fail is so low that this scenario can be excluded as described in the footnote in Section 2.5(1) of the "Safety Requirements for Nuclear Power Plants". The following remarks therefore refer to plants that are supplied with auxiliary service water from large water reservoirs.

The scenario of flooding of the annulus is entered via the reporting section of the operating manual, which contains more detailed information on possible causes of water accumulation in the annulus and the measures to be taken in this regard. If the level increase cannot be attributed to a leak in a refuelling water storage tank (pool), reactor scram should be triggered in power operation at an early stage and the auxiliary service water pump assigned to the affected quadrant should be switched off.

A plant-specific examination is to be carried out to determine whether it makes sense to open the annulus to the reactor auxiliary building in order to drain off some of the water that accumulates there. One of the issues to be judged here is whether this could interfere with equipment that is needed e.g. for borating the primary circuit.

1.2 Analysis of failures

VGB carried out an analysis of the failures of safety-related equipment at an assumed flooding level of 2 m (postulate) on the lower annulus level. The failure of measuring transducers was also considered. Flooding of the measuring transducers can ultimately cause them to fail.

Depending on the processing of the measured values in the logic, the associated reactor protection actions are either triggered or blocked in the event of flooding of the measuring transducers. This applies regardless of whether the criteria for triggering are fulfilled or not, i.e. reactor protection actions required for the respective system status may also be blocked.

In this postulate, the level increase in the annulus does not occur abruptly, but continuously. Due to the arrangement of the measuring transducers at different heights in connection with their assignment to the different redundancies, it can be assumed that an immediate simultaneous triggering or blocking of reactor protection measures will not always take place. Depending on the plant-specific height arrangement of the measuring transducers, this leads to reactor scram with subsequent partial shutdown on the secondary side and to room for manoeuvre for the operating personnel to be able to take appropriate countermeasures. The height arrangement of the measuring transducers was recorded in all plants. From a level of approx. 1.50 m, all measuring transducers would be flooded.

For the different conceivable scenarios, a bounding consideration is possible:

- Since in the case of a postulated flooding of the annulus it can be assumed that the primary circuit has no leakage, the reactor protection signals acting on primary-side safety functions (be it tripping or blocking) are of little significance. A failure of the measuring transducers of the pressuriser level measurement to zero would trigger reactor coolant system isolation, which would be favourable with regard to coolant losses via connecting lines to the reactor coolant system. (In residual-heat removal mode with assumed leakage in a residual-heat removal line in the annulus, this line would also be automatically shut off). Further primary-side safety functions (boration, supplementing of volume contraction) are only required after several hours.

- With regard to the reactor protection signals acting on secondary-side safety functions, the relevant signals are those that can lead to the complete shut-off of the main steam discharge or to the start-up of the emergency feedwater supply system.
  If there is still an option for main steam discharge, this will result in considerably longer non-intervention times. The assumption of a complete "blocking" of the main steam discharge in the initial state of power operation therefore covers this. After the planned triggering of reactor scram and the rapid shutdown on the secondary side when incipient flooding of the annulus is detected, main steam discharge should be restored within approx. ½ h after a possible blocking of the main steam discharge.
  If the emergency feedwater system is triggered (failure of the measuring transducers to zero = SG level < min), there is a tendency to overfeed the SGs as the signals of the overfeed protection are blocked here. Within approx. ½ h, feeding should therefore be interrupted.

For a bounding consideration, it is therefore sufficient to provide for measures to restore main stem discharge in the short term and to limit SG feeding.

1.3 Countermeasures depending on operating phases

1.3.1 Power operation

If an inadmissible water inflow during power operation in more than one quadrant of the annulus, is detected, the section in the operating manual on fundamental safety functions and the measures contained in the operating documentation (in accordance with the section in the operating manual on notification) provide in this case for reactor scram and the immediate shutdown of the plant. Furthermore, the cause of the water inflow must be determined and prevented. Since the onset of a quadrant-wide water ingress occurs well before a pump failure or faulty signals due to flooded measuring transducers, all components required for cooling down the plant and for coolant and boron replenishment will still be available at first.

In order to control the scenario of flooding of the annulus in power operation, heat removal via the secondary side and the avoidance of overfeeding of the steam generators are essential. In the longer term, the primary circuit must be borated and the volume contraction compensated for in order to shut the reactor down.

Reactor core

The following options can be used to restore heat removal through main steam relief:

- The failure of the measuring transducers for the main steam pressure to zero triggers the pressure drop signal with closing of the main steam isolation valves and the shut-off valves upstream of the main steam relief control valves. The signal for shutting off the atmospheric steam dump stations is timed in the excitation (approx. 1 s) so that the set memory for the triggering of the isolation is reset practically immediately and the atmospheric steam dump station can be used again. The pressure reduction is done manually from the control room because the command variable (main steam pressure) for the controller has failed. The shutdown can be controlled via operational main steam pressure measurements as their measuring transducers are located within the containment.

- Moreover, in the event of main-steam relief being blocked, there have for some time (since the KKU event) been procedures in the section of the operating manual dealing with the fundamental safety functions or in the emergency manual. For example, heat removal via the auxiliary steam line into the feedwater tank and from there via the safety valves of the feedwater tank into the turbine building has been added, as has the emergency measure "secondary pressure relief" via main steam safety valves. However, the manual measures required for this are more complex than for the use of the atmospheric steam dump stations.

- Feeding of the steam generators with the emergency feedwater system in the event of failure of the measuring transducers for the SG level to zero is to be expected since the valves of the emergency feedwater system (SG level control valve and emergency feedwater isolation valve) located in the annulus and to be opened in order to isolate the feedwater path are located well above the postulated water level of 2 m. Although feeding would be limited due to the increase of the pressure of the emergency feed pumps to zero pump head as the SG are blocked on the main steam side, by then the SG might be overfed and water might flow into the main steam lines. Although the main steam valves have been proven to be able to cope with blowing off water-steam mixtures, overfeeding should still be avoided. For this purpose, the measured value for the SG level can be simulated (shut-off at > 13.5 m) or the emergency feedwater isolation valves are moved manually on site via adapters. (If the situation requires, emergency feedwater pumps can be disconnected according to the section of the operating manual dealing with external hazards). The feedwater quantity is adjusted to the respective quantities of main steam that are discharged. In case of failed SG level measurements, other signals, e.g. temperature measurements, may be used to judge the approximate level.

With regard to a primary coolant loss, the operators state that limiting functions (coolant mass, pressure and temperature gradient limitation) have also been tested and will not lead to a primary coolant loss after flooding of the measuring transducers. Short-term coolant make-up in the event of postulated flooding of the annulus is possible until the corresponding failures occur.

The maintenance of a safe condition in the long term is to be achieved by adjusting the boric acid concentration CH-K by using HP delivery pumps of the chemical and volume control system. The pumps of the chemical and volume control system can be used because they will not be flooded due to their spatially high installation. Although cooling of the pumps via the closed cooling water system will fail due to flooding of the pumps of the closed cooling water system, it is still possible to use emergency measures to establish a supply to the cooling loads of an injection pump of the chemical and volume control system via hose connections from nearby connections of the fire extinguishing system or the demineralised water system. Since only a good 3 kg/s of water is required for cooling, the connections can be made via fire hoses kept in storage and the discharge can be received by the annulus sump. The connections are accessible even if the annulus is flooded as postulated.

The reconnection of a pump of the chemical and volume control system with injection into the primary circuit requires, in addition to the restoration of cooling, that the pump is switched on again:

- overriding reactor protection signals, e.g. triggering reactor coolant system isolation,
- opening of valves in feed lines and suction pipes,
- overriding shutdown signals from the equipment unit protection and
- addition of the make-up pumps in coolant storage.

For feeding, the operated pump draws from the volume control surge tank, which is refilled accordingly from the coolant storage in the reactor auxiliary building. The required electrical energy is drawn from the D1 busbars or, for the individual valves, from the D2 busbars, with the possibility of these busbars being supplied by the assigned diesels or, due to the long non-intervention time, also via substitute measures (mobile diesel generators or 3rd grid connection).

Since feeding is not required at short notice to ensure subcriticality (boron injection) and to make-up leakages, there are long non-intervention times.

This emergency measure can also be used for feeding against high pressure in the reactor coolant system, if necessary.

Procedures for the implementation of this emergency measure are being tested, in place, or being prepared in plants where the annulus may potentially be flooded.

Spent fuel pool cooling

The heat removal from the spent fuel pool required in the long term is no longer possible with the equipment provided for in the design due to expected system failures in the event of flooding of the annulus. Feeding of the spent fuel pool from areas outside the annulus is possible. Heat removal is achieved via evaporative cooling in combination with a new emergency procedure for containment pressure limitation and thus for limiting the boiling temperature in the spent fuel pool at approx. 120 °C (part of the revised emergency protection concept, among other things for realising additional feed options into the spent fuel pool in combination with evaporation of coolant inventory from the spent fuel pool).

1.3.2 Low-power and shutdown operation

Reactor core

For plant conditions without level reduction in the primary system and a closed primary circuit, the measures from power operation can be considered. The phase-specific regulations of the operating manual specify that at least one steam generator must be available for residual-heat removal until the RCS is opened.

In the case of flooded reactor coolant lines or flooded reactor pools, there are long non-intervention times until measures are necessary to cool the fuel assemblies in the reactor pressure vessel (>> 10 h). The shortest non-intervention times result in the case of mid-loop operation. In this case, the first step is accumulator injection, which can be realised by normalising the system (raising the locking spindles in the injection valves). If the primary circuit is open, heat removal is possible for a longer time via evaporation after the injection of at least four pressure accumulators (available non-intervention time > 10 h with an assumed thermal power in the RPV of 16.4 MW), then coolant injection must take place via the chemical and volume control system. If the primary circuit is still closed pressure-tight, heat removal is carried out via reflux condenser mode after the pressuriser inventories have been injected.

Spent fuel pool cooling

For the condition "All fuel assemblies in the spent fuel pool and slot gate closed", operational demineralised-water injection is to be used to replenish the coolant inventory in the spent fuel pool. No active components in the annulus are required for this. Heat removal from the spent fuel pool is by "evaporative cooling".

2. Flooding of the annulus BWR

The explanations in this Appendix on flooding of the annulus complement the statements in Chapter C.2.5 Concretisation of the recommendation on flooding of the annulus.

VGB made several presentations on this topic:

- VGB presentation of 02 February 2016 [36], slide 15
- VGB letter to the RSK Secretariat of 03 March 2016 [44]
- VGB letter to the RSK Secretariat of 14 March 2017 [46], item. 9
- BStMUV, e-mail of 11 August 2017 [48]
- TÜV-Süd, e-mail of 21 June 2017 [49]

VGB states in [44] and [46] that the auxiliary cooling water trains VE2 and VE3 lead to the intermediate water cooler for nuclear equipment in the corresponding compartments of the annulus of the reactor building. Train VE1 leads to the nuclear services building, where the intermediate water cooler for nuclear equipment of redundant train 1 is located, so that the consequences of a large leak are limited to the reactor auxiliary building and thus have no effect on the availability of redundant trains 2 and 3.

The individual redundant system train areas in the reactor building are sealed off from each other up to a height of 0 m (compartments) so that in the event of flooding, no water can overflow from one redundant system train to the neighbouring redundant train up to this height. The volume of Compartment 2 is approx. 1,529 m³, that of Compartment 3 approx. 1,664 m³. Furthermore, the building joints are sufficiently waterproofed from ±0 m to +1.5 m, so that the flood volume is still correspondingly larger. [48]

There are redundant sump level alarms for monitoring the individual compartments. At a level of > -8.30 m (upper edge of the sump), a safety hazard alarm of class S is issued. The alarm has two channels and complies with KTA Safety Standard 3501. According to the operating manual, Part 4, Chapter 2.7, countermeasures for clearing of a fault have to be taken immediately by the shift personnel.

In case of a further increase, the VE and TF pumps as well as the LP and HP pumps of the TH system affected are automatically switched off at a level > -7.8 m (except in the event of reactor protection system challenges) and the valves located in the water area are closed (compartment protection, safety-relevant limitation). If the reactor protection system and the class S alarm are triggered at the same time, the leak is immediately shut off by manual measures in accordance with the operating manual.

Due to the measures implemented, the effects of flooding remain limited to the affected redundant system train area.

This also applies in the event that water continues to flow into the redundant system train area due to a siphoning effect. A compensatory level is reached below the level of 0 m (0 m corresponds to a water level of the Danube of 433 m above sea level).

The entire reactor protection system is located at the +8.9 m level in the reactor building. Flooding up to this level is to be considered impossible. Therefore, it can be assumed that the RA and RL ISO valves will remain controllable.

Even if all compartments were flooded, the vital safety functions could be fulfilled. The RPV could continue to be fed via the main feedwater system RL. Heat could be removed via the RA main steam system to the ultimate heat sink or by opening the diverse pressure relief valves and releasing steam from the wetwell via the venting system.

If the compartment of redundant system train 2 is flooded, the emergency measure for RPV feeding via mobile pumps is not feasible. Possible measures would be the injection of feedwater and heat removal via the main steam lines and, as measures that can save time, the injection of coolant via control rod drive pumps or coolant injection via the seal water pumps of the coolant recirculation pumps.

On the question of how the water spreads from one compartment to the next, the operator carried out a special inspection with the participation of an authorised expert. The operator explains that between the compartments of redundant system trains 2 and 3 there is the compartment of redundant system train 1 on one side and the YT rooms and the assembly duct on the other side [48]. A ventilation duct runs from 0.00 m to +1.50 m through all three compartments and into the YT rooms, with a fire damper between each compartment. The water entering the ventilation duct first exits in the compartment of redundant system train 1 via ventilation slots and branching ventilation ducts. According to the operator's estimate, the free cross-sections in the ventilation ducts of redundant system train 1 are not sufficient to prevent water from passing between Compartment 2 and Compartment 3 after flooding one of the compartments, although only a partial quantity will reach the other compartment.

Furthermore, the operator has made a probabilistic estimate of the frequency of occurrence of an uncontrolled flooding of the annulus with failure of all countermeasures. Based on plant conditions for which no measures are available to feed the RPV in case of an assumed flooding of Compartments 2 and 3, he derives a frequency for a hazard state of 3.4 E^{-11} 1/a; the 95% percentile is 1.1 E^{-10} 1/a. In the operator's opinion, this shows that the scenario is "extremely unlikely with a high degree of certainty". According to the RSK Safety Philosophy of 2013, the occurrence of an event or event sequence or plant state can thus be regarded as excluded; measures to improve such scenarios can, at best, make minor contributions to risk prevention.

Part F Appendix Load drop (PWR and BWR)

The explanations in this Appendix on load drop complement the statements in Chapter B.1.6/C.2.6 "Concretisation of the recommendation on load drop".

VGB made several presentations on this topic:
- VGB presentation of 11 December 2013 [9], slide 20
- VGB presentation of 04 November 2014, [13] slides 12-25
- VGB presentation of 05 November 2014 [15] slides 4-6
- VGB presentation of 02 February 2016 [36], slides 16-18
- VGB letter of 14 March 2017 [46]
- BStMUV, e-mail of 11 August 2017 [48]

The aim of the considerations of the "NWV" Working Group of the VGB was to show that a drop of heavy loads can be excluded or that the consequences of an assumed drop will not lead to a failure of vital safety functions.

With regard to the vital safety functions, it is essentially a matter of ensuring the fundamental safety function of fuel cooling. The fundamental safety function of reactivity control is not compromised by mechanical impacts due to load drops since they would lead to a "compaction" of fuel assemblies and thus to a reduction in moderation, but not to an increase in moderation and reactivity. The fundamental safety function of activity enclosure can be impaired by mechanical impacts (failure of barriers such as fuel rod cladding tubes), but not to such an extent that "cliff-edge" conditions would occur. This can only be caused by a major coolant loss, which would result in the melting of nuclear fuel on a large scale. (However, large early releases can be ruled out in this case due to the ventilation isolation of the containment implemented in all plants).

Such impacts are practically only conceivable as a result of the assumed drop of loads that, due to their weight, are only moved with the main hoist of the reactor building crane (approx. 30 to 40 operating hours per year).

When discussing the overlap of crane use with other events, e.g. earthquakes, it must be taken into account that the crane with heavy load will only be located in areas that are relevant for the robustness considerations for approx. 20 % of the mentioned operating hours. Furthermore, the hoists are designed according to KTA 3902, Section 4.3 with considerable safety factors in the mechanical dimensioning on the impact and the resistance side, so that a failure of the structures is not to be expected even in case of loads due to design earthquakes. The scenario can therefore be excluded.

The concept of the VGB-WG "NWV" is based

- on the one hand on minimising the frequency of occurrence of load drops that could cause a major coolant loss, and

- on the other hand on using measures with which impaired cooling can be restored to the required extent.

1. Precautions against drop

For the design and operation of the hoisting equipment in the plants with power operation, the requirements according to KTA 3902, Section 4.3 (1983) and KTA 3903 (1982) applied at the time of construction of the plants. In the operating licences for these plants, it was therefore assumed that a drop of heavy loads with safety-related consequences was so unlikely that no further considerations were necessary. In this respect, the VGB-WG-N refers to the documentation paper on the amendment of Safety Standard KTA 3902 for 1983 [documentation paper on the amendment of the Safety Standard, Section 5.1] according to which the failure frequency for hoisting equipment according to Section 4.3 was estimated to be about 10^{-6}/a.

In the 1990s, these Safety Standards were revised, which led in part to additional requirements that were derived, for example, from operating experience and the evaluation of events. Furthermore, these Safety Standards were supplemented by KTA 3905, which formulated additional requirements for the load attachment points. These additional requirements led to more extensive upgrades in the plants. The VGB-WG-N argues that due to these upgrades, the failure frequency has tended to decrease. It is also pointed out that, taking into account other measures such as

- technically: implementation of the more stringent requirements according to KTA 3902 (as of 2011) with new operating and safety controls with diverse/redundant travel path and lifting height limitation (diversity through operational and safety-related limitations),
- administratively: addition of passages in the operating manual regarding the movement of heavy loads as well as operational regulations for the handling/motion of heavy loads in the area of the reactor service floor and in the area of the spent fuel pool and the reactor pool,

not every assumed drop will lead to a risk to vital functions. Accordingly, it could be assumed that a load drop with the potential to impair vital functions has a frequency of occurrence that is at least one order of magnitude lower than the above-mentioned value (see also the following sections).

2. Handling of spent-fuel transport casks

2.1 Procedure

Spent-fuel transport casks (TCs) are moved according to plan in the reactor building only during power operation (concrete ceiling above the reactor pit closed). In the case of the PWR, the TCs are conveyed in a horizontal position on a rail-guided trolley through the material airlock and moved to a position close to the setdown position. The maximum weight of a loaded TC is 127 Mg.

Then the TC is brought into the vertical position next to the setdown position by lifting one end with the building crane and following it with the cart on which the other end rests. Then the TC is lifted vertically a little and then moved sideways at a height of approx. 20 cm above the reactor service floor to the setdown position and is set down there.

After preparatory work on the TC, it is lifted slightly, moved sideways on a direct path and at low speed to the transport cask pool, where it is lowered into the flooded cask pool. The hatch to the spent fuel pool is not closed in this process.

Overshooting of the spent fuel pool or the reactor pool by a TC is prevented by technical measures and administrative requirements (travel range interlocks of the building crane when the TC is attached and monitoring of the process).

After loading with spent fuel assemblies, the TC is removed from the reactor building in the reverse order.

In the case of the BWR, the FA transport cask is raised inside the reactor building to the level of the reactor service floor. However, under the floor of the transport shaft (0 m) there is only the main airlock antechamber (-5 m or -3.5 m), below which there is an empty plant compartment (-8.3 m). Damage to safety equipment due to a drop in the transport shaft is therefore not possible.

Transport and handling operations are regulated in the 40-metre manual and are only carried out inside the reactor building, using the reactor building crane in accordance with an approved step sequence plan over specially designated/proven areas. The highest transport height of approx. 1.30 m is reached when the cask is lifted over the railing at the fuel pool (1.10 m).

Interlocks on the reactor building crane prevent load transport in the pool area under load. Without a key switch, the crane hooks can only be moved unloaded and in the maximum position above the pool. Moving the crane hooks above the pool under load is only possible with a key switch.

2.2 Assessment VGB-WG-N

With the PWR, a drop of the TC when travelling through the airlock on the cart is impossible due to the transport brackets used.

With the movements of the TC on the crane hook at a low height above the reactor service floor, it can be assumed that the massive reinforced concrete structures of the reactor service floor can absorb the impact loads in the event of an assumed drop. Due to the low transport heights, the container is not expected to tip over after a load drop.

The stability of the TC on the setdown position is sufficient to prevent it from tipping over, even in the event of a design earthquake.

If it was assumed that the TC dropped upon being lowered into the transport cask pool, the drop height in the PWR would be at a maximum of approx. 15 m. It is not to be expected that the resulting impact load would be transferred from the bottom of the transport cask pool without any significant cracking. As for the actual spent fuel pool, due to the massive reinforced concrete structures between the spent fuel pool and the transport cask pool - which are supported by a transverse wall arranged underneath (see slides 21 to 25 in the VGB presentation of 04 November 2014) - it is not to be expected that cracks with leakages that could not be overfilled for cooling (see below 5) could occur in the spent fuel pool (bottom of the spent fuel pool and walls).

A leakage of water from the spent fuel pool via the damaged transport cask pool is sufficiently limited by the fact that the structural threshold between the transport cask pool and the spent fuel pool excludes a level drop in the spent fuel pool to below the upper edge of the fuel assemblies.

A failure of the hoisting gear exactly in the short period of time in which the TC is suspended above the opening of the transport cask pool in such a way that it would touch down on the edge of the transport cask pool when dropping and then tip over in the direction of the spent fuel pool is ruled out probabilistically by VGB.

In the case of the BWR, the reactor service floor is designed for a postulated drop in the travel paths provided for the transport cask. Similarly, during construction, the floor of the separate transport cask pool was shown to withstand a postulated drop.

The toppling-over of an FA transport cask into the spent fuel pool (also due to incorrect positioning on the edge of the transport cask pool and subsequent toppling-over into the spent fuel pool) is excluded by VGB since events with overlapping of incorrect travel of the KTA crane and simultaneous failure of the KTA lift rig in terms of the footnote in the SiAnf, item 2.5 (1) need not be assumed.

Furthermore, it was shown for the BWR that unhooking of the hoist in the event of a faulty touchdown on the edge of the transport cask pool with incipient toppling also need not be assumed due to constructive measures in the hoisting gear [48, Annex 1]. Thus, the drop of a transport cask into the spent fuel pool need not be assumed as a whole.

3. Handling of the RPV closure head and the top guide

The weight of the RPV closure head with lifting beam is approx. 150 Mg, that of the top guide with lifting beam approx. 75 Mg. The consideration of the RPV closure head therefore covers the flange area of the RPV with regard to mechanical impacts. (The top guide has a slightly smaller external diameter than the RPV internally, but even a slight skewing, as is practically inevitable in a drop, would cause it to strike the flange area of the RPV and not drop into the RPV).

3.1 Procedure

In the PWR, when the RPV is opened, first the closure head studs are loosened with the stud tensioner, lifted with it and set down at the corresponding location on the reactor service floor. Then, massive guide bolts that are approx. 5 m long are screwed into the RPV flange at three points offset by 120 °, guiding the RPV closure head during lifting and placing. The closure head is subsequently raised above the guide bolts, then moved sideways at the level of the reactor service floor (approx. 10 m above the RPV flange) and set down in its setdown position. While the closure head is being lifted, the water level is also adjusted (approx. 2 m distance) to reduce the dose rate.

When putting the closure head back on, the steps are followed in reverse order.

In the BWR, lifting of the closure head is carried out in a similar way.

3.2 Assessment VGB-WG-N

If it is assumed that the hoisting gear fails while the closure head is still guided by the guide bolts, tilting of the closure head is to be expected, which in conjunction with the raised water level in the reactor pool (PWR) has a braking effect. Nevertheless, if the closure head hits the flange area of the RPV, an impact load is to be expected, which leads to plastic deformations in the suspension and thus to a certain lowering of the RPV (ring girders and support claws). However, as an assessment already carried out within the framework of the German Nuclear Power Plant Risk Study, Phase B, (Main Volume, p. 486ff) has shown, a rupture or failure of the connecting reactor coolant lines is not to be expected since the nearest fixed point is several metres away from the RPV and thus the RCLs can follow the movement of the RPV within the range of the clearance of the penetration through the biological shield. Furthermore, due to the capacity of the emergency cooling systems, fuel cooling could be maintained by injecting coolant even if only one connecting RCL remained functional.

In the case of assumed damage in the connecting RCLs, the level in the RPV would drop to the lower edge of the RCLs. However, due to the significantly lower position of the reactor core and the corresponding water inventory in the RPV, there is sufficient time (approx. 1 h) until the coolant in the RPV has to be replenished to prevent the level from dropping to the area of the reactor core to be cooled. For coolant replenishment in such a scenario, measures were added to the operating documentation for feeding with provided pressure accumulators and residual-heat removal pumps.

If the failure of the hoist is assumed when the closure head has already been moved sideways above the guide bolts, the closure head would hit at least one of the guide bolts and thus become tilted. This would not result in any significantly different conditions than in the scenario discussed above.

For the BWR, it was shown within the framework of a safety analysis that even in the event of a load drop of the RPV closure head, the coolability of the fuel assemblies is maintained and water can be injected into the RPV [15].

4. Handling of other loads

Other loads moved with hoists in the reactor building are covered by the cases discussed above with regard to their weight. In the building areas where these loads are moved, the physical separation of safety-related equipment ensures that, even in the case of assumed consequential damage, a maximum of one redundant train of safety-related equipment would be affected.

If it is assumed that such loads - despite the minimisation of overpasses - drop into the spent fuel pool, the pool lining may be damaged but not the reinforced-concrete structure of the spent fuel pool itself. Any leakage through the damaged lining can be overfed.

If it is assumed that such loads drop into the reactor well/setdown area between the RPV and the spent fuel pool, this applies analogously. Moreover, even if the interlock gate in the spent fuel pool is open, the level there could only drop to the lower edge of the interlock gate so that the fuel would remain sufficiently covered for its cooling (non-intervention time for necessary make-up feeding of several hours).

5. Cooling options

In the case of the PWR, the following are available for the injection of borated coolant:

- Spent fuel pool
  - chemicals injection system (6 kg/s)
  - fuel pool purification system from the flooding pools/tanks of the residual-heat removal system (approx. 20 kg/s)
  - residual-heat removal/spent fuel pool cooling pumps from the flooding pools/tanks of the residual-heat removal system (approx. 180 kg/s)
- RPV
  - chemical and volume control system (approx. 25 kg/s) from the chemicals injection system
  - extra borating system (approx. 8 kg/s)
  - accumulators (eight with an inventory of 34 Mg each)
  - residual-heat removal/spent fuel pool cooling pumps from the flooding pools/tanks of the residual-heat removal system (approx. 360 kg/s or 210 kg/s per pump at 1 bar back pressure)

After injection of the inventories of the residual-heat removal system, the residual heat-removal/spent fuel pool cooling pumps can be switched over to sump operation mode and thus ensure heat removal in the long run.

The various feed options can be activated within time periods that are significantly shorter than the above-mentioned non-intervention times.

Corresponding procedures are available in the operating documents (operating manual, section of the operating manual dealing with fundamental safety functions). Reference is made to emergency measures (in the emergency manual) in the event of a challenge, i.e. in the event of a violation of fundamental safety functions.

With the BWR, there are various options for injecting water into the spent fuel pool and the RPV, see Ch. C.2.9 "Review of the accident management concept with regard to injection options for the cooling of fuel assemblies and for ensuring subcriticality".

Part G Appendix Robustness of the RCP seals (PWR)

The explanations in this Appendix complement the statements in Chapter B.2.3 "Measures to control the loss of the ultimate heat sink".

The design of the sealing sections of the RCPs varies from plant to plant, depending on the pump manufacturer and possibly on the year of manufacture. [10], [33]

1. KSB

1.1 Function of the slide ring seals

- Three hydrodynamic slide ring seals connected in series
- Pressure differences across sealing stages are provided by pipe throttles
- Lubrication and cooling via sealing water from the chemical and volume control system
- Cooling of the seal water circuit via the operating components cooling circuit (HP coolers)
- When the pump shaft is stationary, the stationary seal is closed (by impinging it with nitrogen or via a non-return seal).

1.2 Design of the slide ring seals

- Each of the three hydrodynamic slide ring seals connected in series is capable of providing sealing against full reactor coolant system pressure.

- The stationary or non-return seals are also designed for full reactor coolant system pressure and thus form an additional barrier.

- For the leak tightness of the RCP to the outside, the integrity of the elastomer sealing rings in the area of the last slide ring seal must be ensured.

- The design temperature of the elastomer material EPDM to ensure operational function is 150 °C (current component tests showed an extrapolated temperature resistance of the seal rings under operating pressure conditions up to 260 °C).

- The integrity of the seal section for 10 h autarky was experimentally demonstrated by the manufacturer in 1981 under the following boundary conditions:

    - plant subcritical hot,
    - loss of process cooling water system and volume control system,
    - Closing of the HP leakage drain pipes by emergency-proof locking via the criterion "temperature HP leakage > 100 °C",
    - Initiation of 50 K/h cooldown after ten hours by manual actions.

1.3 Sequence in the event of a loss of the ultimate heat sink in power operation

- Loss of process cooling water system and volume control system,
- manual RCP shutdown or failure after approx. 3 min (equipment unit protection criteria),
- isolation of HP and LP leakage drain pipes (valve supply via D2 system),
- slow warm-up of the RCP sealing casing by means of heat conduction,
- 50 K/h shutdown no later than two hours after reactor trip via atmospheric steam dump station,
- heating of the seal package up to the design temperature of the elastomers (150 °C) can be expected after 1.5 hours. The third seal pair relevant for LOCA exclusion would reach more than 150 °C after ten hours without cooldown. With cooldown, coolant temperature reaches values of less than 150 °C after about six hours. The thermal load is lower than in the 10-h autarky scenario.

1.4 Sequence in the event of a total loss of the three-phase current supply

- Loss of process cooling water system and volume control system,
- no automatic isolation of the HP leakage drain pipes due to the assumed failure of the D1/D2 systems,
- accelerated heat-up of the seal section due to open leakage drain pipes (more than 150 °C after approx. 20 minutes),
- successful execution of the emergency measures of secondary-side depressurisation and steam generator feeding,
- primary system temperatures up to 300 °C in the first 1.5 hours of the accident sequence,
- integrity of the sealing section ensured up to 260 °C in the third sealing stage if,
  - coolant temperature and pressure are reduced within about an hour through secondary-side depressurisation and steam generator feeding or
  - HP leakage drain pipes are isolated after the three-phase power supply has been restored with the mobile diesel unit and appropriate emergency measures have been taken (can be implemented within 1.5 hours).

1.5 Intended additions to operating documents (plant- and unit-specific)

- More detailed information on the loss of the seal water supply and RCP operation with emergency seal water.

- More detailed information on RCP shutdown criteria in the event of a cooling water supply failure.

- More detailed information on the loss of cooling and seal water when the pump is switched off and the primary system is hot: Inclusion of the need to shut down the plant if the component cooling system cannot be brought into operation in the short term.

- Inclusion of the short-term 50 K/h cooldown in the event that process cooling water is to be assumed to have been lost due to an earthquake.

- Inclusion of measures to refill and start up the process cooling water system in the annulus after an earthquake, if required.

- Isolation of the HP leakage drain pipes in the event of a postulated SBO in the course of emergency measures (e.g. secondary-side depressurisation, primary-side depressurisation)

2. Andritz

The Andritz pumps [33] used in plants still in operation have two- or three-stage slide ring seals. For lubrication and cooling of the seals, cold seal water is fed from the volume control system into the seal area. Carbon and tungsten carbide are used as the material pairing for the seal ring and stationary seal ring.

The two or three main seals are followed by a non-contact stationary seal which prevents the escape of primary coolant in the event of failure of the operating seals. In this case, the full system pressure is applied to the stationary seal. This seal is closed when the pump is at a standstill; this is done by applying nitrogen. A position signal indicates that the stationary seal is closed. Secondary seals are provided in the area of the sealing system. These seals are O-rings and backup rings made of elastomers. Otherwise, the features for function, design and procedures in the event of a fault with shutdown of the pumps are comparable to the KSB pumps.

Tests have also been carried out for the Andritz pumps in order to be able to assess up to which temperatures and times sufficient integrity can be assumed for the sealing sections. The integrity of the elastomers when the pump is switched off was verified by Andritz for temperatures up to 300 °C for a period of up to 10 hours.

Further tests carried out recently for comparable seal designs of Andritz pumps have shown that at temperatures up to 300 °C and an exposure time of up to 48 h, there is still no loss of function of the seals.

As a result of the standardised procedure in the event of a total failure of the three-phase power supply with initiation of emergency measures, the temperature effect on the RCP seals is reduced by cooling down the plant. Within the first few hours, it can be assumed that the coolant temperature will drop to well below

300 °C, thus minimising the stress on the seals in the relevant temperature range and maintaining the generally stationary sealing function.

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