Post-Fukushima accident

Germany

Peer review country report

Stress tests performed on European nuclear power plants

ENSREG
European Nuclear Safety Regulators Group
Stress Test Peer Review Board

1 GENERAL QUALITY OF NATIONAL REPORT AND NATIONAL ASSESSMENTS

The accident at the Fukushima nuclear power plant in Japan on 11th March 2011 triggered the need for a coordinated action at EU level to identify potential further improvements of Nuclear Power Plant (NPP) safety. On 25th March 2011, the European Council concluded that the safety of all EU nuclear plants should be reviewed, on the basis of comprehensive and transparent risk and safety assessments; the Stress Tests. The stress tests consist in three main steps; a self-assessment by the licensees, followed by an independent review by the national regulatory bodies, the third phase being an international peer review. The international peer review phase consists of 3 steps; an initial desktop review, followed by three topical reviews implemented in parallel (covering external initiating events, loss of electrical supply, loss of ultimate heat sink, and accident management). There are seventeen individual country peer reviews.

Eighteen German NPP units participated in the EU stress tests. A number of those were shut down in August 2011 as a result of a political decision of the German Government and the related amendment of the Atomic Energy Act.

The following Pressurised Water Reactor (PWR) plants are now permanently shut down:

- Biblis A (KWB-A)
- Biblis B (KWB-B)
- Neckarwestheim 1 (GKN-I)
- Unterweser (KKU)

The following Boiling Water Reactor (BWR) type 69 plants are also permanently shut down:

- Brunsbuettel (KKB)
- Isar 1 (KKI-1)
- Phillippsburg 1 (KKP 1)
- Krümmel (KKK)

The following PWRs are subject to a staggered limitation of the residual operating time ending for the last plants in 2022 at the latest:

- Brokdorf (KBR)
- Emsland (KKE)
- Grohnde (KWG)
- Grafenrheinfeld (KKG)
- Philippsburg 2 (KKP 2)
- Neckarwestheim 2 (GKN-II)
- Isar 2 (KKI-2)

The following BWR type 72 plants are also subject to a staggered limitation of the residual operating time ending for the last plants in 2022 at the latest:

- Gundremmingen B (KRB II-B)
- Gundremmingen C (KRB II-C)

Germany is a federal republic and unless otherwise specified, the execution of federal laws rests in principle within the responsibility of the Federal States (Länder). Execution of the German Atomic Energy Act and statutory ordinances is to a large extent performed by the Länder on behalf of the Federation. The "Regulatory body" is composed of Länder and the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU). Länder are responsible for licensing, supervision, inspection and enforcement as well as specific assessments and reviews of NPPs. The BMU is responsible for the oversight of activities implemented by the Länder. For this reason, the stress test process was implemented with the joint participation of the BMU.

The German National report has been prepared on the basis of thirteen site reports documenting Targeted Safety Re-assessment (TSR), of the 18 nuclear power plant units. The site reports were prepared by the respective license holders and were reviewed by the Länder. The Länder submitted the results of their review to BMU, who summarised the thirteen site reports and five review reports to establish the German National Report.

It should be noted that of the 18 NPPs, Obrigheim NPP is currently under decommissioning, but has spent fuel remaining in a storage pond at the site. Also eight of the NPPs were out of operation at the reference date (30 June 2011). However for the purpose of the TSR these NPPs were considered as being operational. This is due to the amendment to the Atomic Energy Act, which provided permanent shut down for these plants, entered into force on 6 August 2011. The report does not cover the facilities for dry storage of spent fuel, which exist at most sites.

A draft country review report was prepared during the topical review based on the German national report and results from discussions on the three topics. It was delivered to Germany in advance, in order to serve as a basis for the stress-test review team visit. The visit encompassed discussions with the competent authorities of the Federation and the Lander as well as the operators, and included a walk-down of a Grafenrheinfeld NPP. The basis for the discussion during the country visit was the list of open issues, upon which Germany provided appropriate information in advance of the mission. Additional information was also provided at the review meeting itself. The topics for the site visit where delivered to Germany in advance. During the plant visit, in addition to the clarification and explanations, the review team observed specific locations, equipment, arrangements and the use of procedures.

Germany has submitted a comprehensive national report, specifying the analyses undertaken and the results obtained. During the topical review Germany provided extensive answers and explanations to questions that were raised. During the country visit the regulator, the BMU and *Länder* authorities, as well as the RSK, the Technical Safety / Support Organisation (TSO) Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) and representatives from the operating organisations provided comprehensive clarifications and justifications. In the course of the plant visit, all locations requested by the peer review team were made accessible. Unhindered access has been granted to the requested documentation and sufficient explanations provided. With this, the peer review team want to commend both the regulator and the operator, and in particular those colleagues who were directly involved in the review, for their good cooperation and mutual respect.

The Country Report was finalized at the end of the country review, and agreed between the peer review team and the national regulator. The main findings related to the topic are specified at the end of each topical chapter.

1.1 Compliance of the national report with the topics defined in the ENSREG stress tests specifications

The national report is compliant with the topics defined by ENSREG and focuses on the key issues. The structure of the ENSREG specification is followed, which facilitated the review and the understanding of the information. An introductory chapter describing the legal framework and regulatory systems and practices was provided. The peer review team highly appreciated this additional information, as it provided the basis for understanding the structure of the regulatory system.

1.2 Adequacy of the information supplied, consistency with the guidance provided by ENSREG

The information provided is considered to be adequate and takes into account all three NPPs designs (i.e. PWRs, BWRs of type 69 and BWRs of type 72), whilst providing a reasonable level of detail. All scenarios that are required to be analysed, are presented. Information is presented in alignment with the ENSREG guidance. Information on NPPs‘ coping times as well as other technical information, (e.g. simplified diagrams of Structures, Systems and Components (SSC) with hook-up points for mobile equipment), which is required to be provided in the chapter on loss of power supply and loss of Ultimate Heat Sink (UHS) is missing in the National Report. It was indicated by the German counterparts, that this information was not provided due to security concerns. Information on coping times was provided per main NPP design type during the country visit, examples of hook-up points were shown during the KKG plant walk down.

Summaries of the thirteen licensees' reports are attached to the national report. These give brief description of the safety features as to the topics of the stress test for all NPPs.

1.3 Adequacy of the assessment of compliance of the plants with their current licensing/safety case basis for the events within the scope of the stress tests

The legal basis for the peaceful use of nuclear power in Germany is the Atomic Energy Act. Several ordinances are in force, which regulate amongst others, radiation protection, licensing procedure, event reporting and financial security. Regulatory guidelines and documents issued by BMU in consensus with the Länder, concern nuclear safety criteria and other important issues. A comprehensive set of the German Nuclear Safety Standards Committee (KTA) (Kerntechnischer Ausschuss) standards comprising a detailed set of requirements on nuclear safety of NPP systems, structures and components, is in force.

The assessment of compliance of the plants with the current licensing basis for events within the stress test scope was corroborated by Länder on the basis of TSO expert review and is found to be positive in all cases. The report does not identify any cases where the plant is non-compliant with its license conditions in terms of the design bases earthquakes, floods and extreme weather conditions, as well as in terms of protection from loss of power or heat sink.

1.4 Adequacy of the assessments of the robustness of the plants: situations taken into account to evaluate margins

As far as external hazards are concerned, no detailed quantitative assessment has been performed. However, strong arguments were provided during the country visit, that significant margins exist due to conservative hazard assessment and design approaches.

Concerning loss of power supply and loss of ultimate heat sink, the assessment is generally adequate. All situations required by ENSREG stress tests specifications have been taken into account.

The national report addresses all components, which are considered essential for management of severe accidents. According to German counterparts, the plants are robust regarding their capabilities to withstand severe accident risks due to their design and the applied multilevel defence in depth concept. A low frequency level of events exceeding the plant's design basis forms proof of the robustness. The report describes the measures that would be undertaken in case of various severe accident scenarios. In most cases detailed qualitative descriptions of the measures are presented without comment regarding their adequacy or, where relevant, implement ability in extreme conditions.

1.5 Regulatory treatment applied to the actions and conclusions presented in national report

The National report gives a detailed explanation of activities taken by the German regulator (BMU together with Länder authorities) in the framework of stress test. GRS (TSO) on behalf of BMU and the TUVs on behalf of the Länder authorities were deeply involved in the process. Information on the safety review of the Reactor Safety Commission (RSK) performed immediately after the Fukushima event (i.e. before the stress tests) is also provided. RSK has identified nine topics, which are now being discussed in further detail. GRS has issued an information notice covering lessons learned and related improvement measures that are to be considered in the regulatory supervision process. Both the information notice and the RSK evaluations consider not only the Fukushima event, but also the Kashiwasaki-Kariwa seismic event as well as some other operating experiences.

2 PLANT(S) ASSESSMENT RELATIVE TO EARTHQUAKES, FLOODING AND OTHER EXTREME WEATHER CONDITIONS

2.1 Description of present situation of plants in country with respect to earthquake

2.1.1 Design Basis Earthquake (DBE)

2.1.1.1 Regulatory basis for safety assessment and regulatory oversight

The protection against external hazards is based on the *Safety Criteria for Nuclear Power Plants*, *Accident Guidelines*, the RSK guidelines and the relevant KTA safety standards. According to *Safety Criteria for* Nuclear Power Plants, all plant components necessary to shut down the nuclear reactor safely and to remove residual heat or to prevent release of radioactive material shall be designed in such a way that they are able to perform their function even in case of external hazards. According to Accident Guidelines the external natural hazards such as earthquakes, flooding, external fires, have to be considered as design basis accidents.

The protection against Design Basis Earthquake (DBE) is in accordance with the safety standard KTA 2201. Depending on the site, the intensity of the DBE in Germany varies between VI (in North Germany) to - VIII (in South Germany), on the European Macroseismic Scale scale. The corresponding Peak Ground Acceleration (PGA) values are in some cases lower than 0.1g, which is the minimum recommended value by IAEA in SSG-9. The German regulator explained that the use of lower values than 0.1g is justified with consideration of site-specific response spectra. Nevertheless, the peer-review team recognizes that it deviates from the approach recommended by the IAEA. It is recommended that the German regulatory authority should consider the possible impact of this difference on safety.

2.1.1.2 Derivation of DBE

The NPP sites in Germany are located in areas of low to moderate seismicity. Due to this, measurement data for hazard assessments are scarce. However, there is abundant historical information. The leading parameter used for seismic hazard assessment is the macroseismic intensity.

For NPPs, a site-specific deterministic seismic hazard assessment is performed. For all German NPPs the PGA values, macroseismic intensity and the site response spectra for the DBE were determined within the licensing procedures and verified during the Periodic Safety Reviews (PSR)s.

Depending on the site, the horizontal PGA value for DBE varies between 0.05 g and 0.21 g.

The DBE is determined by taking into account the historical data within the radius of at least 200 km. To ensure conservatism of the calculations, it is assumed that all earthquakes in the same seismogenic zone as the NPP site take place in the immediate vicinity of the plant. The earthquakes occurring in the neighbouring seismogenic zones are assumed to take place at the border of the zone as close to the site as possible. The methodology is defined per KTA 2201.1.

2.1.1.3 Main requirements applied to this specific area

According to Part 1 of the nuclear safety standard KTA 2201, a site-specific deterministic seismic hazard assessment is required for NPP sites in Germany. Additionally, in the new revision of this standard just published, the application of probabilistic methods for the hazard assessment is required too. It should be noted that in this context, such probabilistic approaches have been implemented as part of the seismic hazard assessment for all German NPP sites.

The exceedance probability of the DBE according to the revised KTA 2201 is 10-5 per year (median). In the past, the exceedance probability of 10^{-4} per year with the 84^{th} percentile of ground motions parameters was used in parallel. The German experts explained that those two values are likely to lead to comparable loads.

2.1.1.4 Technical background for requirement, safety assessment and regulatory oversight (Deterministic approach, PSA, Operational Experience Feedback)

The DBE is part of the design basis of German NPPs. AllSSC necessary to perform the fundamental functions are classified and designed to withstand the DBE. For NPP sites a site-specific deterministic seismic hazard assessment is required.

Since 2005, seismic Probabilistic Safety Assessments (PSAs) are required as part of the PSRs. So far, those have been performed for four units.

Around 2002, a re-evaluation of the seismic hazard has been performed for all sites in relation with the construction of the interim fuel storage facilities. This re-evaluation included Probabilistic Seismic Hazard Assessment (PSHA) and also a deterministic assessment that considered the updated earthquakes catalogue (that is being continuously refined). The results of this re-evaluation showed only small deviations to the original assessment that were within the expected assumptions.

2.1.1.5 Periodic safety reviews

The periodic safety reviews, which have to be performed every ten years as required by Sect 19a of the Atomic Energy Act, also include a re-evaluation of external hazards and the corresponding protective measures, considering the development of the state of the art. This has been performed for all NPPs.

The decision to perform an entirely new site-specific hazard assessment in the next PSR depends on the following four considerations:

1. If there have been changes to the standards and regulations
2. If a new earthquake catalogue is available,
3. If relevant earthquakes have occurred happened
4. If other scientific findings make a reassessment necessary

If a reassessment of the site-specific seismic hazard is necessary, the assessment is performed in compliance with the requirements of KTA 2201.1.

2.1.1.6 Conclusions on adequacy of design basis

The peer-review identified the process, the methodology and the characteristics of the DBE for all NPPs in Germany. The methodology has been defined following the KTA standards. The reassessment of the DBE has been undertaken within the PSR process. A comprehensive assessment undertaken within the process of construction of new spent fuel storage facilities, confirmed the adequacy of the original DBE estimates.

Both deterministic and probabilistic assessments have been used in the new evaluation of the DBE. The PGA values obtained for specific sites vary between 0.05 g and 0.21 g.

For some sites, the PGA values are therefore below the IAEA SSG-9 recommended value of 0.1 g, which is explained by German experts to be due to the use of site specific response spectra. Nevertheless, the reviewers recommend that the German regulatory authority should consider the possible safety impact of using PGA that is below the internationally recommended value.

2.1.1.7 Compliance of plant(s) with current requirements for design basis

The compliance of the German NPPs with the current requirements for seismic design is verified through the regulatory supervision process, including undertaking regular and specific regulatory inspections in the plants and reviews of technical documents (e. g. to consider lessons learned from Fukushima).

The national report establishes that no deviations were identified related to the compliance with the applicable design basis as relevant for seismic events.

2.1.2 Assessment of robustness of plants beyond the design basis

2.1.2.1 Approach used for safety margins assessment

No systematic assessments of the robustness of plants for earthquakes that are beyond current design basis have been performed. Nevertheless, national assessment concludes that safety margins are available due to both, the conservative hazard assessment methods and robustness of the design itself. As an example, the Konvoi design of a plant located in a very low seismic zone (and thus being even less that 0.1 g PGA) is identical to the design of another one that is located in a high seismic zone (being close to 0.2 g PGA). The engineering judgment on the margins to the seismic events beyond design basis is additionally supported by the fact that all plants that remain in operation are designed to withstand an aircraft crash that is likely to induce loads that might be greater than the seismic loads.

2.1.2.2 Main results on safety margins and cliff edge effects

As discussed in the section above, the actual available margins were not quantified, nor were cliff edge effects identified. The national assessment concluded that the no cliff edge effects are expected even if the DBE should be exceeded. This is justified by the German counterparts to be due to the low level seismicity, conservative site-specific seismic hazard analyses, and inherently conservative design.

The information on eventually available margins varies for each NPP. In some cases margins are linked to the exceedance probability of an earthquake with specific macroseismic intensity. The report also indicates that no cliff edge effects could be identified for loads well above the DBE due to the tectonic and geologic realities in Germany, which would limit the strength of possible earthquakes.

2.1.2.3 Strong safety features and areas for safety improvement identified in the process

The peer-review process identified that the design concept where the plants need to be resistant against an aircraft crash is a strong safety feature, because it offers additional level of protection for other external hazards including seismic events. A strong safety feature is also the availability of fully autarkic and protected (bunkered) buildings containing redundant diesel generators with fuel supply, feed water pumps with stored water supply as well as housing the emergency controls. A strong safety feature identified by the peer review team is a dedicated digger (bagger) to be used to clear the access of rubble (a post Fukushima acquisition).

The peer review process did not identify any safety improvement that has been proposed so far. Nevertheless, some of the measures that are under consideration by RSK and GRS might include improvement measures relevant for this area.

2.1.2.4 Possible measures to increase robustness

Considering the conservatism of the design and the claims of very low seismicity at the NPPs sites, no additional measures to increase the seismic robustness are envisaged by German regulators.

2.1.2.5 Measures (including further studies) already decided or implemented by operators and/or required for follow-up by regulators

Only 4 of the NPPs have performed a seismic PSA. The next round of PSRs might be used to review the seismic hazard and design for all plants, which remain in operation.

As mentioned above, a set of recommendations, that might require analyses and improvement measures, being developed by RSK and GRS might contain some items that are relevant for seismic events.

2.1.3 Peer review conclusions and recommendations specific to this area

The protection of the plants against seismic events is reviewed within the PSR process. The hazard assessment is revisited, when new evidences (e.g. update of earthquake catalogue) become available. For most operating plants the re-assessment has been done within the process of siting of the interim spent fuel storage (2002). The new revision of the KTA standard 2201.1 has been recently published, with additional consideration of probabilistic methods for seismic hazard assessment. Nevertheless the German experts explained that this would probably not result in any significant change in the seismic response spectra for the sites. It has been determined that all German plants are fully compliant with their design basis in this area.

For German NPP sites the PGA values are in some cases lower than 0.1g. The peer-review team recognizes that it deviates from the approach recommended by the IAEA, and thus recommends that the regulator should consider the possible safety impact of this difference.

The margins as well as the cliff edge effects for seismic events have not been determined. It is believed that the robustness of the design and the conservatism in the estimates of the seismic response spectra assures sufficient margin and prevents cliff edge effects.

The peer review process witnessed the existence of a modern and comprehensive seismic monitoring system, with indicators available in the Main Control Room. A procedure, guiding operator actions (e.g. scram the reactor), in a case of seismic events of various intensities/duration has been prepared. The regulator intends to extend the requirement regarding seismic monitoring systems to all plants. No schedule for implementation was provided.

A strong feature is the robustness of the construction of safety relevant buildings that is achieved by the design requirement to assure the resistance to aircraft crashes. Multiple redundancies in the bunkered/protected building give additional support to this statement. Nevertheless, the peer review team considers that the seismic enhancements of the buildings designed e.g to DIN (Deutsches Institut für Normung) 4149 (1980) could be warranted, if they contain/house equipment that might be necessary to manage severe accidents.

Further enhancements have not been made known to the peer-review team. This might change as a result of the assessments following the RSK discussions and GRS Information notice, which are on going.

2.2 Description of present situation of plants in country with respect to flood

2.2.1 DBF (Design Basis Flood)

2.2.1.1 Regulatory basis for safety assessment and regulatory oversight

Since 1982, the requirements for flood protection measures have been specified in the standard KTA 2207, which was revised in 1992 and more recently in 2004. Pursuant to this standard, a permanent flood protection has to be provided for all relevant equipment and facilities, for a spectrum of cases including in particular, dam failures upstream. Nevertheless, temporary protection systems like flooding barriers stored at site are also allowed for flooding events with exceedance probabilities between 10-2 and 10-4 per year.

The latest changes of nuclear safety standard [KTA 2207] compared with the previous version concern in particular the specification and determination of the design basis flood. It is now consistently based on an exceedance probability of 10^{-4} per year. Since then, the amended safety standard has been applied to all modification licences regarding flood protection.

2.2.1.2 Derivation of DBF

Most of the German sites are located at inland rivers. There are no coastal sites. However, some NPPs are sited on rivers with tidal influence, in which case specific attention is paid to its effects.

The site-specific hazard for flooding is determined for all German sites according to KTA 2207, which in its latest revision requires consideration of an exceedance probability of 10-4 per year. The resulting DBF has been provided for all German plants. The methodology to determine the DBF has been described, where the level exceedance probability of 10^{-2} per year is determined by statistical methods reflecting the measured river level. The level related with 10^{-4} per year is determined by extrapolation using Parsons III distribution (river sites) or adding an Extrapolation Difference (ED; sites with tidal influence). This ED is generically derived from water gauges, but is adapted to each of the sites, considering local features. The ED is typically between 100 and 150 cm

2.2.1.3 *Main requirements applied to this specific area*

A site-specific flood hazard assessment is required for NPP sites in Germany according to the nuclear safety standard KTA 2207. This safety standard distinguishes between tide influenced sites and river sites. For both types of sites specific methods for the hazard assessment are stipulated.

In the case of sites on inland rivers, the decisive variable for determining the design basis water levels are based on a flood runoff from a flood with an exceedance probability of 10-4 per year.

In the case of coastal site and sites on tidal rivers the determination of the design-basis water levels are based on a storm-tide water level with an exceedance probability of 10^{-4} per year.

2.2.1.4 *Technical background for requirement, safety assessment and regulatory oversight*

In the revised German PSA guideline (2005), external flooding is one external event that has to be analysed. The required scope of analysis depends on the site-specific flooding hazard. If it can be verified that the sum of contributions of flooding events to the core damage frequency is considerably less than10-6 per year, a more detailed investigation is not necessary.

All plants in Germany except KKP 1, KWB A, and KKE have performed probabilistic analyses for flooding hazards, using different approaches as above. For all plants it has been shown that protection is provided against floods with an exceedance probability of 10-4 per year or less.

To determine the site-specific water level in the vicinity of selected SSCs that needs to be protected, hydraulic analysis are undertaken.

2.2.1.5 *Periodic safety reviews*

The periodic safety reviews, which have to be performed every ten years as required by Sect 19a of the Atomic Energy Act, also include a re-evaluation of external hazards and the corresponding protective measures, considering the development of the state of the art. This has been performed for all NPPs.

The decision to perform an entirely new site-specific hazard assessment within the PSR depends on the following three considerations:

1. If there have been changes to the standards and regulations
2. If relevant flooding events have occurred
3. If other scientific findings make a reassessment necessary

If a reassessment of the site-specific flooding hazard is necessary, the assessment is performed following the requirements of KTA 2207.

2.2.1.6 *Conclusions on adequacy of design basis*

The DBF has been reassessed for all plants in the framework of the PSRs. In general it could be confirmed that the design basis for flood protection is adequate.

2.2.1.7 *Compliance of plant(s) with current requirements for design basis*

There is no evidence that any of the plants are non-compliant with the current licensing basis. It has been reported that, in cases where the deficiencies were identified, necessary back fitting measures were enforced within the normal regulatory process.

2.2.2 Assessment of robustness of plants beyond the design basis

2.2.2.1 *Approach used for safety margins assessment*

The margin assessment focused on safety related buildings, as it was considered that in case of a Beyond Design Basis Flood (BDBF) flooding of non-safety related buildings could be acceptable. For the margin assessment the absolute height of openings in safety related buildings was compared with the DBF.

2.2.2.2 *Main results on safety margins and cliff edge effects*

All of the German NPPs comply with the DBF criteria. Also the majority of plants has safety margin in the case of BDBF. No realistic cliff edge effects have been identified, because the necessary water volumes for such scenarios are regarded to be physically not possible in Germany. Respectively, dyke failures would lead to discharge of large water volumes into retention areas before the water level can reach relevant heights above the DBF at the sites.

2.2.2.3 *Strong safety features and areas for safety improvement identified in the process*

A strong safety feature is that upstream dam failures are addressed in the design of the permanent flood protection measures. Similarly as for the seismic events, a strong safety feature is the availability of fully autarkic and protected (bunkered) buildings containing redundant diesel generators with fuel supply, feed water pumps with stored water supply as well as housing the emergency controls. As measures have already been implemented as a result of the PSR, no additional measures are envisaged for the future to further increase the robustness of the plants.

2.2.2.4 *Possible measures to increase robustness*

An on-going RSK work program concerns reviewing the protection of canals and buildings with respect to the intrusion of water and the buoyancy effect in case of a higher-level flood with a postulated flooding of the plant site.

Three of the four tide-influenced plants (KBR, KKK, KKU) have identified possible additional protective measures in order to increase the margins against floods whit a lower probability than 10-4 per year.

Brokdorf NPP have submitted plans for an increase in the overall protection height (for BDBF) for individual buildings (KBR; already implemented for the emergency feed-water building). In addition, the robustness of the pumps used for water supply to the feed-water tank will be enhanced. Spare parts for the emergency and Residual Heat Removal (RHR) systems will be stored in a flood-protected location.

Krümmel NPP (KKK; now permanently shut down) considered switching from temporary to permanent flood protection measures of safety related buildings for flood events with an exceedance probability of up to 10-4 per year.

Unterweser NPP (KKU; now permanently shut down) considered increasing the dyke height up to +10 m MSL. Temporary measures are planned to increase the protection height of emergency systems.

2.2.2.5 *Measures (including further studies) already decided or implemented by operators and/or required for follow-up by regulators*

As result of PSRs, numerous protection measures have been implemented at individual sites. These measures include increasing of dyke height, the optimisation of periodic inspections of dykes, additional inspections of the check valves in the drainage systems as well as sealing of the cable and piping ducts connecting safety related buildings (e.g. KKB). There is also an on-going review of the accessibility of buildings in case of long-duration flooding on the site.

2.2.3 Peer review conclusions and recommendations specific to this area

The assessment of the DBF and the methods used are based on the KTA standard 2207. The level of a flood up to an exceedance probability of 10^{-2} per year is based on the historical records at the sites. The extension to exceedance probabilities of 10^{-4} per year, which is required for the DBF by KTA 2207, is obtained by extrapolating the historical records using Pearsons III probability distribution. This is mainly applicable for river sites. For tide-impacted sites, the flood level corresponding to an exceedance probability of 10^{-4} per year is obtained by adding a generic Extrapolation Difference, which is adjusted to the specific site. Typically, the extrapolation added is between 100 and 150 cm. All the plants in Germany are in compliance with the design requirements regarding flooding. The protection of the plants against flooding is reviewed every ten years, within the framework of the PSR.

For flooding beyond the DBF, the margins have been assessed, by estimating the difference between the water level of the DBF and the height of openings that could lead to flooding of safety related buildings. Sufficient margins have been determined for safety related buildings and structures. No cliff edge effects have been identified.

Positive safety features include the requirement that all upstream dam failures need to be taken into account when estimating the DBF. Permanent measures have to be in place for such events. Temporary measures are required to be pre-prepared for specific situations where a flood with sufficient lead-time may occur.

Few additional measures have been identified. This is due to the continuous improvement process with respect to flood protection that is inherent to the PSR. Nevertheless, the tide-influenced plants are envisaging possible additional measures that would increase the robustness even further. It is worth noting that the assessment of the accessibility of plants during floods has been undertaken. Still the RSK is considering eventually needed improvements for the accessibility of specific buildings in a case of flooding of long duration.

2.3 Description of present situation of plants in country with respect to extreme weather

2.3.1 Design Basis Extreme Weather

2.3.1.1 Regulatory basis for safety assessment and regulatory oversight

The *Safety Criteria for Nuclear Power Plants* require that all SSCs required for a safe shut down, removal of residual heat or prevention of uncontrolled release of radioactive material shall be designed to enable them to perform their function in the case of extreme meteorological events. Nevertheless, with the exception of lightning (for which the KTA 2206 is applicable) nuclear plants are designed in accordance with the conventional civil engineering standards (e.g. DIN 1055). These standards are typically aimed at loads from events with an exceedance probability of 10-2 per year. In addition, all relevant plant structure are to be designed to meet the requirements of the DIN 25449 (design of nuclear reinforced/pre-stressed concrete structures), giving them an extra level of rigidity. Hazards from heavy precipitation are covered by the design against flooding.

The issue of extreme temperatures can be split into several sub-issues: high and low ambient air temperatures, high and low river/sea water temperatures, and icing. Most plants refer to system specific designs to deal with extreme temperatures. As a protection of the cooling systems against low temperatures most plants recirculate warm (discharge) cooling water. Depending on the site, plants are required to shut down if the cooling water temperatures reach 23°C to 31°C. This is due to water utilization rules (environmental protection) rather than being specific to nuclear safety issues.

The design basis requirements also include the combinations of extreme weather conditions, e.g. ice/snow + storm, ice/snow + low temperatures, low temperatures + storm, low temperatures + low water levels, and high temperatures + low water levels.

2.3.1.2 Derivation of extreme weather loads

The site-specific derivation of the extreme weather loads was not undertaken. Rather the regional or even Germany wide loads as available in DIN standard are used. In the case of heavy rain, the design concept for German plants does not rely on the drainage system to remove the water, rather it relies on the protection of the buildings against ingress of water. Therefore, the extreme rain impact does not need to be analysed on a site-specific basis.

The design against extreme air temperatures is based on the evaluation of historical meteorological data and on the conventional civil engineering standard (e.g. DIN 1055, which is now superseded by

DIN/EN1991). The acceptable Essential Service Water temperatures have been re-evaluated during and after the heat wave in 2003.

2.3.1.3 Main requirements applied to this specific area

No specific nuclear related requirements are applicable to extreme weather events, with the exception of lighting (KTA2206). The overall requirements on the operability of safety systems, envelope extreme weather conditions.

2.3.1.4 Technical background for requirement, safety assessment and regulatory oversight

The requirements for the extreme weather are mainly deterministic, the expected values being established in the standard. There are no requirements for PSAs for extreme weather, nor has such analysis has been performed. Nevertheless, some NPPs have performed screening type assessment for extreme weather events.
The experience feedback is being considered. The heat wave in 2003 initiated the re-evaluation of the impact on such an event on operating NPPs.

2.3.1.5 Periodic safety reviews

The periodic safety reviews, performed every 10 years require that the operational experience or new research findings need to be considered. In this respect, any changes in extreme weather conditions would be picked up and their impact analysed.

2.3.1.6 Conclusions on adequacy of design basis

There is no evidence that the design bases in relation with the extreme weather, as applied to German plants, would not be adequate. There were neither operational events nor any findings through analyses that would question the adequacy of design basis.

2.3.1.7 Compliance of plant(s) with current requirements for design basis

All weather conditions, which are important at the sites, are considered in the design of the plant. The buildings have robust designs, providing protections against a wide range of extreme weather conditions. Where necessary, safety related systems were qualified to withstand specific loads from extreme weather conditions. Therefore, the regulator concludes that the plants in Germany are in compliance with their design basis.

2.3.2 Assessment of robustness of plants beyond the design basis

2.3.2.1 Approach used for safety margins assessment

Within the framework of the stress stets, no additional estimate of the margins to the extreme weather events has been undertaken. The German justification of that is:

1. The lack of any indication that the extreme weather phenomena considered in the past might be inadequate

2. The fact that the robustness of the design of the plants is expected to provide significant margin to extreme weather.

2.3.2.2 Main results on safety margins and cliff edge effects

Margins for extreme weather conditions are not quantified. There are no cliff edge effects reported.

Based on the expert opinion of national experts, safety margins are available for all extreme weather phenomena, and thus no additional measures are necessary. If such should be identified in the future (e.g. through the PSRs), that these would be addressed within the routine oversight process.

It has been stated that the static and dynamic loads, which might be due to extreme weather (e.g. high winds, tornadoes, etc.), are appropriately enveloped by the design requirements with respect to the protection against pressure waves. These would be equivalent to a theoretical wind speed of 790 km/h and/or aircraft crashes.

2.3.2.3 Strong safety features and areas for safety improvement identified in the process

Similarly to other external events, robustness in the design of German plants provides a high level of protection against extreme weather. A strong feature is the design where relevant buildings are protected against surface flooding; therefore no protection against heavy rain is required.

The RSK is currently studying the possible safety issue of impact of extreme low temperature when a NPP is not in operation (and therefore not producing heat). As some of the plants rely on recirculation of condenser/service water for the heat-up of water intake, this could become an issue during long periods of extreme low temperature.

2.3.2.4 Possible measures to increase robustness

KKB and KKK mention specific measures to increase the robustness: KKB investigated measures to improve robustness against heavy rainfall events, and KKK planned to implement improvements for the event "icing of ventilation openings of emergency diesel generators during extreme weather conditions".

2.3.2.5 Measures (including further studies) already decided or implemented by operators and/or required for follow-up by regulators

For the moment no measures have been decided or implemented for extreme weather conditions.

The Federal Ministry BMU has initiated research activities and specific assessments to evaluate the potential impact of extreme weather conditions on German NPPs. Depending on the results of these activities regulatory actions (e. g. new requirements and revision of safety standards) will be considered to improve the safety of German NPPs.

2.3.3 Peer review conclusions and recommendations specific to this area

The design basis for extreme weather is based on national engineering standards rather than on specific nuclear related requirements. Nevertheless, those standards have been assessed to be adequate for this purpose. There is no evidence that any of the plants do not comply with the design requirements for extreme weather.

Specific assessments of the extreme weather conditions beyond those that are the design basis have not been undertaken. This is due to the fact that such conditions are not believed to be possible in Germany.

The margins to the extreme weather beyond design basis have not been estimated within the process. In general, the margins are likely to exist mainly due to the design of German NPPs covering also events such as explosions and aircraft crashes that are enveloping the conditions of extreme weather events (e.g. wind and tornadoes). Margins for heavy rain are likely to exist due to design features to assure the protection of important safety buildings. The margins with respect to extreme temperatures are likely to exist due to the design concept that allows for operability in a range of conditions.

It has been reported that some operators are considering additional assessments and improvements to further reduce risk. BMU has also initiated research activities in the area of extreme weather, which may result in new requirements.

The peer review team recommends considering the assessment of margins with respect to extreme weather conditions exceeding the design basis (e.g. by extending the scope of future PSRs).

3 PLANT(S) ASSESSMENT RELATIVE TO LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

3.1 Description of present situation of plants in country

3.1.1 Regulatory basis for safety assessment and regulatory oversight

The legal base for the peaceful use of nuclear power in Germany is the Atomic Energy Act. Several ordinances are in force, regulating among other issues, radiation protection, licensing procedure, event reporting and liability. Sublegal regulatory guidelines and documents issued by BMU in consensus with the Länder, concern nuclear safety criteria and other important issues. A comprehensive set of KTA standards comprising a detailed set of requirements on nuclear safety of NPP systems, structures and components, is in force.

3.1.2 Main requirement applied to this specific area

The main requirement for the emergency power supply systems is Criterion 7.1 of the NPP Safety Criteria:

"In addition to the electrical power supply from the main supply and the main generator, reliable emergency power supply systems shall be provided for plant components of safety-related significance, which guarantee the electrical power supply to these plant components in the events of the main supply and the main generator failing. Independent, redundant emergency power generators and distribution systems shall be provided for the emergency power supply, such that, even during maintenance procedures and a simultaneous occurrence of a single failure, an emergency power supply sufficient from the standpoint of safety is ensured. The redundancy of the emergency power generation and distribution systems shall correspond to the redundancy of the mechanical systems. It shall not be possible, to put all emergency power supply systems simultaneously out of operation during and after external events. It shall be ensured that, before the maximum allowable time period for uninterrupted operation of the emergency power generators has lapsed, the emergency power can be supplied by other sources."

The main requirement for residual-heat removal during specified normal operation is Criterion 4.2 of the NPP Safety Criteria:

"A reliable, redundant system shall be provided for the removal of residual heat during specified normal operation and it shall be in such a condition that, even after interrupting the heat removal from the reactor to the main heat sink, limits specified respectively for the fuel elements, the reactor coolant pressure boundary, and the containment are not exceeded, even if a single failure should occur in the RHRS."

More specific requirements on the systems, structures and components are comprised in the German Nuclear Safety Standards (KTA standards).

3.1.3 Technical background for requirement, safety assessment and regulatory oversight

It is evident from the National Report that deterministic approach complemented by PSA is the main basis for the regulatory requirements, assessment and oversight. Detailed explanation is provided on how operating experience (namely TMI and Chernobyl accidents), was taken into account in the updating process of regulatory requirements.

3.1.4 Periodic safety reviews

Periodic safety reviews are a legally required process; reviews are performed in ten years periods. The defence in depth and the fundamental safety functions have to be reassessed using current site conditions and impacts conceivable at the plant site. These regular safety reviews address also enhanced protection against hazards as well as the implementation of on-site or plant internal preventive and mitigative accident management measures.

3.1.5 Compliance of plants with current requirements

It is confirmed in the National Report that all German plants are in compliance with current requirements.

3.2 Assessment of robustness of plants

3.2.1 Approach used for safety margins assessment

All situations, which are required to be taken into account within the ENSREG stress tests specifications, are analysed in the National Report. Sequential loss of lines of defence is assumed with the exception of mobile diesels and pumps. Both Operators and the Regulator in Germany considered the use of accident management measures to cope in the most challenging scenarios and plants modes, including the ENSREG stress tests scenarios. These accident management measures are described in the plant specific accident management manuals and include complete timelines with the grace period and time needed to implement the measures. These figures were developed during the setup of accident management measures and were used to address the ENSREG stress tests specification on potential cliff-edge effects, coping times and associated measures. Information summarising coping times and the associated measures per plant type, per accident scenario, and per relevant operating state was provided and discussed during the country visit.

3.2.2 Main results on safety margins and cliff edge effects

Power Supplies – design features

All German NPPs have standard power supply scheme and provide house load operational mode. There are several strong design features at all German NPPs, at least three off-site power supply possibilities, always with a qualified underground cable connection (e.g. to a nearby hydroelectric power plant). Furthermore, additional emergency diesel generators are available in almost all German NPPs. Also in most NPPs a mobile diesel generator is available onsite to recharge the batteries or to supply selected pumps and components. It is planned to provide mobile diesel generators for all plants.

An acceptable defence in depth level is also available at the NPP Obrigheim (KWO – plant in decommissioning stage) where there is only the nuclear fuel stored in a spent fuel pool within a special bunkered building. All relevant power supply systems are double redundant there.

3.2.2.1 Loss of Off-site Power

If the main grid is not available, all German NPPs have the ability of load rejection to house-load operation. If that load rejection is unsuccessful the station supply is automatically switched over to the standby grid connection. If this connection is also unavailable the emergency power system of the plants automatically takes over the electrical supply of the safety-related trains.

Emergency power systems of the German NPPs have at least four emergency diesel generators (first level of emergency power supply). Their fuel and oil capacity are sufficient for at least 72 hours without any manual measures (with the exception of the plants KKP-1 where manual measures are necessary after 24 hours and KKB, where manual measures are necessary after 34 hours). With manual measures the operating time can be increased. With support of these emergency power systems the plants can be shutdown (to 'cold shutdown') and the residual heat can be removed.

The residual heat removal, from the spent fuel pools will be carried out by the corresponding spent fuel pool cooling system, in this case electrically supplied by the emergency power system.

Peer review concluded that sufficient safety margins exist, no cliff edge effects were identified, no damage to the fuel can occur in this situation.

3.2.2.2 Loss of off-site power and loss of the ordinary back-up AC power sources

- *PWR construction line 4 and 3*

An additional emergency power system (bunkered D2-system; second level of emergency power supply), equipped with four D2 emergency diesel generators is designed to cope with the loss of off-site power and loss of the ordinary back-up AC power sources situation.

The fuel and oil capacity of the four D2 additional emergency diesel generators are sufficient for at least 24 hours without manual measures. With manual measures the operating time can be increased to at least further 48 hours. The D2-diesels are cooled by the emergency feed water system. The emergency feed water systems of the different plants have sufficient water for at least 10 h to supply the steam generators and to cool the D2-diesel power engines, and additional measures for refilling the water tanks are foreseen. With support of the D2-system the plant can be cooled via the steam generators.

The residual heat removal from the spent fuel pool will be carried out by the essential service water system in this case electrically supplied by the D2-system.

The batteries of the 'first level' emergency power system (D1-system) supply a secured DC power for at least 2 hours. The batteries of the D2-system will be continuously charged from the D2-diesel generators.

- *PWR construction line 2*

Various systems are designed also in PWRs construction line 2 which can ensure, as a minimum, feeding the steam generators, cooling down the units and remove the heat in the spent fuel pools. These systems are electrically supplied with the additional emergency diesel generators and are of various degree of redundancy. The minimum coping time (fuel and oil reserves for the additional emergency diesel generators) without manual measures mentioned is 24 hours.

In the case of KWB-A/B, connections to the 380 V emergency standby switchgears of the other unit are used, and furthermore there are two additional emergency diesels on the site that can operate for 10 hours without additional measures.

The batteries of the 'first level' emergency power system of the PWRs construction line 2 supply a secured DC power for at least 3 hours. The batteries of the additional emergency power systems will be continuously charged from the additional emergency diesel generators.

- *BWR construction line 72 (KRB II-B/C plant):*

One additional emergency diesel generator for each unit is available within the additional RHRS to cope with loss of off-site power and loss of the ordinary back-up AC power sources situation. Each unit has a dedicated and fully independent additional RHRS -. Inside each system one diverse emergency diesel generator is available. This emergency diesel generator is protected against earthquake and flooding and is physically separated from the other emergency diesel generators with respect to an airplane crash. Furthermore, five direct connections to the corresponding emergency power train of the other unit exist.

The fuel and oil capacity of the additional RHRS emergency diesel generator is sufficient for at least 72 hours without manual measures. With manual measures the operating time can be increased. With support of additional RHRS the plant can be cooled down and the residual heat can be removed.

The residual heat removal from the spent fuel pool, which is outside the containment, occurs by evaporation. The evaporation losses can be made up by mobile pump(s).

The batteries of the 'first level' emergency power system of the BWRs construction line 72 supply a secured DC power for at least 2 hours. The batteries of the additional RHRS - will be continuously charged from the additional RHRS-diesel generator.

- *BWR construction line 69:*

– *KKP-1*

For KKP-1 two USUS (the independent sabotage and accident protection system) emergency diesel generators are available. USUS is arranged in two trains, which are built physically separated and functionally independent inside the USUS building. Each train includes one dedicated additional emergency diesel generator. The USUS building is protected against external hazards. Furthermore, the steam-driven high-pressure coolant injection system will be started. The control system of the steam driven pump is dependent on batteries.

The fuel and oil capacity of the two USUS emergency diesel generators are sufficient for at least 72 hours without manual measures. With support of USUS the plant can be cooled down and the residual heat can be removed.

The residual heat removal from the spent fuel pool, which is outside the containment, occurs by evaporation. The evaporation losses can be made up by fire pump(s) or mobile pump(s). Additionally, in the frame of emergency measures, the circulation pump of the operational heat removal systems from the spent fuel pool can be supplied from USUS.

– *KKI-1, KKK*

The emergency grid connection to the hydroelectric power plant, which has a black start capability, is available. Furthermore, the steam-driven high-pressure coolant injection system will be started. The control system of the steam driven pump is dependent on batteries. For KKI-1 these batteries can be recharged with an accident management measure by a mobile diesel generator stored on-site.

The capacity of the emergency grid connection is sufficient to cool down and to remove the residual heat.

The residual heat removal from the spent fuel pool, which is outside the containment, will be carried out by the R, which is in this case electrically supplied by the emergency grid connection.

– *KKB*

For KKB two UNS (the independent emergency system), emergency diesel generators are available. Furthermore, the steam-driven high-pressure coolant injection system will be started. The control system of the steam driven pump is dependent on batteries.

The fuel capacity of the two UNS emergency diesel generators is sufficient for at least 72 hours without manual measures. The oil capacity of the two UNS emergency diesel generators is sufficient for at least 60 hours without manual measures. With support of UNS the plant can be cooled down and the residual heat can be removed.

The residual heat removal from the spent fuel pool, which is outside the containment, will be carried out by the RHRS in this case electrically supplied by UNS.

The batteries of the 'first level' emergency power system of the BWRs construction line 69 supply a secured DC power for at least 3 hours. The batteries of the additional emergency power systems will be continuously charged from the additional emergency diesel generators.

Peer review team concluded that sufficient safety margins exist, no cliff edge effect were identified.

3.2.2.3 Station Blackout (SBO)

It was emphasized in the National Report that in addition to the normal off-site power supply and the emergency power systems ('first level' and 'second level'), each plant unit has a safety-classified emergency grid connection available through manual operations (third grid connection).

If all above described and analysed power supply alternatives fail, all plants have an additional battery secured DC and AC power supply. This enables, together with accident management measures, the removal of the residual heat. Also in most NPPs a mobile diesel is available to recharge the batteries or to supply selected pumps/components.

- *PWR construction line 4, 3 and 2:*

In this situation all operational and safety-relevant systems for steam generator feeding are unavailable and thus the accident management measure 'secondary bleed and feed' will be applied. When ‘secondary bleed and feed’ is not successful, the accident management measure ‘primary bleed and feed’ will be applied. It is proved that for the accident management measures corresponding to the accident management manual the plant personnel can perform independently these measures and that the available time is sufficient. The corresponding times for the different feeding options described in the National Report differ from plant to plant due to different features of the plant design (e.g. water inventory on secondary side of the steam generators, water inventory of the feed water system with or without feed water tank) and different assumption for the calculation of these times.

The preparation time for the accident management measure ‘secondary bleed and feed’ is up to 50 to 70 minutes. A time-span of 2 to 7.5 hours can be gained by this measure with the content of the feed water lines or tank. For long-term heat removal feeding of at least one steam generator with a mobile pump is necessary. This pump is combustion engine driven and is available on site. It can take the water from different water storages. With supply of fuel, this pump could assure cooling even beyond 72 hours. The ‘secondary bleed and feed’ measure is also possible in case of complete loss of DC power supply (batteries), because all components needed either do not need power, or are designed with fail-safe principle. Manual operation of valves inside the emergency feed water system is also possible. The rate of the ‘secondary bleed and feed’ must be manually controlled through adjustment of the feed water flow, which will ensure sufficient cooling, while preventing reactivity issues if the cooling rate is too high. The operation procedures needed for this operation were presented during the plant visit at KKG.

If the measure ‘secondary bleed and feed’ is not successful, the accident management measure ‘primary bleed and feed’ is a further option to gain time until the core heat-up for restoration of the power supply and the active core cooling systems. By this measure the grace period is between 90 and 100 minutes. The preparation for this measure occurs during the ‘secondary bleed and feed’-phase. For the above mentioned measure primary bleed and feed or for their preparation the battery secured power supply is necessary to open the pressurizer relief and safety valves.

During SBO, if the reactor pressure vessel is not closed, the residual heat will be removed by evaporation. The grace period without any accident management measure for such an open reactor pressure vessel during mid-loop operation (which is the most penalising case), is about 30 minutes until the start of boiling in the primary circuit, and between 1 and 3 hours until the water level drops to the top of fuel elements. The accumulators can be used to make up the evaporation losses. In the example of KKG, the current procedure now involves the local manual opening of the accumulator valves inside the containment, but procedures are being developed to recover the motor operated valves by restoring electrical power (either directly to the relevant 380V busbar in the switchgear building using a small mobile generator, or by using the mobile emergency diesel generator to supply the D2 busbars). Ultimately, also fire-fighting water can be used for primary water make-up.

Information on main coolant pumps (MCPs) seals at PWRs was provided during the country visit, according to which MCPs seal leakage, by their design, is not an issue at German PWRs (at least for some time), given that the unit can be cooled down using accident management measures.

The residual heat from the spent fuel pool is removed by evaporation. Mobile pump(s) can be used for the make up of evaporation losses. The grace period without any accident management measure for the spent fuel pool (immediately after core unloading) is between 15 and 18 hours until a temperature of 80 °C is reached and between 55 and 100 hours until the water level has dropped to the top of the fuel elements.

The discharge times of the additional batteries (‘second level’), in the different plants are between 2 and 5 hours, except for GKN-II the discharge time is 11 hours and for KWB-A/B 20 hours .

- *BWRs construction line 72:*

In case of a SBO the reactor and turbine scram is initiated, the containment is isolated and the steam will be released to the wetwell through the reactor pressure vessel safety valves. After depressurization of the reactor pressure vessel (either automatically or manually), feeding of the reactor pressure vessel will continue with cooling water from the feed water tank due to the developed pressure difference between tank and vessel. In addition mobile pumps are available to feed the reactor vessel in the low-pressure range.

For containment heat removal and to avoid containment over-pressurization failure the filtered containment venting is used, which was installed as accident management measure with no need of power supply.

The residual heat from the spent fuel pool, which is outside the containment, can be removed by evaporation. Mobile pump(s) can be used for the make up of evaporation losses.

The discharge time of the additional RHRS -batteries is at least 8 hours.

- *BWRs construction line 69:*

In case of a SBO the reactor and turbine scram is initiated, the containment is isolated and the steam will be released to the wetwell through the reactor pressure vessel safety valves. The feeding of the reactor pressure vessel will be carried out by the steam-driven high-pressure coolant injection system (with cooling water from the wetwell). The control system of the steam-driven pump is dependent upon batteries. If the wetwell achieves the boiling point or batteries fail the feeding of the reactor pressure vessel has to be continued with a mobile pump from the demineralized water inventory. To reach the low-pressure range, the pressure in the reactor vessel has to be reduced using two main depressurisation valves. Additionally diverse motor-driven valves (battery supplied), are available which can also be used for pressure relief if necessary.

For containment heat removal and to avoid containment over-pressurization failure the filtered containment venting is used (without power supply possible, with exception of the plants KKK and KKB where battery-supply is needed).

The residual heat from the spent fuel pool, which is outside the containment, can be removed by evaporation. Mobile pump(s) can be used for the make up of evaporation losses.

The discharge time of the additional ‘second level’ batteries (if available) is at least 3 hours.

3.2.2.4 Loss of Primary Ultimate Heat Sink

PWR plants:

Power operation or plant shutdown, primary circuit closed:

Various solutions of emergency feed water system with sufficient level of redundancy and resilience are available for heat removal via the steam generators to the atmosphere. The Essential Service Water System (ESWS) is not required to ensure the residual heat removal. The emergency feed water tanks can be refilled from different sources.

Plant shutdown, primary circuit open:

In some plants (KKE, GKN-II, GKN-I, KKP-2, KWB-A, KWB-B) an alternate UHS is available for the heat removal from the primary circuit. KKE, GKN-II: in case of complete failure of the 4x50% cell cooling towers the river is the alternate heat sink or vice versa;

In other plants (GKN-I, KKP-2, KWB-A, KWB-B) the well feeds the coolers of ESWS. In the plants GKN-I, KWB-A, KWB-B additional operator actions, i.e. installation of flexible tube connections (well – fighting water pump – heat exchanger) are necessary to continue the RHR.

In other plants (KKI-2, KKG, KBR, KWG, KKU) accident management measures are available or in planning to ensure the residual heat removal. The residual heat can be removed by evaporation of the primary coolant and injection of water from different sources (flooding with RHR-system or mobile pumps).

The spent fuel pools cooling:

In the plants KKE, GKN-II, GKN-I, KKP-2, KWB-A, KWB-B an alternate ultimate heat sink is available for the heat removal from the spent fuel pool. KKE, GKN-II: in case of complete failure of the 4x50% cell cooling towers the river is the alternate heat sink or vice versa. In the plants GKN-I, KKP-2, KWB-A, KWB-B the well feeds the coolers of ESWS. In the plants GKN-I, KWB-A, KWB-B additional operator actions, i.e. installation of flexible tube connections (well – fighting water pump –

heat exchanger) are necessary to continue the spent fuel pool cooling. In the plants KKI-2, KKG, KBR, KWG, KKU accident management measures are available or in planning to ensure the residual heat removal. The residual heat can be removed by evaporation of the spent fuel pool coolant and injection of water from different sources (RWSTs via RHR-system, mobile pumps).

KWO (in decommissioning):

In case of loss of the primary ultimate heat sink, very long time spans are available for counter measures, 5 days to 60°C coolant temperature in the spent fuel pool, 75 days to uncovering of fuel elements.

BWR plants:

In all plants with the exception of KKK an independent emergency RHRSis installed. The heat sink is ensured by various means. Coolant loss due to evaporation can be replenished from different sources (e.g. mobile pump, water supply system).

In KKI plant the case of complete loss of the access to cooling water from the river is considered, for this scenario accident management measures are available to ensure the decay heat removal (depressurization of the reactor coolant system, water injection from different sources, e.g., by a turbo pump, fire fighting water pumps or mobile pumps, filtered containment venting).

In KKK plant accident management measures are available to ensure the decay heat removal depressurization of the reactor coolant system, water injection from different sources e. g. by turbo pump, fire fighting water pumps or mobile pumps, filtered containment venting.

Spent fuel pool cooling:

The spent fuel pool cooling can be always ensured either by independent emergency RHRS, or the mobile pumps can be used.

3.2.2.5 Loss of Primary and Alternate Heat Sink

PWR plants:

The heat removal via the steam generators to the atmosphere can be ensured by the accident management measure secondary bleed and feed. If the accident management measure secondary bleed and feed is not successful, additional time for further measures can be obtained by the accident management measure primary bleed and feed.

Plant shutdown, primary circuit open:

In case of complete loss of the primary ultimate heat sink and the alternate heat sink, accident management measures are available to ensure heat removal from the primary circuit. The decay heat can also be removed by evaporation of the reactor coolant and injection of water from different sources (flooding with RHRS-, mobile pumps).

Spent fuel pool cooling:

In case of complete loss of the primary ultimate heat sink and the alternate heat sink, accident management measures are available to ensure heat removal from the spent fuel pool. The decay heat can also be removed by evaporation of the spent fuel pool coolant and/or injection of water from different sources (e. g. from the RWSTs via RHRS, mobile pumps).

KWO (in decommissioning):

In case of loss of the primary ultimate heat sink, very long time spans are available for counter measures, 5 days to 60°C coolant temperature in the spent fuel pool, 75 days to uncovering of fuel elements.

BWR plants:

Accident management measures are available to ensure heat removal from the reactor coolant circuit. The decay heat can be removed by evaporation of the reactor coolant and injection of water from different sources (depressurization of the reactor cooling system, water injection from different sources e. g. injection by mobile pumps, heat removal by filtered containment venting).

Spent fuel pool cooling:

In case of complete loss of the primary ultimate heat sink and the alternate heat sink, accident management measures are available to ensure heat removal from the spent fuel pools. The decay heat can be removed by evaporation of the spent fuel pool coolant and injection of water can be provided from different sources with mobile pumps or fire fighting water pumps.

3.2.2.6 Loss of UHS & SBO

Noting that SBO effectively results in loss of the primary UHS, the combination of loss of UHS and SBO is enveloped by the SBO. The discussion is presented in the National Report on the effects/consequences in the section on SBO. Review team agreed on acceptability of this approach.

3.2.3 Strong safety features and areas for safety improvement identified in the process

The emergency feed water buildings and the additional emergency diesel generators which both are protected against external hazards (PWR/BWR) are in the view of peer review team strong safety features of many German NPPs.
Activities are on-going to draw conclusions from the Fukushima event, namely discussion for the need to implement an alternate heat sink in NPPs which do not have an alternate heat sink and discussions on measures to further increase the robustness of the power supply and the autarky of the plant.

3.2.4 Possible measures to increase robustness

It was confirmed at the EU stress tests, that because of the robust design with all measures taken in the defence in depth concept, the possibilities for additional measures to further increase the robustness are limited. Nevertheless, the analysis performed by RSK as well as the information notice prepared by GRS indicate some areas where improvement might be possible.
In most of the units no further measures are foreseen to increase the robustness of the plants in case of loss of the ultimate heat sink.
After the Fukushima event some plants brought mobile diesel generators on the site with the goal to increase the robustness of the plants. This activity is continuing at other plants.

3.2.5 Measures (including further studies) already decided or implemented by operators and/or required for follow-up by regulators

The following activities are in progress (for plants which are in shutdown now the measures are halted):

- At KBR plant, feasibility studies to increase the AC power supply robustness.

- KKU applied for license for measures aimed at using a fire water pump to sustain low-pressure feed to the emergency feed water system or to the emergency condition diesel system even under harsh ambient conditions.
- The assessment of the Länder shows potential to improve the DC current supply of the emergency power system at the KKB plant.
- At KKI-1 plant plans for installing two new emergency diesel generator buildings and for replacing the water-cooled emergency diesel generator with new air-cooled, diverse units.

The GRS information notice (WLN 2012/02) contains 22 recommendations; all NPP operators are expected to provide information on whether and how these recommendations will be taken into account.

The information notice includes, e.g., the following topics:

- SBO coverage for at least 10 hours
- Additional emergency power generator available within 10 hours
- Diverse ultimate heat sink
- Two feeding points for connection of mobile equipment to supply the essential component cooling system

The RSK review is assessing the following areas related to the loss of safety functions:

- Implementation of diverse UHS
- Optimisation of power supply
- Increasing of Spent Fuel Pools (SFPs) cooling capabilities
- Effects of flooding of PWRs annulus
- Multiunit events analysis

3.3 Peer review conclusions and recommendations specific to this area

The peer review team accepted the conclusions of German regulator that no weaknesses have been found related to this topic of the stress tests.

The safety precautions against loss of electrical power are specified in KTA safety standards. Three off-site grid connections, a design of n+2 redundancies for the emergency power supply backed up by additional emergency power diesel generators and battery support for at least 2 hours is required. The implemented design features as reported by the licensees indicate considerable robustness against loss of off-site power.

For the safety precautions against loss of ultimate heat sink, the situation regarding the design of the component cooling systems and essential service water systems in the German nuclear power plants differs from site to site. The regulations principally demand an n+2 redundant design for active components of the safety relevant (essential) service water systems. The implemented precautionary measures, within the design were confirmed by the Länder authorities to be in line with the normal regulatory quality standards. The protection of the plants against the loss of ultimate heat sink and possible consequences is confirmed.

Because of the robust design with all measures taken in the concept of defence in depth, the possibilities for additional measures to further increase the robustness seem to be limited. Nevertheless, the investigation by RSK and the GRS information notice offer areas for further enhancement.

4 PLANT(S) ASSESSMENT RELATIVE TO SEVERE ACCIDENT MANAGEMENT

4.1 Description of present situation of plants in Country

4.1.1 Regulatory basis for safety assessment and regulatory oversight

First requirements for a Severe Accident Management (SAM) program were published in 1988 as the result of RSK discussions. The primary intention was the prevention of severe accidents starting at power operation although some selected mitigative measures for dominating phenomena were proposed as well. Those first requirements were followed by RSK discussions, decisions, positions and recommendations regarding filtered containment venting systems, containment isolation, N2 inertisation of the BWR containment, additional Reactor Pressure Vessel (RPV) injection or refilling options (BWR), electrical power supply, secondary-side and primary-side bleed and feed (PWR), diverse RPV pressure limitations for BWR, hydrogen recombination and development of an Accident Management Manual.

The operators´ accident management Programs for the beyond-design-basis and severe accidents have been implemented since the late 1980s on a voluntary basis. All above-mentioned measures have been implemented. In the context of legally required PSR every ten years, the Accident Management measures are now stipulated.

An RSK/SSK Recommendation and a document by the Länder committee for nuclear technology define alert criteria to be used in case of an emergency and for the organization of external provisions. The RSK recommendations have been supplemented by nuclear safety standards (KTA) for severe accident management.

4.1.2 Main requirements applied to this specific area

For the beyond design basis accidents, a description of recommendations issued in the last 15-20 years is provided in the national report.

According to RSK recommendations, plant-internal accident management measures for beyond design basis events are aimed at preventing serious degradation of the reactor core and the release of radioactivity into the environment. In case of these events, the goal of protecting the environment must have priority over other goals, such as protection of components.

With increasing deviation from the design range, the protective measures must be designed to cover a wide spectrum of event sequences and their design shall be based on generally valid scientific engineering principles. Event sequences can be identified with the aid of probabilistic safety analyses, of operating experiences, of results from reactor safety research and of postulated damages in the plant. Possible accident management measures shall be carefully planned, specified in an Accident Management Manual and trained for. Accident management measures shall be analyzed for their effectiveness, for the feasibility of their implementation and for their compatibility with the plant safety concept.

Specific RSK requirements exist, concerning for example hydrogen countermeasures and filtered containment venting. German nuclear safety standards concern requirements for the Operating and Accident Management Manuals, stationary systems for monitoring the local dose rate within NPP's, the monitoring of radioactivity in the inner atmosphere of NPP's as well as of the discharges of gaseous, aerosol-bound and liquid effluent bound radioactive substances during normal operation as well as during design basis accidents, instrumentation for determining the dispersion of radioactive substances in the atmosphere, accident measuring systems, communication means for NPP's and NPP control rooms, remote shutdown stations, local control stations.

4.1.3 Technical background for requirement, safety assessment and regulatory oversight

Comprehensive safety reviews have been conducted by the RSK after the TMI 2, Chernobyl and Fukushima accidents resulting in recommendations for strengthening the defence in depth of the operating reactors. The results of the German Risk Study „Deutsche Risikostudie Kernkraftwerke - Phase B“ (1981-1989), the first large comprehensive study including deterministic and probabilistic results for severe accidents based on a PWR reference plant, significantly influenced the development with respect to severe accident management in Germany. After the TMI accident regulatory actions have been undertaken related to the development of guidance for strengthening the role of PSA in safety reviews and to the consideration of beyond design conditions and related regulatory research; following the Chernobyl accident regulatory actions were focused on the development and implementation of preventive and mitigative accident management.

Level 1 PSA has to be performed for all plant operational states and covering plant internal events as well as plant internal and external hazards. In addition, a Level 1 PSA for low power shutdown states as well as a Level 2 PSA has to be performed considering only internal events. The PSA is used to assess strengths and weaknesses, in particular vulnerabilities and cliff edge effects, in the design and operation and to identify potential improvements. Furthermore, a process of learning generic lessons from national as well as international operational experience is practiced by the BMU in cooperation with the *Länder* authorities (GRS Information Notices).

4.1.4 Periodic safety reviews

In Germany, safety reviews have been carried out periodically every 10 years of plant operation according to standardized German national criteria. The defence in depth and the fundamental safety functions have to be reassessed using current site conditions and impacts conceivable at the plant site. One of the safety review guidelines specifies a set of beyond-design-basis scenarios to be analyzed and to be covered by the Accident Management Manual.

4.1.5 Compliance of plants with current requirements

The German Federal office for Radiation Protection (BfS) compiled and continuously updates a report on the implementation of Accident Management (AM) measures. The results of this process have been summarized in the national report, showing full compliance of the plants with related RSK recommendations. Furthermore, the license holders have submitted Accident Management Manuals to *Länder* within the regulatory oversight process.

4.2 Assessment of robustness of plants

4.2.1 Adequacy of present organizations, operational and design provisions

4.2.1.1 Organization and arrangements of the licensee to manage accidents

In the short term following the beginning of an accident, the general responsibility for accident management lies mainly with the shift personnel. The shift staff is large enough to ensure the initial management of the event by applying the Procedures from the Operational and later the Accident Management Manual. After initiation of the Emergency Response Organization (ERO), it takes over the responsibility. General responsibility for the accident management lies with the plant manager.

In case of a beyond design basis accident and once an emergency alert is established, the ERO replaces the normal organization. The ERO consists of the Emergency Response Team and the Deployment Units. The Emergency Response Team consists of the emergency response team leader, the heads of the relevant sections and other key officers. Plant personnel staff the Deployment units. Additional staff are recalled to the plant by means of special alerting procedures and devices.

Technical support on the site can be provided by different institutions and companies (especially the Kerntechnischer Hilfsdienst GmbH), including the vendor. The AREVA crisis team consists of own experts and supports the operators` emergency organization. The utility corporate crisis centre supports the Emergency Response Team of the NPP at all safety relevant, strategic and technical questions and assumes corporate relevant decisions. Safety relevant decisions about measures to be taken at the plant are made only by the Emergency Response Team. The NPPs do have cooperation contracts so that mobile equipment can be transported from one site to another.

All NPPs have main control rooms and emergency control rooms located in seismically qualified and flooding protected buildings. Access to the main control room building and to the emergency control room is generally possible via different access routes, including underground. A mobile filtered air supply system for the main control room is available.

Emergency rooms (without filtered air supply) for the Emergency Response Team are available on-site.

Off-site alternative Technical Support Centres within adequate distance from the plant are also available in case of high contamination on-site. These off site emergency facilities have been meant for emergency response organization, although sorting and decontamination equipment is not available there. Such equipment can be installed in case of accident.

All plants have different means of communication available, including several satellite telephones to be used (also indoors in specific locations).

4.2.1.2 Procedures and guidelines for accident management

Emergency organization and procedures are described in the Accident Management Manual. The emergency team directs the deployment forces that operate on the site.

In all German NPPs, AM measures - addressed to remove decay heat from the fuel assemblies in order to prevent core melt - are defined by appropriate procedures described in the Accident Management Manual. Main relevant measures for PWR plants are secondary bleed and feed (by feed water pumps or passive injection from feed water system/tank or by mobile pumps) and primary bleed and feed. For BWR's separate systems to inject water at high, intermediate and low pressure is used. Mobile pumps are foreseen as well (injection after RPV depressurization). There is also the possibility to use filtered containment venting as an additional ultimate heat sink.

Procedures to describe separate accident management measures following a core melt situation are included in the Accident Management Manual. It is planned to develop Severe Accident Management Guidelines (SAMGs) aimed at mitigating severe accidents through a more comprehensive and systematic approach. These SAMG's will include details on essential instrumentation etc. and cover the involvement of the vendor crisis teams. Validation and verification of the use of Accident Management measures is performed during regular exercises, including simulators. The development of generic SAMG is scheduled for completion by the end of 2012, following which time plant specific adjustments will be made. SAMG have already been implemented for the PWR in GKN-I. For low power and shutdown states no specific SA measures have been implemented so far, but some measures from other existing manuals can also be used in these states. Additional procedural improvements for low power and shutdown states SAM are foreseen within the framework of systematic development and implementation of SAMGs.

4.2.1.3 Hardware provisions for severe accident management

In the case of **fuel damage inside the RPV** the following accident management measures are foreseen:

- In PWR NPPs, measures to inject water into the reactor vessel following fuel damage can be used if the necessary equipment is recovered. Passive autocatalytic recombiners (PARs) are installed in the containment. Filtered containment venting (FCV) is installed to counteract containment pressure build-up. The status of the plant can be determined with accident proof instrumentation which however depends on the availability of power
- For BWR type 72 plants, flooding the drywell of the containment is an accident management measure but it cannot be stated that melt-through of the RPV can be prevented for all situations. The wetwell is inerted by nitrogen and PARs are installed in addition in drywell

and wetwell; a FCV system to limit pressure build up is installed and connected to the wetwell; one common system is used for both NPPs at the site

- For BWR type 69, the possibility for flooding the containment to cool the RPV was foreseen to be considered in the SAMG implementation before these plants have been shutdown; the wetwell and the drywell are inerted by nitrogen; a FCV system to limit pressure build-up is installed and connected to the wetwell
- It has been taken into consideration that control rods in PWR and BWR plants are destroyed or do melt at lower temperatures than fuel rods and that this fact may be of importance during the re-flooding. To avoid the possibly resulting re-criticality, different possibilities exist to inject boron, which is also stored on the sites

In the case of **failure of the RPV**,

- For PWR plants a dry phase of molten core-concrete interaction occurs in the reactor pit. Also in this phase, the hydrogen concentration is controlled by PARs; after several hours of erosion the melt gets automatically into contact with sump water
- In the case of BWR type 72 plants, PARs, N2-inertisation of wetwell and FCV are also in place. Flooding the containment in order to ensure cooling of the molten fuel and reduce the risk of MCCI is also envisaged; in most cases water is already present in the cavity at the time of RPV failure
- For BWR type 69 plants, N2-inertisation of wetwell and drywell and FCV are also in place; before those plants were shutdown, further measures were foreseen to be considered in order to prevent containment failure due to contact with molten fuel in the lower part

The following measures are in place to **ensure containment integrity:**

- Prevention of fuel damage/core melt sequences at high pressure is ensured through pressurizer relief and safety valves in PWR NPPs and safety relief valves in the BWR type plants. The actuation mode of all depressurization valves (RV, SV) has been upgraded (electromotor and solenoid) and several power supplies for valve operation are foreseen, including batteries;
- The hydrogen risk is adequately covered in the different phases of a severe accident progression by the presence of PARs and by N2 inertization for BWR plants. The PARs are seismically qualified;
- Containment failure resulting from continuous pressure increase is prevented by the availability of a FCV. Seismic qualification of this system has not been requested; the system is typically installed inside seismically qualified buildings.
- With regard to the prevention of basemat melt-through, in BWR 72 the melt is always covered by water because an overflow exists from wetwell to drywell and flooding the drywell and the wetwell by different means is implemented in AM manual; in PWR no direct injection into the cavity is possible before RPV failure; after failure it is possible through the ECCS and failed RPV. In addition after several hours of erosion the melt gets automatically into contact with sump water. Injection of water into PWR containment is also possible using the fire extinguishing system (as per design, German PWRs do not have containment spray systems).- Conclusions on the efficiency of melt cooling vary in PSA Level 2 analyses.

The implementation of key measures to maintain containment integrity, like containment isolation and containment venting can be performed without AC power. Provisions for the management of hydrogen risks are passive (PARs) and do not need any energy supply or are implemented within a very short time (N2-inertisation in BWRs).

Measuring and control instrumentation is qualified for design basis accident conditions according to KTA safety standards. Instrumentation requested for severe accident management includes the pressure and temperature measurement and hydrogen concentration measurement in the containments of PWR and BWR NPPs. The instrumentation is connected to a battery backed power train and therefore, if batteries or any other power supply to the DC-train is restored, the instrumentation can be restored.

Further studies on the use of instrumentation under severe accident conditions will be done during the development of the SAMGs.

4.2.1.4 Accident management for events in the spent fuel pools

It is assumed that due to long grace periods and the adoption of appropriate measures to ensure cooling of the fuel elements in the pool, damage to fuel elements can be excluded. Severe Accident measures for coolant injection into spent fuel storage pools are described in operator's documents (BHB/NHB). Mobile equipment will be used early enough to avoid fuel uncover (1m of water is sufficient for necessary radiation protection). Coolant injection to the fuel pool from the outside via permanently installed provisions is currently under consideration.

For the PWR plants, spent fuel pools are located inside the containment and the PARs will operate in case of hydrogen production but severe accidents in SFP have not been studied yet. . Pool conditions can be detected by direct pool measurements. Sampling in the containment is also possible. The control room is protected from direct radiation by the concrete shielding of the reactor building.

For BWR plants the spent fuel pools are located in the reactor building. Severe accident measures are focused on coolant injection which avoids fuel degradation and ensures preservation of the radiation shielding but severe accidents in SFP have not been studied yet; SFP water level and temperature measurements are available.

The control room is in a separate building and is protected from direct radiation by the concrete shielding Location and filtered air intake of the control room ensure its availability.

4.2.1.5 Evaluation of factors that may impede accident management and capability to severe accident management in multiple units case

Depending on the severity of impairment due to severe natural events, different means of transportation to the site including the use of helicopters from the technical aid or civil defence forces are possible. Additionally the KHG, the German nuclear interventions force "Kerntechnischer Hilfsdienst GmbH", is contracted by each NPP and is the relevant intervention force for nuclear emergency assistance. Also, operational hoisting gear (tractors, fork-lift trucks) is available at different locations in the plant for moving debris or snow masses.

Different possibilities exist to relocate the Emergency Response Team if the Emergency Response Room can no longer be used.

Concerning the multiple units case, at present, there is only one plant with two operating units which are fully independent from each other except for the common FCV. The case of simultaneous core melt in multiple units was not studied. The use of FCV in such a scenario is under study.

4.2.2 Margins, cliff edge effects and areas for improvements

4.2.2.1 Strong points, good practices

During the peer review process, the following general strong points for the German NPPs have been identified:

- SSAM measures including significant hardware modifications have been in place for many years.
- In case of simultaneous accidents in different units, generally accident measures can be performed independently in each unit.
- SAMGs for full power states exist for one NPP (GKN-I) and are being developed for all operating NPPs.
- Nuclear intervention force (Kerntechnischer Hilfsdienst GmbH) exists since 1977.
- Hydrogen combustion in the venting system is avoided through prior inertisation.

- Main control room habitability during filtered venting is ensured.
- Emergency response organization could be housed in different buildings. Alternative support centre is part of concept.
- Containment isolation has been qualified to withstand most probable beyond-design-basis conditions.
- Filtered containment venting Including retention of organic iodine

In addition, the following more specific strong points have been identified:

For PWRs:

- Secondary side Bleed & Feed including mobile pumps to feed the SG
- Primary side Bleed & Feed
- PARs

For type 72 BWRs

- Diverse RPV depressurization and injection systems
- Mobile pumps for RPV injection
- N2 inertisation of the wetwell
- PARs in both drywell & wetwell

For type 69 BWRs

- Diverse RPV depressurization and injection systems
- Mobile pumps for RPV injection
- N2 inertisation of the wetwell and drywell

4.2.2.2 Weak points, deficiencies (areas for improvements)

During the peer review process, the following possible areas for improvement have been identified:

- The accident-proof instrumentation is battery secured for only 2 to 3 hours
- Monitoring system for hydrogen in PWR is not seismically qualified (which is in compliance with German regulations)
- Consequences of fuel meltdown in the spent fuel pool have not been considered
- No low power or shut down SAMGs existing (some guidance in operational manuals)
- No long-term SAM procedures (to bring back the plant in a controlled and stable stage) have been developed so far

4.2.3 Possible measures to increase robustness

4.2.3.1 Upgrading of the plants since the original design

Since many years and based on a RSK Recommendation of 1989, SAM features are implemented in all plants. For the identification of accident conditions diverse measuring systems for RPV-Level and other measurements have been installed. For PWRs the available accident management measures are mainly primary and secondary feed and bleed with multiple ways for coolant injection and the use of pre-installed systems for the injection by mobile pumps into the SGs. For BWRs multiple systems for coolant injection have been implemented as well as diverse systems for depressurization of the reactor pressure vessel. Additional measures to restore the power supply have been implemented. In the area of mitigative measures for protecting the containment integrity, the Severe Accident Management Program concentrated on the use of passive autocatalytic recombiners in PWRs and BWRs of type 72.

Inertisation by nitrogen is done in BWRs 69 for the entire containment, and in BWRs 72, for the wetwell only. Filtered containment venting is installed in all German plants. To ensure the habitability of the Main Control Room under all circumstances, the measure "filtering of the supply air in the control room – maintaining overpressure" has been realized in all NPPs. All German NPPs do have an Emergency Control Room to safely shut down the reactor. All accident management measures, preventive and mitigative, as well as additional possibilities for the use of operational systems have been described in detail in the symptom-based accident management manual. Accident management manuals for treatment of beyond design basis accidents have been introduced in all NPPs.

4.2.3.2 Ongoing upgrading programmes in the area of accident management

Recently, RSK started a renewed discussion on the implemented severe accident management measures in Germany. This resulted in the publication of new and extended recommendations: "Basic recommendations for the planning of emergency control measures by the licensees of nuclear power plants" dated 14th October 2010.

4.2.4 New initiatives from operators and others, and requirements or follow up actions from Regulatory Authorities: modifications, further studies, decisions regarding operation of plants

4.2.4.1 Upgrading programmes initiated/accelerated after Fukushima

Based on the results of the plant-specific safety review of German nuclear power plants in the light of the events in Fukushima-1, the RSK concluded that there is a need for further efforts concerning, among others:

- Development of the accident management concept under external hazard conditions (re-establishment of the supply of three-phase alternating current, injection possibilities for the cooling of fuel assemblies, identification of available safety margins, consideration of wet storage of fuel assemblies, etc.).
- Review additional requirements on accident management and the optimisation of available measures.
- More specific RSK recommendations with regard to SAM are as follows:
- Systematic inclusion of internal/external hazards into the AM Program (including operability of mobile equipment)
- A need for re-check/extension of a number of AM measures (also in connection with external hazards), which are:
- Long-term energy supply (in particular to mobile systems)
- Long-term heat removal from reactor core and spent fuel pool
- Long-term heat removal from the wetwell of a BWR
- Safe release of off-gases containing combustible gases by the existing filtered containment venting system
- Use of AM measures under long-term SBO conditions
- Development of AM measures to protect the building structure surrounding the spent fuel pool in a BWR, which is outside the containment, against hydrogen combustions or to prevent them

The GRS Information Notice WLN 2012/2 contains additional recommendations as a follow-up of the Fukushima accident.

4.2.4.2 Further studies envisaged

Additional preventive and mitigative procedures and especially SAM guidelines for full power and low power and shutdown states as well as the cooling of the spent fuel pool are to be developed. Also, the extension and revision of the Accident Management concepts for NPPs, which do not continue the power operation, shall be performed.

4.2.4.3 Decisions regarding future operation of plants

The Amendment introduced the following main modifications of the Atomic Energy Act:

- The granting of further electricity production rights according to the 11th amendment of the Atomic Energy Act was cancelled. The licences for power operation of the seven oldest nuclear power plants (Biblis A, Neckarwestheim I, Biblis B, Brunsbüttel, Isar I, Unterweser, Philippsburg and the Krümmel NPP were terminated with the entry into force of the amended Atomic Energy Act on 6th August 2011.
- For the three youngest plants, the licences for power operation will expire in 2022 at the latest; for the other plants on a step-by-step basis until 2015/2017/2019/2021 at the latest.
- The transfer of electricity volumes will still be possible, provided that the respective end times are adhered to

4.3 Peer review conclusions and recommendations specific to this area

The German regulatory basis concerning severe accident management has been extensively described. This has been complemented by adequate information on the technical requirements and its compliance. Also, hardware provisions for severe accident management are described in detail.

The country review, including the plant walk down, provided comprehensive additional information.

It appears that severe accident management concept in Germany is focused on the use of preventive measures, to be followed by mitigative measures such as filtered containment venting systems. The Accident Management measures and the related plants internal organization appear to be well structured, adequate to cope with accidents with different levels of severity, including severe core damage and adequately supported by procedures and guidelines. Although guidance exists in the Accident Management Manuals, SAMGs have been developed for one plant only. It is expected that SAMGs will be available at all plants. The operability of instrumentation essential to severe accident management should be systematically evaluated for severe accident conditions. This should be achieved in the course of SAMG development. Improvements in this respect might be necessary.

List of Acronyms

AM	Accident Management
BDBF	Beyond Design Basis Flood
BMU	Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
BWR	Boiling Water Reactor
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DIN	Deutsches Institut für Normung
ERO	Emergency Response Organization
ESWS	Essential Service Water System
FCV	Filtered containment venting
GKN	Nuclear power plant Neckarwestheim
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit mbH
KKB	Nuclear power plant Brunsbüttel
KKE	Nuclear power plant Emsland
KKG	Nuclear power plant Grafenrheinfeld
KKI	Nuclear power plant Isar
KKK	Nuclear power plant Krümmel
KKP	Nuclear power plant Philippsburg
KKU	Nuclear power plant Unterweser
KRB	Nuclear power plant Gundremmingen
KTA	Kerntechnischer Ausschuss
KWB	Nuclear power plant Biblis
KWG	Nuclear power plant Grohnde
KWO	Nuclear power plant Obrigheim
MCP	Main Coolant Pumps
PAR	Passive autocatalytic recombiners
PGA	Peak Ground Acceleration
PSHA	Probabilistic Seismic Hazard Assessment
PSR	Periodic Safety Reviews
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
RHR	Residual Heat Removal
RSK	Reactor Safety Commission
RPV	Reactor Pressure Vessel
SAMG	Severe Accident Management Guidelines
SAM	Severe Accident Management
SBO	Station Blackout
SFP	Spent Fuel Pool
SSC	Structures, Systems and Components
TSR	Technical Safety Reassessment
UHS	Ultimate Heat Sink
UNS	the independent emergency system
USUS	Independent sabotage and accident protection system