Federal Ministry for the
Environment, Nature Conservation
and Nuclear Safety

EU Stresstest
National Report of Germany

Progress Report of September 15, 2011

Executive Summary

The European Council concluded in March that the safety of all EU nuclear plants should be reviewed on the basis of a comprehensive and transparent risk assessment ("stress test").

The German Bundestag (Federal Parliament) called upon the German Federal Government on 17th March 2011 to conduct a comprehensive review of the safety requirements for the German nuclear power plants. The competent Federal Ministry asked its advisory body, the RSK, to perform this review. The findings of the RSK safety review were presented to the public on 17th May 2011.

ENSREG published the scope and modalities for comprehensive risk and safety assessments of EU nuclear power plants on 13th May 2011. This "Declaration of ENSREG" determines the concept, methodology and time schedule of the EU stress test.

The BMU as the federal regulator in Germany asked the *Länder* authorities to initiate the EU stress tests according to the ENSREG Declaration. The "stress tests" were started by all German licensees with the self-commitment to deliver the progress report by 15th August 2011 and the final report by 31st October 2011.

The structure of the German national report follows mainly the chapters according to the ENSREG Declaration. In addition detailed insights from the broader scope and specific methodology of the RSK safety review are also included in the chapters. In chapter 6 are in particular insights from the RSK safety review related to initiating events caused by man-made hazards, such as an aircraft crash, terrorist attack or cyber attacks.

The licensee progress reports and the certificates by the competent federal state regulators reflect an interim state. Further work is planned and should take additional guidance to be given by ENSREG into account.

The Federal State authorities have so far reviewed the licensee reports in terms of completeness, the adequate application of the ENSREG methodology and the correct categorisation of the referenced documentation (if already done by the licensee). In summary, the adequate application of the ENSREG methodology was confirmed. The correctness of licensee statements were assessed for design basis measures and plausibility checks were performed for additional measures in the beyond design basis area where possible.

This progress report gives an overview of the work planned and of results already achieved. It would be premature to summarize results of the EU stress test. Based on the progress made so far the completion of the final licensee reports and of the report of the German regulatory body seems to be achievable.

Content

List of abbreviations

AC	Alternating current
AMM	Accident Management Measures
BfS	Federal Office for Radiation Protection
BMI	Federal Ministry of the Interior
BMU	Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
D1/D2	Emergency Power Grid 1/2
DC (power supply)	Direct Current
ENSREG	European Nuclear Safety Regulator Group
GFZ	German Research Centre for Geosciences in Potsdam
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
IAEA	International Atomic Energy Agency
KTA	The Nuclear Safety Standards Commission
MSK scale	Medvedev-Sponheuer-Karnik scale
NPP	Nuclear Power Plant
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
RSK	Reactor Safety Commission
SAMG	Severe Accident Management Guideline
SBO	Station Blackout
SR	Safety Review
TMI	Three Mile Island
TSO	Technical Safety Organisation
WENRA	Western European Nuclear Regulators' Association

0 Regulatory Framework and Infrastructure

The Federal Republic of Germany is a Federation with 16 Federal States. There are 18 nuclear power plants at 13 sites that fall under the EU stress test as requested by the European Council on 23/24 March 2011. These sites are situated in five Federal States. These plants are operated by four different utilities. The German nuclear regulatory body consists of the regulatory authority of the Federation and the regulatory authorities of the Federal States (*Länder*). Each regulatory authority is a division of a ministry. To understand how the stress test process is implemented information on cooperation of the different regulatory authorities and of their interaction with the licensees is necessary.

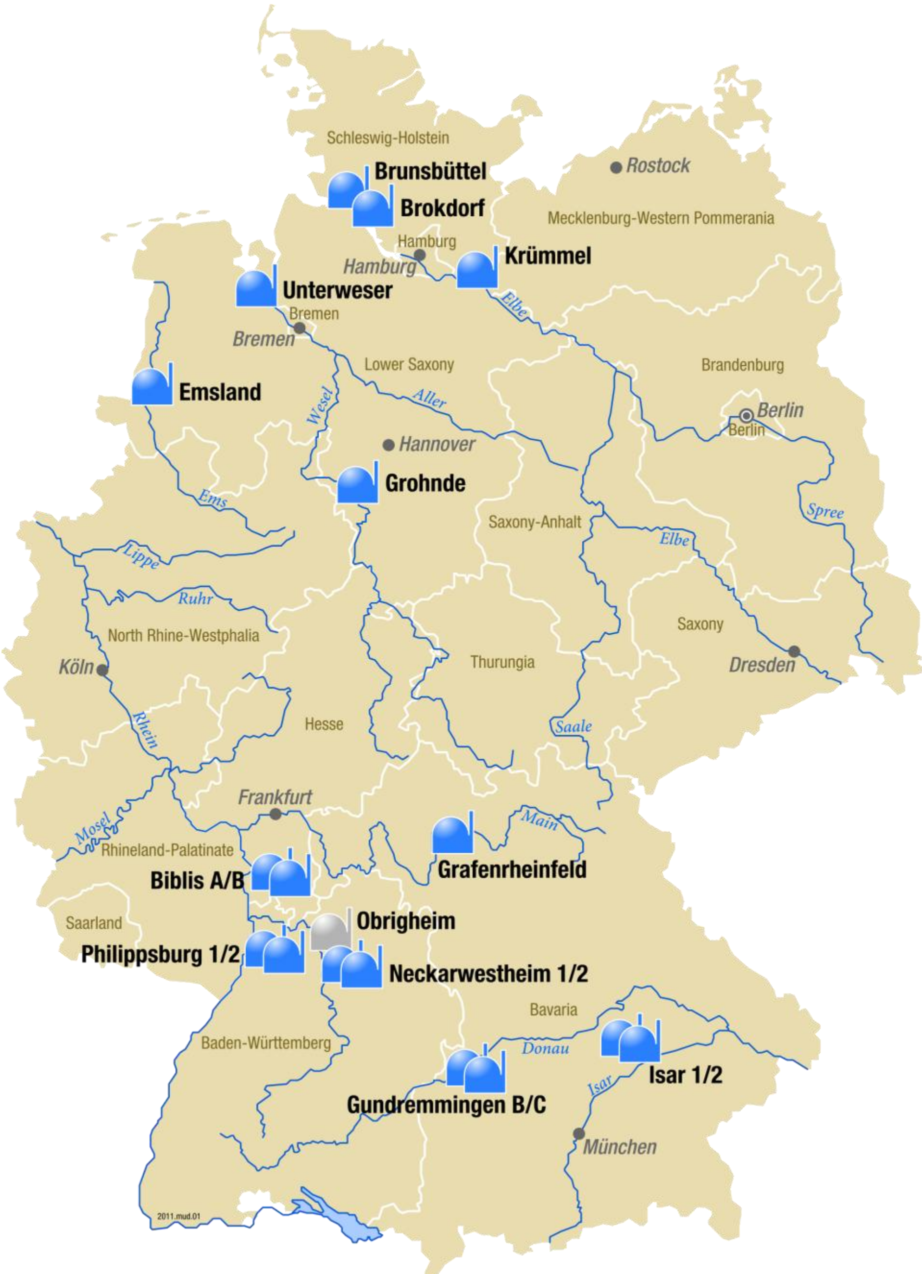

Figure 1: Sites of Nuclear Power Plants in Germany which are considered in the "stress test"

0.1 Regulatory body in Germany

Responsibilities for legislation and execution are assigned to the organs of the Federation and the Federal States - the *Länder* - according to their scope of functions. Specifications are given by provisions of the Basic Law /I/ of the Federal Republic of Germany.

The Federal Parliament has the legislative competence for the peaceful use of nuclear energy. The legal base for the peaceful use of nuclear power in Germany is the Atomic Energy Act /II/. The Atomic Energy Act is executed - with some exceptions - by the *Länder* on behalf of the Federal Government. In this respect, the *Länder* authorities are under the supervision of the Federation with regard to the lawfulness and expediency of their actions.

The "Regulatory body" in Germany is therefore composed of authorities of the Federal Government and authorities of the *Länder* governments.

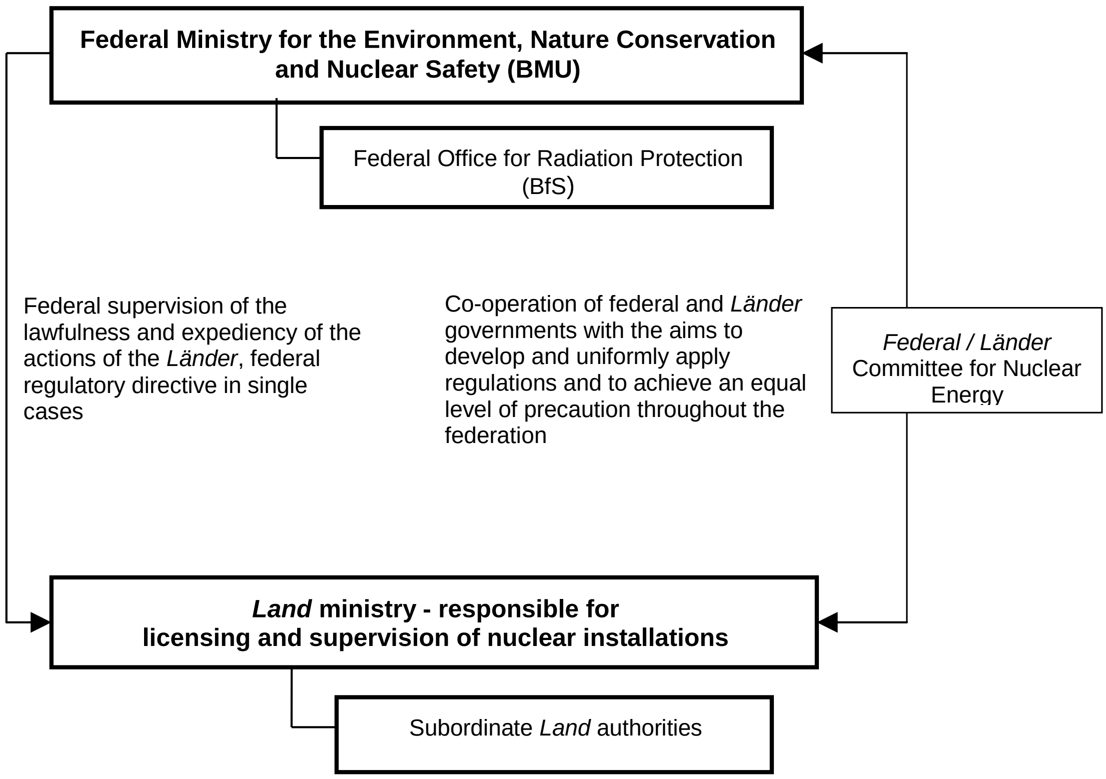

Figure 2: Organisation of the Regulatory Body

By organisational decree, the Federal Government specifies the Federal Ministry competent for nuclear safety and radiation protection. In 1986, this competence was assigned to the then newly established Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU).

Hence the BMU is the supreme regulatory authority in charge of nuclear safety and security in Germany.

Licensing and supervision of nuclear power plants are executed by the *Länder* on behalf of the Federation. In this respect, the *Länder* authorities are under the oversight of the Federation with regard to the legality and expediency of their actions.

Table 1: The *Länder* Licensing and Supervisory Authorities for Nuclear Installations

Land	**Nuclear Installations**	**Licensing Authority**	**Supervisory Authority**
Baden-Württemberg	Obrigheim Neckarwestheim 1 Neckarwestheim 2 Philippsburg 1 Philippsburg 2	Environment Ministry in agreement with Economics Ministry and Interior Ministry	Environment Ministry
Bavaria	Isar 1 Isar 2 Grafenrheinfeld Gundremmingen B Gundremmingen C	State Ministry of the Environment, Public Health and Consumer Protection In agreement with State Ministry of the Economy, Infrastructure, Transport and Technology	Bavarian State Ministry of the Environment and Public Health
Hessen	Biblis A Biblis B	Ministry of the Environment, Energy, Agriculture and Consumer Protection	
Lower Saxony	Unterweser Grohnde Emsland	Ministry for Environment and Climate Protection	
Schleswig-Holstein	Brunsbüttel Krümmel Brokdorf	Ministry of Justice, Equality and Integration (MJGI)	

0.2 Framework and arrangements for response to nuclear accidents

In Germany there are comprehensive precautions and arrangements for emergency preparedness as well as for operational experience feedback. These systems have been presented in the German report to the 5th Review meeting under the Convention on Nuclear Safety /III/.

Nuclear emergency preparedness comprises on-site and off-site planning and preparedness for emergencies.

On-site emergency planning is realised by technical and organisational measures taken at nuclear power plants to control an event or to mitigate its consequences (e.g. preventive and mitigative accident management measures).

Off-site emergency planning comprises disaster control and precautionary radiation protection. Disaster control serves for averting imminent danger. Precautionary radiation protection aims at coping with consequences of unplanned radiological releases below reference levels for short-term measures by means of precautionary protection of the population and serves for preventive health protection.

In case of severe nuclear accidents in foreign plants after - in addition to emergency response activities as appropriate - comprehensive safety investigations by licensees, regulatory authorities supported by their independent expert organizations and safety reviews by the RSK are well established practice in Germany.

In the past systematic, comprehensive safety reviews were performed by RSK after the TMI 2 and Chernobyl accidents.

Main regulatory actions based upon results and recommendations of the RSK safety review after the TMI accident in 1979 were concerned with:

- measures to enhance preventive safety
- human and organizational factors
- role of PSA as investigative tool
- consideration of beyond design conditions and related regulatory research

The regulatory actions based upon the RSK safety review after the Chernobyl accident in 1986 were focused on:

- preventive and mitigative accident management
- periodic safety reviews (PSR): safety status, PSA, security status

Due to these reviews, comprehensive action plans have been performed with the aim to extend the level of protection beyond the base on which the plants were originally licensed. In the context of the now legally required PSR any ten years safety and protection levels have to be reassessed using current site conditions and impacts conceivable at the plant site.

0.2.1 General procedure in Germany for safety reviews in response to safety significant incidents and accidents in NPP

After any safety significant incident or accident safety reviews are performed. Depending on the severity of the related event three action levels are distinguished:

- First level: Have there any deficits been revealed that need immediate actions to ensure safety of operating plants.
- Second level: Are there any lessons to be learned from the specific events or event sequence for improving safety precautions for a reliable prevention of similar event sequences. Such measures have to be addressed in corrective action programmes.
- Third level: Are there generic lessons to be learned from phenomena or root causes that have led to the accident. Such generic lessons are used to enhance safety features and to strengthen or extend the defence in depth concept.

Such reviews have to be performed by licensees as well as by nuclear safety regulators resulting in appropriate consequential actions. Depending on the severity of an accident such reviews can also lead to reassessments of the risks associated with nuclear power plant operation and related political and governmental considerations and decisions.

The safety reviews after Fukushima now try the new approach of a stress test as practiced in other areas such as the banking sector. The current stress test will enable a review of the validity of current design assumptions and of possible areas for additional protective measures.

0.3 Realisation of the EU stress test in Germany

Shortly after the Fukushima accident a safety review of the nuclear power plants was requested by the Federal Parliament as well as by Federal Government and Federal State Governments. The competent Federal Minister, the BMU, asked its advisory body, the RSK, to perform such a comprehensive safety review for preparing political and regulatory decision. This RSK safety review was carried out from 15th March to 15th May 2011. This review covered not only Fukushima related events but a broader spectrum of impacts due to initiating events conceivable at the site such as aircraft crash. For this safety review the RSK received reports from the licensees and statements from the Federal State regulators. RSK was supported by GRS, the expert organisation of the BMU.

The implementation of the EU “stress test” follows the methodology and the schedule as laid down in the ENSREG declaration. Regarding the technical scope of the “Stress test” for the progress report there was no uniform European interpretation of inclusion of extreme weather conditions and of the assessment of the loss of safety functions provoked by indirect initiating events, for instance large disturbances from electrical power grid impacting AC power distribution systems or forest fire, airplane crash.

The risks due to security threats will be treated by a different process at the EU level. The outcome of this process can only be addressed in future reports.

The progress report takes of course advantage of the RSK safety review. Findings of this review are summarized each chapter. Meanwhile licensee reports have been and will be further developed, strengthening the input for regulatory review and certification. On the other hand it is necessary to further streamline the process on the EU level to enable adequate consistency and quality of the final licensee and national reports.

Scope and methodology of the EU “stress tests” are described in the following chapter.

1 Scope, methodology and general aspects of the licensee reports

1.1 Scope and methodology of post Fukushima safety reviews in Germany

The safety review methodology as applied by the RSK /IV/ or as specified in the ENSREG declaration /V/ evaluates the plant response to extreme external and internal impacts and aggravating conditions in the environment of the plant combined with the assumption of additional losses of safety functions. In these scenarios, the sequential loss of safety functions and lines of defence is assumed in a deterministic approach, irrespective of the probability of such failures and losses. Based on the current plant status (for the EU stress test status at the 30.06.2011) and behaviour as verified under the supervision of the regulator and supplemented by additional analyses and engineering judgment, any possible weaknesses are to be identified. Measures that can be taken under such conditions to prevent or mitigate severe consequences are systemically analysed, including the assessment of robustness of design features, adequacy of protective measures and possible cliff edge effects.

Such scenarios have not been considered in the licensing and supervisory procedures. Instead, all measures were to be taken and assured that such extreme challenges can practically be excluded. The current stress test will allow a review of the validity of current design assumptions and of possible areas for additional protective measures.

In Germany, these comprehensive safety and risk assessments are performed following two different approaches:

- the approach of the RSK as specified by requirements and assessment criteria
- the approach as specified by WENRA and ENSREG and agreed on 12th-13th May 2011.

The methodology, scope and depth of these two approaches show conceptional differences. The methodology of the RSK approach was based on the concept of robustness levels. To assess the robustness four levels (basic and level 1 to 3) have been defined by the RSK for any topic to be analysed. These levels reflect the assurance of the required safety functions to prevent „cliff edges“. The RSK based its review on licensee reports that have been prepared on the basis of a questionnaire.

In the approach for EU-“stress test” the approach and methodology for the reassessment are presented by short instructions to any issue to be reported for the different topics listed. Furthermore, the margins in view of robustness shall be described, but no further classification is foreseen.

Concerning the scope of the assessment the RSK considered also the robustness of precautionary measures. Furthermore, man made hazards including aircraft crash, blast wave and toxic gases were analysed. Moreover, also terrorist and cyber attacks have been considered and have been quoted in chapter 6.

- **The progress report**

The progress report will present the current status and results achieved so far from both approaches. The final reports shall reflect further progress achieved by the licensee reports, regulatory reviews and of an improved common understanding on the EU level regarding content, scope and methodology.

The reports by the German licensees used a structure that has been agreed in principle among the European utilities. This structure has been adapted by the German licensees to the German situation.

In each chapter, the results achieved by the RSK so far will be quoted in a concise form. The RSK is continuing its work on issues of special interest identified so far. The results and the on-going work programme are available on the homepage of the Reactor Safety Commission (http://www.rskonline.de/English/index.html).

1.1.1 The safety review performed by the Reactor Safety Commission (RSK)

In connection with the events in the Japanese Fukushima-I plant, the German Bundestag (Federal Parliament) called upon the German Federal Government on 17th March 2011 to conduct a comprehensive review of the safety for the German nuclear power plants. On the request by the competent, federal nuclear regulator the BMU, its advisory body, the RSK, performed this review. The RSK endorsed the catalogue of requirements for plant-specific reviews of German nuclear power plants in the light of the events in Fukushima-I - Japan. The insights gained from the accident sequence in Japan were to be considered in particular with respect to whether the current design limits had been defined correctly and how robust the German nuclear power plants are regarding beyond-design-basis events.

- **General considerations**

On the basis of the information available so far about the accident sequence in Japan, the RSK derived the following need for review for the German nuclear power plants:

- Examination of to what extent the fundamental safety functions "reactivity control", "cooling of fuel assemblies in the reactor pressure vessel as well as in the fuel pool" and "limitation of the release of radioactive substances (maintaining of the barrier integrity)" are fulfilled in the event of impacts beyond the design requirements applied so far.
- Examination of to what extent the system functions for fulfilling the fundamental safety functions remain available for assumptions going beyond the scenarios postulated so far.
- Review of the necessary scope of accident management measures and their effectiveness.

One focus of the review regarding the robustness of all installations and measures was on the identification of an abruptly occurring aggravation in the event sequence (cliff edges) and, if necessary, on the derivation of measures for its avoidance (example: exhaustion of the capacity of the batteries in the event of a station blackout). Included in the scope of the review were:

- Natural hazards such as earthquakes, flooding, weather-related effects as well as possible simultaneous occurrences.
- Postulates that are independent of concrete event sequences, such as failures affecting several redundant system trains, (common-cause failures, systematic failures), station blackout for longer than two hours, long-lasting loss of auxiliary service water supply.
- Aggravating boundary conditions for the performance of accident management measures, such as non-availability of electricity supply, hydrogen formation and explosion risk, restricted availability of personnel, inaccessibility due to high radiation levels, impairment of external technical support.

- ***Conclusions of the RSK safety review***

Based on the robustness levels determined for each issue, the RSK came to the following conclusion:

"It follows from the insights gained from Fukushima with respect to the design of these plants that regarding the electricity supply and the consideration of external flooding events, a higher level of precaution can be ascertained for German plants.

The RSK has furthermore reviewed the robustness of German plants with respect to other important assessment topics.

The assessment of the nuclear power plants regarding the selected impacts shows that for the topic areas considered, there is no general finding for all plants in dependence of type, age of the plant, and generation.

The existing plant-specific design differences according to the current state of licencing were only partially considered by the RSK. Plants that originally had a less robust design were backfitted with partly autonomous emergency systems to maintain the fundamental safety functions. In the robustness assessment performed here, this selectively leads to evidentially high degrees of robustness.

The RSK has derived first recommendations for further analyses and measures from the results of the plant-specific review:"

- ***Continuation of RSK consultations***

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 agreed on the topics to be further dealt with:

Earthquake

Consideration of all conditions of low-power and shutdown operation (e.g. flooded reactor cavity during refuelling).
New curves for the determination of the probabilities of seismic acceleration loads at concrete sites that might lead to a higher level of design earthquakes.

Flood

Protection of canals and buildings regarding the intrusion of water and the floating resistance in the case of a higher level flood. Assumed postulate: flooding of the plant site.
Accessibility of the plant buildings in the case of longer-term flooding.

Station blackout

Specific examination of low-power and shutdown operation and storage of the fuel assemblies in the fuel pool. Battery capacities, safety margins of the plants, demand for 10 hours of availability.

Loss of offsite power

Long-lasting loss of offsite power, superimposition of an aftershock with operation of the emergency diesels.

Loss of service water supply

Robustness of the existing service water supply requirements under consideration of account current operating experience, also taking into account the cooling of the fuel assemblies both in the fuel pool and in the reactor core during low power and shutdown operation.

Precautionary measures

In-depth examination of precautionary measures to prevent load crashes in the area of the primary system and the fuel pool.
Generic aspects of “flooding of the annulus” in PWR plants

Accident management measures

Further 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.).
Supplementation of the requirements on accident management (SAMG).
Optimisation of available measures.

Aircraft crash

Consequential mechanical effects due to an aircraft crash that lead to a limited loss of coolant.
Protection of the fuel pool of plants in decommissioned.

Release of explosive and toxic gases in the vicinity of plants

Verification of adherence to safety margins in the case of blast waves and site-specific consideration of toxic gases.

Effects of an accident in one power plant unit on the neighbouring unit

Based on the damage states of a power plant unit, the consequences for the maintaining of the fundamental safety functions of the unaffected unit are to be examined.

Generic issues

Superimposition of events with system operating conditions of short duration (e.g. superimposition of earthquakes with loaded fuel assembly transport casks attached to a crane).
Long-term operation and post-operational phase of the fuel pools.
Impact on grid stability.

The RSK has requested their expert committees to resume consultations on the respective topics. The results of these consultations will be considered for the final report.

1.1.2 The „stress test“-specifications of WENRA and ENSREG

The European Council concluded on 24th and 25th March that the safety of all EU nuclear plants should be reviewed on the basis of a comprehensive and transparent risk assessment ("stress test"). ENSREG and the Commission were invited to develop as soon as possible the scope and modalities of these tests.

In response to this request, ENSREG made use of the WENRA proposal of 22th-23th March as a basis for the definition of the "stress test" and ENSREG published the scope and modalities for comprehensive risk and safety assessments of EU nuclear power plants on 13th May 2011. This "Declaration of ENSREG" /V/ determines the concept, methodology and time schedule of the elaboration of this report.

The further proceeding was discussed at different opportunities and will be addressed by different task forces to streamline reporting and peer reviews. The ENSREG Declaration uses some terms which are important for the assessment. The current understanding of these terms is as follows.

"Stress test" is defined as a targeted reassessment of the safety margins of nuclear power plants in the light of the events which occurred at Fukushima: extreme natural hazards challenging the plant safety functions and leading to a severe accident.

"Cliff edge" is defined as a step change in the event sequence. Examples are the exhaustion of the capacity of the batteries in the event of a station black out or exceeding a point where significant flooding of the plant area starts after water overtopping a protection dike.

These terms need a more precise and applicable definition in the context of the defence in depth concept and the related safety and design concept as applied t plants.

1.1.3 The performance of the EU stress test in Germany

The BMU as the Federal regulator asked by letter of 30th May 2011 the *Länder* authorities to initiate the EU stress according to the ENSREG Declaration, for those NPPs under their regulatory supervision that fall under the ENSREG Declaration. The "stress tests" were started by all German licensees latest on 1st June 2011 with the self-commitment to deliver the progress report until 15th August 2011 and the final report until 31st October 2011 as requested by the ENSREG Declaration.

On invitation by the BMU a joint meeting of the regulatory authorities of the federation and the federal states, of their designated safety experts and the licensees was held on 30th June 2011. The necessary activities of the parties involved, the timeframes of the activities and issues of process implementation were discussed and agreed.

The basic procedure of the EU stress test in Germany is shown in figure 3:

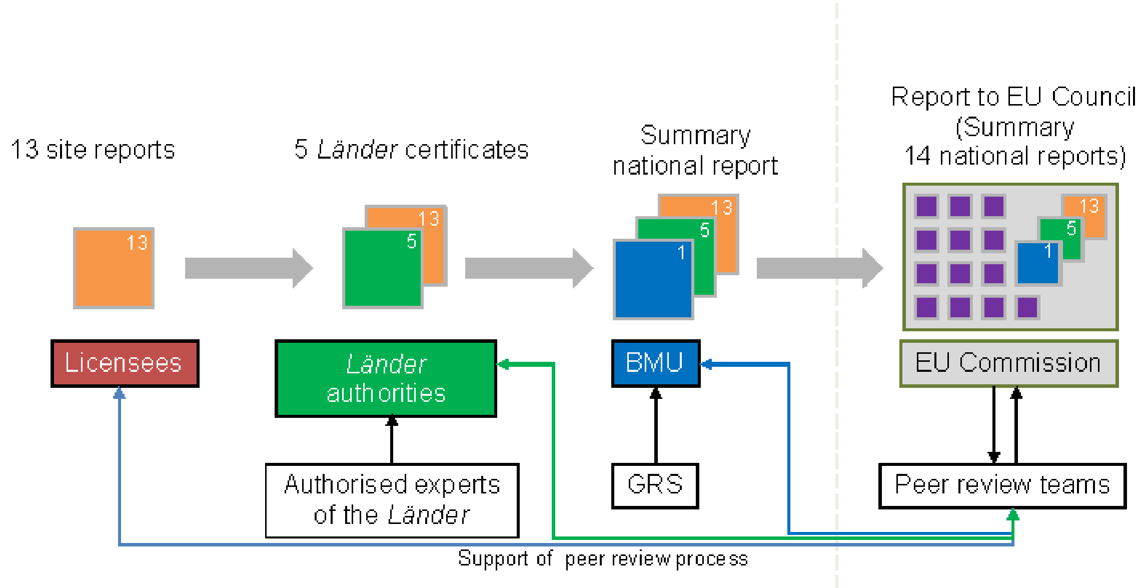

Figure 3: Procedure of the EU stress test in Germany

1.1.3.1 The approach of the German licensees

As agreed at the meeting the licensees proposed a common structure of the licensee's reports which covered the requirements and methods of the ENSREG Declaration to be applied for all 13 sites in Germany. This structure was proposed after coordination among NPP licensees in the EU. The proposed structure was accepted by the regulators in principle with some amendments regarding assumptions, methods and - regarding chapter 6 - the issues covered by the RSK safety review. Due to the tight time frame these amendments could not be taken into account for the progress report by the licensees.

In the licensee's reports, the respective plants of each site are examined site by site, with special consideration of the site-specific conditions. By the deadline of 15th August 2011, the licensees had submitted their progress reports with an extent of 100 to 200 pages to the *Länder* supervisory authorities /10-27/. This national progress report makes a first, but due to time constraints limited use of the licensee reports and of the related reviews by the *Länder* supervisory authorities.

1.1.3.2 Activities of the *Länder* authorities

It was commonly agreed that the *Länder* regulatory authorities prepare review certificates on the respective licensees' reports. In these certificates the following review aspects should be addressed:

Review aspect	Progress report	Final report
1. Completeness of responses	X	X
2. Adequate application of the ENSREG methodology	X	X
3. Correct classification of the referenced documentation	X	X
4. Assessment of the engineering judgement (plausibility)		X
5. Assessment of improvements proposed to increase the robustness		X
6. Short overall appraisal	X	X

Reviews of the submitted reported and references by the dedicated experts have already been initiated from the beginning. In this context the design base against external hazards as verified by the regulatory authorities shall be highlighted to enhance the common understanding in the EU. Compared to the RSK safety review presented on 16th May 2011 meanwhile further supporting documentation and verifications have been and will be presented by the licensees. These will enable more detailed regulatory review and assessment for the final report.

The licensees have reported on design base measures and on beyond design base measures that have already been reviewed in the supervisory process. For these the correctness of statements has been reviewed. For additional measures in the beyond design base area only a plausibility review will be performed for the final report. The reviews refer to the plant status of 30th June 2011.

Safety documentation referenced by the licensees is categorized as follows:

- category 1: reviewed and confirmed by experts in a licensing or supervisory procedure
- category 2a: formally submitted for a licensing or supervisory procedure
- category 2b: not formally submitted for a licensing or supervisory procedure, but with quality assurance by the licensee.

The review certificates for the licensee progress reports were forwarded to the BMU by the federal state regulators until 2nd September 2011. In these reports the main topics such as earthquakes, flooding, emergency power case and accident management are covered.

The final reports of the licensee reports, due on 31st October, will be reviewed by the federal state regulators and their dedicated experts and discussed with the licensees until 18.11.2011. The summary reports of the federal state regulators will be sent to the BMU until 02.12.2011.

The reviews of the licensee reports by the federal state regulators can be based on continuous supervision of the plant's safety status, operational experience and safety records. As reported for the 5th Review Meeting under the CNS over their entire lifetime - from the start of construction to the end of decommissioning with the corresponding licences - nuclear installations are subject to continuous regulatory supervision in accordance with the Atomic Energy Act and accessory nuclear ordinances.

Supervision is performed by the *Länder* authorities. The *Länder* are assisted by independent authorised experts. The decisions on supervisory measures to be performed are taken by the regulatory authority. The supervisory authority pays particular attention to

- the fulfilment of the provisions, obligations and ancillary provisions imposed by the licence notices,
- the fulfilment of the requirements of the Atomic Energy Act, the nuclear ordinances and the other nuclear safety standards and guidelines, and
- the fulfilment of any supervisory order.

To ensure safety, the supervisory authority monitors, also with the help of its authorised experts or by other authorities,

- the compliance with the operating procedures,
- the performance of in-service inspections of components and systems important to safety,
- the evaluation of reportable events,
- the implementation of modifications of the nuclear installation or its operation,
- the radiation protection monitoring of the nuclear power plant personnel,
- the radiation protection monitoring in the vicinity of the nuclear installation, including the operation of the independent authority-owned remote monitoring system for nuclear reactors,
- the compliance with the authorised limits for radioactive discharge,
- the measures taken against disruptive action or other interference by third parties,
- the trustworthiness and technical qualification and the maintenance of the qualification of the responsible persons as well as of the knowledge of the otherwise engaged personnel in the installation, and
- the quality assurance measures.

In accordance with the Atomic Energy Act, the authorised experts called in by the supervisory authority have access to the nuclear installation at any time and are authorised to perform necessary examinations and to demand pertinent information

1.1.3.3 Activities of BMU

The BMU prepares the national progress report by 15th September and the final national report by 31th December 2011 on the basis of the licensees' reports and the review certificates provided by the *Länder* regulatory authorities with expert support by the GRS.

The structure of the final reports is guided by the structure of the licensees' reports. In a preparatory phase until the end of August BMU and GRS elaborated a common framework and the general background for each chapter taking the licensee’s progress reports into account. The draft national progress report was exchanged between the participants for comment and improvement.

After submittal of the national progress report to the European Commission on 15th September 2011 all parties involved will discuss the further working programme to complete the agreed work programme and to assure timely submittal of the final report of Germany on 31.12.2011.

1.1.3.4 Structure of the German national report

The structure of the German national report follows mainly the chapters according to the ENSREG Declaration and the agreed proposal by the licensees. In addition to these chapters, the insights from the broader scope and specific methodology of the RSK safety review are also included in chapter 6.

1.2 Sites and main plant data

Within the framework of the European stress test, 18 nuclear power plants at 13 sites were analysed according to their status as at 30th June 2011. Of these plants, 17 had an operating license until that date. The Obrigheim plant has been in permanently shut down since 2005, but spent fuel is still stored in the spent fuel pool. Hence, according to the definition of the “General Aspects” in the ENSREG Declaration, this plant also has to be considered in the “stress test”.

Table 2 depicts the main data of the nuclear power plants.

Table 2: Main data of German NPP subjected to EU "stress test"

Name	Type; Construction line	thermal power [MW]	1. Criticality	Location of spent fuel storage
Location and number of units			License holder	
Biblis A*	PWR 2	3517	16.07.1974	in containment
Biblis B*		3733	25.03.1976	
Two similar units at river upper Rhein			RWE Power	
Brokdorf	PWR 3	3900	08.10.1986	in containment
Single unit at river lower Elbe			E.ON Kernkraft	
Brunsbüttel*	BWR 69	2292	23.06.1976	in reactor building outside containment
Single unit at river lower Elbe			Kernkraftwerk Brunsbüttel	
Emsland	PWR 4	3850	14.04.1988	in containment
Single unit at river Ems			Kernkraftwerke Lippe-Ems	
Grafen-rheinfeld	PWR 3	3765	09.12.1981	in containment
Single unit at river Main			E.ON Kernkraft	
Grohnde	PWR 3	3900	01.09.1984	in containment
Single unit at river Weser			E.ON Kernkraft	
Gund-remmingen B	BWR 72	3840	09.03.1984	in reactor building outside containment
Gund-remmingen C			26.10.1984	
Two similar units at river Donau			Kernkraftwerk Gundremmingen	
Isar 1*	BWR 69	2575	20.11.1977	in reactor building outside containment
Isar 2	PWR 4	3950	15.01.1988	in containment
Two different units at river Isar			E.ON Kernkraft	
Krümmel*	BWR 69	3690	14.09.1983	in reactor building outside containment
Single unit at river Elbe			Kernkraftwerk Krümmel	
Neckar-westheim 1*	PWR 2	2497	26.05.1976	in containment

Neckar-westheim 2	PWR 4	3950	29.12.1988	
Two different units at river Neckar			EnBW Kernkraft	
Philippsburg 1*	BWR 69	2575	09.03.1979	in reactor building outside containment
Philippsburg 2	PWR 4	3950	13.12.1984	in containment
Two different units at the upper Rhein			EnBW Kernkraft	
Unterweser	PWR 2	3900	16.09.1978	in containment
Single unit at the lower Weser			E.ON Kernkraft	
Obrigheim	PWR 1	1050	22.09.1968	in containment
Single unit in final shutdown at the Neckar			EnBW Kernkraft	

* shut down since German moratorium

1.3 Overview of main safety significant differences of units

According to the time of their construction, the nuclear power plants with pressurised water reactors can be classified according to four construction lines, whereas those with boiling water reactors belong to two different construction lines. The construction line is given for each plant in the second column of table 2.

The plants of the 1st construction line of pressurised water reactors (Obrigheim and Stade) have in the meanwhile been shut down. The 2nd construction line consists of PWRs which went into operation in the end of the 70ties. These have been succeeded by the so called “pre-Konvoi” plants of construction line 3 in the 80ties. The 4th construction line consists of three plants of the Konvoi type.

Concerning BWRs two construction lines are present, construction line 69 and 72.

The construction lines illustrate the continuous development in safety technology. The 1st and 2nd construction line of PWR and the 69 construction line can be dedicated to generation 2 of the international categories of NPP and the other construction lines to generation 3.

Design characteristics important to the safety of these construction lines have been summarized and presented in the reports of Germany for the reports under the Convention on Nuclear Safety.

The design underlying the safety concept shall be included in the final report. Before that it has to be clarified on the EU level to what extent and level of detail the systems for providing or supporting main safety function shall be addressed in the final national report.

1.4 Scope and main results of safety assessments SR/PSA

Since the beginning of the 1990s, Safety Reviews (SR) have been carried out every 10 years of plant operation according to standardized national criteria. They consist of a deterministic safety status analysis, a probabilistic safety analysis (PSA) and a deterministic analysis on physical protection of the plant.

The performance of safety reviews is stipulated in the amended version of the Atomic Energy Act of April 2002 and based on the respective current national guidelines for the deterministic and probabilistic safety analysis. Hence, the PSAs performed so far are part of the SR.

1.4.1 General aspects of PSA in Germany

The probabilistic safety analysis (PSA) has to be performed as part of the SRs according to a PSA guideline. Supplementary technical documents ("PSA Methods" and "PSA Data") to the regulatory guideline provide methods and data to be applied. The PSA guideline was revised in November 2005 in view of an extended scope within the framework of the Safety Review.

According to the guideline, a PSA for power operation is to be performed up to and including Level 2 by applying methods corresponding to the state-of-the-art of science and technology. Preferably plant-specific data have to be used.

A PSA for low power and shutdown states is to be performed up to the Level 1. Internal initiating events as well as internal and external hazards obligatory have to be analysed.

The PSA analyses and quantifies the plant response to initiating events conceivable at the site and plant. PSAs are used to assess strengths and weaknesses in particular vulnerabilities and cliff edge effects in design and operation and to identify improvements. Generally, relative criteria and not absolute criteria are used when comparing the results to those from deterministic safety analyses and engineering judgement. PSA results are also used to assess the determining factors and their significance contributing to vulnerabilities of a plant and to assess the balance of plant design and operation.

The frequency of operational occurrences and accidents due to internal and external causes as well as potential faults and failures of the safety systems/components are analysed. Further, erroneous human actions are addressed. Event sequences that lead to plant states which cannot be controlled according to the designed safety features are called hazard states. Additionally or instead core damage states have been analysed and presented as end states of level 1 PSAs. At such end states also measures for preventive accident management as specified in the emergency manual can be taken into account.

Every plant in Germany has performed a PSA according to these requirements; some plants make use of it as a Living-PSA. In the following chapter the main results are summarised.

1.4.2 Results of the PSAs

All of the values given in the PSA result tables are taken from the “stress test”-reports of the licensees and have to be regarded as preliminary values. They may alter in the final report. /10-29/. The values have been cross-checked partly by the *Länder* supervisory authorities. Some results are still preliminary and need further review in the supervisory process.

The following tables show the results of PSA level 1 and level 2 for each NPP in Germany:

- In the tables “HS” is the abbreviation for the hazard states as outlined in the chapter before and “CD” stands for core damage; “LERF” means large early release frequency and “LRF” stands for large release frequency.
- The table contents is based on the tasks which must be performed according to the German PSA-Guide:
  PSA Level 2 must be performed for internal initiating events in case of power operational states only;
  PSA Level 1 must be performed for internal initiating events in case of power operation, low power operation and shutdown states;
  PSA Level 1 must be performed for internal and external hazards in case of power operation only.
- A sum of the several CDF contributions to PSA Level 1 is only provided if the sum is also mentioned in the “stress test”-report of the licensee. In this case it should be an upper bound of the sum of all parts of the PSA level 1.

<table>
<tr><th></th><th></th><th colspan="2">GKN I</th><th colspan="2">GKN II</th><th colspan="2">KKP 1</th><th colspan="2">KKP 2</th></tr>
<tr><th></th><th></th><th>HSF [1/a]</th><th>CDF [1/a]</th><th>HSF [1/a]</th><th>CDF [1/a]</th><th>HSF [1/a]</th><th>CDF [1/a]</th><th>HSF [1/a]</th><th>CDF [1/a]</th></tr>
<tr><td rowspan="11">PSA Level 1</td><td>Power operation</td><td>2,20 E-05</td><td>2,20 E-06</td><td>1,40 E-06</td><td>5,80 E-07</td><td>2,90 E-06</td><td>1,40 E-06</td><td>1,60 E-06 [3)]</td><td>7,60 E-07 [3)]</td></tr>
<tr><td>Low power / shutdown</td><td>4,20 E-06</td><td>4,20 E-06</td><td>4,50 E-06</td><td>4,50 E-06</td><td>1,50 E-05</td><td>3,50 E-06</td><td>1,80 E-06</td><td>1,80 E-06</td></tr>
<tr><td>Fuel pool cooling</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td>7,00 E-08</td></tr>
<tr><td>Fire</td><td>1,50 E-05</td><td>8,60 E-07</td><td>2,10 E-06</td><td>1,90 E-07</td><td>8,30 E-06</td><td>1,50 E-06</td><td>5,50 E-07</td><td>5,50 E-07</td></tr>
<tr><td>Internal flooding</td><td></td><td></td><td></td><td></td><td>1,40 E-07</td><td>7,70 E-08</td><td colspan="2">see power operation</td></tr>
<tr><td>Airplane crash</td><td>1,40 E-08</td><td>1,40 E-08</td><td>2,60 E-09</td><td>2,60 E-09</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Flooding</td><td>1,50 E-07</td><td>4,80 E-08</td><td>8,20 E-08</td><td>3,40 E-09</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Explosion pressure wave</td><td>2,30 E-09</td><td>2,30 E-09</td><td>< 1,00 E-07</td><td>< 1,00 E-07</td><td></td><td></td><td></td><td></td></tr>
<tr><td>Earthquake</td><td>1,60 E-08</td><td>1,90 E-09</td><td>5,80 E-09</td><td>9,30 E-10</td><td></td><td></td><td>1,50 E-07</td><td>1,50 E-07</td></tr>
<tr><td>Extreme weather conditions</td><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Sum</td><td></td><td>7,30 E-06</td><td></td><td>5,40 E-06</td><td></td><td>6,50 E-06</td><td></td><td>3,30 E-06</td></tr>
<tr><td rowspan="2">PSA Level 2
Power Operation</td><td>LERF [1/a]</td><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td><td colspan="2"></td></tr>
<tr><td>LRF [1/a]</td><td colspan="2">5,6 E-08 [1)]</td><td colspan="2">3,2 E-08 [1)]</td><td colspan="2"></td><td colspan="2">2,10 E-09 [2)]</td></tr>
</table>

[1)] Frequency of releases with more than 5% of core inventory of Caesium

[2)] Frequency of releases with more than 1% of core inventory of Caesium

[3)] Including internal flooding

		KKK		KKI 1		KKI 2		KKG	
		HSF [1/a]	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**
PSA Level 1	Power operation	6,80 E-06 [1)]	1,30 E-06 [1)]		9,50 E-08 [8)] 3,70 E-06 [9)]		3,20 E-07 [1)]		9,20 E-07 [9)]
	Low power / shutdown	1,70 E-06	4,60 E-07		1,40 E-06 [10)]		1,10 E-07 [10)]		4,90 E-07 [10)]
	Fuel pool cooling	2,00 E-07 2,00 E-06 [2)]							
	Fire	1,45 E-07	1,40 E-07						
	Internal flooding	see power operation							
	Airplane crash	6,80 E-09 [3)]							
	Flooding		1,80 E-10		1,90 E-07 [7)]		< 1,00 E-07		negligible
	Explosion pressure wave	< 2,50 E-08							
	Earthquake	3,80 E-06 [6)]			negligible		negligible		negligible
	Extreme weather conditions				negligible		negligible		negligible
	Sum		**< 8,0 E-06**						
PSA Level 2 Power Operation	**LERF [1/a]**	8,10 E-08 [1),4)]		6,30 E-10 [11),12)]		3,10 E-11 [11),12)]		1,80 E-10 [11),12)]	
	LRF [1/a]	1,30 E-07 [1),5)]		4,70 E-09 [12)]		8,70 E-10 [12)]		2,80 E-10 [12)]	

[1)] Includes the initiating events: transients, LOCAs, internal flooding
[2)] Low power shutdown
[3)] Coarse analysis
[4)] < 10h after start of initiating event, more than 1% Caesium inventory
[5)] immediately after start of initiating event, more than 1% Caesium inventory
[6)] Intensity < 6 (MSK), only frequency of IE has been calculated.
[7)] Frequency of dam failure [1/a]
[8)] In the frame of PSA Level 2: internal events only
[9)] Internal and external events
[10)] Internal events
[11)] Early: until 10 h after start of IE
[12)] Large: min 1% of caesium core inventory

		KKB		KRB B/C		KBR		KWB A	
		HSF [1/a]	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**
PSA Level 1	Power operation	7,10 E-06 [1]	1,90 E-06 [1]	3,11 E-07	1,78 E-07		1,20 E-06 [2]	1,63 E-06	
	Low power / shutdown	3,50 E-05	3,30 E-06	4,85 E-07		1,40 E-06	1,10 E-06		
	Fuel pool cooling	2,50 E-07 [2] 2,50 E-06 [3]							
	Fire	1,70 E-07	3,80 E-08			9,20 E-07		< 1,00 E-07 [13]	
	Internal flooding						In „Power operation“ enthalten	1,80 E-07	
	Airplane crash	1,10 E-07 [4]					<< 1,00 E-07		
	Flooding		<< 1,00 E-06 [7]				1,40 E-07		
	Explosion pressure wave	1,10 E-06 [5]					< 7,00 E-08		
	Earthquake	3,60 E-06 [6]					negligible		
	Extreme weather conditions						negligible		
	Sum		**< 1,00 E-05**					**< 3,00 E-06** [11] **< 3,00 E-05** [12]	
PSA Level 2 Power Operation	**LERF [1/a]**			2,00 E-09 [14]		1,30 E-10 [8),9)]		< 1,00 E-09 [10]	
	LRF [1/a]					2,00 E-10 [9]			

[1] Includes the initiating events: transients, LOCAs, internal flooding
[2] Including external hazards
[3] Low power shut down
[4] Coarse analysis
[5] Only frequency of IE has been calculated
[6] Intensity < 6 (MSK), only frequency of IE has been calculated
[7] Estimation (Frequency of IE 1,00 E-04, water intake for design flood << 1,00 E-02)
[8] Early: until 10 h after start of IE
[9] Large: min 1% of caesium core inventory
[10] PSA of level 2 not yet finished, results comparable to KWB B expected
[11] Confirmed by reviewer in 2005
[12] A conservative estimation of the regulator, including earthquake and low power shutdown
[13] For selected critical rooms only
[14] Large: > 10% Iodine inventory; early: less than 10 h after start of initiating event

		KWB B		KKE		KWG		KKU	
		HSF [1/a]	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**	**HSF [1/a]**	**CDF [1/a]**
PSA Level 1	Power operation	8,11 E-07	2,33-07	1,52 E-06	5,4 E-07		8,10 E-07 [3)]		9,00 E-07 [3)]
	Low power / shutdown	6,30 E-07		4,40 E-07			7,70 E-07	1,00 E-06	
	Fuel pool cooling								
	Fire	3,63 E-08		3,65 E-6	6,70 E-07				
	Internal flooding	1,95 E-09							
	Airplane crash	1,10 E-08							
	Flooding	negligible					negligible	6,50 E-07/a	
	Explosion pressure wave			negligible					
	Earthquake	1,70 E-07					3,60 E-08	negligible	
	Extreme weather conditions						negligible	negligible	
	sum								
PSA Level 2 Power Operation	**LERF [1/a]**	< 1,00 E-09 [1)]		2,21 E-08 [2)]		3,30 E-10 [4),5)]		not completed	
	LRF [1/a]					6,30 E-09 [5)]			

[1)]< 10h after start of initiating event, more than 1% Iodine or Caesium inventory

[2)]Large: > 10% Iodine inventory, Early: less than 10 h after IE

[3)]Including external hazards

[4)]Early: until 10 h after start of IE

[5)]Large: min 1% of caesium core inventory

2 Earthquakes

2.1 Generic aspects

All nuclear power plants in Germany were designed to withstand the natural external hazards, such as wind and snow. In addition, flooding and earthquakes were taken into account depending on the site specific hazard. For flooding, earthquake and lightning nuclear safety standards are available, whereas the design against other natural hazards is based on conventional civil engineering standards.

Design against earthquake

Since 1990, the design against earthquakes is based on a design basis earthquake (formerly called "safe shut-down earthquake") in accordance with safety standard [KTA 2201.1] /2.1/. The so-called operating basis earthquake, formerly to be considered additionally according to the previous version of 1975, was replaced by an "inspection earthquake" where only the plant condition has to be checked. The design basis earthquake has the largest intensity that, under consideration of scientific findings, could occur in a wider vicinity of the site (up to a radius of about 200 km). Depending on the site, the intensity of the design basis earthquake varies between less than VI and a maximum of VIII on the MSK scale.

In the power plants of older construction lines, the seismic qualification of civil structures, components and plant equipment was partly based on simplified (quasi-static) methods which delivered the basic values for the corresponding design specifications. In more recent nuclear installations, the newly developed dynamic analyses were also applied.

Review by the regulatory authority

After the applicant had pre-selected a site, a regional planning procedure was initiated which preceded the nuclear licensing procedure. This took into account all impacts of the individual project on the public, on traffic ways, regional development, landscape protection and nature conservation. Besides the site characteristics, the design of the nuclear installation against external hazards was checked in the nuclear licensing procedure.

Re-evaluation of the site-specific conditions:

The safety reviews which have to be performed every ten years also include a re-evaluation of the protective measures against external hazards, considering the development of the state of the art. As a result of these reviews, measures have been taken or planned as far as necessary.

The protection against external hazards is assessed on the basis of the Safety Criteria for Nuclear Power Plants /2.2/, the RSK guidelines /2.3/, accident guidelines /2.4/ and the relevant KTA safety standards /2.1/.

The Safety Criteria for Nuclear Power Plants /2.2/ require that all plant components necessary to safely shut down the nuclear reactor, to remove residual heat or to prevent uncontrolled release of radioactive material shall be designed such that they are able to perform their function even in the case of external hazards.

As regards the design against external hazards, the accident guidelines /2.4/ distinguish between hazards to be treated as design basis accidents in the sense of the guidelines and hazards which, on account of their low occurrence probability, are not considered as design basis accidents, and for which measures are taken to minimise the risk. Accordingly, the external natural hazards (earthquake, flood, external fire, lightning and other natural hazards) have to be considered as design basis accidents.

For some nuclear installations at sites with relevant seismicity, a re-evaluation of the seismic safety has been performed due to the on-going development of methods for seismic hazard analysis and for design verification in particular in the context of periodic safety reviews. In general, the re-evaluations with regard to the design of components showed that, on the basis of more precise seismic input and modern verification methods, the technical equipment of the plants partly has considerable margins with respect to seismic loading.

2.2 Results of the RSK safety review

In the framework of the RSK safety review the safety of the German NPPs was assessed in particular with respect to the external hazards ‘earthquake’ and ‘flooding’. This assessment was mainly focused on the robustness of the NPPs, i. e. the safety margins available for beyond design basis events. Due to this focus, the appropriateness of the design basis itself was not re-evaluated. From a comparison of the site specific hazard and the design of the NPPs as well as additional questions regarding potential effects of beyond design basis events and available accident management measures for such events conclusions on the safety margins were drawn by the RSK. To quantify the resilience of the plants the RSK defined robustness levels. For earthquakes these robustness levels were defined as follows:

“Basic level: The safety of the plant is demonstrated for an earthquake with an exceedance probability of 10-5/a.

Level 1: Design margins with respect to the earthquake determined plant-specifically according to the state of the art in science and technology – basis: exceedance probability of 10-5/a – are shown such that the fundamental safety functions are maintained also in the case of an intensity increase by <u>one</u> intensity level. Effective accident management measures may also be taken into account.

Level 2: Design margins with respect to the earthquake determined plant-specifically according to the state of the art in science and technology – basis: exceedance probability of 10-5/a – are shown such that the fundamental safety functions are maintained also in the case of an intensity increase by <u>two</u> intensity levels. Effective accident management measures may also be taken into account.

Level 3: Earthquakes with an intensity level <u>greater than 2</u> are practically to be excluded at the plant site.

Alternatively: Design margins with respect to the earthquake determined plant-specifically according to the state of the art in science and technology – basis: exceedance probability of 10-5/a – are shown such that the fundamental safety functions in the case of an intensity increase by two intensity level. Effective accident management measures may also be taken into account. This is ensured by available safety systems."

The definition of the robustness levels on the basis of adding fixed values in terms of macroseismic intensity is problematic from a scientific point of view, as the macroseismic intensity scale cannot be considered linear and therefore, the addition of one step in intensity can result in completely different additional loads (i. e. forces, stresses etc.). Depending on the site specific hazard level used for the determination of the design basis, one step in macroseismic intensity will also correspond to different exceedance probabilities. Therefore, the robustness levels for different plants are not comparable: For one plant Level 1 could mean resilience for a beyond design basis earthquake with an exceedance probability of 10^{-6}/a whereas for another plant it could correspond to an exceedance probability of 10^{-7}/a.

Based on the licensee reports in April 2011 the RSK concluded:

"[…] that regarding the seismic design there partly exist considerable safety margins and that the arguments put forward by the licensees in this respect are principally plausible. This judgement is based i.a. on conservative assumptions in the calculation chains and the knowledge gained from the seismic PSAs performed so far for the individual plants. The RSK estimates the potential for safety margins in the magnitude of one intensity level.

It could not be explicitly identified from the documents whether all conditions of low-power and shutdown operation were considered (e.g. flooded reactor cavity during refuelling). The RSK considers a discussion of this topic necessary. It shall add this point to its working programme and deal with the resulting issues.

More recent curves are available for the determination of the probabilities that seismic acceleration loads may be exceeded at concrete sites; these result from a service provided on the Internet by the GFZ German Research Centre for Geosciences in Potsdam. The RSK considers a discussion of this topic necessary. It shall add this point to its working programme and deal with the resulting issues."[1] /2.2/

The robustness levels assigned by RSK give a qualitative indication of the potentially available safety margins. A reasonable quantification of the available margins would require new site specific hazard assessments for beyond design basis earthquakes, i. e. earthquakes with exceedance probabilities smaller than 10^{-5}/a (median) and design re-evaluations for higher loads than assumed in the current design basis. The typical time scale for such assessments is on the order of two to three years.

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[1] With respect to the generic hazard curves mentioned in the last paragraph, there is a widespread consensus in the seismo-engineering community, that such generic curves cannot be applied to NPPs as they do not duly account for the site specific seismicity and the local soil conditions.

2.3 Status related to Earthquakes and statement of the federal regulatory body

2.3.1 Documents available to the federal regulator

- **Progress reports of the licensees**

So far, all licensees have provided a document dealing with the status related to earthquake. Nearly all licensees have addressed almost all points of the ENSREG requirements. However, there are differences in the degree of detail. Some topics are still undergoing internal review. For verification, documentation was referenced explicitly in several reports.

The intensity of the design basis earthquake was determined and shown in licensee reports both deterministically and probabilistically. The existing design basis was assessed in all reports that have been evaluated so far as appropriate and as providing safety margins for beyond-design-basis events. It showed that various postulates are covered by the design basis earthquake and that safety margins exist up to an exceedance probability of 10^{-6} per year.

In most cases, the operating manual and the emergency manual are referred to as the most relevant plant operating rules. Some licensees additionally refer to an emergency response staff manual.

Indirect effects of an earthquake were considered, and most licensees reported that no consequences for coping with the design basis earthquake could be identified. Building structures and components that have no seismic design are not needed for controlling safety functions after an earthquake event.

If offsite power is lost, the available emergency power supplies are challenged. In general, those are designed to withstand the design basis earthquake. In some cases, additional mobile generators are available.

Due to the low intensity of the design basis earthquake (I $\approx$ VI - VIII), the assumption is mostly that the infrastructure can continue to be used after the earthquake. Hence no prevention or delay of the access of personnel and equipment is expected.

To ensure that the conditions of the valid operating licenses remain fulfilled, the licensees mostly refer to the operating manual, recurrent inspections, the inspection manual, and the periodic safety review. In some cases, reference was also made to the respective licensee's own management system as well as to the evaluation of national and international operating experience.

No external mobile equipment is necessary to control the design basis earthquake. One licensee reports that even in beyond-design-basis scenarios that would require accident management measures, no external mobile equipment is necessary. For support in emergency situations, the licensees have taken technical and personnel precautions in the form of the Kerntechnischer Hilfsdienst GmbH, a nuclear emergency support organisation that is available at any time.

Regarding the safety-relevant systems and structures, there are so far no safety-relevant deviations from the licensed conditions known.

Following the accident at the Fukushima nuclear power plant, some licensees have initiated additional reviews. So far, these have revealed that there are no modifications required to increase the robustness of the plants in the event of an earthquake. Other licensees report that this issue is currently being dealt with internally.

The majority of the licensees in Germany rate the earthquake intensity range and the seismic risk as relatively low. Several licensees report that on the basis of the design basis earthquake, the potential for safety margins lies in the range of about one intensity level. In some individual cases, the spent fuel pool was also considered. Here, no significant effects are expected. Other licensees report this issue is currently being dealt with internally.

A few licensees report that at present no weak points or cliff edge effects can be identified. Most other licensees refer to on-going internal review of the issue. This also applies in a similar way to the provisions made to avoid these cliff edge effects or to enhance the robustness of the plants as well as the safety margins for the integrity of the containment system.

The assessment of the earthquake intensity range which the plant can withstand without a loss of containment integrity is still on-going internally at most of the licensees'. The same applies to a beyond-design-basis earthquake and a beyond-design-basis flood caused by it. In a few individual cases, a beyond-design-basis flood caused by an earthquake was assessed to be physically impossible.

2.3.2 Certificates of the *Länder*

The *Länder* authorities have so far reviewed the licensee reports in terms of completeness of responses, the adequate application of the ENSREG methodology and the correct classification of the referenced documentation (if already done by the licensee). In general the adequate application of the ENSREG methodology was confirmed. The design base as referenced and categorized has been confirmed. The review of licensee reports showed some needs for amendments and harmonisation of content. /10-27/

2.3.3 Further work planned for the final report

The licensees identified as the main topics for the final report an update of the chapters on robustness, the provision of information about cliff edge effects and the evaluation of possible plant improvements. Also references and their categorization have to be extended. Additional guidance from the EU level would be useful.

The competent authorities will cooperate to harmonise the degree of detail of the different reports and therefore work on a common structure for their certificates.

3 Flooding

3.1 Generic aspects

All nuclear power plants in Germany were designed to withstand the natural external loads, such as wind and snow. In addition, flooding and earthquakes were taken into account depending on the site specific hazard. For flooding, earthquake and lightning nuclear safety standards are available, whereas the design against other natural hazards is based on conventional civil engineering standards.

Depending on the overall cooling concept for the nuclear power plant, the system design has to comply with requirements important to safety for the cooling water supply. It has been verified for the individual site conditions that the cooling water supply is ensured even under unfavourable conditions, e.g. low water in the river or failure of a river barrage.

Design against flooding:

Since 1982, the requirements for flood protection measures have been specified in nuclear safety standard [KTA 2207] /3.1/, revised in the years 1992 and 2004. Pursuant to this standard, a permanent flood protection has to be provided.

Under special boundary conditions, protection against the difference between the water levels of a flood with an exceedance probability of 10^{-2}/a and the design basis water level of 10^{-4}/a may also be provided by temporary measures.

The sites of the nuclear power plants are mostly located inland at rivers and, in some cases, at estuaries with tidal influences. In most of the cases, sites have been selected which are located sufficiently high. In all other cases, the structures important to safety were sealed for water tightness and were built with waterproof concrete. Furthermore, the openings (e.g. doors) are located above the level of the highest expected flood. If these permanent protective measures should not be sufficient, mobile barriers are available to seal the openings.

Review by the regulatory authority

After the applicant had pre-selected a site, a regional planning procedure was initiated which preceded the nuclear licensing procedure. This took into account all impacts of the individual project on the public, on traffic ways, regional development, landscape protection and nature conservation. Besides the site characteristics, the design of the nuclear installation against external hazards was checked in the nuclear licensing procedure.

Re-evaluation of the site-related factors:

The safety reviews which have to be performed every ten years also include a re-evaluation of the protective measures against external hazards, considering the development of the state of the art. As a result of these reviews, measures have been taken or planned as far as necessary.

The protection against external hazards is assessed on the basis of the Safety Criteria for Nuclear Power Plants /3.2/, the RSK guidelines /3.3/, accident guidelines /3.4/ and the relevant KTA safety standards /3.1/.

The Safety Criteria for Nuclear Power Plants /3.2/ require that all plant components necessary to safely shut down the reactor, to remove residual heat or to prevent uncontrolled release of radioactive material shall be designed to be able to perform their function even in the case of external hazards.

The design requirements specified in the regulatory guidelines /3.4/ for external hazards distinguish between hazards to be treated as design basis accidents and hazards which, on account of their low occurrence probability, are not considered as design basis accidents, and for which measures must be taken to minimise the risk. Accordingly, the external natural hazards (earthquake, flood, external fire, lightning and other natural impacts) are considered as design basis accidents.

In order to standardise the procedure for flood protection, safety standard [KTA 2207] /3.1/ was revised and has been available since November 2004 in an updated version. The changes 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}/a. Since then, the amended safety standard has been applied to all modification licences regarding flood protection.

3.2 Results of the RSK safety review

In the framework of the RSK safety review the safety of the German NPPs was assessed in particular with respect to the external hazards ‘earthquake’ and ‘flooding’. This assessment was mainly focused on the resilience of the NPPs, i. e. the safety margins available for beyond design basis events. Due to this focus, the appropriateness of the design basis itself was not re-evaluated. From a comparison of the site specific hazard and the design of the NPPs as well as additional questions regarding potential effects of beyond design basis events and available accident management measures for such events conclusions on the safety margins were drawn by the RSK. To quantify the resilience of the plants the RSK defined robustness levels. For flooding these robustness levels were defined as follows:

“Basic level: The safety of the plant is demonstrated for a design basis flood (10,000-yearly flood).

Level 1: Design margins with respect to the design basis flood (10,000-yearly flood) determined plant-specifically according to the state of the art in science and technology are shown such that maintaining of the fundamental safety functions is ensured for river sites in the case of a water discharge increased by the factor 1.5 and for tide sites in the case of flood higher than one metre with respect to the de-sign basis flood and in the case of postulated failure of barrages due to a common cause failure of dikes or similar structures. Effective accident management measures may also be taken into account.

Level 2: In addition to Level 1, design margins with respect to the design basis flood (10,000-yearly flood) determined plant-specifically according to the state of the art in science and technology are shown such that maintaining of the fundamental safety functions is ensured for river sites in the case of a water discharge increased by the factor 2.0 and for tide sites in the case of flood higher than two metres with respect to the design basis flood and the resulting water level. Effective accident management measures may also be taken into account.

Level 3: Owing to the topography and the plant design considering the assessment criteria of Level 2, a failure of fundamental safety functions is practically excluded. Temporary measures are not taken into account." /3.2/

The definition of the robustness levels on the basis of adding fixed values to the water level of the design basis flood or multiplying the discharge corresponding to the design basis flood by fixed values is problematic from a scientific point of view for various reasons: (1) Depending on the site specific conditions (e. g. topography, hydrological condition in the catchment basin etc.) already Level 1 (i. e. adding 1 m water level or multiplying the discharge by a factor of 1.5) could lead to unrealistic (physically impossible) assumptions. (2) The safety of a plant w.r.t. floods depends on the water level at the location of the site. The discharge in the river is not directly linked to this water level, because higher discharges could - in extreme cases - lead to lower water levels at the site. E. g., if upstream or downstream of the plant civil flood protection measures fail and retention areas are opened during extreme flood events this could reduce the water level at the site even to values below the design basis flood level. (3) Depending on the site specific hazard level used for the determination of the design basis, the addition of one meter in water level or the multiplication of the discharge by a factor of 1.5 might correspond to completely different exceedance probabilities (c. f. Section 2.2.1 on the results of the RSK safety review for earthquakes).

Based on the licensee reports in April 2011 the RSK concluded:

"As for the fulfilment of the robustness criteria regarding impacts caused by flooding, the assessment by the RSK showed for all plants that there are significant design margins with respect to the 10,000-yearly flood postulated according to the current state of the art in science and technology. The extent of these margins differs from plant to plant. A final judgement of what relevance these differences have is not possible in this first step of the safety review as site-specific conditions for an increase in the volumetric flow rate or a rise in the water level, especially also taking the transgression probabilities into account, are not considered in the criteria.

The accessibility of the premises of several plants is restricted in the case of the water levels considered here. In the case of some plants, their premises will already be flooded if the design flood occurs. The RSK recommends in these cases that the assurance of the safety of the plant during the course of a longer-lasting flood should be reviewed as part of the supervisory procedure.

Owing to a lack of information, the RSK could not consider the protection of canals and the floating resistance of building structures under these increased impacts.

The Biblis A and B plants as well as the Emsland plant are classified by the Reactor Safety Commission as having the highest robustness level (Level 3) due to their topographical location and plant layout. The Isar 2 and Krümmel plants achieve Level 2 in the assessment. The Isar 1 plant reaches Level 1. All other plants can reach Level 1 or higher if corresponding supporting documents are provided. According to the documents presented, the Unterweser plant cannot fulfil the criteria to reach Level 1." /3.2/

The robustness levels assigned by RSK give a qualitative indication of the potentially available safety margins. A reasonable quantification of the available margins would require new site specific hazard assessments for beyond design basis floods, i. e. for floods with exceedance probabilities smaller than 10^{-4}/a, and re-evaluations of the flood protection measures for higher loads (water levels, dynamic forces etc.) than assumed in the current design basis. Due to the limited availability of historic data on water levels and the inhomogeneity of the data sets due to the impact of river regulations, the determination of flood hazard curves covering exceedance probabilities smaller than 10^{-4}/a involves very large uncertainties that do not allow a reliable hazard assessment.

3.3 Status related to flooding and statement of the federal regulatory body

3.3.1 Documents available to the federal regulator

- **Progress reports of the licensees**

So far, all licensees have provided a document dealing with the status related to flooding. Nearly all licensees have addressed almost all points of the ENSREG requirements. However, there are differences in the degree of detail. Some topics are still undergoing internal review. For verification, documentation was referenced explicitly in several reports.

The height of the design basis flood has been given in all reports evaluated so far. Although the licensees refer to different flood levels (ranging from the 100-year flood up to the 100,000-year flood) for the controlling flood event at the site, all licensees have shown that their design basis flood complies with the requirements of KTA 2207 /3.1/, i. e. the 10,000-year flood event is covered by the design. The failure of a dike and the failure of a river barrage as well as precipitation with snowmelt were also considered.

According to most licensees, the appropriateness of the design follows from conservative calculation methods and is confirmed by current analyses, such as the periodic safety review. One licensee reports large safety margins compared with the design.

The licensees have listed the most important structures, systems and components that are necessary for reaching a safe shutdown condition and that are to remain available after the flood. One licensee reports the presence of temporary installations for protecting premises and buildings against flooding.

Several licensees do not require specific provisions for maintaining the water intake function. In some cases, owing to expected arising of dirt, provisions have to be made to ensure the operation of the fine screens and travelling screens in the intake structure.

Most licensees do not require specific provisions for maintaining emergency power supply as the emergency power system is designed against the design basis flood or is located in a position that is safe from flooding.

According to the licensees' reports, permanent and temporary flood protection measures exist. The permanent measures mentioned include backfills which provide that the site is at or above the level of the design basis flood and the construction of dikes.

Temporary protection measures are taken in dependence of the flood level. One licensee also mentions constantly filled auxiliary service water lines in order to avoid floating of un-operated auxiliary service water lines upon flooding.

In most cases, the operating manual and the emergency manual are referred to as the most relevant plant operating rules. Some licensees additionally refer to the in-service inspections (testing manual), emergency response staff manual, and communication exercises.

Secondary effects of floods were also considered: Restrictions of or damage to the external water and electricity supplies have to be reckoned, and the same applies to accessibility, leading up to supplies being possible exclusively from the air. In the case of one plant, it was assumed that as a consequential event of a design basis flood, the plant has to deal with large amounts of flotsam and offsite power would be lost. The consequential events assumed had no effects on the availability of safety functions.

Some licensees need not to presume a loss of offsite power as the external power supply is designed against the design basis flood. One licensee mentions restrictions and possibly loss of the external power supply in the event of flooding.

Accessibility of the plants during a design basis flood is possible in most cases. Access roads to the plant premises are mostly trafficable, transport routes on the plant premises and access to buildings is not significantly restricted. Some licensees reckon with limitations of the accessibility of the plant under certain circumstances. However, it is possible to reach the plant by watercraft or from the air. One licensee reports that although the plant premises are accessible, the surrounding area of the power plant are not.

To ensure that the conditions of the valid operating licenses remain fulfilled, the licensees mostly refer to the operating manual, the emergency manual, the inspection manual, and the periodic safety review. In some cases, references were also made to the respective licensee's own management system as well as to the evaluation of national and international operating experience.

Control of the design basis flood is mostly ensured by design basis measures alone, so that no mobile devices, accident management measures or external equipment are needed. In plants that do use mobile equipment, this mobile equipment is regularly inspected and monitored according to the operating manual/inspection manual. No external equipment is needed.

According to the licensees' reports, there are so far no safety-relevant deviations from the licensed conditions with respect to flooding known.

Following the accident at the Fukushima nuclear power plant, some licensees initiated additional reviews. So far, these have revealed no deviations from the licensing requirements. On this basis, in most cases there are no modifications required to increase the robustness of the plants in the event of a flood. One licensee has initiated measures to further enhance flooding protection in case of a beyond design basis flood. Other licensees report that this issue is currently being dealt with internally.

Re-evaluations of the height of a possible flood have been made. These estimates range from assumptions that there will be no deviation from the design requirements up to the exclusion of an occurrence of flood events exceeding the design basis flood. In some plants, this re-evaluation is still pending.

The advance warning times for additional protection measures are such that sufficient time is available for implementing the measures.

Some licensees report that at present no weak points or cliff edge effects can be identified e.g. (1) since no flooding higher than the design basis flood can occur at the site due to the physical conditions or (2) since even in the event of a 1,000,000-year flood, all buildings protected against flooding – including the safety-relevant systems housed inside them – will remain available. Other licensees refer to on-going internal review of the issue.

The review of the provisions to avoid cliff edge effects or to enhance the robustness of the plant is in most cases still on-going. As a first result, most licensees report that at present, no potential for plant modifications, improvements or additional provisions can be derived. One licensee has applied for an increase in the height of the flooding protection for the buildings of the emergency diesel generator system and the central water chiller, the reactor auxiliary building, the reactor building annulus, and the main-steam and feedwater valve annex.

3.3.2 Certificates of the *Länder*

The *Länder* authorities have so far reviewed the licensee reports in terms of completeness, the adequate application of the ENSREG methodology and the correct classification of the referenced documentation (if already done by the licensee). In general the adequate application of the ENSREG methodology was confirmed. The design base as referenced and categorized has been confirmed. The review of licensee reports showed some needs for amendments and harmonisation of content. /10-27/

3.3.3 Further work planned for the final report

The licensees identified as main topics for the final report an update of the chapters on robustness, the provision of information about cliff edge effects and the evaluation of possible plant improvements. Also references and their categorization have to be extended. Additional guidance from the EU level would be useful.

The competent authorities will cooperate to harmonise the degree of detail of the different reports and therefore work on a common structure for their certificates.

4 Loss of electrical power and loss of ultimate heat sink

4.1 Loss of electrical power

4.1.1 Generic aspects

The generic requirements for the electrical power supply in nuclear power plants in Germany are contained in KTA 3701 /4.1/. According to this safety standard, the plant's own unit generator has to be capable of supplying the safety-relevant consumers („load rejection to auxiliary station supply“ automatic systems have to be available); also, two offsite connections have to exist for electrical power supply from which the electrical power for all trains of the emergency power system can be obtained (main grid connection and standby grid connection). If possible, these two connections should be functionally separate from each other and decoupled with regard to their protection, and they should also be linked either to separate grid system switchgears or to different voltage levels. The connection of the standby grid connection in case of a challenge has to take place automatically. If the above-mentioned supply options are not available, emergency power generators with diesel engines and batteries have to be additionally provided on site to ensure the electrical power supply of the emergency power consumers. Furthermore, there has to be at least one further supply option providing the electrical power required as a minimum to supply one residual-heat removal system train to ensure residual-heat removal (e.g. the emergency power system connection). It has to be possible to add this emergency power system connection manually on demand.

The requirements for the design of the emergency power generators are regulated in KTA 3702 /4.2/. From this, it follows that there has to be a n+2 redundant design. Requirements for the storage of auxiliary and operating materials are also specified. Following an RSK recommendation /4.3/, the requirements for battery capacity were laid down to be greater than 2 hours. As regards the protection of the plant against man-made external hazards, either specially protected additional emergency power generators were installed or existing installations received protective cover.

The emergency power system design in German plants has been described in the German report to the 5th Review Meeting on the Convention on Nuclear Safety in 2011 /4.4/ presenting in particular the following tables:

Table 3: Electric power supply, PWR

Design characteristics	Construction line 1	Construction line 2	Construction line 3	Construction line 4
Number of independent off-site power supplies	At least 3			
Generator circuit breaker	Yes			
Auxiliary station supply in the case of off-site power loss	Yes, load rejection to auxiliary station supply			
Emergency power supply	2 trains with 3 diesels altogether, or 4 trains with 1 diesel each	4 trains with 1 diesel each		
Additional emergency power supply for the control of external impacts	2 trains	1 - 2 trains, unit support system at one double-unit plant	4 trains with 1 diesel each	
Uninterruptible DC power supply	2 x 2 trains	4 trains (except for 1 plant with 2 x 4 trains)	3 x 4 trains	
Protected DC power supply	2 hours			
Separation of trains	Intermeshed emergency power supply, physical separation of the emergency power supply grids	Partially intermeshed emergency power supply, physical separation of the emergency power supply grids	Largely non-intermeshed emergency power supply, physical separation of the emergency power supply grids	

Table 4: Electric power supply, BWR

Design characteristics	Construction line 69	Construction line 72
Number of independent off-site power supplies	At least 3	
Generator circuit breaker	Yes	
Auxiliary station supply in the case of off-site power loss	Yes, load rejection to auxiliary station supply	
Emergency power supply	2 - 6 trains with at least 1 diesel each	5 trains with 1 diesel each
Additional emergency power supply for the control of external impacts	2 - 3 trains with 1 diesel each	1 - 3 trains with 1 diesel each
Uninterruptible DC power supply	2 x 2 trains or 4 x 2 trains	2 x 3 trains
Protected DC power supply	At least 2 hours	
Separation of trains	Partially intermeshed emergency power supply, physical separation of the emergency power supply grids	Largely non-intermeshed emergency power supply, physical separation of the emergency power supply grids

4.1.2 Results of the RSK safety review

On request of the BMU, the RSK examined the robustness of the German plants against the occurrence of a station blackout (SBO) and in the event of a long-lasting (> 2 hours) SBO as part of the plant-specific safety review. To start with, the focus was on power operation as initial condition, and the failure of the power supply systems of the so-called basic level was postulated. A review of this postulate is subject to the supervision procedure.

The basic level was postulated as follows:

The following installations exist to prevent SBO: grid connection, standby grid connection, supply via the unit's own generator, an emergency power generating facility that fulfils the requirements of nuclear safety standards KTA 3701 /4.1/ and 3702 /4.2/, and an additional independent three-phase alternating current supply that will be available in the short term (e.g. assured power supply) or mutual unit support.

For the assessment of robustness, the following assessment criteria were drawn up:

Level 1

Power supply for the necessary safety installations (other than emergency installations) for maintaining of the fundamental safety functions can be ensured by an additional diverse and redundant (at least n +1) emergency power system.

Alternatively:

For the postulated failure of the installations of the basic level, maintaining of the fundamental safety functions can be ensured for at least up to 10 hours with available battery capacities as well as procedural measures to maintain residual-heat removal that correspond to the respective power supply (e.g. steam-driven injection pumps, fire pumps). Accident management measures can be implemented during this period by means to supply sufficient power.

Level 2

In addition to the basic level, there is another diverse supply for emergency power consumers complying with the requirements for safety systems with at least n+1 and that is also protected against rare external hazards (aircraft crash, etc.), e.g. D2-emergency power grid, emergency system.

Level 3

In addition to Level 2, there are battery capacities for at least 10 hours as well as procedural measures to maintain residual-heat removal that correspond to the respective power supply (e.g. steam-driven injection pumps, fire pumps) to ensure the necessary safety functions. Accident management measures can be implemented during this period by means to supply sufficient power supply.

RSK-findings:

“In principle, it can be stated that the electrical power supply of the German nuclear power plants is very robust compared to the majority of the plants world-wide including Fukushima I. All German plants dispose of at least one additional assured supply and several emergency power generators, of which at least two are protected against external hazards.

As concerns a station blackout during low-power and shutdown operation and the effects on the cooling of the fuel assemblies in the fuel pool, there were only isolated licensees's statements available that could be included in the RSK safety review. Both the RSK and the supervisory authorities will follow this up.

Hence, in the assessment of the licensees' answers to the questions relating to the "long-lasting station SBO" by means of the robustness criteria, the RSK concentrated on power operation as initial plant condition.

In summary, the following can be stated for the individual plants:

As for the Biblis A and B, GKN 1, Isar 1 and Krümmel plants, it is considered possible that they can fulfil level 1 pending further verification. This concerns in particular additional verifications to prove the effectiveness of additional grid connections or a cross-connection to the neighbouring unit.

Apart from the D1 diesels (basic level), the Konvoi and pre-Konvoi plants dispose of diverse and redundant D2 emergency diesels for steam generator feeding and for the power supply of further fundamental safety functions. The D2 emergency diesels supplying the emergency feedwater system are protected against external hazards, including an aircraft crash. Hence these plants fulfil the robustness criteria according to level 2.

All other plants fulfil the robustness criteria according to level 2 by way of diverse, redundant emergency diesels supplying the emergency feedwater system or by means of second-level emergency systems for residual-heat removal in combination with an emergency power supply from the neighbouring unit or a further grid connection. In these cases, the protection against external hazards, including an aircraft crash, is also achieved by structural measures or by physical separation of the different emergency power supply installations.

All licensees operating PWRs and BWRs have provided information about battery capacities, measures for core cooling, and accident management measures to re-establish power supply. In most cases, the information about the battery discharge times is so far not sufficient to allow an assessment of whether in the event of a total loss of the three-phase AC power supply, the battery discharge times will be long enough to maintain necessary safety functions over a period longer than10 hours or more in combination with measures.

Regarding the robustness of the design against a loss of offsite power lasting longer than 72 hours, the following can be summarised on the basis of the review so far:

According to the licensees' information, contracts or oral agreements exist with respect to the supply of auxiliary and operating materials. In most cases, no information was provided within the framework of the RSK review about the times of delivery of the auxiliary and operating materials nor about any damage caused by natural external hazards.

According to the licensees, in some cases there are considerable supplies of oil and fuel stored on the plant site. For some plants, operation over several weeks is thereby possible. There is no information available about the protection of these materials against natural external hazards, nor about transport protection.

With a few exceptions, all plants have access to mobile emergency power generators in their surroundings. In these cases, the times until the availability of the mobile emergency power generators lie clearly below 72 hours."

4.1.3 Status related to loss of electrical power and statement of the federal regulatory body

4.1.3.1 Documents available to the federal regulator

- **Progress reports of the licensees**

So far, the licensees have described the design of the plants, as required by the ENSREG specifications. Furthermore, provisions for a long-lasting loss of offsite power with and without external support have been described. In addition, the designs of the plants against a loss of offsite power and an additional loss of the normal standby power supply source, the corresponding battery capacities, and – where appropriate – the times until the occurrence of fuel damage have been described. Also, a large number of the plants have already provided information about the design against a loss of off-site power, an additional loss of the normal standby power supply source, and a failure of other diverse facilities for AC current supply. Some individual subchapters (e.g. time before damage of fuel becomes unavoidable) were interpreted differently by the licensees, leading to certain issues being mentioned in different places. Regarding the description of cliff edge effects and the robustness of the plant, the majority of the licensees refer to the final report. In individual cases, improvement measures are mentioned, like the setting-up of mobile facilities. As regards the classification of documents, many licensees refer to the final report. /1-9/

4.1.3.2 Certificates of the *Länder*

The *Länder* authorities have so far reviewed the licensee reports in terms of completeness of responses, the adequate application of the ENSREG methodology and the correct classification of the referenced documentation (if already done by the licensee). In general the adequate application of the ENSREG methodology was confirmed. The design base as referenced and categorized has been confirmed. The review of licensee reports showed some needs for amendments and harmonisation of content. /10-27/

4.1.3.3 Further work planned for the final report

The licensees identified as main points for the final report an update of the chapters on robustness, the provision of information about cliff edge effects and an evaluation of possible plant improvements. Also references and their categorization have to be extended. Additional guidance from the EU level would be useful.

The competent authorities will cooperate to harmonise the degree of detail of the different reports and therefore work on a common structure for their certificates.

4.2 Loss of ultimate heat sink

4.2.1 Generic aspects

In the German nuclear power plants, the situation regarding the design of the circulating water and auxiliary service water systems differs from site to site. The regulations principally demand a n+2 redundant design for active components of the safety-relevant auxiliary service water systems. So far, there is no requirement in the regulations for a diverse heat sink; nevertheless, for some plants the possibility exists to remove the heat to a heat sink that is independent of the river, such as a well or a multiple-cell cooling tower.

In the case of PWR plants, it has to be taken into account in the event of a loss of the auxiliary service water supply during power operation that there is no challenge of the residual-heat removal chain as steam generator feeding is carried out by corresponding systems. In shutdown condition, the residual heat is removed via a residual-heat removal chain to the river water. The same applies to the heat generated in the fuel pool and the heat loss involved in the operation of safety-relevant components such as diesels and electric motors. The supply units such as pumps, diesel engines and pipes are protected by physical separation or bunkering in such a way that in the event of an external impact, at least one train will remain available for residual-heat removal (emergency residual-heat removal chain). The electric power supply of the emergency residual-heat removal chains comes from the installations of the emergency standby buildings (PWR) or from the emergency diesels that are protected against external hazards (BWR). /4.3/, /4.5/, /4.6/, /4.7/, /4.8/, /4.9/, /4.10/

4.2.2 Results of the RSK safety review

The loss of the auxiliary service water system was also part of the RSK safety review.

The assessment showed that the loss of the auxiliary service water system is controlled in all plants by corresponding accident management measures. The GKN 2, KKE and KKP 2 plants dispose of diverse heat sinks. In the KKB and KKP 1 plants, independent diverse and redundant auxiliary service water system trains are available to maintain the fundamental safety functions.

Not all plants provided information about the postulated loss of the auxiliary service water supply, which is necessary for the assessment of the robustness of the cooling of the fuel assemblies in the fuel pool. This assessment will be followed up within the framework of the licensing procedure. Moreover, one partial aspect of the failure assumptions, namely the complete loss of the circulating water return in areas with CCF potential (e.g. entry of the circulating water return lines into a building), was as a rule not dealt with in the answers presented by the licensees'. The RSK recommends that if there is any CCF potential, corresponding accident management measures should be provided for all operating phases in the plants concerned.

4.2.3 Status related to loss of ultimate heat sink and statement of the federal regulatory body

4.2.3.1 Documents available to the federal regulator

- **Progress reports of the licensees**

So far, the licensees have described the designs of the plants to deal with the loss of the ultimate heat sink and the safety-relevant auxiliary service water system, as required by the ENSREG specifications. Furthermore, the plant designs with regard to a loss of the ultimate heat sink and the safety-relevant auxiliary service water system as well as an alternative heat sink were described. There was also a description of the loss of the ultimate heat sink in combination with a station blackout. Some individual subchapters (e.g. time before damage of fuel damage becomes unavoidable) were interpreted differently by the licensees, leading to certain issues being mentioned in different places. Regarding the description of cliff edge effects and the robustness of the plant, the majority of the licensees refer to the final report. As regards the classification of documents, many licensees refer to the final report. /1-9/

4.2.3.2 **Certificates of the *Länder***

The *Länder* authorities have so far reviewed the licensee reports in terms of completeness of responses, the adequate application of the ENSREG methodology and the correct classification of the referenced documentation (if already done by the licensee). In general the adequate application of the ENSREG methodology was confirmed. The design base as referenced and categorized has been confirmed. The review of licensee reports showed some needs for amendments and harmonisation of content. /10-27/

4.2.4 Further work planned for the final report

The licensees identified as main topics for the final report an update of the chapters on robustness, the provision of information about cliff edge effects and the evaluation of possible plant improvements. Also references and their categorization have to be extended. Additional guidance from the EU level would be useful.

The competent authorities will cooperate to harmonise the degree of detail of the different reports and therefore work on a common structure for their certificates.

5 Severe accident management

5.1 Generic aspects of severe accident management

Following the TMI accident in 1979 and as a consequence of Chernobyl accident in 1986, numerous studies were carried out looking into the feasibility and effectiveness of measures to prevent or limit the consequences of beyond-design-basis events with core melt. The results of the "German Risk Study Nuclear Power Plants – Phase B" (1981-1989) in particular had an important influence on the decisions taken by the RSK, e.g. the first recommendations of the 238th meeting on 23th November 1988 on accident management as well as the comprehensive RSK recommendations of the 273th meeting on 9th December 1992.

The licensees subsequently committed themselves to implement the measures recommended by the RSK in their plants as far as the given conditions allowed. In contrast to many other concepts developed during the 1990s, numerous plant upgrades were performed, also concerning selected preventive and mitigative accident management measures for relevant plant states.

In the context of safety reviews as required today the licensees have to report also on accident management measures. The corresponding guideline specifies a set of beyond design base scenarios to be analysed and covered by the emergency manual. Currently additional Guidance (Severe Accident Management Guidance) is under consideration and already partially implemented in plants.

Table 5 shows the status of implementation of important accident management measures in BWRs and Table 6 the status of implementation of accident management measures in PWRs. /5.1/

Table 5: Implementation of accident management measures in BWRs

5/2010

Measure	KKB	KKI 1	KKP1	KKK	KRB B	KRB C
Emergency manual	●	●	●	●	●	●
Independent injection system	●	●	●	●	☐	☐
Additional injection and refilling of the reactor pressure vessel	●	●	●	●	●	●
Assured containment isolation	●	●	●	●	✓	✓
Diverse pressure limitation for the reactor pressure vessel	●	●	●	●	●	●
Filtered containment venting	●	●	●	●	●	●
Containment inertisation	●	●	●	●	● *	● *
Supply-air filtering for the control room	●	●	●	●	●	●
Emergency power supply from neighbouring plant	☐	☐	●	☐	●	●
Increased capacity of batteries	●	✓	●	●	✓	✓
Restoration of off-site power supply	●	●	●	●	●	●
Additional off-site power supply (underground cable)	●	●	●	●	●	●
Sampling system in the containment	❍	●	●	❍	●	●

* wetwell inerted, drywell equipped with catalytic recombiners

✓ design ● realised through backfitting measures ❍ applied for ☐ not applicable

Table 6: Implementation of accident management measures in PWRs

5/2010

Measure	KWB A	GKN 1	KWB B	KKU	KKG	KWG	KKP 2	KBR	KKI 2	KKE	GKN 2
Emergency manual	●	●	●	●	●	●	●	●	●	●	●
Secondary-side bleed	●	●	●	●	●	●	●	●	●	✓	✓
Secondary-side feed	●	●	●	●	●	●	●	●	●	●	●
Primary-side bleed	●	●	●	●	●	●	●	●	●	●	●
Primary-side feed	●	●	●	●	●	●	✓	●	●	✓	✓
Assured containment isolation	●	●	●	●	●	✓	●	●	●	✓	✓
Filtered containment venting	●	●	●	●	●	●	●	●	●	●	●
Catalytic recombiners to limit hydrogen formation	●	●	●	●	●	●	●	●	●	●	●
Supply-air filtering for the control room	●	●	●	●	●	●	●	●	●	✓	●
Emergency power supply from neighbouring plant	●	●	●	□	□	□	●	□	□	□	●
Sufficient capacity of the batteries	●	●	●	✓	●	✓	●	●	●	●	●
Restoration of off-site power supply	●	●	●	●	●	●	●	●	●	●	✓
Additional off-site power supply (underground cable)	●	●	●	●	●	●	●	●	●	●	●
Sampling system in the containment	❍	●	❍*	●	●	●	●	●	●	●	●

✓ design ● realised through backfitting measures ❍ applied for □ not applicable
* installation of sampling system in revision 2011

- **Justification, concepts and implementation of the plant internal accident management measures and approaches**

Probabilistic Risk Analyses performed since the late 70ies in Germany have highlighted the following two key findings:

- Due to existing safety margins, operating and safety systems for the control of design-basis accidents can provide additional and reliable protection against core melt in many beyond-design-basis events if they are correspondingly considered.
- The operating personnel can contribute essentially to controlling even beyond-design accident sequences and thus to preventing core melt. Accident analyses and risk studies have shown that in many cases, the operating personnel has in general sufficient time available to act appropriately.

Hence the logical conclusion was to evaluate the potential for flexible use of the engineered systems with consideration of their safety margins by the operating personnel in postulated beyond-design-basis events.

In this context, the RSK recommendations of the years 1987 and 1988 comprised:

1. the preparation of an emergency manual[2],
2. measure to ensure core cooling,
3. measures to retain activity and to maintain the integrity of the reactor pressure vessel, and
4. measures to ensure emergency power supply.

In the non-mandatory guidance instruments, a joint recommendation of RSK and SSK on the complex of accident management/emergency preparedness serves as a reference for the "Alarm criteria". In the recommendation[3], the terms pre-alert and disaster alarm are defined as follows

- Pre-alert:
  A pre-alert is triggered if in an event in a nuclear installation there has so far been no or only a slight effect on the environment compared with the criteria for triggering a disaster alarm, but if due to the plant state it cannot be excluded that effects might occur that correspond with the criteria for triggering a disaster alarm.
- Disaster alarm:
  A disaster alarm is triggered if following an accident in a nuclear installation, a hazardous release of radioactive materials into the environment has been ascertained or is threatening to occur.

This also includes the so-called criteria concept, which is subdivided as follows:

- a general dose criterion consisting of
  - general plant criterion (see Table 7) and special plant criterion
  - general release criterion with emission criterion and immission criterion (see Table 8)

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[2] "The emergency manual is a manual that is an open document. New insights can lead to its extension." (Statement of the RSK at the 244th meeting on 24th May 1989)

Table 7: General plant criterion

Measure	Plant state	Alarm level
The fundamental safety objectives are fulfilled.	Safe condition	No alert/alarm
The fundamental safety objectives cannot be fulfilled by the means provided for this purpose by the design.	The plant state is beyond the design condition, i.e. the safety state is not as per design.	Pre-alert
The fundamental objectives cannot be fulfilled even by means of accident management measures.	A plant state prevails where hazardous releases have been ascertained or are threatening to occur.	Disaster alarm

Table 8: General release criteria

Criterion		Pre-alert	Disaster alarm
Immission criterion	In the vicinity of the plant	0.1 mSv/h over several hours	1 mSv/h over several hours
Emission criterion	Nuclide-specific effective dose (iodine, noble gases and suspended mater) and thyroid dose (iodine)	At one tenth of the reference values for disaster alarm	NPP and nuclides specific reference values, which lead to effective doses higher than 10mSv or a thyroid dose of 50mSv

5.2 Results of the RSK safety review

Regarding the accident management measures the Reactor Safety Commission has derived the following review needs for the German nuclear power plants:

"Review of the necessary scope of accident management measures (AMM) and their effectiveness. Here, the extent and the quality of pre-planning for postulated event sequences, such as the non-availability of the cooling chain for cooling of the fuel assemblies in the reactor core as well as in the fuel pool, the non-availability of electricity supply, and any massive fuel assembly damage that may occur up to core meltdown, have to be assessed. Furthermore, a substantial destruction of the infrastructure and inaccessibility due to high local dose rates as well as the availability of personnel also have to be assessed".

Furthermore, the scope of the review has to include:

"Aggravating boundary conditions for the performance of accident management measures, such as non-availability of electricity supply, hydrogen formation and explosion risk, restricted availability of personnel, inaccessibility due to high radiation levels, aggravation of external technical support"

An extensive list of issues was set up and served as a basis for the review procedure. The objective was to clarify,

- to what extent the existing accident management measures are effective even under further-reaching assumptions regarding aggravated boundary conditions caused by external hazards or with respect to failure postulates, and
- to what extent additional accident management measures for a further minimisation of the residual risk might be useful.

The Reactor Safety Commission concludes that the answers of the licensees supplied to the list of questions are presently not sufficient to allow a consistent allocation of the plant-specific AMM to the different levels according to the defined criteria. With respect to the events at Fukushima, following the evaluation of the answers and other information provided, the RSK has therefore derived generic key aspects for further considerations.

The accident management concept should be further developed so as to ensure the effectiveness of the AMM even in the event of external hazards. Here, the following aspects following/during external hazards have to be considered:

- limitations of the accessibility of the power plant area and power plant buildings,
- operability of the AMM,
- availability of the remote shutdown and control station.

The availability of alternating current is a necessary prerequisite for the majority of the AMM by which safety functions can be ensured or re-established. Against this background, the accident management concept should be developed further so that in a postulated SBO the supply of alternating current can be re-established within a plant-specifically determined grace period. From the point of view of the RSK, this includes:

- external-hazard-protected layout of standardised feed points on the outside of the buildings for the supply of the emergency power busbars and, where necessary, of emergency power busbars supplying the emergency feedwater system (interconnectable in the building).
- external-hazard-protected provision of mobile emergency power generators with sufficient capacity for supplying one redundant residual-heat removal train or for recharging batteries.

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

- External-hazard-protected provision of mobile pumps and other injection equipment (hoses, connectors, couplings, etc.) as well as of boric acid, with required grace periods for provision and delivery at the scene.
- Assurance of a water intake that is independent of the receiving water and available even after an external impact (physical separation if necessary).
- Possibilities of injecting water into the steam generators, reactor pressure vessel and the containment (in the latter case also with consideration of higher back-pressures) without the need to enter areas with high risk potential (dose rate, de-

bris load) and to be able to compensate local destruction (e.g. by permanent and physically separated injection paths).

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

The safety margins still available in the beyond-design-basis range have to be identified on the basis of corresponding analyses and can also be used by application of procedures developed on this basis. This should be taken into account in connection with the implementation of the so-called Severe Accident Management Guidelines (SAMG).

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

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

5.3 Status related to Severe Accident Management and statement of the federal regulatory body

5.3.1 Documents available to the federal regulator

- **Progress reports of the licensees**

The licensees have delivered in the progress reports a description of the Severe Accident Management organization and the available measures in their plants. /1-9/ The descriptions follow a uniform structure taking into consideration the ENSREG-Specifications:

The **Severe Accident Management Concept** has been described in detail regarding organisation and management.

The available preventive and mitigative Accident Management Measures (AMM) for **core cooling** and for the protection of the containment integrity have been described in further detail.

The AMM currently in place at the various stages of loss of the core cooling function have been described for the requested situations according to the ENSREG specification i.e. before and after occurrence of fuel damage in the RPV.

Also AMM and plant design features for protecting the integrity of the containment function after occurrence of fuel damage have been described. In German NPP inertisation by nitrogen and /or an autocatalytic recombiners are used to prevent H2 deflagration or detonation as a challenge for the containment. A long-term over-pressurization of the containment is prevented by filtered venting.

Adverse conditions are treated so far in the reports as boundary conditions for the execution of the measure and as far as these conditions are considered in the design of the measure.

In summary it can be stated that the licensees have presented a detailed description of the Severe Accident Management in their plants considering the ENSREG-Specifications. Some licensees make reference to the final report for issues which are in progress. The issue identification of cliff edge effects has not been handled uniformly by the licensees: Some licensees argue that sharp criteria for cliff-edge effects and the time before it (e.g. a specific failure pressure of the containment) can not be determined due to the conservative plant design and due to different applicable accident management measures. In fact, plant components and measures have failure probabilities with different band widths, which do not lead directly into a catastrophic behaviour of the plant. Other licensees argue that the German safety philosophy considered in the plant design excludes a sudden loss of a fundamental safety objective (which results from a cliff edge effect).

Also the issue behaviour of accident management measures under adverse conditions has not been handled uniformly by the licensees: Some licensees evaluate the behaviour considering what the situation could be on site, other one focus the evaluation more on the design of the measures and one licensee argues that without a previous event sequence analysis the failure of an accident management measure can not be stated.

Long-term post-accident activities for accident management measures (ENSREG-Specification) were not yet considered in the licensee progress reports. However, some licensees report about the possible extension of accident management concept in the frame of the development of SAMG (Severe Accident Management Guidance) to further mitigate the consequences of severe accidents.

5.3.2 Certificates of the *Länder*

The *Länder* authorities have so far reviewed the licensee reports in terms of completeness of responses, the adequate application of the ENSREG methodology and the correct classification of the referenced documentation (if already done by the licensee). In general the adequate application of the ENSREG methodology was confirmed. The review of licensee reports showed in some cases needs for amendments of the licensee reports, in particular additional statements to the specified and already installed AM-measures.

5.3.3 Further work planned for the final report

The licensees identified as main points for the final report an update of the chapters on robustness, the provision of information about cliff edge effects and already installed AM-measures and an evaluation of possible plant improvements. Also references and their categorisations have to be extended. The review of licensee reports showed some needs for amendments and harmonisation of content. /10-27/ Additional guidance from the EU level would be useful.

The competent authorities will cooperate to harmonise the degree of detail of the different reports and therefore work on a common structure for their certificates.

6 Consequences of loss of safety functions from any initiating event conceivable at the plant side

In the technical scope of the ENSREG Declaration it is mentioned (page 4) that "Furthermore, the assessment of consequences of loss of safety functions is relevant also if the situation is provoked by indirect initiating events, for instance large disturbance from the electrical power grid impacting AC power distribution systems or forest fire, airplane crash".

In this sense in the national RSK safety review man made hazards have been analysed. Based on this analysis the related procedure, results and insights are summarized.

6.1 Man made hazards

The following man made hazards have been considered in the RSK safety review:

- Aircraft crash
- Gas release including blast wave
- Terrorist attacks including attacks on computer-based controls and systems

In addition to the man made hazards also the effects of an

- impact of an accident in a power plant unit on the neighbouring unit

have been considered.

6.1.1 Catalogue of requirements for plant-specific reviews

The following catalogue of requirements, listed in keywords, was set up for the review. The catalogue refers to the entire reactor complex, including the fuel pools, and covers all operating conditions.

- Topic "aircraft crash"

Review of maintaining of the fundamental safety functions in case of commercial aircraft or military aircraft crash (accidental, deliberate) with consideration of the following aspects:

- Crash scenarios taking into account aircraft type, speed, loading, impact location, etc.
- Structural reserves in case of loads caused by aircraft impact
- Mechanical impacts including impact of wreckage
- Fuel fire effects
- Effectiveness of spatial separation

- Leak as consequential event (e.g. due to induced vibrations)
- Feasibility and effectiveness of accident management measures with consideration of impacts on infrastructure and personnel

- Topic “gas release”

Review of the boundary conditions for the determination of the site-specific impacts caused by toxic and explosive gases and blast wave

- Topic “terrorist attacks”

Review of maintaining of the fundamental safety functions or accident management measures in case of

- Loss of individual infrastructures or buildings (parts thereof)
- Selective local destruction of systems

- Topic “external attacks on computer-based controls and systems”

Review of maintaining of the fundamental safety functions in case of external attacks on computer-based controls and systems

- Topic “impact of an accident in a power plant unit on the neighbouring unit”

Review of the impact of a beyond design basis event in a power plant unit on the neighbouring unit.

6.1.2 Levels of Protection

The following levels of protection have been defined to assess the degree of robustness in case of the man made hazard.

6.1.2.1 Aircraft crash

For the aircraft crash a difference is made between the mechanical impact (impact of the aircraft) and the thermal impact (kerosene fire).

Mechanical Degree of protection 1

Maintaining of the fundamental safety functions in the event of a military aircraft crash (Starfighter type).

Thermal Degree of protection 1

Maintaining of the fundamental safety functions in case of accident-caused kerosene fire due to crash of a military aircraft at least of the Starfighter type.

Mechanical Degree of protection 2

Maintaining of the fundamental safety functions in case of an aircraft crash considering the load-time function according to the RSK guidelines and a load-time function of a medium-size commercial aircraft.

Thermal Degree of protection 2

Maintaining of the fundamental safety functions in case of accident-caused kerosene fire due to crash of a medium-size commercial aircraft.

Mechanical Degree of protection 3

Maintaining of the fundamental safety functions in case of an aircraft crash considering the load-time function according to the RSK guidelines and a load-time function of a large commercial aircraft.

Thermal Degree of protection 3

Maintaining of the fundamental safety functions in case of accident-caused kerosene fire due to crash of a large commercial aircraft.

6.1.2.2 Gas release

The assessment on this topic has been subdivided since different issues are concerned that cannot be dealt with together.

- The blast wave is to be assumed directly at the buildings.
- The release of flammable gases may also have other impacts (e.g. on the service water, power supply installations).
- Toxic gases may have a different profile of detectability and effects.

6.1.2.2.1 Blast wave

Degree of protection 1

With respect to robustness it is ensured that in the event of related impacts, the fundamental safety functions are maintained also if taken into account potential consequential damages and potential impact-related personnel un-availabilities - in accordance with the requirements of the guideline on blast waves issued by the Federal Ministry of the Interior (BMI).

Degree of protection 2

In the event of a blast wave 20% higher (pressure distribution curve) compared to Degree of protection 1, the fundamental safety functions are maintained also if taken into account potential consequential damages and potential impact-related personnel unavailabilities. Further, destruction of the infrastructure has to be taken into account also considering potential consequential damages. Accident management measures may be considered if those are designed against such impacts or can be provided from outside the plant in due time.

Degree of protection 3

In the immediate vicinity and at the plant site, sources for explosive gases - both stationary and temporary with release potential - leading to a failure of the fundamental safety functions is practically excluded.

6.1.2.2.2 Flammable gases

Degree of protection 1

Flammable gases are detected and isolations are installed to maintain the fundamental safety functions.

Degree of protection 2

In addition to Degree of protection 1:
Automatic isolations are installed to maintain the fundamental safety functions.

Degree of protection 3

In addition to Degree of protection 2:
Automatic isolations with a more sophisticated design (e.g. redundant, diverse) are installed to maintain the fundamental safety functions.

6.1.2.2.3 Toxic gases

Degree of protection 1

The toxic gases that might be present in the surrounding area and at the site were determined and plant-specific protection measures are provided. For the identified toxic gases, isolations of supply air for the main control room are installed.

Degree of protection 2

In addition to Degree of protection 1:
Automatic isolations of supply air for the main control room are installed.

6.1.2.3 Terrorist attacks

6.1.2.3.1 Failure of the fundamental safety functions depending on the effort required for destruction

Considering the physical protection measures that are currently in place, the protection measures of the plants against external hazards (blast wave, aircraft crash) also represent at the same time a far-reaching status of protection against terrorist attacks by external intruders. In addition, a wide spectrum of possible destructions of essential system functions through terrorist attacks is covered by the consideration of the effects of postulates concerning the loss of the electricity and coolant supplies.

Within the time-frame set for this safety review, the RSK is not able to perform a robustness assessment of the plants regarding the necessary overcoming of staggered protection measures. Due to the high level of confidentiality regarding physical protection measures, the results of an assessment would only be available to a restricted group of persons.

6.1.2.3.2 External attacks on computer-based controls and systems

At present, no software-based systems are in use in the reactor protection systems of German nuclear power plants.

Software-based systems are partly used in limitation systems and operational systems. Despite the defence-in-depth concept it is therefore necessary to examine the effects of such attacks with regard to the robustness of these systems.

This is currently being done within the supervisory procedures of the *Länder* as a result of the Information Notice issued by GRS.

6.1.2.4 Effects of an accident in one power plant unit on the neighbouring unit

Degree of protection 1

A failure of the fundamental safety functions of the unaffected neighbouring unit is prevented by permanently installed systems and the monitoring of their operation. These systems at least comply with the requirements for emergency systems, and measures are provided that the required operating personnel can perform the necessary activities at least over a period of 1 week in shift operation without violation of the radiation exposure limits for the operating personnel.

Degree of protection 2

A failure of the fundamental safety functions of the unaffected neighbouring unit is prevented by permanently installed systems and the monitoring of their operation. These systems at least comply with the requirements for emergency systems, and measures are provided that the required operating personnel can perform the necessary activities in the long term.

6.2 Results of the RSK Safety Review

- **Procedure of the robustness assessment**

The RSK prepared a "Catalogue of requirements for plant-specific reviews of German nuclear power plants in the light of the events in Fukushima-I (Japan)". To classify the results of the safety review, the RSK defined graded criteria regarding robustness for the review topics mentioned in this catalogue and applied these criteria for the assessment (referred to in the following as assessment criteria).

Such a review of the plants with respect to their behaviour in the event of impacts beyond the design basis and upon postulated un-availabilities of safety system in terms of a safety review is carried out for the first time. The assessment criteria postulated by the Reactor Safety Commission serve solely for a topic-specific differentiation with regard to the existing safety margins and do not represent any regulatory requirements. With the time available, it was not possible to generate these assessment criteria with regard to the quantitative approaches on the basis of scientific limit analyses for this first statement by the Reactor Safety Commission.

Similarly, the different approaches in the assessment criteria could not be systematically reviewed with regard to their consistency with each other nor with regard to their relevance for the existing defence-in-depth concept of the plants. The different assessment aspects will thus always have to be assessed specifically to the particular topic. The RSK considers summarising or compensatory assessments for one plant to be methodically incorrect.

- **Man made hazards**

The assessment criteria for a postulated **aircraft crash** differ in three degrees of protection. Here, a difference is made between the mechanical impact (impact of the aircraft) and the thermal (kerosene fire) degree of protection according to the consideration of the crash of an aircraft comparable to a Starfighter (Degree of Protection 1), the load-time diagram of the RSK Guidelines (Phantom), or the crash of a medium-size commercial aircraft (Degree of Protection 2) and additionally of a large commercial aircraft (Degree of Protection 3).

Consequential mechanical effects due to an aircraft crash that lead to a limited loss of coolant, e.g. leaks in small pipes, have so far not been postulated and could not be assessed within the framework of this review. The RSK will include this in its working programme and deal with the resulting issues.

For all pre-Konvoi and Konvoi PWR plants as well as for the BWR plants KKK and KRB B/C, proof has been furnished that the requirements resulting from the load assumptions according to the RSK Guidelines (Phantom military aircraft) are fulfilled (Degree of Protection 2). As regards the crash of civil aircraft, further proof of its possible control has to be furnished for a confirmation of Degree of Protection 2 and 3.

For the KKU, KKI 1 and GKN 1 plants, the criteria of Degree of Protection 1 are demonstrably fulfilled. To fulfil Degree of Protection 2, further proof is necessary; Degree of Protection 3 cannot be reached on the basis of the documents presented.

Regarding the KWB-A and B, KKB and KKP 1 plants, fulfilment of the mechanical Degree of Protection 1 – for KKB and KKP1 also fulfilment of the thermal Degree of Protection 1 – depends on the presentation of further proof.

Regarding the capacity of withstanding loads from **blast waves,** the assessment by the Reactor Safety Commission shows that the Degree of Protection 1 can be confirmed for all German NPPs, with the exception of the plants mentioned in the following, with regard to the assumed load (pressure according to the BMI Guideline).

As for the adherence to safety margins, there is also confirmatory information in some cases. In other cases, however, no clear statement can be derived from the information provided with respect to the adherence to safety margins. A corresponding review within the framework of this RSK safety review was not possible. The RSK therefore recommends that such reviews should be carried out additionally within the framework of the supervisory procedure.

In the case of the KWB-A, KKP 1, KKI 1 and GKN 1 plants, lower load were assumed, justified by site-specific conditions. Whether the Degree of Protection 1 is fulfilled depends on the presentation of additional proof and its confirmation.

According to the BMI Safety Criteria, the entry of **explosive materials** into the plant has to be prevented. Here, the site-specific boundary conditions have to be taken into account. Having implemented measures to fulfil this requirement, all plants reach Degree of Protection 1. Against the background of the site-specific conditions, however, the plant-specific implementations of these protection measures differ from each other. As regards an isolation of the ventilation system upon a gas alarm, automatic ventilation isolation is implemented in the KBR, KKB, KKE, KWG, KKK and KKU plants (Degree of Protection 2).

The site-specific consideration of **toxic gases** is part of the design concept of German nuclear power plants. Having implemented measures to fulfil this requirement, all plants reach Degree of Protection 1. An automatic detection of such gases in terms of Degree of Protection 2 has not generally been installed; only in the Unterweser nuclear power plant is it planned to install an automatic detection system with resulting automatic ventilation isolation. The RSK considers a discussion of this topic necessary. It shall add this point to its working programme and deal with the resulting issues.

Regarding the **effects of an accident in one power plant unit on the neighbouring unit**, no specific questions were posed by the RSK. Hence there is no information that might be evaluated available on this topic area. Against the background of the experience gained from Fukushima, the RSK recommends that an analysis of this issue should be carried out as part of the supervisory procedure for the twin-unit plants concerned. Based on the postulated damage states of the neighbouring unit (i.a. fires, activity releases, core damage states, core meltdown), this analysis has to examine the consequences and assess the maintaining of the fundamental safety functions of the unaffected unit.

- **Terrorist attacks**

Failure of the fundamental safety functions dependent on the effort required for destruction

Considering the security measures that are currently in place, the protection measures of the plants against external hazards (blast wave, aircraft crash) also represent at the same time a far-reaching status of protection against terrorist attacks by external intruders. In addition, a wide spectrum of possible destructions of essential system functions through terrorist attacks is covered by the consideration of the effects of postulates concerning the loss of the electricity and coolant supplies.

Within the time-frame set for this safety review, the RSK was not able to perform a robustness assessment of the plants regarding the necessary overcoming of staggered protection measures.

External attacks on computer-based controls and systems

At present, no software-based systems are in use in the reactor protection systems of German nuclear power plants.

Software-based systems are partly used in limitation systems and operational systems. Despite the defence-in-depth concept it is therefore necessary to examine the effects of such attacks with regard to the robustness of these systems.

This is currently being done within the supervisory procedures of the *Länder* as a result of the Information Notice issued by GRS.

7 Conclusions

On the basis of the “Declaration of ENSREG”, reviews of the nuclear power plants were initiated in Germany. The topical areas earthquake, flooding, loss of electrical power and loss of ultimate heat sink as well as severe accident management have been addressed. The results have initially been compiled generically for all plants in this progress report. A more detailed plant-specific review will be presented in the national final report. The reviews so far as well as the findings of the RSK safety review show a high level of robustness for the German nuclear power plants with respect to the hazards concerned. However, measures for further enhancement of the robustness of German nuclear power plants are currently under consideration.

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Berichts-Nr.: 101 / 2011 / 08 vom 15.08.2011

/26/ EnBW Kernkraft GmbH
Betreiberbericht für die Europäischen Stresstests
Kernkraftwerk Philippsburg KKP 2
Berichts-Nr.: 102 / 2011 / 08 vom 15.08.2011

/27/ EnBW Kernkraft GmbH
Betreiberbericht für die Europäischen Stresstests
Kernkraftwerk Obrigheim KWO
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/28/ RWE Power AG
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Chapter 2: Earthquake

/2.1/ Kerntechnischer Ausschuss (KTA)
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/2.2/ Sicherheitskriterien für Kernkraftwerke vom 21. Oktober 1977 (BAnz. 1977, Nr. 206)

/2.3/ RSK-Leitlinien für Druckwasserreaktoren
3. Ausgabe vom 14. Oktober 1981 (BAnz. 1982, Nr. 69a) mit den Änderungen: in Abschnitt 21.1 (BAnz. 1984, Nr. 104), in Abschnitt 21.2 (BAnz. 1983, Nr. 106) und in Abschnitt 7 (BAnz. 1996, Nr. 158a) mit Berichtigung (BAnz. 1996, Nr. 214) und den Anhängen vom 25. April 1979 zu Kapitel 4.2 der 2. Ausgabe der RSK-LL vom 24. Januar 1979 (BAnz. 1979, Nr. 167a)
Anhang 1: Auflistung der Systeme und Komponenten, auf die die Rahmenspezifikation Basissicherheit von druckführenden Komponenten anzuwenden ist
Anhang 2: Rahmenspezifikation Basissicherheit; Basissicherheit von druckführenden Komponenten: Behälter, Apparate, Rohrleitungen, Pumpen und Armaturen (ausgenommen: Einbauteile, Bauteile zur Kraftübertragung und druckführende Wandungen < DN 50)

/2.4/ Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV (Störfall-Leitlinien) vom 18. Oktober 1983 (BAnz. 1983, Nr. 245a)

Chapter 3: Flooding

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Anhang 1: Auflistung der Systeme und Komponenten, auf die die Rahmenspezifikation Basissicherheit von druckführenden Komponenten anzuwenden ist
Anhang 2: Rahmenspezifikation Basissicherheit; Basissicherheit von druckführenden Komponenten: Behälter, Apparate, Rohrleitungen, Pumpen und Armaturen (ausgenommen: Einbauteile, Bauteile zur Kraftübertragung und druckführende Wandungen < DN 50)

/3.4/ Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV (Störfall-Leitlinien) vom 18. Oktober 1983 (BAnz. 1983, Nr. 245a)

<u>Chapter 4: Loss of electrical power and loss of ultimate heat sink</u>

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Chapter 5: Severe accident management

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