**Note:**
**This is a translation of Chapter 1 of the RSK statement entitled "Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Kernkraftwerke unter Berücksichtigung der Ereignisse in Fukushima-I (Japan)".**

**In case of discrepancies between the English translation and the German original, the original shall prevail.**

# Plant-specific safety review (RSK-SÜ) of German nuclear power plants in the light of the events in Fukushima-1 (Japan)

(…)

## 1 Summarising assessment and recommendations

In connection with the events in the Japanese Fuhkushima-1 plant, the German Bundestag called upon the German Federal Government on 17-03-2011 to

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*conduct a comprehensive review of the safety requirements for the German nuclear power plants. For this purpose, an independent expert commission is to be tasked with carrying out a new risk analysis of all German nuclear power plants and nuclear installations with consideration of the knowledge available about the events in Japan – especially also with respect to the safety of the cooling systems and the external infrastructure – as well as of other extraordinary damage scenarios;* [1]

…

On 17-03-2011, the Federal Environment Ministry (BMU) called upon the Reactor Safety Commission (RSK) at its 433th meeting to draft a catalogue of requirements for a safety review of the German nuclear power plants and to assess the results of the review carried out on this basis. Here, the insights gained from the accident sequence in Japan are to be considered in particular with respect to whether the current design limits have been defined correctly and how robust the German nuclear power plants are regarding beyond-design-basis events. According to the task given by the BMU, the report by the RSK was to be presented on 15-05-2011.

Within the framework of the plant-specific safety review of German nuclear power plants hereby presented, the RSK has performed a robustness assessment for selected essential aspects. The RSK has not yet carried out a review of to what extent the current design limits have been defined correctly.

### Essential insights gained from the accident sequence in Japan

The Reactor Safety Commission has gained the following provisional insights from the accident in Japan, which affected plants that were in operating as well as plants that were shut down for refuelling and overall maintenance inspection. Here, it has to be stated that until this day, there is not yet full clarity about all

[1] 96th sitting of the German Bundestag on 17-03-2011; motion for a resolution of the CDU/CSU and FDP fractions on the issue of a government policy statement by the Federal Chancellor on the current situation in Japan, printed paper 17/5048

aspects of the accident sequence, the design requirements (application of the Japanese regulations), the method of updating the plants to new levels of knowledge, and the scope and content of accident management procedures at Fukushima I. However, it appears that the following points in particular are important with regard to an assessment of the robustness of a defence-in-depth concept.

The earthquake event in Japan caused damage to the infrastructure and thus also power system failures in wide areas. According what is known so far, the safety systems of the nuclear reactor units at Fukushima I initially maintained their functions to ensure the supply of emergency power and cooling water.

Upon the impact of the tsunami approximately one hour later, the emergency power supply – with the exception of the batteries – as well as the service water system failed; in addition, there was further damage to the infrastructure. According to the information available, this was due to the inadequate design of these plants to withstand tsunami impacts. The tsunami loads led to grave consequences at Fukushima I as important safety systems such as the emergency power generation system and the service water system had not been laid out sufficiently flood-protected. At Fukushima I, the two emergency power generators of each reactor unit are accommodated in the basement of the turbine building, so that when the plant area and the turbine building were flooded, the failure of the emergency power generators was inevitable.

The depressurisation of the reactor coolant systems carried out to allow injection by means of fire pumps was performed clearly too late for a prevention of core damage. With the current level of information, it is not possible to judge whether this was due to inadequate organisational structures, accident management procedures or insufficient numbers of personnel due to the effects of the tsunami or event to other influences. The fact that the depressurisation and injection with fire pumps came too late was then essential for the core damage that occurred at Fukushima I, Units 1 to 3, with the consequence of hydrogen formation and the loss of at least one activity barrier in several units. Several explosions destroyed barrier functions and possibly also further safety installations, contributing to the aggravation of the accident sequence. Regarding the organisation and effectiveness of accident management measures, the destruction of the infrastructure had not been adequately considered.

Obviously, installations and measured to prevent hydrogen explosions in the buildings (venting, recombiners, leaktightness of the systems, barriers) were not effective or did not exist.

The unavailability of the emergency power and service water supplies led furthermore to the loss of cooling of spent fuel assemblies in the fuel pools, with the consequence of further activity releases from fuel assemblies of which some had already been removed from the reactor pressure vessel for a very long time.

**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 unavailabilities of safety system in terms of a stress test is carried out for the first time. The assessment criteria established 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, but they could generally only be postulated.

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 backgrounds will thus always have to be assessed specific to the particular topic. Hence the RSK considers summarising or compensatory assessments to be methodically incorrect.

Moreover, the assessment criteria were prepared for the first time and within a very narrow time frame and were thus not yet available at the start of the review. Due to these circumstances, the catalogues of questions generated at the start are not in all cases in tune with the assessment criteria. This is why at the time of the assessment, licensee's answers were not available with respect to all basement criteria, or the answers did not address the assessment criteria sufficiently.

The RSK was given a large amount of information in heterogeneous form. On the basis of this array of information it was not possible to achieve at this point in time a consistently reliable allocation to the robustness levels or degrees of protection. Hence the present results of the robustness assessment often also include indications regarding the need for further analysis and assessment.

A graduation was applied to the assessment criteria. The higher the safety margins that can be demonstrated against impacts on the plant regarding the fulfilment of the safety objectives, the higher is the degree of robustness. Here, within the framework of the robustness assessment, a differentiation is made between **robustness levels** regarding natural hazards, postulates, precautionary measures and accident management measures and **degrees of protection** for the man-made hazards to be additionally considered according to the RSK Catalogue of Requirements.

The concept of the design of German nuclear power plants is based as a priority on the prevention of events or of any safety-relevant consequences of events. This means that regarding redundancy, diversity and barriers, designs of more recent reactor generations tend to fulfil stricter requirements. This is why the technical realisations in the plants with respect to the robustness below the assessment criteria described here are also different. This is not generally addressed in the assessment.

As a basis for the robustness assessment, the RSK presupposes that the plants correspond to their current licensed condition and that the improvement measures identified as safety-relevant in the safety reviews regularly carried out in accordance with the Atomic Energy Act (AtG) or as a result of other regulatory processes have been implemented and any possible deficits regarding safety demonstrations have been removed. These assumptions also include that preventive and mitigative accident management measures according to the recommendations of the RSK and the state of the art in Germany are implemented and that corresponding procedures are provided in the accident management manual and are regularly exercised. The RSK did not verify as part of this robustness assessment whether these conditions are actually fulfilled. Confirmation of the fulfilment of these conditions is one of the regular tasks of the licensing and supervisory authorities.

As a statement on the robustness of the plants quite substantially also depends on to what extent these conditions are actually fulfilled, the Reactor Safety Commission recommends that the competent supervisory authorities demonstrate the state of implementation in the individual plants.

As regards the assessment criteria, there are generally – specific to each topic – three levels or degrees of protection each defined. The aim is here to query the assurance of the requisite function to avoid "cliff edge" conditions (e.g. with the consequence of massive fuel assembly damage, releases requiring evacuation).

With the differentiated representation of the degrees of robustness, not only deterministic criteria, such as increase of the hazard, diversity and redundancy requirements, but also probabilistic criteria, such as the occurrence frequency of events, are used as far as these represent reliable criteria. At the highest level, i.e. Level 3, a violation of the safety objectives is practically excluded.

The assessment of the robustness of the plants is based on the fulfilment of basic levels defined specifically for each topic. In the case of man-made hazards, degrees of protection were defined for the criteria. The term "degree of protection" was already introduced by the RSK in 2001 for the assessment of safety against the crash of a commercial aircraft. This definition, which differs from the other events/postulates, is also useful since internationally and throughout Europe, man-made hazards are assessed separately, especially taking terrorist hazards into account.

**Assessment**

Considering the information available and the scope of the topics considered, the following can generically be stated for the German nuclear power plants when drawing a direct comparison with the causes and consequences of the accidents at Fukushima I:

Initiating events that may lead to such tsunamis are practically excluded for Germany according to current knowledge. At Fukushima I, the design of the plants was inadequate for a tsunami with an occurrence frequency of approx. $10^{-3}$/a to be considered on the basis of the literature available. In the area of external natural hazards, the effects to be considered according to the state of the art in science and technology in connection with occurrence frequencies of approx. $10^{-3}$/a, especially those that may lead to "cliff edge" effects, are taken into account throughout in the designs of German nuclear power plants.

The electricity supply of the German nuclear power plants is more robust throughout than at Fukushima I. All German plants have at least one additional assured incoming supply and more emergency power generators, with at least two of them being protected against external impacts.

**Natural hazards**

The RSK is of the opinion 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 base i.a. on the conservativities in the calculation chains and the knowledge gained from the seismic PSAs performed so far for the individual plants. The RSK sees the potential for safety margins in the magnitude of one intensity level.

It could not be explicitly seen 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.

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 volume of water flowing 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 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 demonstrations are provided. According to the documents presented, the Unterweser plant cannot fulfil the criteria to reach Level 1.

As **other natural hazards** are largely covered by other external hazards considered and by the consideration of extended postulates with regard to their effects on the safety-relevant building structures and the vital functions, the RSK is of the opinion that the analysis and assessment need not be performed as part of this safety review and is therefore not an object of this statement.

**Postulates**

Accident control and the limitation of the accident consequences at the Japanese Fukushima I nuclear power plant have been considerably hampered by the loss initially of the grid supply and all emergency diesels (Station Blackout – SBO) and later on of the DC voltage supply via the batteries as well as by the long-lasting loss of the service water supply.

In the plant-specific safety review (stress test), the RSK has therefore examined the robustness of the German plants in the event of the occurrence of an SBO or in a long-lasting (> 2 hours) SBO as well as in an assumed loss of the service water supply. It has furthermore examined how robust the plants are in the case of a long-lasting (> 72 hours) loss of offsite power.

In its assessment of the answers of the licensees to the question s relating to a "long-lasting SBO" by means of the robustness criteria, the RSK has confined itself to power operation as initial plant state.

Regarding the Biblis A and B, GKN 1, Isar 1 as well as the Krümmel plant, it is considered possible that these can fulfil the criteria for Levels 1 if further proof is furnished. This concerns especially additional proof to confirm the effectiveness of further grid connections or a cross-connection to the neighbouring unit.

Apart from the D1 diesels (basic level), the Konvoi pre-Konvoi plants have diverse and redundant D2 emergency diesels for steam generator feeding and for the electricity supply needed to maintain further vital functions. The D2 emergency diesels are protected against external impacts, including 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 diverse, redundant emergency diesels or by emergency systems for residual-heat removal in combination with an emergency electricity supply from the neighbouring unit or a further grid connection. The protection against external hazards, including aircraft crash, is also achieved in these cases by structural measures or by physical separation of the various emergency power supply installations.

All licensees of PWR and BWR plants have provided details about battery capacities, process-based measures for core cooling, and emergency measures to re-establish electricity supply. The information about the discharge times of the batteries is so far mostly insufficient to allow an assessment of whether it is possible to maintain vital safety-related functions with their help in combination with process-based measures in the event of a complete loss of the AC power supply over a longer period of time, i.e. for 10 hours and more.

The evaluation of the licensees' answers to the questions relating to the **"long-lasting loss of offsite power"** shows that according to the licensees, written contracts or oral agreements exist on the supply of and operating materials. There are mostly no statements on the delivery times of and operating materials nor on the consideration of damage caused by natural hazards.

The licensees account for sometimes considerable oil and fuel stocks on the plant premises. For some plants, this allows the operation of emergency diesels over several weeks. There is no information about the protection of these materials against natural hazards and about their safe transport. With a few exceptions, all plants have access to mobile emergency power generators in the vicinity of the plant. In these cases, the times until the availability of the mobile emergency power generators lie clearly below 72 hours.

For the postulated **loss of the service water supply**, information needed for the assessment of the robustness of the cooling of the fuel assemblies in the fuel pool is not available throughout. According to the Catalogue of Requirements of the Reactor Safety Commission, these require specific examination, which, however, could not be carried out for this statement out due to the extent of the documents and the time frame.

Also, a partial aspect in connection with the failure assumptions, namely the complete failure of the cooling water return system in areas with CCF potential (e.g. entry of the cooling water return pipes into a building), was generally not covered by the answers provided by the licensees. The RSK recommends that in the case of existing CCF potential, corresponding emergency measures are provided for all operating phases in the plants concerned. In the assessment of the fulfilment of the requirements of Level 1, this aspect was not considered due to the lacking data base.

The plant-specific assessment showed that the loss of the service water supply can be controlled in all plants by corresponding emergency measures (Level 1). The GKN 2, KKE and KKP 2 plants have diverse heat sinks (Level 2). In the KKB and KKP1 plants, autonomous diverse and redundant service water supply trains are available for maintaining vital functions (Level 3).

**Robustness of precautionary measures**

Precautionary measures are understood as measures that for accident analyses are assessed as not failing. If in the robustness assessment their failure cannot be practically excluded, then their failure bears in itself a potential for "cliff edge" effects.

Due to the very specific character of precautionary measures (PM), a specific assessment that is specially suited for to each PM has to be made. In many cases, an assessment of individual PMs by means of the RSK assessment criteria (Levels) on the basis of the information available and in the light of the short time available was not possible. The following statements can therefore only be seen as a first and provisional step of an overall assessment. In the scope of this statement, mainly PMs to prevent flooding were dealt with. In this context, PWR and BWR were assessed separately.

Regarding PWR plants, it was found that flooding in the containment will not lead to a loss of vital functions due to sufficient dimensioning of the volume of the reactor building sump. This means that Level 3 is achieved by all plants.

Flooding in the reactor building annulus of a PWR may lead to the loss of vital functions if the cliff edge level is exceeded. With the exception of the Biblis site, control of this situation by accident management or higher-order measures was not demonstrated. It was not examined to what extent interventions in the flooded areas are necessary for the accident management measures as provided at Biblis.

Owing to the importance of the generic aspects of "flooding of the annulus in PWR plants", the RSK will include an in-depth consideration of this matter in its working programme and deal with the resulting issues.

The questions regarding the other precautionary measures included in the scope of the assessment were answered by the licensees at very different levels of detail. On this basis, a reliable classification of these precautionary measures could either be made only to a limited extent or not at all within the time frame given. Based on a first overview, it can be said that in the event of a failure of the above-mentioned precautionary measures postulated in terms of a robustness assessment, no obviously existing cliff edge effects could be identified.

However, in the opinion of the RSK, the precautionary measures to prevent load crashes in the area of the primary system and the fuel pool, which are also footed on administrative measures, require further in-depth examination with regard to their consequences. It will included this in its working programme and deal with the resulting issues.

Regarding the BWR plants, there are two cases that have to be considered with respect to the PMs to prevent flooding with the potential of a loss of vital functions. The most extensive inflows of water into the reactor building ensue from leaks in the connecting lines of the pressure suppression pool or from leaks in service water system lines (potentially unlimited with operating pumps). In the case of leaks in the connecting lines of the pressure suppression pool, not only the possible consequences of the flooding but also the loss of the pressure suppression pool as heat sink and water reservoir for RPV feeding have to be considered.

Regarding postulated leaks in the service water system, the potentially most extensive inflows of water will be into the reactor building. In the KKB, KRB II and KKP 1 plants, autonomous emergency systems for residual-heat removal are available to maintain vital functions in the event of flooding (Level 2). In the KKI 1 plant, two pumps of the safety system for residual-heat removal are designed against flooding (Level 2). For the KKK plant, further proof is required to show that in a postulated failure of the PMs a leak in the service water system can be controlled by accident management measures (Level 1).

As regards postulated leaks in connecting lines of the pressure suppression pool, the loss of the pressure suppression pool as heat sink and water reservoir for RPV feeding is the most relevant safety-related consequence. In the KKB, KKI-1 and KKP-1 plants, the timely initiation of cooldown operation by manual action is necessary; if this is unsuccessful, vital functions are at risk. In the shortness of time, it was not possible to derive reliable assessments regarding accident management measures that may possibly be available and effective in this case. On the basis of the information available, an allocation to a particular Level is not possible. In the KKK plant, a containment return system (Level 2) is available should the timely initiation of cooldown operation by manual actions be unsuccessful. Only if the former fails as well are vital functions at risk. In the KRB II plant, a pressure suppression pool water level that is sufficient for residual-heat removal operation is ensured by structural (passive) measures (Level 3).

For a range of events (LOCA inside or outside the containment, transients involving a considerable drop in the water level, inadvertent opening of main-steam valves), the accident control concept of a BWR is based on the successful isolation of the main-steam lines.

With the exception of KKI-1, the individual plants have given no details in the documents presented for the robustness assessment regarding the control of leaks or breaks in main-steam lines in the event of a failure of steam line isolation. As far as the RSK is aware, events involving the failure of steam line isolation are not treated in the operating documents (operating manual or accident management manual) of all plants.

Against this background it is currently not possible to confirm the fulfilment of individual levels.

**Aggravating boundary conditions for the implementation of accident management measures (AMM)**

Additional to the existing design of the plants regarding the first three levels of defence of the defence-in-depth concept in German nuclear power plants, possibilities were created with the introduction of accident management measures to prevent any serious consequences for the environment even in the case of beyond-design-basis assumptions and scenarios, so that with these measures, the robustness of the defence-in-depth concept was further enhanced.

The objective of this safety review has been 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 ascertains that the answers 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 t the define 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 three-phase alternating current is a necessary prerequisite for the majority of the AMM by which vital 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 three-phase 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, debris 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 meltdown (maintenance of a sufficient pressure suppression capability at increased pressure suppression pool temperature).

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 planned and currently effected introduction 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
- Measures for the limitation of releases from the fuel pool in BWRs in the postulated event of severe fuel assembly damage, considering possible $H_2$ formation.

**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 included 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) 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 distribution according to the BMI Guideline with a maximum excess pressure of 0.45 bar). 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 for the maintenance of the vital functions of the unaffected unit.

**Terrorist attacks**
**Breach of vital functions in dependence of the effort required for destruction**

Taking the security measures that are currently in place into account, 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.

**Recommendations**

The Reactor Safety Commission has formulated different recommendations within the framework of this "Plant-specific safety review (RSK-SÜ)" The issues addressed in this context are of differing safety relevance. The formulated recommendations make no claims of being complete.

**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 result 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 ensure vital 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.**

**(…)**