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January 11, 1989

POLICY ISSUE
(Information)

SECY-89-004

For: The Commissioners

From: Harold R. Denton, Director
Office of Governmental and Public Affairs

Subject: TRANSMITTAL OF REPORT ON THE SAFETY REVIEW OF NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY

Purpose: To inform the Commissioners of a report by the Reactor Safety Commission on the results of their two-year safety review of all the nuclear power plants in operation or under construction in the Federal Republic of Germany.

Discussion: This review was requested by the Federal Minister of the Interior ("BMI") in June 1986 and later confirmed by the Minister for the Environment, Nature Conservation and Nuclear Safety ("BMU").

A summary of results begins on page 7, some highlights of which include the following:

- Safety Concept: Because large-scale engineered facilities cannot be operated entirely troublefree in spite of high quality design, manufacture, and construction, the concept of safety at several levels is needed. This includes 1) quality design and manufacture as well as good plant management, 2) use of control and limitation equipment to maintain the plant within design limits when malfunctions occur, 3) plant design to withstand design basis accidents (DBA), and 4) establishment of accident management measures.

- Safety-Review: Plants did not reveal any deficiencies which would require immediate action. Backfits have brought older plants up to current levels of safety. Accident management measures, some of which need to be implemented, have increased flexibility to control events far beyond the spectrum of DBAs. General safety assessments for plants should be done every ten years.

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- o Operating Safety: The RSK reports that LWRs in the Federal Republic of Germany had an average availability of about 87% from 1985 to 1987. At an international level, this is a high percentage. Fluctuations were moderate and scrams showed a downward trend. Average radiation exposures were low (less than 4 person-Sv per unit and year) and discharge of radioactive substances during normal operation were well below approved limits.

Two related areas of the report that may be of particular interest to you are the discussions on Unusual Events (page 13) and Procedure for the Evaluation of Safety-Related Operating Experience (page 15).

According to the report, all plants evaluate safety-related operating experience obtained from the plant in question, other plants in the FRG, and foreign plants reported through bilateral agreement information and the incident reports from such organizations as the Nuclear Energy Agency, International Atomic Energy Agency, and the Institute of Nuclear Power Operations. Coordination and active participation of all plant personnel in the evaluation is an important part of the feedback process. The evaluation of operating experience by the supervisory authorities (similar to the NRC) and the Technical Supervisory Inspectorates (TUV) guarantees an independent review of the subsequent corrective measures of the licensees.

Additionally, the RSK found that none of the events involved a danger to the population, the environment or plant personnel as a result of the release of radioactive substances. More specifically, the majority of events did not have any impact on power operation, safety systems were required in only a small number of events and operated as designed, and less than one percent of the events involved an increased discharge of radioactive substances which in no case exceeded the annual limit approved for normal operation.

It is interesting that although the report discussed significant events which occurred at all plants, it does not mention the notable events at Biblis and Stade nuclear power plants that were recently publicized.

Harold R. Denton, Director
Office of Governmental and
Public Affairs

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Ihr Zeichen	Ihr Schreiben	Unser Zeichen	Tel.-Durchwahl	Datum
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Dear Dr. Denton,

the Reactor Safety Commission (RSK) has completed the safety review of all nuclear power plants in the Federal Republic of Germany, which was initiated in 1986.

The results of the safety review and the final report have been published few days ago by the Federal Minister for the Environment, Nature Conservation and Reactor Safety (BMU).

On behalf of the RSK chairman Prof. Dr. Birkhofer and the BMU I am sending for your information one copy of the final report. An english translation of the report is also enclosed.

Sincerely yours,

- Armin Jahns -
Executive Secretary, Reactor Safety Commission

enclosure

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Form 007

Geschäftsstelle der
Reaktor-Sicherheitskommission

Final Report

Results of the Safety Review of Nuclear Power Plants in the Federal Republic of Germany by the RSK

Recommendation by the Reactor Safety Commission (RSK)

November 23, 1988

This is a translation of german original. In case of discrepancies the german text shall prevail.

Contents

A. Objectives, Procedure and Summary of Results

1. Advisory assignment

The Federal Minister of the Interior ("BMI") asked the Reactor Safety Commission ("RSK") in June 1986 to carry out - apart from the analysis and evaluation of the accident at the Chernobyl Nuclear Power Plant - a safety review of all the nuclear power plants in operation or under construction in the Federal Republic of Germany. This assignment was confirmed and further extended by the Minister for the Environment, Nature Conservation and Nuclear Safety ("BMU") who took over responsibility in mid-1986.

The safety review carried out by the RSK consisted of an examination along the lines of the current further development of the engineered safety features of nuclear power plants, considering operating experience as well as new results of both research projects and risk studies.

In this context, it was also examined whether operating experience and findings derived from unusual events would be indicative of possible safety-related improvements in nuclear power plants. Within the scope of the permanent advisory assignment of the RSK, the review also included the question whether, and if so which, measures of accident management will be reasonable, i.e. how the low probability of reactor accidents can be decreased further and how their consequences can be further reduced.

Furthermore, the RSK was asked to submit a suggestion with respect to requirements for future periodic safety reviews of nuclear power plants as a supplement to the established supervision by the authorities.

2. Procedure

In August 1986, the RSK started asking the manufacturers and licensees of nuclear power plants for comprehensive plant-specific information as a basis for its safety review. This request was supplemented further in the course of its discussions. The respective procedure was based on a comprehensive discussion plan.

The discussions concerning the individual nuclear power plants were held within the RSK and its respective committees on the basis of the documents submitted, while hearing both manufacturers and licensees, and with the participation, with respect to a number of topics discussed, of the authorized experts called in for the nuclear licensing and supervisory process. The safety re-

view mainly concerned questions of fundamental importance and is, therefore, not as detailed as the reviews carried out within the scope of the expert assessment under the nuclear licensing and supervisory procedures. Expert opinions prepared on behalf of the state authorities within the scope of the latters' supervision under nuclear law were taken into account as far as their basic statements were concerned, e.g. the safety analysis of Stade Nuclear Power Plant prepared by TÜV Norddeutschland or the expert opinion on the nuclear facilities in the State of North Rhine Westphalia prepared by Elektrowatt Ingenieurunternehmung AG ("EWI"). As far as Unterweser and Biblis A Nuclear Power Plants are concerned, comprehensive safety analyses are under way within the scope of the nuclear supervisory processes of the states in question. The RSK was informed on the respective status during its safety review.

In the course of the safety review, discussion results with respect to topics of both generic and plant-specific importance were approved by the RSK in the form of either comments or recommendations (see compilation in Appendix 1).

Operational questions of the safety review of LWR nuclear power plants were dealt with by an ad hoc working party specially set up for this purpose. On the basis of a list of topics established for pressurized water and boiling water reactors, the operational questions were discussed in the respective plants while at the same time hearing the licensees. Inspection rounds of the plants were a major tool in support of the discussions.

Scope and depth of the safety review of the individual nuclear power plants varied because of the following reasons:

Nuclear power plants which, at the time of the discussion, were still under construction or whose nuclear commissioning had not yet been initiated although their construction was completed:

- Brokdorf (KBR)
- Isar 2 (KKI-2)
- Emsland (KKE)
- Neckarwestheim 2 (GKN-2)

Nuclear power plants which, at the time of the discussion, were in the process of nuclear commissioning or had started nuclear operation shortly beforehand:

- Grohnde (KWG)
- Philippsburg 2 (KKP-2)

- Mülheim-Kärlich (KMK)
- Gundremmingen B and C (KRB B/C)
- Hamm-Uentrop (THTR-300)

With respect to these plants, comprehensive RSK discussions on their safety-related equipment and on operational questions have been held recently prior to the grant of the operating license and were finalized by recommendations. For this reason, the safety review concentrated on questions of accident management. Individual safety-related and operational questions were discussed in addition.

Nuclear power plants which have been in operation for a longer period of time:

- Obrigheim (KWO)
- Stade (KKS)
- Biblis A and B (KWB A/B)
- Neckarwestheim 1 (GKN-1)
- Unterweser (KKU)
- Grafenrheinfeld (KKG)
- Würgassen (KWW)
- Brunsbüttel (KKB)
- Isar 1 (KKI-1)
- Philippsburg 1 (KKP-1)
- Krümmel (KKK)

With respect to these plants, all relevant groups of subjects were discussed again within the scope of the safety review.

The major aspects of the generic statements made in Part B. I. with respect to light water reactors also apply to the THTR. Specific recommendations for the THTR are contained in Part C. Kalkar Nuclear Power Plant (SNR-300) is dealt with in a separate recommendation. The experimental reactor at Karlsruhe with its very low output - the so-called "Kompakte Natriumgekühlte Kernreaktoranlage" (KNK II), or compact sodium-cooled nuclear reactor plant - will be included in the safety review at some later time, as may be other nuclear facilities.

3. Summary of results

- Safety concept

Experience shows that large-scale engineered facilities cannot be operated entirely troublefree in spite of high quality in terms of design, manufacture and construction. Therefore, the decisive thing is to take precautions against malfunctions, incidents and accidents within the scope of a foresighted safety concept. To achieve a

sufficient degree of protection, an in-depth safety concept is employed at several safety levels. This concept is made up of a well-balanced combination of priority measures for the prevention of malfunctions and incidents and measures for their control up to the limitation of the consequences of accidents.

At the operating level (first safety level), design and manufacturing quality as well as the care exercised by plant management contribute to good availability. With a view to safety-related considerations, this is of importance insofar as malfunctions and incidents are thus avoided at the same time.

Nevertheless, malfunctions of components or systems may turn up. Typical examples include component failures such as the failure of a pump in the primary system or also in the feedwater/steam circuit. When such operational malfunctions occur, control and limitation equipment is used to maintain the plant within admissible design limits for specified normal operation (second safety level). This equipment is supported by a utilization of inherent safety properties of the reactor plant. The control and limitation equipment responds in a differentiated way and in accordance with the respective malfunction, e.g. by way of power reduction. When the cause of the malfunction has been eliminated, it is right away possible to continue operation of the plant.

At yet another level, the third safety level, nuclear power plants are also designed to withstand postulated accidents (design basis accidents) as a precaution against damage. The design basis accidents are defined in such a way that each of them is representative of a group of events taking a similar course, i.e. they are the basis for the representative loads for the respective group of events for plant planning purposes. To cope with the design basis accidents, safety systems are installed which are characterized by reliability, redundancy and, to a far-reaching extent, diversity and which fulfill their tasks even if the external power supply fails. The efficiency and reliability of these systems is demonstrated in detail in the licensing procedure. The safety features are also supported by inherent safety properties of the reactor plant, such as negative temperature and/or power coefficients. A meaningful approach in terms of engineering always includes a strengthening of preventive measur in response to the potential improvements identifi in the area described.

In the past few years, the further development of safety engineering has always been oriented to a strengthening of prevention, i.e. the avoidance of serious core

damage. The concept of staggered in-depth safety levels has proven its worth. It is a well-balanced concept which does not require any further extension or modification from the point of view of the RSK.

The procedure for the designing of safety systems leads to an over-dimensioning of components and systems and, as a result of the application of the single-failure criterion, to a redundant system design. When considering things realistically, and when using the safety reserves of components, the systems are considerably more efficient so that they can be applied in a flexible approach in order to cope with events exceeding the design limits (severe accidents). This concept also constitutes the precaution against core meltdown accidents, even if a hypothetically postulated failure of safety systems occurs.

Within the framework of safety studies, reactor safety research and risk studies, the safety potential which exists at nuclear power plants is further investigated systematically and utilized in a selected approach in the form of the derived measures of accident management. The inclusion of accident management creates a fourth safety level, which is independent of the preceding levels. This fourth level permits to prevent serious core damage and to warrant the integrity of the containment even in the event of hypothetical failures of safety systems.

- Safety review

In its safety review, the RSK has been dealing with the safety-related equipment of the individual nuclear power plants. In doing so, it also considered the design basis accidents to be postulated in accordance with today's practice and examined adherence to the protective aims. It included operating experience and the feedback of malfunctions and incidents as well as experience made in other countries. The operational organization in the individual nuclear power plants was another important element. Moreover, the RSK has dealt with events beyond the design basis.

For the evaluation of the results of its safety review, the RSK set up and adhered to the following categories:

- Deficiencies which require immediate action at the plants concerned;

- Indications of improvements as a result of an evaluation of the operating experience relevant for the plant and of the further development of safety engineering;

teristics were determined on the basis of the maintenance documents of nuclear power plants, and surveys were performed with respect to special components such as emergency diesels. All in all, a satisfactory reliability of the components was determined on the basis of this information. The RSK has repeatedly dealt with this question. Inasfar as there was reason to carry out improvements, it has made corresponding suggestions, for example with respect to pilot valves of internal fluid-controlled valves, with respect to the main steam isolation valves of PWRs and with respect to other components. In this context, the RSK underlines the importance of in-service inspections in line with the requirements.

Within the scope of the safety review, the RSK has also been dealing with the reliability of the containment isolation (cf. Appendix 1). As a result of the review, design changes of the containment isolation dampers were carried out in a number of plants.

1.3 Discharge of radioactive substances during normal operation and occupational radiation exposure

With respect to all plants, the discharge of radioactive substances with the exhaust air and with liquid wastes during normal operation was far below the approved values, in most cases by about 2 orders of magnitude, over their entire operating lives. In the course of the last 10 years, the discharge of radioactive substances has further decreased. This is the outcome of a number of improvements which were made, among other things, in the design of fuel elements and in ventilation and waste gas systems.

Occupational radiation exposure is first of all due to work during inspections. The decisive factors in this context are the scope of work on activity-containing systems and the dose rates involved. It is in particular due to backfitting measures on the primary circuit that higher values may occur in individual years. For example, the exchange of pipes in BWRs involved radiation exposures of approx. 20 person-Sv. During the time from 1985 to 1987, the radiation exposures quoted in the following table occurred at light water reactors. What is quoted is the average collective dose of a plant, and both plant personnel and outside personnel are considered. Apart from the averaging over all plants, a difference has been made between plants which have been in operation for some time already (commissioning prior to 1984) and more recent plants.

Year	All Plants No.	All Plants Collective Dose/NPP (Sv)	Commissioned prior to 1984 No.	Commissioned prior to 1984 Collective Dose/NPP (Sv)	Commissioned in 1984 or later No.[a)]	Commissioned in 1984 or later Collective Dose/NPP (Sv)
1985	14	3.5	11	3.8	3	2.4
1986	16	3.0	11	3.8	5	1.4
1987	16	2.9	11	3.5	5	1.4

a) Only plants which have been in operation all year round

In spite of the scope of the nondestructive examinations carried out, the radiation exposures are low. This is due to a number of measures and in particular also to the automation of examinations. Thus, even in the case of the nuclear power plants which have been in operation for a longer period of time, a considerable decrease in occupational radiation exposure was achieved, if compared on a long-term basis, despite longer overall operating lives. The RSK has been dealing with this question for many years. In its yearly audits in connection with the evaluation of the operating reports it has worked towards a lowering of the doses and had quoted a reference value of 4 person-Sv per unit and year in previous discussions. As is shown by the figures quoted, this reference value was not reached if the average of the plants is applied. In individual plants, the reference value was exceeded in the years from 1985 to 1987, among other things as a result of comprehensive work on the primary system. The highest annual value in a plant during this time was 7.55 person-Sv. The lower values for recent plants are not only due to shorter operating lives, but also to the consistent translation, already in the planning stage, of the operating experience gained at other plants.

1.4 Unusual events

The reportable events which the licensee of a nuclear power plant has to report to the supervisory authority in charge are laid down in the reporting criteria for unusual events which are uniform throughout the Federal Republic. Irrespective of the measures taken by the state governments concerned, these unusual events are gathered and evaluated in a central system by GRS on behalf of the BMU.

Within the scope of its permanent advisory assignment, the RSK regularly evaluates unusual events. It finds that none of these events involved a danger to the population in the environment or the plant personnel as a result of radioactive substances. As a result of its discussions with respect to unusual events, the RSK has, in a generic approach, recommended a number of improvements.

For its discussions, the RSK has available the annual reports of the licensees which contain the reports of all unusual events to GRS, and also the reports through the Incident Reporting Systems of the Organisation for Economic Co-Operation and Development / Nuclear Energy Agency (OECD/NEA) and of the International Atomic Energy Agency (IAEA), as well as the transmittal reports prepared by GRS with respect to selected events. Once a year the RSK discusses the operating experience of all the German nuclear power plants on the basis of the annual reports, and in particular the unusual events, and several times a year it discusses a selection of events which also includes relevant events in foreign plants. Current events are discussed according to the prevailing requirements, irrespective of these routine discussions.

The experience with respect to unusual events is summarized as follows by the RSK:

- The by far overwhelming majority of the unusual events did not have any impact on power operation; they were e.g. reportable findings during in-service inspections.
- Safety systems were required in a small number of events, with a typical value of 1 or 2 times per year and plant.
- Insofar as safety systems were required during malfunctions they operated as designed.
- Out of the unusual events, less than 1% involved an increased discharge of radioactive substances. In no case has there been a discharge of radioactive substances as a result of such an event that exceeded the annual limit approved for normal operation. In a total of nine cases, short-term limits for normal operation were exceeded.

As a result of the routine discussions of unusual events by the RSK, a separate evaluation within the scope of the safety review was not necessary. Instead, major emphasis was placed on the information about measures which were recommended and taken on the basis of events,

also with a view to the translation of the experience gathered at other nuclear power plants.

1.5 Procedure for the evaluation of safety-related operating experience

In all plants, organizational measures have been taken in order to evaluate safety-related operating experience from the plant in question and from other plants. The information which the licensees have at their disposal with resp to the experience gathered in other plants include:

Experience from the Federal Republic of Germany:

- Event reports, transmittal reports and other reports distributed by GRS.
- Reports distributed within the scope of the "Technische Vereinigung der Großkraftwerkbetreiber" (VGB).
- Reports distributed by the manufacturers.
- Exchange of experience in the corresponding VGB organs.

Experience obtained from foreign countries:

- Incident reports within the Incident Reporting Systems of the Organisation for Economic Co-Operation and Development / Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA) as well as the related GRS evaluations.
- Reports within the scope of the Institute of Nuclear Power Operations (INPO) and USERS.
- Information as a result of bilateral agreements.

The evaluations are handled differently at the various plants. However, in all cases the professional preparation is made in the departments responsible for the technical field concerned. The reports distributed by GRS with respect to domestic and foreign events are also available to the supervisory authorities and their authorized experts. The supervisory authorities ask their authorized experts and the licensees to submit comments on a case-by-case basis. In many cases, the licensee is also obliged to submit comments with respect to its own plant in a routine procedure in response to all events occurring at German nuclear power plants.

The RSK is of the opinion that the relevant German operating experience is covered by the sources of information

referred to. Similarly, the Incident Reporting Systems of OECD and IAEA and the reports by INPO and USERS cover information on foreign experience. As far as the reports of the Incident Reporting Systems are concerned, a preliminary evaluation of their applicability to German plants and a preliminary selection of the reports which are relevant for German conditions, including the preparation of additional information, are effected centrally by GRS on behalf of the BMU. The RSK found that a systematic evaluation of the reports by INPO and USERS is more difficult, as there is no corresponding service. It holds that the evaluation at the individual plants should be supported by a generic preliminary selection and preliminary discussion of the reports by INPO and USERS. It awaits corresponding suggestions to be made by the licensees. Furthermore it points out that the evaluation of operating experience beyond the events which have to be reported, remains one of the major tasks of licensees.

That the reports are dealt with by the respective departments at the nuclear power plants is an adequate procedure, since it is there that the importance of a fact and the consequences to be drawn, if any, can be judged best. The organization of handling shall be as dictated by the circumstances prevailing at the respective plant. The RSK recommends to make sure that all the departments concerned, and in particular the operating personnel, are called in and that further action is coordinated. The RSK underlines that, irrespective of the form of organization chosen, the interdepartmental follow-up of the evaluation requires a sufficient number of staff disposing of solid knowledge of the plant.

The active participation of plant personnel in the licensee bodies for an exchange of experience is an important part of the feedback of experience. The RSK welcomes these activities on the side of the licensees.

The evaluation of the reports on operating experience by supervisory authorities and Technical Supervisory Inspectorates guarantees an independent review of the measures taken by the licensees.

The RSK will continue to deal with the question of the evaluation of operating experience and, in doing so, will also treat the contents of the reporting criteria and the classification of reportable events in terms of further improvement.

It recommends to collect, in a central and independent approach for the entire territory of the Federal Republic, not only the events, but also the remedial measures taken.

2. Operating and incident instructions, training issues of the plant personnel

2.1 Operating and incident instructions in the operating manual

With respect to more recent plants, the scope for the safe operation of a nuclear power plant is laid down in the safety specifications. For older plants, comparable provisions exist. These instructions are part of the operating manual.

It is the licensee's responsibility to see to it that the plant is operated in accordance with these provisions and that the highest possible reliability of the safety system is ensured. The organizational and administrative precautions for the performance of reviews and tests are also laid down. Should there be special occasions for reviews or tests which necessitate certain modes of operations which have to deviate from these provisions, the consent of the supervisory authority shall be obtained.

In the incident chapter [1], the operating manual contains the description of the automatic measures in the short-term phase as well as further instructions for the plant personnel to enable it to cope with the incident. For design basis accidents, there are so-called event-oriented instructions each of which is tailored to a group of event sequences characterized by a design basis accident. Possible variations within the event sequence are taken into consideration by ramifications within an instruction or by more than one instruction (e.g. SG heating tube rupture). The application of these instructions presupposes that the type of incident is identified by the plant personnel. For this purpose, the incident chapters contain a compilation of the identification criteria for the various types of incidents.

These event-oriented instructions are supplemented by state-oriented (also referred to as protective aim or symptom-oriented) instructions. They do not require any knowledge of the type of incident (event) that has occurred in order to cope with the incident but are oriented to the observed state of the plant and are exclusively directed towards the protective aims. The superordinate protective aims to be quoted include the assurance of subcriticality, sufficient core cooling and the safe confinement of the radioactive substances. For each protective aim, the operating manual quotes the parameters which can be used to check whether the aim is adhered to, as well

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[1] This chapter covers design basis accidents

as the measures which are provided for its adherence and for coping with the incident.

With the exception of the KMK plant, the operating manuals currently contain the event-oriented incident instructions. The state-oriented supplements are already available at a number of plants whereas they are still being prepared with respect to others. The KMK incident chapter is mainly characterized by a state-oriented structure. Event-oriented descriptions are also provided as supplements for the design basis accidents. These documents are available to the control room personnel and serve above all training purposes.

The introduction of an additional state-oriented chapter as a supplement ot the event-oriented instructions corresponds to the RSK recommendations. In this way, actions are also made available for cases where the personnel does not succeed in identifying an incident or where an incident takes an unexpected course. In part, this covers also event sequences which were not postulated for the design of the plant. Furthermore, personnel training is supported by making the safety-related background of the measures to be taken in accordance with the respective event more transparent.

The RSK has repeatedly discussed the measures and instructions for coping with incidents, and in particular the state-oriented supplement of the operating manual for pressurized water reactors as well as the state-oriented procedure for the KMK. The discussions of the corresponding concept for BWRs are being continued. A detailed review of the incident instructions would have been beyond the scope of the safety review. The RSK assumes that this is carried out by the supervisory authorities and authorized experts. It takes it for granted that the experience derived from actual events is taken into consideration in the operating manuals.

Accident management measures which can be taken against events exceeding the design basis up to serious hypothetical accidents are described in a separate emergency manual and not in the operating manual. This is further discussed in Section 9.2.

2.2 Simulator training

In the past, questions of the training of plant personnel have been treated in detail, e.g. by way of the suggestions of the RSK when laying down the training contents or the information on the level of training at the time of the commissioning of nuclear power plants. Within the scope of the safety review, the RSK has mainly been dealing with the scope of the simulator training.

Training on the simulator is part of the plant-specific training of the responsible shift personnel. Among other things, simulator training has confirmed its necessity as a result of unusual events where human failure played an essential role.

Both power plant engineering and simulator engineering are in a process of continuous development. The simulators which are in operation are backfitted in accordance with the current requirements. New simulators were made available for a number of plants.

With respect to the scope of simulation it can be said that the PWR simulators are practically capable of simulating their reference plants 100%. They cover all the major parts, systems and components of the plant (e.g. the nuclear steam generation system and the auxiliary, ancillary and supply systems).

Moreover, all major operating states and operating functions (such as burnup states, startup phases) can be simulated. Anomalous operation and incidents are taken into consideration by more than 150 basic events. In addition, numerous combinations of malfunctions and incidents may be simulated. As far as the scope of simulation is concerned, it can be said with respect to the PWR plants that the full-scale simulators used in the Federal Republic of Germany correspond to the international state of the art in terms of control room and modelling engineering.

The layout of the training concept is such that in spite of partially different plant data and plant engineering as compared to the reference plants, the learning aims can be achieved by substitute measures, if necessary. A plant-specific simulator is available for the plant at Mülheim-Kärlich.

As far as the BWR plants are concerned it can be said that the scope of simulation of the BWR-1 Simulator is approximately comparable to that of the PWR-1 Simulator used for the older German PWRs. The contract for a plant-specific simulator (BWR-2) for the KRB-II twin unit plant has already been awarded.

As a result of its discussions within the scope of the safety review, the RSK arrives at the following statements with respect to simulator training in the Federal Republic of Germany:

- The RSK is of the opinion that a continuous further development of the simulation quality is necessary for both PWR and BWR simulators.

- Another major aim of future activities should be the treatment of events beyond the design basis. In this context, the preventive measures dealt with in the emergency manuals (cf. Section 9.2) should be taken into consideration. A decision on the extent of participation of the internal plant emergency staff in emergency excercises which also include the simulator must be made in due time.

- The program of simulator training should be followed up by an independent authorized expert.

- Substitute training events should be organized in particular if the full extent of the training objective for a specific plant cannot be achieved with the available simulators (e.g. because of plant-specific features).

- During a first-time training on the simulator, about 50% malfunction/incident situations should be dealt with as a rule. During retraining measures, the portion of malfunction/incident training should clearly prevail. In order to use the time spent on simulator training as effectively as possible, a preparation in the plant which should be in keeping with the intended simulator training is necessary and appropriate.

In its further discussions of the training of personnel, the RSK will again deal with the duration of retraining. For this purpose, it will also deal further with the question of simulator use, and in particular with the appropriateness of specific simulators at the nuclear power site in question.

3. Information in the control room

In connection with the safety review, the RSK asked the licensees to submit plant-specific inventories with respect to the information which is available in the control room and/or at the emergency control room and with respect to the equipment of the control room and/or emergency control room areas with a view to longer stays of personnel.

In nuclear power plants in the Federal Republic of Germany, the control room is, as a rule, allocated to the switchgear building. Because of its comprehensive surveillance and intervention possibilities, the control room is the central location at which accident management measures can be taken in the case of events beyond the design basis in order to keep the plant in a safe condition or transfer it into a safe condition (cf. Section 9).

As a matter of principle, there is a physically separated emergency control room which, as a rule, is protected against external impacts and is provided with an independent power supply. This station is used to keep the plant in a safe condition if a control room failure occurs.

For the ergonomic design of the control room, the control room personnel was, in many cases, included in the development. The RSK will continue to deal with this subject, considering the development of the technology of information display and information processing.

The RSK points out that, as far as a few older plants are concerned, improvements should be made within the area of incident and wide-range instrumentation in accordance with the requirements of the RSK Guidelines and KTA 3502 (Incident and Wide Range Instrumentation) within a reasonable period of time. In individual plants, improvements have already been planned and/or implemented.

In how far the existing control room instrumentation is suitable for measures of accident management will have to be dealt with when considering the individual measures.

4. Quality of pressure-retaining components and systems including containment and pressure suppression system

4.1 Quality status

The requirements with respect to the quality of the pressure boundary and of the outer systems are laid down in the RSK Guidelines for Pressurized Water Reactors, including the General Specification "Basic Safety". The same requirements are made with re ect to the components of boiling water reactors.

Within the scope of its discussions accompanying the commissioning of new plants, the RSK satisfied itself in each individual case of the adherence to the requirements for pressure-retaining components and systems as contained in the Guidelines. With respect to the plants constructed after 1979, the requirements of basic safety are met.

In the older plants, numerous safety-related pressureretaining components and systems have been replaced and substituted by components and systems of a higher quality in terms of design, materials and processing which fulfill the basic safety requirements. The necessary backfitting measures were carried out either already during the con-

struction phase or during inspection outages. Examples include vessels of the outer systems of PWRs and BWRs, the exchange of containment plates and measures taken with respect to the pressure suppression system and pipes containing reactor coolant, above all in the case of BWRs (also see Section 8).

With respect to components and systems which were not replaced, the results of the BMFT research and development projects for component safety and the results of the investigations initiated by the BMU showed that the safety level with respect to the required plant monitoring and the in-service inspections is ensured.

Moreover, the refitting and backfitting measures led to a considerable improvement in the preconditions for the successful application of nondestructive examinations. This applies above all to the extension of accessibility, also for the use of manipulators and for the processing of surfaces.

In all other respects, the in-service inspections so far carried out with regard to the pressure-retaining walls of the reactor pressure vessels and the reactor coolant pipes of the PWRs and refitted BWRs have so far not indicated any operation-related damage.

4.2 Maintenance of the quality of components and systems during operation

The pressure-retaining components are designed for the safe absorption of mechanical and thermal stresses, also considering neutron irradiation as well as operating and incident conditions, including anticipated transients, for their entire operating lives.

The input data of fatigue analyses are checked by representative comprehensive measurements of operating stresses.

The demonstration concerning adherence to the fatigue conditions has to be made on the basis of actual operating conditions. If necessary design improvements or an adjustment of the operating mode are made. In the reactor pressure vessels of all plants, the neutron fluence was and/or is monitored, among other things, by suspended accelerated irradiation capsules of the materials. It is thus ensured that the safety margin against component failure is questioned at no time.

With respect to the influence of neutron irradiation on the wall of the reactor pressure vessel it has to be stated that the investigations of accelerated irradiation capsules carried out so far, together with the results

of research projects, have confirmed the decrease in toughness postulated in the safety analyses, with the exceptions mentioned in the plant-specific part of this recommendation.

The integrity of the steam generator heating tubes is ensured by a suitable mode of operation and monitored by means of nondestructive examinations performed during in-service inspections. The Incoloy 800 material used in the U tube steam generators of German PWR plants has been found to be, to a far-reaching extent, insusceptible to operational damage caused by stress corrosion cracking. As a result of the wall thickness wastage identified during the nondestructive examinations at the beginning, measures were taken by which it was possible to mitigate the deposition of corrosionpromoting wastage in the secondary chamber above the tube plate. In particular, the water quality was improved at most of the plants by discontinuing the use of phosphates. The procedures preferred for the non-destructive examination, and in particular the multifrequency eddy current process, have proven their worth as reliable testing methods.

The RSK will continue to deal with the questions involved in the influences of neutron irradiation and, in doing so, concentrate above all on the anisotropy of material properties and the spectral distribution of the neutron energy.

For the detection of corrosion and erosion influences, monitoring measures are taken, including nondestructive examinations. In this context, no inadmissible corrosion-supported damage has so far been identified on the large components of the pressure boundary, and in particular on the pressure vessels and the reactor coolant pipes (PWRs, BWRs), which meet the requirements of the RSK Guidelines and of the other engineering codes and guides, or which have been upgraded for adjustment to the stipulations laid down in these codes and guides.

The corrosion-supported cracks identified on a few other components which had not been refitted gave rise to the far-reaching elimination of the preconditions for the occurrence of strain-induced stress corrosion cracking. These include

- further refitting measures for adjustment to the conditions of basic safety;

- reduction of the amount and frequency of thermal loads (optimization of the operating mode);

- improvement of the water quality (in particular adherence to a low oxygen content);

- aimed selection of the locations to be examined during in-service inspections as a result of the loading collective and the existing damage.

Besides, the RSK constantly deals with the questions of the influences of corrosion on pressure-retaining components of light water reactors and their internals.

On the basis of the previous experience of many years of operation and the results of selected monitoring measures, improvements have been implemented against the damage due to erosive corrosion in the secondary circuit. Countermeasures concerned e.g. the use of chromium-alloyed ferritic or austenitic materials, the alkalinity of the coolant in the secondary circuit of pressurized water reactors, the prevention of high local flow speeds and pronounced flow disturbances.

The tightness of the valves between drywell and wetwell of BWRs is also reviewed within the scope of the in-service inspections. Inadmissible values did not result in any one case. The RSK requests the submission of analysis results with respect to the development of the differential pressure between drywell and wetwell in the course of a loss-of-coolant accident for the individual BWRs. The review has left no doubt that the closed position of the valves in the early phase of a loss-of-coolant accident is ensured in any case.

5. Safety systems

5.1 Electric power supply

At the nuclear power plants in the Federal Republic of Germany, the electric power supply to the safety-related equipment during normal operation is effected from the auxiliary power supply of the power plant, which is in turn supplied from its own unit generator or the connected power grid. At all nuclear power plants, at least two decoupled grid connections are available for supply from the grid. In addition, most of the nuclear power plants dispose of another grid connection which can be used to get emergency power. For the remaining nuclear power plants, the refitting of the further grid connection is being planned or implemented at present. Should the supply from the auxiliary power supply system fail, the emergency diesels of the emergency power system with its redundant layout will assume power supply to the equipment that is important in terms of safety.

- Grid connection

Operating experience shows that the connections to the high-voltage grid (110 kV to 380 kV) constitute a reli-

able potential supply for the important safety-related equipment of nuclear power plants. This reliability is achieved in particular by means of engineered and organizational precautions for the control of grid disturbances and for the restoration of the grid following a regional or supraregional failure of the grid.

The grids of the interconnected partner organizations are characterized by a sturdy design. They are remotely controlled to a far-reaching extent and are monitored automatically. Faulty means of operation and faulty grid sections can be isolated in a selective approach and bypassed for power transmission purposes. In the case of an imminent grid breakdown, loads will be separated from the grid under a five-stage plan in order to be able to continue the operation of the grid in the area of the nuclear power plants. For the restoration of the grid, a sufficient number of power plants are available which can be started up without power supply from the grid (pumped-storage power plants, gas-turbine power plants). The sites of these power plants are widely distributed geographically so that power transmission to the nuclear power plant will be possible via at least one route.

In the case of disturbances, the protection of the auxiliary power supply of the nuclear power plants is given top priority. The necessary measures are laid down. In addition, there are specifications for the interconnected partners in the Federal Republic of Germany and in Western Europe for the control and limitation of disturbances in the interconnected grid and the restoration of the grid following a grid breakdown.

As a result of their precautions and investigations the licensees have demonstrated that the nuclear power plants can be supplied again with auxiliary power from the grid within one to two hours following a large-scale failure of the grid caused by an electric disturbance.

As far as mechanical damage in the area of the overhead transmission routes is concerned, the RSK requests that in the case of a steel tower collapse and the postulated consequential damage, at least the further connection to the grid will remain intact so that this can be used by the respective nuclear power plant to get power in order to supply the necessary emergency power loads.

For this reason, the RSK considers a two-hour planning value for accident management measures as realistic and sufficient, measured from the beginning of the failure of supply until the restoration of supply to the nuclear power plants from the grid.

- Auxiliary power station

During normal operation, feeding into the emergency power system will be from the auxiliary power station. In the case of a failure of supply from the auxiliary power station, the connections to the emergency power system are opened, and the autonomous emergency power diesel generator of the emergency power system will assume supply to the emergency power loads. In a few older nuclear power plants, there is no consistent physical separation in the area of the auxiliary power stations, partly only at certain voltage levels. To the extent this has not yet been done, the RSK recommends to check at these nuclear power plants and/or to make sure by means of additional measures that even in the case of an internal failure-initiating event in the area of the auxiliary power station (e.g. fire) a short-term substitution of the emergency power diesel generators by some other kind of supply (e.g. grid or adjacent unit) remains a possibility.

Moreover, the RSK recommends - to the extent this has not yet been implemented - that a chronologically graduated addition of loads be provided for the long-term changeover inside the auxiliary power system, or that a demonstration be submitted showing that the admissible voltages for the loads are adhered to in changeover processes.

- Emergency power system

In the Federal Republic of Germany, each unit of a nuclear power plant has its own autonomous emergency power system. At the individual nuclear power plants, the design (power capability, number of trains) of the emergency power systems basically corresponds to the design of the process systems to be supplied.

At some older nuclear power plants, several trains of the emergency power system are not characterized by a consistent physical separation. The RSK recommends that - to the extent this has not yet been implemented - it should be checked, while considering the respective emergency system, which consequences an internal failure-initiating event (e.g. fire) can have with respect to the plant (process engineering, instrumentation, displays in the control room). It must be made sure that in case such an event occurs not only the superordinate aims of protection (shutdown, residual heat removal, long-

term subcriticality) are fulfilled, but also that the plant condition is displayed sufficiently.

In addition, the RSK recommends that - to the extent this has not yet been implemented - emergency power operation be initiated via two diverse activation criteria (as a rule, voltage and frequency) and that overvoltage protection for direct voltage equipment be protected with diode-decoupled load feeding against single failures.

Moreover, the RSK will continue to deal with the admissible periods of time for maintenance and servicing procedures in the emergency power system.

5.2 Instrumentation and control equipment of the safety system

The instrumentation and control system of the safety system comprises the pickup and processing of measured values, the logic and evaluation circuits as well as the control of the active safety features. It monitors and evaluates the process variables which are of importance for the safety of a nuclear power plant and its environment and controls the active safety features in the case of malfunctions and incidents in order to maintain the state of the reactor facility within safe limits.

In all nuclear power plants, the instrumentation and control equipment of the safety system is designed redundantly and is reviewed at regular intervals. Important events are discussed within the RSK. All in all, operating experience with the instrumentation and control equipment of the safety system is good.

The instrumentation and control equipment of the safety system is designed in accordance with the requirements derived from the incident analysis. It consists of several subsystems which are redundant to each other and which are, in general, separated physically and electrically. The demonstration of suitability is furnished by type and qualification tests, by operation and by in-service inspections at periodic intervals. In this context, plant-specific features with respect to electrical, mechanical and process impacts on the instrumentation and control equipment of the safety system are taken into consideration.

The RSK attaches great importance to the independence (electrical and physical separation) of the subsystems of the instrumentation and control equipment of the safety system, including their decoupling from the operational instrumentation and control equipment. For this area, and within the scope of the safety review, it recommen-

ded investigations by authorized experts at the older nuclear power plants. The results of these investigations have not yet been made available completely. The RSK will continue to deal with this subject.

In addition, the RSK will deal with the instrumentation and control equipment of the safety system with a view to sufficient insensitivity (sturdiness) with respect to environmental and disturbing influences.

The RSK points out that the licensees are required to make special efforts to maintain the specific competence for servicing the instrumentation and control equipment of their plants.

5.3 Incident resistance of electrical equipment

The electrical equipment which is necessary for the operability of the safety system during incidents must perform its function also during the incident conditions then prevailing. This electrical equipment is designed in accordance with the state of the art at the time of the construction of the respective nuclear power plant. Electrical equipment was adjusted to the updated state of the art, also with respect to its capability of withstand incidents, when modifications and backfitting measures were carried out.

Within the scope of its safety review, the RSK recommended that differences which may still exist at individual nuclear power plants as compared with modern plants be identified and described in investigations carried out by authorized experts. For some nuclear power plants, opinions by authorized experts are already available. In addition, the licensees have prepared and submitted reports on the capability of their plants to withstand incidents.

In general, these reports show that there is a sufficient incident resistance of the electrical equipment. For a further support - in particular of the useful life of the incident-resistant components - the RSK recommends a systematic review of the incident-resistance of the electrical equipment of the safety system and the incident instrumentation with a view to the incident conditions to be postulated, the selection of components and circuits to be designed in an incident-resistant way, and the test requirements for the demonstration of incident-resistance over their useful lives in nuclear power plants.

The RSK recommends to lay down, as a result of these investigations, whether and which groups of components

should be subjected to practical tests on a random sample basis in order to establish their resistance to incidents.

An essential precondition for the maintenance of the incident-resistance of components is their suitable and sturdy design. The RSK recommends that the suitability and sturdiness of the respective design be evaluated as well when the incident-resistance is reviewed.

5.4 Reactor scram system

The reactor scram system of the PWR uses control elements which are kept in the withdrawn position, i.e. at the upper edge of the reactor core, by means of electromagnets during power operation of the reactor. If a reactor scram is initiated, the power supply to the holding magnets is interrupted and the control elements, as a result of their dead weight, drop into the core. The effect of the scram system is fail-safe, i.e. it reaches the safe state in the case of a power failure.

In the case of the BWR, the reactor scram is effected by a hydraulic insertion of the control rods. The required pressure is generated by means of nitrogen blankets in water tanks. The insertion is initiated by opening the reactor scram valves. Following a reactor scram, a spindle nut is repositioned as an additional means to keep the shutdown rods in the reactor core.

Within the scope of the safety review, there have been no doubts with respect to the efficacy and reliability of the reactor scram systems.

In connection with the discussion of the failure of the automatic reactor scram system at Salem 1 nuclear power plant (USA) in 1983, the RSK recommended to check the operating nuclear power plants with pressurized water reactors as to whether the reliability of the reactor scram system could be further increased by means of a second independent possibility of activating the reactor scram. As a result of this recommendation, a second activation feature was backfitted. The safety review has confirmed this approach.

5.5 Heat removal systems

Nuclear power plants are provided with heat removal systems which remove the heat from the reactor, both during specified normal operation and after incidents. The safety-

related systems are redundant and provided with diverse drives in some instances, depending on the plant concerned. These systems are connected to the emergency power supply system.

In particular in the case of leaks, the emergency and residual heat removal system assumes the function of cooling the reactor sufficiently. The emergency and residual heat removal system is made up of several subsystems which are capable of feeding in against the system pressures resulting as a function of the incident in question. For the BWR, the pressure suppression system is also of importance in this context. Below certain leak sizes in the case of a PWR, additional heat must be removed via the steam generators for a certain period of time. Emergency feedwater systems are available to feed into the steam generators in these cases.

In the case of a failure of operational systems or parts of the safety systems as a result of external impacts, emergency systems or other suitable features will assume the residual heat removal.

The above-mentioned systems differ for different plants and series. The RSK finds that, in some cases, even comprehensive backfitting measures were carried out in this area, in particular with respect to older plants. The safety review has not resulted in any objections.

5.6 Exhaust air system

Areas of a nuclear power plant where the room air may be contaminated are ventilated by ventilation systems. An uncontrolled discharge of radioactive substances to the environment is prevented by the maintenance of subatomospheric pressure and graduated pressures and/or directed flows, or also by the closing of isolation dampers. Exhaust air from restricted access areas is monitored in accordance with the RSK Guidelines and, if necessary, directed through aerosol filters for the elimination of particulate radioactive substances and through iodine sorption filters for the elimination of gaseous idoine. Exhaust air from compartment areas involving a greater contamination potential, such as plant compartments of the containment of a pressurized water reactor, is filtered continuously. Incident filter systems are available in addition to the operating filter systems although they are not used during normal operation so as to have available the highest possible degree of filter efficiency. Inadmissible operating states are prevented by additional components such as droplet separators, heaters and pressure surge valves.

Both number and efficiency of the exhaust air systems were increased and improved with the increasing output of the nuclear power plants, the extended requirements and as a result of the further development of the state of the art. In older nuclear power plants, filter systems were backfitted in some cases so that additional requirements were fulfilled, such as the filter systems for the exhaust air of the turbine hall in a number of boiling water reactors, or the possibility of compartmentwise filtering of the exhaust air of the reactor auxiliary building in the case of pressurized water reactors, as recommended by the RSK. As far as the older pressurized water reactors with single-train annulus exhaust air handling (incident filters) are concerned, a further possibility for annulus exhaust air handling was provided in the majority of cases by the installation of a filter system operating when required. However, some of the filter systems which operate when required are not provided with droplet separators and secondary heaters and are neither designed for making available the separating efficiency of the incident filters. A differing status of the exhaust air systems can be found which is, among other things, also due to the fact that more recent plants are provided with larger and, in some cases, additional filter systems, also for reasons of operation, e.g. in order to save time when preparing refueling.

Adherence to the requirements of the guidelines was checked by the RSK in its discussions with respect to the commissioning of new plants.

As recommended by the RSK, additional features are being installed at the nuclear power plants within the scope of accident management (see Section 9.3).

5.7 Individual issues

5.7.1 Interface between high-pressure and low-pressure systems

On the basis of findings made in the course of the safety review, it shall be checked in all nuclear power plants how the low-pressure systems connected to the reactor cooling circuit are protected against an inadmissible application of pressure by means of isolation valves, safety valves, pressure measurements, position indicators of the valves, etc. The RSK requests information to be submitted with respect to connections between reactor cooling circuit and low-pressure systems.

The RSK is of the opinion that it is necessary to ensure that any internal overpressurization of pipes or other com-

ponents is prevented whose failure may involve a loss of coolant outside the containment. The RSK asks for suggestions with respect tr such improvements. It will discuss the matter on the oasis of the suggestions made by the licensees.

5.7.2 Protection of safety equipment against flooding

As far as the question of a flooding of building areas is concerned which accommodate safety equipment, the RSK finds that a necessity of short-term operator intervention may result in the case of large-size leaks. This questions has been discussed repeatedly in the past, and selected precautions have been taken. In order to get a survey of the circumstances prevailing at all the nuclear power plants, the RSK asks for plantspecific information concerning precautions against flooding, identification possibilities and statements covering the resulting consequences. The RSK will discuss this subject as soon as the information is available.

5.7.3 Design of pipes inside the annulus

In PWRs of more recent design, the ventilation pipes in the annulus are designed to withstand pressures. This means that even in the case of a hypothetical combination of a loss-of-coolant accident and the postulated failure of the containment isolation, the safety systems will retain their operability although they are not designed against corresponding temperatures and the humidity of the containment atmosphere in such a case. The RSK will continue to deal with this question for plants where the ventilation pipes in the annulus are not of a pressure-resistant design. In this context, it will also check whether this question is of importance with respect to other kinds of pipes as well.

5.7.4 Containment penetrations

Following a penetration isolation in the case of a loss-of-coolant accident, the water volume between the double isolation features is heated up and expands as a result of the heat conduction from the containment atmosphere. The RSK asks all nuclear power plants to check the resulting stresses on the respective penetrations and, if necessary, provide relief possibilities for the section between the isolations.

The RSK wishes to be informed on the results of these checks.

5.7.5 Design of the reactor building crane

The RSK gathered information as to how the design and mode of operation of the reactor building crane are used to rule out, render sufficiently improbable or cope with a crash of a fuel element transport cask or of any other heavy load onto the spent fuel pit. It was found in this context that the reactor building crane was designed and/or backfitted in accordance with the relevant Safety Standard KTA 3902 in a number of plants. The RSK takes note of the fact that backfitting is planned for a couple of further plants. It asks the licensees of those plants for which no backfitting has so far been planned to state how the abovementioned protective aims are achieved (cf. Part B. II.).

5.7.6 Mode of operation during steam generator heating tube failure

When designing PWR nuclear power plants, damage to steam generator heating tubes shall also be considered, and measures to control such damage shall be provided. The RSK dealt with this issue in its discussions of the safety concept and the commissioning of plants. It discussed again in detail special precautions for steam generator heating tube damage at plants commissioned in 1984 or later. It will again discuss the question with respect to plants commissioned before that date.

6. Fire protection

The measures taken to ensure fire protection have gained in importance in the course of the technical development of the nuclear power plants, in particular the development towards greater output. In the course of time, an integrated fire protection concept was developed which is subdivided into the categories of fire protection measures in terms of civil engineering, fire protection measures in terms of plant engineering and fire protection measures in terms of plant operation. As this concept and the requirements laid down by it have undergone a certain development in the course of time, its present form could not have been used in its entirety at the plants which were the first to be constructed.

Within the scope of this safety review, the RSK asked the licensees, also with the assistance of GRS, in how far the requirements were met at the individual plants.

The RSK finds that the conceptual differences are not indicative of differing safety levels. It points out that, in the meantime, the fire protection measures for the ol-

der plants now in operation are being adjusted step by step to the current state of the art. As a result of the generally unchanged structural features, additional plant engineering measures were applied to enable an early detection and fighting of fires.

In addition, the RSK finds that it is in particular the emergency systems either in existence or under construction that make a contribution to the improvement of the incident situations initiated by fires. Another essential improvement is the inerting of the containments of BWRs.

As a matter of principle, it has to be made sure that even in areas which are not easily accessible a suitable release will provide early fire fighting. In this context, it has to be checked in how far the older plants are provided with a sufficient number of effective fire extinguishing equipment at the safety-related locations.

This global review carried out by the RSK - which cannot replace a detailed review by authorized experts - has not led to any basic complaint concerning the fire safety of these nuclear power plants. Nevertheless, the RSK will continue to deal with this question and considers it necessary for the licensee to describe for each plant the fire protection measures in terms of civil engineering, plant engineering and plant operation.

7. Protection against external impacts

Of the various possibilities of external impacts affecting a reactor plant, those which hold the potential of massive damage to plant components are of particular safety-related importance. They may be the effects of an earthquake, an explosion of chemicals, an aircraft crash or third-party impacts.

All the nuclear power plants have been designed against earthquakes on the basis of the requirements valid at the time of their construction and/or the site-specific circumstances. After 1975, a uniform design approach was made on the basis of a KTA safety standard (KTA 2201). Moreover, research work such as the shaker tests on the superheated steam reactor (HDR) has shown that the mechanical equipment in the plants is based on a very conservative design and is not endangered.

Man-made external impacts such as an aircraft crash on the reactor building or an explosion of chemicals involving potential damage to safety-related features are characterized by a very low probability of occurrence.

Because of their low risk, these cases are not among the design basis accidents. Nevertheless, since the early seventies, the RSK has recommended an increased structural protection against external impacts, including in particular also protection against third-party impacts and against the crash of military aircraft traveling at high speeds.

Older plants are not fully provided with the structural protection which is a feature of modern plants. However, it should be considered that, in the past, features for an autonomous residual heat removal were backfitted at a number of older plants. Within the scope of its further discussions concerning accident management, the RSK will also deal with the question in how far these plants may use accident management measures to limit the consequences of an aircraft crash and/or of third-party impacts.

In addition, the RSK recommends that the restrictions concerning flights over nuclear power plant sites be tightened and monitored more effectively.

8. Major safety-related backfitting measures

Within the scope of its safety review, the RSK obtained a survey of the major safety-related backfitting measures in the various nuclear power plants. In the past, detailed discussions were held in the RSK for the evaluation of measures of particular importance. The examples to be quoted include the exchange of main steam and feedwater pipes in the nuclear power plants with boiling water reactor, the backfitting of emergency systems, the exchange of vessels with large energy content, the exchange of the steam generators at the Obrigheim nuclear power plant, modifications of the high-pressure emergency core cooling injection system of pressurized water reactors for the prevention of excessive thermal stresses on the reactor pressure vessel wall when feeding in as well as features supplementing the exhaust air systems.

The RSK finds that the backfitting measures it has recommended in the course of time, e.g. following the TMI accident, have in the main been implemented. As far as the backfitting measures as a whole are concerned, it states further that, in the course of the operating lives of the nuclear power plants, the backfitting measures have resulted in an adaptation to the state of more recent safety considerations. As an example for this aspect, the independent autonomous emergency systems are quoted again which were constructed in response to the development of the consideration with respect to the physical separation of systems and the independence of redundant system for coping with both internal and external events acting upon the plants.

9. Accident management

9.1 Importance of accident management and integration into the design concept of nuclear power plants

As a precaution against damage, nuclear power plants are also designed against incidents to be postulated (design basis accidents). The design basis accidents are defined in such a way that each of them is representative of a group of similar events, i.e. they constitute the representative stresses for these groups of events for purposes of plant planning (cf. Guidelines for the Assessment of the Design of PWR Nuclear Power Plants against Incidents, so-called design basis accidents, pursuant to Sec. 28, para. (3) of the Radiological Protection Ordinance; December 1983). In order to cope with the design basis accidents, safety systems are installed which are reliable, redundant and, to a far-reaching extent, diverse and which perform their functions even if the outside power supply fails. The efficiency and reliability of these systems is demonstrated in detail in the course of the licensing process. A meaningful approach in terms of engineering always involves the utilization of identified potential improvements by strengthening preventive measures. Thus, the further development of safety engineering in recent years has always been oriented to the strengthening of preventive measures. This design principle also includes sufficient precautions against core meltdown accidents. The concept of incident control has proven its worth. It is a well-balanced concept and does not need any further extension or mofidication from the point of view of the RSK.

Irrespective of this aspect, the consequences of hypothetical system failures and combinations of failures, which have not been taken into account explicitly when designing the plant (events beyond the design basis, severe accidents), were and are also being investigated within the scope of safety studies, reactor research and risk studies.

Analyses for the flexible use of existing systems and additional measures within the scope of accident management are based on the results of these investigations. The starting point for the analyses with respect to such measures is the existing safety potential of nuclear power plants which results from the design of the plant for safe operation and against postulated design basis accidents. Because of deterministic postulates in the analyses, which constitute the design basis, such as the single failure concept or the postulate of the inefficiency of operating systems for the control of design basis accidents, and because of additional pessimistic analysis boundary condi-

tions, the existing systems, if viewed realistically, show considerably higher efficiencies than determined in the analyses. This means that the existing systems, including the operating systems, can also be used for coping with events beyond the design basis.

The recommendation of accident management measures does not mean that the safety features installed in the plants are insufficient. Such measures further add to the flexibility of the plants when coping with events far beyond the spectrum of design basis accidents (boundary considerations, area of severe accidents). Thus, they have to be allocated to the fourth level of the in-depth safety concept (safety levels).

In accordance with their protective aims, accident management measures permit an early control of the state of the plant and the retention of fuel and fission products in the reactor pressure vessel and in the primary circuit with a high degree of reliability even if the events exceed the design basis. And even if this were not successful, the broad spectrum of accident management measures provides for a decisive limitation of fission product release and the prevention of long-term contamination.

The implementation of additional measures of accident management is thus, in the opinion of the RSK, not a precondition for the safe operation of the plant. These measures are not part of the design basis accident concept, but plant-internal measures in the sense of a supplement to emergency planning (accident management).

9.2 Planning of measures within the scope of accident management

Measures of accident management must be analyzed in particular with respect to their efficacy, implementability, and compatibility with the safety concept. On the basis of the analyses, the decision-making and organizational structures required for such measures will have to be laid down. For the individual measures, implementation instructions have to be prepared which shall be laid down in a document separate from the operating manual, i.e. the emergency manual. The RSK will continue to deal with these questions.

Within the scope of the safety review, it has discussed the concept of the emergency manuals in accordance with which such instructions are being prepared at the various plants. In a number of plants, this work has already progressed very far. The personnel training is also extended to events exceeding the design basis, with accident mana-

gement measures being included in accordance with the emergency manual. The discussions of the concept of the emergency manual will also be continued.

9.3 Accident management measures

9.3.1 Preconditions for their implementation

Within the scope of the safety review, the RSK has first dealt with the creation of the preconditions for the implementation of accident management measures. In the main, these concern the specification that, because of the comprehensive supervisory and intervention possibilities, the control room should be the central location inside the plant where accident management measures are planned, initiated, implemented and supervised. The importance of the emergency control room and/or auxiliary shutdown station remains unaffected.

Moreover, the preconditions for the necessary power supply must exist. As a result of experience with respect to the reliability of the power supply while using the manifold possibilities of the interconnected grid, the RSK specified that, as a planning basis for accident management measures, the restored availability of an external power supply may be assumed again two hours after failure of this supply (cf. Section 5). In this context, the RSK assumes that the personnel at the grid switchgear stations have suitable instructions available for the switching of lines for the priority supply of the nuclear power plants. The RSK recommends that the operator of the grid carry out a responsible check of the appropriateness of the instructions and of the planned switching measures.

In the case of a total failure of the auxiliary power system, or in the case of a failure of the main and standby grids in the near range of the nuclear power plant, those loads which are needed for the safety of the facilities within the scope of accident management shall be capable of being fed by a cable buried in the near range of the nuclear power plant. The cable connection must be physically separated from the main and standby grids in such a way that a simultaneous failure is practically excluded. The licensees were asked to submit a plant-specific concept for their respective plants.

In addition, following a separation of the power plant unit from the grid, a fast return to the grid must be possible as soon as this is available again. It shall be investigated and specified for each nuclear power plant which measures have to be implemented after an interruption of the supply from the auxiliary power supply system (grid and

generator) in order to be able to establish a connection to the grid again as soon as this is available, even if the emergency power diesels are not available. In this context, the two cases of the emergency power diesels failing to start and failing after having started shall be considered. The RSK recommends to plan and design these measures in such a way that the connection to the grid can be established at short notice as soon as the grid is again available.

To secure the direct voltage supply for accident management measures, the RSK recommended to design the discharge times of the batteries in the emergency power system in such a way that the loads can be supplied by the batteries alone for about 2 to 3 hours. Corresponding measures have been implemented or initiated.

The investigation of the feasibility of the measures of accident management also comprises, with respect to each individual measure, the review of a sufficient instrumentation and availability of information as well as an unambiguous specification of the criteria to be derived for the initiation of accident management measures.

9.3.2 Measures planned and/or already implemented

As a result of the discussion in the RSK, the licensees of German nuclear power plants with light water reactors have so far planned or already implemented the following accident management measures for which the RSK has submitted comprehensive recommendations or comments (cf. the list in Appendix 1). Aspects which have turned up in the course of the implementation of the individual measures will be addressed in the following.

9.3.2.1 Concept for secondary-side and primary-side depressurization and coolant injection in the case of PWRs

The primary aim in the implementation of measures for the flexible use of existing systems is the prevention of a core meltdown accident, or at least the retention of a damaged reactor core inside the reactor pressure vessel, in order to prevent any further progressing of the accident. Moreover, in the case of such events, pressures in the primary system within the range of the response pressure of the pressurizer valves shall be prevented. An early depressurization increases the possibilities for flexible measures for coolant injection into the primary system.

The licensees of the KWU pressurized water reactors in the Federal Republic of Germany have developed a concept which is based on the complete failure of the secondary side heat sink as a result of the failure of all operating and safety-related injection systems of the steam generators. Priority is given to measures for a depressurization of the steam generators and injection into the depressurized steam generators, e.g. using the feedwater tank as an accumulator, or mobile systems. As a backup measure, the opening of the pressurizer valves on the primary side is considered. For this purpose, modifications in the activation and, in the case of older plants, modifications of the valves and pipes are necessary in order to cope with the water loads.

The RSK agrees to the concept. It asks to be informed on the further development steps.

As far as the KMK plant is concerned, the licensee has so far not finalized an investigation of a primary-side depressurization as the time history of the KMK plant differs from that of the KWU PWR plants. The RSK asked the licensee to submit investigations of the adequateness and possibilities of a primary-side depressurization.

9.3.2.2 Hydrogen distribution and hydrogen combustion inside the containment

- BWR

By means of a selected reduction of the oxygen content in the atmosphere of the containment, inadmissible stresses acting on the containment as a result of hydrogen/oxygen recombination in hypothetical events with a great production of hydrogen can be excluded. This is of particular importance for the boiling water reactors of the 69 series, because of the relatively low containment volume. An inerting concept was developed and has already been implemented in a number of plants. This provides for continuous inerting during operation and also takes into consideration the accessibility requirements with respect to the containment during normal operation.

The containments of the BWRs of the 72 series (KRB B/C) differ considerably from those of the BWRs of the 69 series. At present, the licensee of the KRB is in the process of developing an inerting concept and a pressure suppression concept which will take account of the different circumstances.

The RSK discussion will follow as soon as the corresponding documents are available.

- PWR

The inclusion of accident management measures also constitutes a far-reaching precaution for the prevention of core meltdown accidents and the related formation of hydrogen.

If it is assumed as a hypothesis that in spite of the existing redundant and diverse safety features and possible accident management measures the reactor core will remain insufficiently cooled for a longer period of time, and cannot be retained inside the reactor pressure vessel in the further course of the accident, the production of great amounts of hydrogen has to be anticipated as a result of steam/metal reactions during the first few hours and the melt/concrete interaction in the long run. The earlier the ignition and combustion of the hydrogen, the lower the stresses acting on the containment and its internals.

If the production times of the hydrogen are considered and the fact that in such an accident scenario various ignition sources exist for the hydrogen/air/steam mixtures (e.g. hot surfaces, electrostatic charges by gas/particle flows), an early uncontrolled combustion of the hydrogen inside the containment at an uncritical point in time may be assumed. Together with the inerting effect of the steam, this combustion would not jeopardize the integrity of the containment.

In addition, comprehensive investigations and developments as suggested by the RSK were carried out for the early elimination of hydrogen inside the containment by means of a controlled ignition with the aid of autonomous fuses and films having a catalytic effect. At present, the RSK is in the process of evaluating the results of these development activities and will make a respective recommendation in a few months' time.

9.3.2.3 R & D program for the investigation of the hypothetical melt/concrete interaction

If it is assumed as a hypothesis (cf. Sec. 9.3.2.2) that, in spite of the existing redundant and diverse safety features and in spite of possible accident management measures, the reactor core will remain insufficiently cooled for a longer period of time, that it cannot be retained in the reactor pressure vessel in the subsequent course of the accident, and that the molten core penetrates the foundations as a result of thermal and chemical interaction with the concrete, it will spread both axially downward and radially inside the concrete.

To achieve a better understanding of these phenomena, the RSK will assess at a later time the results of respective research and development projects currently under way.

9.3.2.4 Sampling system for accident situations

The RSK considers it necessary for all nuclear power plants to have a corresponding system for drawing samples from the containment atmosphere and from the coolant following design basis accidents.

The RSK is of the opinion that the determination of concentrations of radionuclides in the containment atmosphere and in the sump allows to draw conclusions with respect to the condition of the reactor core following an event beyond the design basis. It should therefore be examined how a corresponding measuring system can be implemented.

It is the opinion of the RSK that when conceiving a sampling system for events beyond the design basis it must be clearly understood from which compartment areas and/or sumps the samples are drawn so that meaningful measuring results are obtained. At present, the licensees are preparing a concept which will be discussed in detail in the RSK.

9.3.2.5 Depressurization of the containment of pressurized water and boiling water reactors following events beyond the design basis

In December 1986, the RSK specified the requirements to be put forward, from its point of view, for a depressurization system for the containment of PWRs, to be followed in June 1987 by those for BWRs, with respect to design and modes of operation, stresses to be taken into account, and layout.

The licensees of the nuclear power plants have taken up the RSK suggestions and, in the majority of cases, have already submitted suggestions for the implementation of a depressurization system with filters.

The RSK also discussed various filter systems which permit an effective retention of aerosols and iodine (cf. Appendix 1).

With respect to the safety-related importance of a measuring system for the monitoring of the emissions of radioactive substances during depressurization of the containment following an event beyond the design

basis, the RSK feels that corresponding measuring values should not be used in the decision on the depressurization, since unrestricted priority is given to the assurance of the integrity of the containment, and thus the time for opening the corresponding relief valves is determined by the pressure buildup inside the containment.

In the main, the emission values determined shall be used for the implementation of accident management measures and for a subsequent preservation of evidence. Details of the measuring equipment are still being discussed, and a separate comment will follow.

9.3.2.6 Follow-up of work relating to accident management

The RSK follows the results of the investigations carried out under BMU assignments with respect to measures of accident management, the results of work on risk studies for PWRs and BWRs, as well as further research results which are of relevance for the area of accident management. In due time, it will discuss the corresponding conclusions.

B. II. Plant-specific results

In Part B. I., the RSK reported on the results of the review and made recommendations which apply to all plants. For Grohnde (KWG), Philippsburg 2 (KKP-2) and Brokdorf (KBR) nuclear power plants as well as for the so-called convoy plants, there are no additional recommendations. The following statements have to be made with respect to the other plants:

1. Nuclear power plants with pressurized water reactor

1.1 Obrigheim Nuclear Power Plant (KWO)

- Reactor pressure vessel

In the reactor pressure vessel of Obrigheim Nuclear Power Plant, the copper content in part of the weld metal is higher than at the plants constructed after KKS. Therefore, the toughness reduction as a result of neutron irradiation has progressed more rapidly in part of the beltline circumferential weld of the vessel at the beginning than had been assumed in the design. By means of selected operational measures, and in particular the use of shielding dummy fuel elements and an adequate arrangement of the fuel elements in their different burnup states, the fluence increase could be reduced to such an extent that the toughness decrease over the scheduled lifetime will remain within limits which do not pose any problem in terms of safety.

As in the case of a loss-of-coolant accident caused by a small leak, i.e. involving only a slow pressure reduction, the high-pressure emergency injection will preferably take place via the hot-side pipe and the emergency coolant is preheated to at least 70 °C in the case of a possible injection on the cold side after completion of the additional safety injection, the cold water injection loading case, which is decisive for the demonstration of the fracture safety of the beltline circumferential weld, can be excluded. The condition of the beltline circumferential weld is monitored by fluence measurements and nondestructive examinations. Analyses have shown that the toughness remaining over the lifetime will be sufficient, even under incident conditions, with respect to faults which cannot be detected during the nondestructive examination. This means that the partial toughness reduction in the beltline weld, which is caused by neutron fluence, is sufficiently protected.

- Pipes of the pressure boundary

The RSK expects that, considering the analyses still to be completed with respect to the reactor coolant pipe, an application of the leak postulates pursuant to the latest version of the RSK Guidelines will be justified.

- Electrical equipment

The RSK takes note that at Obrigheim Nuclear Power Plant a short-term replacement of the emergency power diesel generators of emergency grid 2 by another supply remains possible in the case of an internal failure-initiating event in the area of the auxiliary power system (e.g. by fire).

At Obrigheim Nuclear Power Plant, the two trains of emergency grid 1 of the emergency power system are not consistently separated physically. While considering the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also displayed sufficiently.

In addition, the RSK recommends to initiate emergency power operation in both emergency grids via two diverse activation criteria (voltage and frequency).

- Exhaust air system

At present, there is only one train of the annulus exhaust air handling system. The RSK takes affirmative note of the fact that a backfitting to 2 x 100% (incident filters with droplet separator and secondary heater) is planned.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Fire protection

At present, improvements are being made with respect to compartmentalizations, and a stationary fire extinguish-

ing system is being backfitted. The RSK makes reference to its statements listed in B. I. 6.

- Depressurization of the containment following events beyond the design basis

The pipes of the system have been implemented. No final decision has as yet been made with respect to the filter concept. The RSK will discuss this as soon as the concept documents are available.

- Filtering of supply air to the control room

A mobile filter system near the control room is ready for operation. The RSK requests that it be informed about the concept.

1.2 Stade Nuclear Power Plant (KKS)

- Reactor pressure vessel

In the reactor pressure vessel of Stade Nuclear Power Plant, the copper content of the weld metal is higher than at the plants constructed later. Therefore, the toughness reduction as a result of neutron irradiation has progressed more rapidly in the beltline circumferential weld of the vessel at the beginning than had been assumed in the design. By means of selected operational measures, and in particular a careful operation of the reactor and a suitable arrangement of the fuel elements with their different burnup states, the fluence increase could be reduced to such an extent that the toughness decrease over the scheduled lifetime will remain within limits which do not pose any problem in terms of safety.

As in the case of a loss-of-coolant accident caused by a small leak, i.e. involving only a slow pressure reduction, the high-pressure emergency injection will take place via the hot pipe, the cold water injection loading case, which is decisive for the demonstration of the fracture safety of the beltline circumferential weld, can be excluded. The condition of the beltline circumferential weld is monitored by fluence measurements and nondestructive examinations. Analyses have shown that the toughness remaining over the lifetime will be sufficient, even under incident conditions, with respect to faults which cannot be detected during the nondestructive examination. This means that the toughness reduction in the beltline weld, which is caused by neutron fluence, is sufficiently protected.

- Main steam pipes

The licensee notified the RSK that it will replace the pipes in the main steam system between steam generator and valve station (feedwater head) outside the reactor building. Moreover, an internal valve station with control bypass will be available. The RSK takes affirmative note of this.

- Electrical equipment

At Stade Nuclear Power Plant, there is no consistent physical separation in the area of the auxiliary power system. The RSK recommends to check and, if necessary, to make sure by additional measures that even in the case of an internal failure-initiating event in the area of the auxiliary power system (e.g. fire) a short-term replacement of the emergency power diesels by another supply system will remain possible.

At Stade Nuclear Power Plant, the two trains of emergency grid 1 of the emergency power system are not consistently separated physically. While considering the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. fire). It has to be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also displayed sufficiently.

In addition, the RSK recommends to initiate emergency power operation in both emergency grids via two diverse activation criteria (voltage and frequency).

- Exhaust air system

At present, there exists only one train of the annulus exhaust air handling system. There is a multi-train filter system which responds when required, although it does not have any additional features for the reduction of the humidity of the air. The RSK recommends the backfitting of one of the trains of the existing filter system, which can be switched to the annulus, with droplet separator and secondary heater. As an alternative, a second complete filter system for annulus air handling (incident filters) can be backfitted.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Design of the reactor building crane

The RSK takes affirmative note of the fact that it is planned to backfit the reactor building crane in accord ance with the requirements of KTA 3902.

1.3 Biblis Nuclear Power Plant (KWB, Units A and B)

- Pipes of the pressure boundary

The RSK expects that, considering the analyses still to be completed with respect to the reactor coolant pipe, an application of the leak postulates pursuant to the latest version of the RSK Guidelines will be justified.

- Emergency system

KWB A/B do not have unit-allocated emergency systems (of their own). Support is from the adjacent unit. In the safety analysis mentioned (cf. Section A. 2.) for Unit A, this will be assessed. The RSK requests that it be informed about the results.

- Design of the reactor building crane

It was explained to the RSK how the crash of a fuel element transport cask or of any other heavy load on the spent fuel pit is ruled out, rendered sufficiently improbable or coped with. The RSK will discuss this.

- Electrical equipment

At Biblis A Nuclear Power Plant, there is no consistent physical separation in the area of the auxiliary power system. In connection with the discussions of the emergency power case on April 19, 1988, the RSK recommended to make sure by additional measures that even in the case of an internal failure-initiating event in the area of the auxiliary power system (e.g. fire) a short-term replacement of the emergency power diesels by another supply system (e.g. grid or adjacent unit) will remain possible. In addition, the RSK recommended to arrange the main grid connections for Unit B along two separate overhead routes.

At Biblis A Nuclear Power Plant, it is planned to arrange two trains each of the four-train emergency power system in pairs and to provide consistent physical separation. While considering these changes and the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but also that the state of the plant is displayed sufficiently.

In addition, the RSK recommends to initiate emergency power operation in both units via two diverse activation criteria (voltage and frequency) and to protect the excess voltage protection for the 220 V direct voltage systems with diode-decoupled load feeding against single faults.

- Exhaust air system

At present, the KWB A and B Plants only have a single-train annulus exhaust air handling system. The exhaust air filtering for the annulus and the reactor auxiliary building is effected only via aerosol filters.

The RSK takes affirmative note of the fact that a filter system which responds when required will be backfitted.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Depressurization of the containment after events beyond the design basis

A concept for the depressurization of the containment will be extended to cover iodine filtration. The RSK asks for the concept to be submitted to it.

- Supply air filtering for the control room

A concept for the supply air filtering for the control room exists and will be dealt with by the RSK.

1.4 Neckar-1 Joint Nuclear Power Plant (GKN-1)

- Pipes of the pressure boundary

The RSK expects that, considering the analyses still to be completed with respect to the reactor coolant pipe, an application of the leak postulates pursuant to the latest version of the RSK Guidelines will be justified.

- Fuel pit cooling system

A third fuel pool cooling system will be backfitted during one of the next inspection outages. The RSK requests that corresponding documents be submitted to it.

- Exhaust air system

The RSK takes affirmative note of the fact that a filter system which responds when required will be backfitted.

- Control room and emergency control room

Following the modifications still under way, the incident instrumentation will correspond to KTA 3502.

- Electrical equipment

The RSK recommends to provide a chronologically graduated addition of loads for the long-term changeover in the auxiliary power system, or to demonstrate that the admissible voltage values for the loads are adhered to when changing over. In this context, particular attention shall be paid to power supply to the heat sink.

In addition, the RSK recommends to initiate emergency power operation via two diverse activation criteria (voltage and frequency).

- Design of the reactor building crane

The RSK requests an explanation of how the crash of a fuel element transport cask or of any other heavy load on the spent fuel pit is ruled out, rendered sufficiently improbable or cope with.

1.5 Unterweser Nuclear Power Plant (KKU)

- High-pressure injection from the containment sump

As a substitute measure for high-pressure injection from the containment sump, the licensee provides the sump / residual heat removal system / volume control system pathway which permits injection at a rate of 14 kg/s.

- Electrical equipment

The RSK takes note of the fact that a long-term change-over feature exists in the auxiliary power system. The graduated addition of loads must be implemented in such a way that it does not fall below the admissible voltage limits nor trip the overcurrent protection. In this context, particular attention shall be paid to power supply to the heat sink.

At Unterweser Nuclear Power Plant, there is no consistent physical separation in either of the two ± 24V trains in emergency grid 1 of the emergency power system. While considering the emergency system concerned, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also displayed sufficiently.

In addition, the RSK recommends to initiate emergency power operation in both emergency grids via two diverse activation criteria (voltage and frequency) and to protect the excess voltage protection for the 220 V direct voltage systems with diode-decoupled load feeding against single faults.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

1.6 Grafenrheinfeld Nuclear Power Plant (KKG)

- High-pressure injection from the containment sump

The RSK takes affirmative note of the fact that corresponding backfitting measures are planned.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Electrical equipment

The RSK takes affirmative note of the fact that the excess voltage protection for the 220 V direct voltage systems is protected against single faults with diodedecoupled load feeding.

1.7 Mülheim-Kärlich Nuclear Power Plant (KMK)

- Exhaust air system

The annulus exhaust air handling system of the KMK Plant is a two-train system. The exhaust air of the annulus and of the reactor auxiliary building is passed only through an aerosol filter. The systems for the maintenance of subatmospheric pressure in the containment and in the annulus as well as the recirculation air filter systems in the containment are equipped with both aerosol and activated charcoal filters.

The possibility of iodine filtering for the compartments of the reactor auxiliary building with corresponding contamination potential shall be reviewed.

- Accident management

The RSK will again deal with the concept of depressurization on the primary side (cf. B. I., Sec. 9.3.2.1).

2. Nuclear poer plants with boiling water reactor

2.1 Würgassen Nuclear Power Plant (KWW)

- In-service inspections

With respect to the recirculation water circuits, the scope of the nondestructive examinations has been extended considerably since 1982. Most of the corresponding baseline measurements have been completed. The testing areas with ultrasonic measurement displays which must be recorded will be included in the program of the in-service inspections and monitored for modifications.

- Electrical equipment

At KWW, there is no consistent physical separation in the area of the auxiliary power systems. In the case of failure-initiating events in the area of the auxiliary power system (e.g. a fire), the safety of the plant is ensured by the independent residual heat removal system.

The RSK takes affirmative note of the fact that the feeder provided to supply the independent residual heat removal system is independent of the auxiliary power system.

At KWW, two or three trains of emergency grid 1 of the emergency power system are not consistently separated physically. While considering the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also displayed sufficiently.

In addition, the RSK recommends to protect the excess voltage protection for the 220 V direct voltage systems by means of diode-decoupled load addition against single failures.

- Exhaust air system

The purge air system of the KWW is only a single-train system. Filtering of the exhaust air from the turbine hall is not possible. In the reactor building, exhaust

air filtering of the operating platform is possible. The RSK recommends to backfit, and provide for incidents, a second filter system which responds when required and can be changed over to the reactor building or the turbine hall.

- Control room and emergency control room

The RSK takes affirmative note of the fact that adaptation to the requirements of KTA 3502 is planned.

- Emergency measures

Concepts for an additional reactor pressure vessel injection and possible make-up feeding within the scope of accident management are being prepared. The RSK requests that the concepts be submitted in due time.

2.2 Brunsbüttel Nuclear Power Plant (KKB)

- Pressure relief system

Concerning the diversity of the safety and relief valves a concept of the licensees of the Series 69 BWR is being prepared. It is planned to install bypass valves supplementing the main valves. The RSK requests that the concept be submitted. Diverse pilot valves exist on two safety and relief valves.

- Postulated loss of water from the wetwell into the reactor building within the scope of incident control

As a precaution against the postulated loss of water from the wetwell into the reactor building, it is planned to depressurize the reactor and transfer it into the residual heat removal mode of operation before the orifice nozzle relief pipes of the relief valves emerge. The operator is alarmed to a lack of water in the wetwell by way of a hazard signal. The measures to be taken are described in the operating manual. The RSK requests that it be informed of the times available for manual intervention and that a draft concept be submitted as to how a postulated leak in the wetwell which cannot be isolated can be controlled, if necessary without manual intervention that would have to be effected within a relatively short period of time. This will be discussed further by the RSK.

- Electrical equipment

At KKB Nuclear Power Plant, there is no consistent physical separation of the trains of emergency grid 1 of the emergency power system. While considering the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also displayed sufficiently.

The RSK recommends to protect the excess voltage protection for direct voltage systems with diode-decoupled load feeding against single faults.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Design of the reactor building crane

The RSK takes affirmative note of the fact that a refitting of the reactor building crane in accordance with the requirements of KTA 3902 is planned.

- Emergency measures

Concepts for an additional reactor pressure vessel injection and make-up feeding within the scope of accident management are being prepared. The RSK requests that the concepts be submitted in due time.

2.3 Isar-1 Nuclear Power Plant (KKI-1)

- Pressure relief system

Concerning the diversity of the safety and relief valves, the licensees of the 69 BWR Series are preparing a concept. The RSK takes note of this fact. It is planned to install two diverse pilot valves on two safety and relief valves.

- Postulated loss of water from the wetwell into the reactor building within the scope of incident control

As a precaution against the postulated loss of water from the wetwell into the reactor building, it is planned to depressurize the reactor and transfer it into the residual heat removal mode of operation before the orifice nozzle relief pipes of the relief valves emerge. The operator is alarmed to a lack of water in the wetwell by way of a hazard signal. The measures to be taken are described in the operating manual. The RSK requests that it be informed of the times available for manual intervention and that a draft concept be submitted as to how a postulated leak in the wetwell which cannot be isolated can be controlled, if necessary, without manual intervention that would have to be effected within a relatively short period of time. This will be discussed further by the RSK.

- Electrical equipment

At KKI-1 Nuclear Power Plant, there is no consistent physical separation of two trains each of the four-train emergency power system (two trains are protected against external impacts). While considering the emergency system, the RSK recommends to check which consequences may result for the plant (process engineering, instrumentation, displays in the control room) in the case of an internal failure-initiating event (e.g. a fire). It must be ensured that in the case of such an event not only the superordinate aims of protection are fulfilled (shutdown, residual heat removal, long-term subcriticality), but that the state of the plant is also adequately displayed.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Design of the reactor building crane

The RSK takes affirmative note of the fact that it is planned to refit the reactor building crane in accordance with the requirements of KTA 3902.

- Emergency measures

Concepts for an additional reactor pressure vessel injection and make-up feeding within the scope of accident management are being prepared. The RSK asks for submission in due time.

2.4 Philippsburg Nuclear Power Plant (KKP-1)

- Pressure relief system

Concerning the diversity of the safety and relief valves, the licensees of the 69 BWR Series are preparing a concept. It is planned to install bypass valves for the main valves. The RSK requests that the concept be submitted. Diverse pilot valves exist on two safety and relief valves.

- Electrical equipment

The RSK takes affirmative note of the fact that a second diverse activation criterion (frequency) is provided for the initiation of emergency power operation.

- Design of the reactor building crane

The RSK took affirmative note of the fact that it is planned to backfit the reactor building crane in accordance with the requirements of KTA 3902.

- Control room and emergency control room

The RSK takes affirmative note of the fact that an adaptation to the requirements of KTA 3502 is planned.

- Emergency measures

Concepts for an additional reactor pressure vessel injection and make-up feeding within the scope of accident management are being prepared. The RSK requests that the concepts be submitted in due time.

2.5 Krümmel Nuclear Power Plant (KKK)

- Pressure relief system

Concerning the diversity of the safety and relief valves, the licensees of the 69 BWR Series are preparing a concept. It is planned to install bypass valves on the main valves. The RSK requests that the concept be submitted. A diverse pilot valve system already exists on two safety and relief valves.

2.6 Gundremmingen Nuclear Power Plant (KRB B/C)

- Pressure relief system

The RSK asks for submission of the suggestions for a diversification of the safety and relief valves.

- Emergency measures

With respect to accident management, tha RSK is awaiting the submission of concepts by the licensee.

C. Results of the Safety Review of Hamm-Uentrop Nuclear Power Plant (THTR-300)

1. Systems engineering (status/backfitting)

1.1 Operating and safety systems, incident control

1.1.1 Components and pipes

o Prestressed concrete vessel

The primary circuit of the THTR-300 with its continuous operating pressure of 39 bar is contained in a prestressed concrete pressure vessel the interior of which is lined with a cooled steel liner. The design pressure is 46 bar. The following statements are made concerning its safety against failure: As a result of the distribution of the tensile stresses to several hundreds of tendons each consisting of approx. 150 individual wires, even the failure of individual tendons (which need not be postulated) would not lead to a failure of the vessel. The failure of tendons would be identified by continuous monitoring of representative tendons as a result of the increase in stress before the load-bearing capability of the concrete support structure would be endangered. The integrity of the vessel closures is such that they withstand 1.5 times the design pressure. The integrity of the liner was demonstrated at 64 bar.

These facts and circumstances are evidence of the considerable safety potential of the prestressed concrete vessel.

o Steam generators, main steam and feedwater pipes including connecting pipes (secondary circuit)

The heat is removed through 6 steam generator/circulator units which are located in the prestressed concrete vessel. With respect to the decisive areas, the water/steam circuit meets the criteria of basic safety.

1.1.2 Shutdown systems

The THTR-300 has 2 independent and diverse shutdown systems, i.e. the reflector rods and the core rods. For reasons of safety, all reflector rod and core rod drives are embedded in the ceiling of the prestressed concrete vessel.

The reflector rod system is used for reactor control and scram purposes. The 36 reflector rods can move freely in vertical boreholes of the side reflector.

They are moved by an electric motor and a chain. For scram purposes, they will drop into the reflector by gravity. The same applies in the event of a power failure.

The core rod system is used for long-term shutdown and can render the reactor subcritical from any state of operation. The system consists of 42 core rods which are pneumatically inserted into the pebble bed.

Any fast ejection of either core or reflector rods can be ruled out for reasons of design.

In the case of an inadmissible reactivity gain, if any, or in the case of a failure of the reactor scram system when required (ATWS), the physical design (negative temperature coefficient of reactivity), in combination with the high temperature strength of the reactor core, ensures the limitation of the reactor power.

Within the scope of the safety review, no doubts resulted concerning either the efficiency or the reliability of the shutdown systems (also see Sec. 1.2.2).

1.1.3 Residual heat removal systems

During incidents where the operating heat removal system and the startup grid supply are available, heat removal is effected via the operating circuit within the scope of an automatic fast shutdown procedure.

In the case of incidents which lack the conditions of the fast shutdown procedure, a decay heat removal procedure is automatically activated via the residual heat removal system. This consists of two cooling loops (capacity of 100% each in a pressure relief incident and 200% each with the reactor pressurized) each of which holds two of each of the active components.

To cover the case of a long-term interruption of the residual heat removal as a result of a failure of the two cooling loops, the operating manual contains instructions ("LUNWA") as to how a minimum emergency cooling chain can be made operative by manual intervention. For this purpose, 3 hours are available after the interruption of the heat removal.

The safety review did not lead to any complaints.

1.1.4 Pressure relief incident and ingress of air into the reactor core

In the case of an ingress of air into the primary circuit following a pressure relief, it is ensured that the amounts of air entering as a result of gas contraction in the primary circuit, air pressure fluctuations and natural convection will remain limited to levels which are insignificant in terms of safety. The formation of a stack draft effect can be ruled out.

The residual heat removal systems provide for a rapid cooling down of the primary circuit to temperatures at which the graphite corrosion of the fuel elements which are affected most becomes insignificant. The resulting release of fission products can be neglected; the formation of ignitable mixtures of gases can be ruled out.

Even in the case of the extremely unlikely combination of a pressure relief incident and a failure of the residual heat removal systems, the calculations performed showed that the graphite corrosion of the bottom reflector which may be caused by the ingress of air will remain insignificant.

1.1.5 Exhaust air system

The exhaust air system is designed in such a way that a negative pressure, as compared with the remaining compartments, is maintained in those compartments where radioactivity may occur as a result of operational leakages out of the reactor cooling system.

The measurements which have so far been carried out with respect to the activity in the primary circuit (coolant gas) resulted in levels for the various groups of nuclides which are far below the design levels.

1.1.6 Electric power supply

It is a plant-specific characteristic of the THTR-300 that two trains each of the four-train emergency power system have not been consistently separated physically. To the extent this has not yet been done, the RSK recommends to check which consequences a failure-initiating internal event (e.g. a fire) may have on the plant (process engineering, instrumentation, displays in the control room). In the case of such an event, it must be ensured that not only the superordinate aims of protection are fulfilled, but also that the state of the plant is displayed adequately.

1.1.7 Fire protection

This global review carried out by the RSK - which cannot replace a detailed review by an authorized expert - did not result in any serious complaints concerning fire protection. Nevertheless, the RSK will further deal with this question and considers it necessary for the licensee to submit the essential fire protection measures in terms of structures, plant engineering and operation.

1.1.8 Protection against external impacts

Within the scope of its further discussions of accident management at this plant, the RSK will also deal with the question in how far accident management measures will permit a mitigation of the consequences of an aircraft crash or third-party impacts.

1.2 Accident Management

1.2.1 Filtering of air supply to the control room

The licensee showed that it will be possible to make a mobile filter system available which can be used when required. The RSK recommends to implement this measure.

1.2.2 Investigations of reactivity accidents

Within the scope of the safety review, no doubts resulted with respect to the efficiency and reliability of the shutdown systems, including the associated control and instrumentation equipment.

The licensee investigated the behavior of the plant in the case of a failure of the core rods when required. This investigation covered a representative spectrum of initial states of the plant. The licensee showed that, if the long-stroke and short-stroke drive systems fail, the core rods can be inserted if pressurized gas cylinders (He or N_2) are connected in time to the long-stroke cylinders of 2 core rods each. Moreover, a recriticality could be delayed by an injection of nitrogen into the primary circuit.

In addition, it is found that even a recriticality will be compatible with the system, since the reactor, as a result of its negative temperature coefficient of reactivity, will get stabilized at a level where the respective power can be removed by means of the residual heat removal system.

1.2.3 Depressurization of and re-injection into the steam generators

The licensee showed that, beyond the procedures of the automatic residual heat removal and the procedure of heat removal after 3 - 5 hours of interruption (LUNWA procedure), various steam generator valves can be opened by manual intervention within the scope of the accident management so that both pressure relief of and re-injection into the steam generators will be possible.

1.2.4 Steam generator heating tube failure without steam generator isolation

In the course of the licensing procedure, a hypothetical water ingress accident without steam generator isolation was investigated. The analysis of this acccident had been required prior to the decision with respect to the omission of a safety valve on the prestressed concrete vessel. The maximum pressure was calculated at 54.3 bar and does not constitute any danger to the prestressed concrete vessel (cf. Secs. 1.1.1 and 1.2.7).

1.2.5 Emergency injection into the liner cooling system

The occurrence probability of core heatup accidents with a simultaneous failure of the liner cooling system is determined by the long-term failure of the emergency power supply system. It is smaller by approximately one order of magnitude than that of core heatup accidents with an operative liner cooling system. A minimum period of 24 hours is available for the restoration of the liner cooling.

The licensee explained the possibility of emergency injection into the liner cooling system from a fire-fighting pipe. The fire extinguishing water system is independent of the power supply system of the THTR-300. After the implementation of a few modifications of the pipe network of the liner cooling system, an emergency injection would be possible within 1 to 2 hours after being required.

The RSK recommends to effect those changes of the pipe network of the liner cooling systems as are necessary to start emergency injection into the liner cooling system, when required, from the fire extinguishing system.

1.2.6 Investigations with respect to core heatup accidents

If it is postulated as a hypothesis that, in spite of the existing redundant and diverse safety features and in spite of possible emergency measures, the reactor core will remain insufficiently cooled over a longer period of time, then the reactor core will continue to heat up and, with an increasing temperature gradient, more and more heat will be transferred via the core surface to the surrounding components.

If the liner cooling system is in operation the prestressed concrete vessel will remain at operating temperature. The liner cooling system constitute a heat sink. This means that all the temperatures in the primary circuit will gradually decrease again after 7 days. During this period of time, the mean coolant gas temperature will reach a maximum of 560 °C. In this context, the thermal insulation of the liner of the prestressed concrete vessel is not endangered; the tendons will remain cold. The possible failure of the top reflector and of the top thermal shield as a result of a failure of their suspensions, would not lead to any additional release of fission products.

During the heating-up phase, the most essential aspect is the limitation of the pressure buildup associated with the heating-up of the coolant gas. This is done by reconnecting the gas purification plant to the primary circuit by manually opening again the penetration valves of the plant which are closed automatically during this accident. This measure has been prepared both technically and administratively. The gas purification plant is provided with a number of safety valves which limit the pressure to approx. 50 bar and discharge filtered gases (aerosols) to the stack.

The release of fission products to the environment was investigated with respect to core heatup accidents for the period following the accident and without consideration of a filtering of the primary gas discharged via the stack. Even in the case of an unfavorable combination of release mechanisms for dominant fission products, the plant remains accessible for accident management measures. This applies even if a long-term failure of the liner cooling system is postulated in addition.

The RSK finds that the gas purification plant is of importance for the mitigation of the consequences of a core heatup incident. The RSK considers the following emergency measures as meaningful:

- let primary helium flow into the purified gas store,

- discharge filtered gas from the primary circuit to the stack.

It recommends that the preconditions required for implementing these accident management measures be fulfilled within a closed concept. In this context, measures delaying the accident sequence, such as the moving of circulator isolation devices, should be taken into consideration.

1.2.7 Activity confinement

The most important activity barrier lies in the spherical fuel element itself and in particular in the coated fuel particles. Even in the case of high temperatures due to an accident, this barrier maintains its retention capability to a far-reaching extent. Thus, the accessibility of the plant for accident management measures is also ensured in the accident scenarios under review.

The reactor core and the helium as the primary coolant are confined in the prestressed concrete vessel. Experiments and investigations were carried out with respect to the behavior of the prestressed concrete vessel under extreme stresses which would only have to be anticipated in the case of a long-term failure of the liner cooling system. When a pressure of 70 bar is reached, the tightness of the vessel closures would be lost, and the vessel would begin to discharge. The load-bearing capability of the prestressed concrete support structure with a reduced concrete strength as a result of the temperature was estimated at 135 bar at the time of the latest opening of the pressure vessel closures. The liner would fail only at even higher pressures, due to its large strain reserves. Without any pressure relief, a maximum pressure of 110 bar would not be exceeded in the course of the accident, and the load-bearing capability of the reinforced concrete structure would always be greater than the pressure in question.

The release of steam and gas (CO_2) when heating up the concrete was also investigated. On the basis of the results obtained it can be assumed that for both steam and gas there are sufficient pathways - e.g. along the armoured tubes and between liner and concrete as well as, at higher temperatures, through porosities and the system of cracks beginning to form in the concrete - so that any inadmissible pressure buildup between liner and concrete can be ruled out.

The RSK is of the opinion that although this vessel has considerable safety reserves against accident-related stresses, further in-depth investigations should follow with respect to the discharge of steam and gases which

are discharged from the concrete during the heating-up process and might lead to a pressure buildup between concrete and liner.

2. Operation

2.1 Operating manual, training and preservation of qualification

The RSK was informed as to the measures concerning the acquisition of qualification by the responsible nuclear power plant personnel and the proof of such qualification as well as the measures for the preservation of qualification. Both training and the measures for the preservation of qualification are in compliance with the guidelines of the BMU for, among other things, the proof of qualification, the content of the qualification examination and programs for the preservation of the qualification of the responsible shift personnel at nuclear power plants. Accordingly, the knowledge referred to as qualification is trained repeatedly at regular intervals. It is intended that all shift supervisors, deputy shift supervisors and reactor operators attend, as a rule, a minimum of 100 hours of such events a year. The training measures also include the observation of startup and shutdown procedures in the control room as well as the planning and practising of emergency measures.

Instead of the simulator training, the licensee places particular emphasis on the regularly repeated theoretical instructions and practical exercises.

In its further discussions of personnel training, the RSK will again address the duration of retraining. In this context, it will also continue to deal with questions of simulator use, and in particular with the suitability of a plant-specific simulator for the THTR-300.

As far as the state-oriented approach in incidents and accidents is concerned, the licensee showed on the basis of a draft of the revised Chapter 3.1 "Incident treatment" how it intended to apply the state-oriented approach for the control of design basis accidents. The fulfillment of the protective aims of reactor shutdown, residual heat removal and activity confinement is reviewed, and instructions for action in the case of nonfulfillment are given, on the basis of flow charts.

In principle, the RSK is in agreement with the licensee as far as the state-oriented approach is concerned. It recommends to also take into consideration another aim of protection, i.e. to finish as quickly as possible an

ingress of water into the primary circuit during a steam generator leakage. It requests that the draft of the modified operating manual be submitted to it again for discussion as soon as the the assessment will have been finalized. In this context, it is also necessary to prepared an emergency manual which will have to be a separate document from the operating manual.

Manufacturers and licensee informed the RSK on the preparation and implementation of maintenance and modification work. The preparation and implementation of maintenance and modification work are laid down in the operating manual. Prior to their implementation, all the essential modification and maintenance activities will be evaluated in terms of safety by licensee, authorized expert and supervisory authority.

2.2 Operating experience at the THTR-300

The RSK was informed on the operating experience with the THTR-300. Following the successful commissioning operation the plant was handed over to the licensee on June 1, 1987. The RSK has discussed in detail the THTR-300 events that have occurred so far, and it will continue this discussion. The following were noteworthy events:

- Relatively frequent automatic shutdown with residual heat removal via the residual heat removal system

The "automatic shutdown with residual heat removal", which occurred relatively frequently at first, was released, among other things, by limits which had been set too narrow. Thereafter, unjustified releases of the automatic residual heat removal procedure could be avoided by resetting the limits on the basis of commissioning experience.

- Increased occurrence of damaged operating elements (absorber, graphite and fuel elements) beyond what had been anticipated

The increased occurrence of damaged operating elements beyond what had been anticipated is attributed by both vendor and licensee to processes during the commissioning phase of the reactor. Because of trials, the core rods were inserted into the core more frequently than is

normal, and the core had undergone a certain densification as an insufficient number of spheres had been circulated in the meantime. This meant that unusually strong stresses acted on the operating elements. In addition, one core rod trial was performed without injection of ammonia which is normally added as a lubricant. The withdrawal of the damaged operating elements from the reactor takes more time than had originally been anticipated, since the flow behavior of the reactor core, as quoted by both vendor and licensee, differs from the calculations insofar as the operating elements traverse the reactor core more quickly in the center, and more slowly towards the rim, than had been calculated. According to the licensee's estimate, operating elements which were damaged during the abovementioned trials will continue to be encountered for at least another six months of full-load operation when withdrawing the operating elements from the reactor.

- Difficulties when withdrawing operating elements from the reactor

When ciruclating the operating elements of the reactor, it was found in the course of the power increase to 60% of nominal power that the operating element throughput is considerably reduced by the singulizer / helical scrap separator. The cause was considered to be a coolant gas stream which cools the sphere discharge tube and counteracts the movement of the spheres (operating elements) to be discharged from the reactor core. For reasons of process engineering, this coolant gas stream was further increased with the rise in reactor power, since this also involved an increase in the speed of the coolant gas circulator.

By means of design changes during the 1987 inspection outage, the coolant gas stream was redistributed in such a way that the withdrawal of the spheres at the necessary sphere withdrawal rate will also be possible at rated reactor power.

- Failure of a helical scrap separator for the identification and separation of damaged and undamaged operating elements

There was a break in operation in April 1987, as one of the helical scrap separators did no longer withdraw any spheres. Its operability could be restored by a minor change in design.

- Emergency power case

In the early phase of commissioning, an emergency power case occurred. The event initiating the emergency power case was the attempt to switch the electric feedwater pump over from a supply bus to a redundant bus. The immediate cause for the separation of the transformer supplying the bus was identified as the short sequence of approx. 1 s between the cutoff and the renewed addition of the pump. This resulted in several changes in certain details, optimizations and process-related specifications.

The RSK gathered information on the operating experience with and events so far encountered at the THTR-300 and is in agreement with the measures taken by the licensee. It does not see any reasons that would speak against the licensee's explanation of the increased occurrence of damaged operating elements which has so far continued. The RSK requests that it be also informed about any further occurrence of damaged operating elements and recommends an intensive investigation into the causes.

Within the scope of the current phase of inspection, the licensee has examined 6 hot gas channels and parts of the hot gas plenum by means of a probe. In doing so, it was found that the heads of some Incoloy 800 central bolts and of two Incoloy 800 corner bolts, which are used for the fixation of metallic insulation packages, were missing in the channels.

In addition, an inspection of 4 hot gas channels showed that a total of 5 graphitic parts were found to have become detached from the graphitic areas of three hot gas channels.

Following the submission of detailed information the RSK will discuss these findings.

D. Requirements for Future Periodic Safety Reviews of Nuclear Power Plants

1. Introduction

The BMU asked the RSK to suggest requirements for future periodic safety reviews. On the basis of the results of the current safety review and the continuous further development of safety engineering by taking into consideration both operating experience and unusual events as well as new results of research projects and risk studies, the RSK makes the following suggestion:

2. Objectives

The periodic safety review is to supplement the current review within the scope of the supervision of operation under nuclear law.

The periodic safety reviews are to be carried out approximately three times during the operating life of a plant. A first-time comprehensive review is to be carried out about 10 years after commissioning. Thereafter, further reviews are to be carried out at intervals of 10 years.

The RSK intends to deal with generic results of the safety reviews of the individual plants.

3. Content and Scope of the Periodic Safety Review

The RSK is of the opinion that the periodic safety review should have the following scope:

- safety status of the plant
- evaluation of safety status and service record
- probabilistic safety analysis.

3.1 Safety status of the plant

The safety status of the plant shall be described and explained by the licensee:

- Systems engineering

  . operating and safety systems
  . accident management

- Operation

  . normal operation, unusual events and incidents
  . accident management

- Multi-Unit Aspects
  (as far as relevant)

- Fulfillment of safety-related requirements

  . status report on the fulfillment of the applicable safety-related requirements

- Consideration of developments

  . description of how the further development of safety engineering and of accident management has been taken into consideration

- Backfitting measures

  . compilation of the backfitting measures carried out and their justifications.

3.2 Evaluation of safety status and of service record

The safety status of the plant as shown by the licensee shall be evaluated.

On the basis of the available documents and by intensive discussions (audits) with the licensee, the areas to be evaluated with a view to the service record shall include, among others:

. plant management
. technical qualification
. organization
. in-service inspections
. maintenance
. feedback of experience
. radiological protection
. emergency planning
. plant security.

3.3 Probabilistic safety analysis

These safety analyses are to supplement the safety evaluation referred to in Sec. 3.2 of the engineered safety features of the nuclear power plant with progressing operation on the basis of probabilistic methods.

The RSK is of the opinion that a probabilistic safety analysis shall be performed for each nuclear power plant. For this purpose, it is necessary to carry out event sequence and reliability analyses (so-called level 1 analyses). The analyses shall be performed on the basis of an accepted methodology and using proven computer codes. The RSK considers it advisable that these analyses be carried out under the licensee's responsibility.

If results of analyses are to be transferred to similar plants using the "delta approach" (e.g. identical design series, identical system layout, identical physical arrangement of systems), the applicability remains to be checked in each individual case.

With a view to the review of the probabilistic safety analyses, the reliability data determined (reliability of components, occurrence frequencies of initiating events, probabilities for the failure of system functions) shall be recorded at a central location for all the individual plants.

In the interest of a uniform and competent implementation and evaluation of the probabilistic safety analyses it is advisable to call in an independent central institution which has had many years of experience in the implementation and evaluation of probabilistic safety analyses. The authorized experts called in during the supervisory process should also participate.

As a result of more recent findings in connection with the operation, with unusual events and with reactor safety research, the probabilistic safety analyses shall be updated.

4. Chronological implementation of future periodic safety reviews

4.1 Probabilistic safety analyses

The RSK is of the opinion that the probabilistic safety analyses should be started as soon as possible. The RSK expects the analyses to be completed for all nuclear power plants in about 10 years time.

4.2 Periodic safety review

The RSK will still deal with the separate setting up of a time schedule for the implementation of the periodic safety reviews. In doing so, it will consider the aim of performing such a safety review approximately three times during the lifetime of a plant and approximately

every 10 years. The chronological order, i.e. the staggering of the time schedule, should be effected considering the past operating lifetime of the plants concerned.

Appendix 1

Comments and Recommendations
made so far within the scope
of the safety review

Status: November 23, 1988

Comments and recommendations made so far within the scope of the safety review

A. Generic Comments and Recommendations

1. COMMENT dated October 15, 1986 (216th Meeting)

   Evaluation of the results of the experts' meeting on the accident at the Chernobyl Nuclear Power Plant and further RSK discussions

   - among other things, a compilation of the desired information for the safety review

2. COMMENT dated September 17, 1986 (215th Meeting)

   Filtration of the exhaust air during the depressurization of the containment following a core meltdown accident

3. RECOMMENDATION dated December 17, 1986 (218th Meeting)

   - Assurance of the containment isolation (for LWRs)
   - Equipment of control room and auxiliary shutdown station with a view to accident management (for LWRs)
   - Depressurization of PWR containments through aerosol filters in the case of core meltdown accidents
   - Inerting of the containment (of BWRs, 69 Series)
   - Reliability of the turbo injection pump (of BWRs, 69 Series)

4. RECOMMENDATION dated June 24, 1987 (222nd Meeting)

   Nuclear power plants with boiling water reactor of the 69 Series (KWW, KKB, KKI-1, KKK)

   - Depressurization of the containment via a filter system

5. RECOMMENDATION dated October 21, 1987 (226th Meeting)

   Accident management for nuclear power plants with light water reactors
   - State of discussions: October 1987

6. COMMENT dated April 20, 1988 (231st Meeting)

   Specifications for filter systems in the depressurization sections of the containments of PWRs and BWRs

7. COMMENT dated May 18, 1988 (232nd Meeting)

   Formation and combustion of hydrogen following hypothetical core meltdown accidents in light water reactors

8. COMMENT dated November 23, 1988 (238th Meeting)

   Depressurization of the containment with the use of iodine sorption filters with molecular sieves

B. Plant-Specific Comments and Recommendations

1. COMMENT dated September 17, 1986 (215th Meeting)

   Brokdorf Nuclear Power Plant (KBR)
   - Review of safety in connection with the accident at the Chernobyl Nuclear Power Plant

2. COMMENT dated April 15, 1987 (220th Meeting)

   Evaluation of the accident at the Chernobyl Nuclear Power Plant with a view to the Kalkar Nuclear Power Plant (SNR-300)

3. COMMENT dated June 24, 1987 (222nd Meeting)

   Krümmel Nuclear Power Plant (KKK)
   - Inerting of the containment

   Krümmel (KKK) and Brunsbüttel (KKB) Nuclear Power Plants
   - Depressurization of the containment through a filter system

4. COMMENT dated July 30, 1987 (223rd Meeting)

   Brunsbüttel Nuclear Power Plant (KKB)
   - Inerting of the containment

5. COMMENT dated November 25, 1987 (227th Meeting)

   Accident management measures for the convoy plants in accordance with the RSK Recommendation dated December 17, 1986

6. COMMENT dated November 25, 1987 (227th Meeting)

   Stade (KKS), Unterweser (KKU) and Grohnde (KWG) Nuclear Power Plants
   - Containment isolation
   - Control room equipment
   - Containment depressurization

7. COMMENT dated December 16, 1987 (228th Meeting)

   Isar 1 (KKI-1) Nuclear Power Plant
   - Depressurization of the containment through a filter system
   - Inerting of the containment
   - Supply air filtering of the control room, maintenance of overpressure

8. RECOMMENDATION dated March 16, 1988 (230th Meeting)

   Investigations with respect to event sequences for nuclear power plants with boiling water reactor, including measures of accident management, using Krümmel (KKK) Nuclear Power Plant as an example

9. COMMENT dated March 16, 1988 (230th Meeting)

   Philippsburg 1 (KKP-1) Nuclear Power Plant
   - Depressurization of the containment through a filter system
   - Inerting of the containment
   - Supply air filtering of the control room, maintenance of overpressure

10. COMMENT dated March 16, 1988 (230th Meeting)

    Philippsburg 2 (KKP-2) Nuclear Power Plant
    - Supply air filtering of the control room, maintenance of overpressure

11. COMMENT dated June 22, 1988 (233rd Meeting)

Mülheim-Kärlich (KMK) Nuclear Power Plant
Accident management
- Depressurization of the containment in the case of accidents
- Control room supply air filtration

12. COMMENT dated June 22, 1988 (233rd Meeting)

Würgassen (KWW) Nuclear Power Plant
Accident management
- Inerting of the containment
- Depressurization of the containment
- Control room supply air filtration

13. COMMENT dated September 21, 1988 (234th Meeting)

Grafenrheinfeld (KKG) Nuclear Power Plant
- Depressurization of the containment in the case of accidents
- Control room supply air filtration

14. COMMENT dated September 21, 1988 (234th Meeting)

Philippsburg 2 (KKP-2) Nuclear Power Plant
- Depressurization of the containment in the case of accidents

15. COMMENT dated November 23, 1988 (238th Meeting)

Neckar GmbH (GKN-1) Nuclear Power Plant
- Depressurization of the containment through filter systems
- Control room supply air filtration

Appendix 2

1. List of Subjects for the 1988 RSK Safety Review

2. Systems engineering

2.1 Operating and safety systems

- Components and pipes
  - o Pressure boundary
  - o Main steam and feedwater pipes
  - o Connecting pipes to the pressure boundary

- Heat removal system
  Primary side, secondary side (PWR)
  Pressure suppression system (BWR)
  Residual heat removal and emergency core cooling systems
  Closed cooling water system, drywell and wetwell spray cooling systems (BWR)

- Shutdown system

- Emergency system

- Electrical equipment
  - o Control and instrumentation equipment of the safety system
  - o Power supply to the safety system
  - o Incident resistance of the electrical equipment of the safety system and of the incident instrumentation
  - o Independence (physical and electrical separation) of the control and instrumentation subsystems (groups of redundancies, plates)

- Containment

- Exhaust air system

- Control room and auxiliary shutdown station

- Fire protection

- Protection against external impacts

- Structures

- Sampling system (incidents)

- ATWS

2.2 <u>Accident management</u>

- Emergency measures for injection purposes
- Depressurization of the containment
- Power supply
- Sampling system
- Hydrogen distribution and combustion (PWR)
- Supply air filtration for the control room
- Inerting of the containment (BWR)

3. <u>Operation</u>

3.1 <u>Normal operation, unusual events and incidents</u>

- Operating manual
- In-service inspections
- General operating experience
- Unusual events
- Technical qualification
  . Personnel training
  . Training simulators and training on the training simulator

3.2 <u>Accident management / disaster control</u>

- Emergency manual
- Stay of emergency personnel
- Communication equipment

4. <u>Multi-Unit Aspects</u>

Appendix 3

List of Abbreviations

ATOG	Abnormal Transient Operation Guidelines
ATWS	Anticipated Transients Without Scram
BHB	Operating Manual
BMFT	Federal Ministry for Research and Technology
BMI	Federal Ministry of the Interior
BMU	Federal Ministry for the Environment, Nature Conservation and Reactor Safety
DWR	Pressurized Water Reactor
F&E	R&D
GKN	Neckar Joint Nuclear Power Plant
GRS	Gesellschaft fur Reaktorsicherheit
HDR	Großwelzheim Superheated Steam Reactor
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operations (USA)
KBR	Brokdorf Nuclear Power Plant
KKB	Brunsbüttel Nuclear Power Plant
KKE	Emsland Nuclear Power Plant
KKG	Grafenrheinfeld Nuclear Power Plant
KKI	Isar Nuclear Power Plant
KKK	Krümmel Nuclear Power Plant
KKP	Philippsburg Nuclear Power Plant
KKS	Stade Nuclear Power Plant
KKU	Unterweser Nuclear Power Plant
KMK	Mülheim-Kärlich Nuclear Power Plant
KNK	Karlsruhe Compact Sodium-Cooled Reactor Facility
KRB	Gundremmingen Nuclear Power Plant
KTA	Nuclear Safety Standards Commission
KWB	Biblis Nuclear Power Plant
KWG	Grohnde Nuclear Power Plant
KWO	Obrigheim Nuclear Power Plant
KWU	Kraftwerk Union (Siemens)
KWW	Würgassen Nuclear Power Plant
LWR	Light Water Reactor
OECD/NEA	Organisation for Economic Co-operation and Development / Nuclear Energy Agency
RSK	Reactor Safety Commission
SNR	Kalkar (SNR-300) Nuclear Power Plant
SWR	Boiling Water Reactor
THTR	Hamm-Üentrop (THTR-300) Nuclear Power Plant
TMI	Three Mile Island (USA)
TÜV	Technical Supervisory Inspectorate
VGB	Association of Large Power Plant Licensees