ENVIRONMENTAL POLICY

Convention on Nuclear Safety

Report by the Government of the Federal Republic of Germany for the Third Review Meeting in April 2005

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Contents

Chapter

List of Figures

List of Tables

Abbreviations

BfS	*Bundesamt für Strahlenschutz* Federal Office for Radiation Protection
BMU	*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit* *- Bundesumweltministerium -* Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
BMWA	*Bundesministerium für Wirtschaft und Arbeit* Federal Ministry of Economics and Labour
BWR	Boiling Water Reactor
GRS	*Gesellschaft für Anlagen- und Reaktorsicherheit*
ICRP	International Commission on Radiological Protection
KTA	*Kerntechnischer Ausschuß* Nuclear Safety Standards Commission
LAA	*Länderausschuß für Atomkernenergie* Federal States Committee for Nuclear Energy
OECD/NEA	Organisation for Economic Co-operation and Development/ Nuclear Energy Agency
PWR	Pressurised Water Reactor
RSK	*Reaktor-Sicherheitskommission* Reactor Safety Commission
SSK	*Strahlenschutzkommission* Commission on Radiological Protection
SR	Safety Review
VGB	VGB Power Tech, Technische Vereinigung der Großkraftwerksbetreiber
WANO	World Association of Nuclear Operators

Introduction

In view of the constitutional protection of human life as well as the protection of public health, the Government of the Federal Republic of Germany decided to end the use of nuclear energy for commercial electricity production in Germany in an orderly manner. This decision is implemented by limiting the standard lifetime of the nuclear power plants to 32 years from the date of commissioning.

Safe operation of the nuclear power plants has to be ensured for their remaining operating lives in accordance with the provisions of the Atomic Energy Act. An essential condition for that is an efficient and a competent nuclear regulatory supervision. To ensure this, the government agencies responsible in Germany will guarantee the necessary financial resources, the technical competence of their personnel, the required number of personnel as well as an expedient and effective organisation.

The Federal Government will continue to meet Germany's existing international obligations. This particularly applies to the fulfilment of the Convention on Nuclear Safety.

In the Federal Republic of Germany, the principles of a democratic social order, namely the government's responsibility to protect life and health and natural resources needed to sustain life, the separation of powers, the independence of licensing and supervisory authorities and the supervision of administrative actions by independent courts are established by the Basic Law. The legislation, administrative authorities and jurisdiction created specifically for the peaceful use of nuclear energy establish the framework of the system for assuring the protection of life, health and property of the directly employed and the general public from the hazards of nuclear energy and the damaging effects of ionising radiation. In accordance with the legal requirements in the field of nuclear technology, the assurance of safety is given priority over economic considerations. The nuclear safety regulations are in compliance with the internationally accepted safety standards as specified, for example, in the "Safety Fundamentals" of the IAEA.

With the submission of this third report, the Federal Republic of Germany demonstrates that it complies with the Convention on Nuclear Safety. Anyhow, there is still need for action for the future to maintain the safety level of the German nuclear power plants required during the remaining operating lives. Above all, the challenges known for a quite a while, such as ageing of the nuclear power plants, the liberalisation of the electricity market and the risk of a decrease of safety-engineering competence in a field of technology which is phased out, have to be met efficiently.

This report follows the Convention both in structure and content and the associated Guidelines Regarding National Reports. However, it is not only limited to changes since the Second Review Meeting, but has been drafted in a comprehensive form. The numbering of the chapters corresponds to the numbering of the articles in the Convention. Articles that do not contain obligations for contracting parties are not discussed in this document. Each commitment is individually commented on. As suggested in the Guidelines Regarding National Reports, statements made in the report are basically generic in nature, however, plant specific details are presented wherever necessary to support the statement that requirements of the Convention are being met. In accordance with the Guidelines, it is additionally stated at the end of the chapter which changes took place since the previous report, which measures have been taken and which measures will be taken in the future (see also → Future Activities of the Federal Regulator). The history of the use of nuclear energy in Germany is presented in Chapter 6. Here, the research reactors were included additionally. Although research reactors do not represent nuclear installations as defined by the Convention, they have been included in this report because the Federal Government wants to take into account the recommendations from the IAEA "Code of Conduct on the Safety of Research Reactors" from 2004.

In order to demonstrate compliance with the commitments, the relevant national laws, ordinances and standards are referred to, and it is described how the essential safety requirements are met. In this third national report, special emphasis is again put on describing the licensing procedure and state supervision as well as the nuclear safety measures.

The Appendix to this report contains a list of the currently operating and decommissioned nuclear power plants and research reactors, a compilation of design basis and beyond-design basis accidents to be referred to for safety reviews, a survey of design characteristics important to safety for the operating nuclear power plants (nuclear installations as defined by the Convention), sorted according to type and design generation. Also included is a comprehensive list of the legal and administrative provisions of the nuclear safety standards and guidelines which are relevant to the safety of nuclear installations as defined by the Convention and which are referred to in this report.

6 Existing Nuclear Installations

History

Research and development in the field of civil use of nuclear energy has been initiated in Germany in 1955 after the Federal Republic of Germany officially had renounced the development and possession of nuclear weapons and had become a sovereign state. The research and development programme was based on an intensive international co-operation and included the construction of several prototype reactors, the elaboration of concepts for a closed nuclear fuel cycle and for the final storage of radioactive waste in deep geological formations.

In 1955, the Federal Government established the Federal Ministry for Nuclear Affairs and Germany became founder member of EURATOM and the Nuclear Energy Agency (NEA) of the OECD. With the help of US manufacturers, German power utilities began to develop commercial nuclear power plants (Siemens/Westinghouse for PWR, AEG/General Electric for BWR).

In the following years, the West German nuclear research centres were founded:

1956 in Karlsruhe (*Kernforschungszentrum Karlsruhe* KFK),
in Geesthacht (*Gesellschaft für Kernenergieverwertung in Schiffbau und Schiffahrt* GKSS) and
in Jülich (*Kernforschungsanlage Jülich* KFA);
1959 in Berlin (*Hahn-Meitner-Institut für Kernforschung* HMI) and
in Hamburg (*Deutsches Elektronen-Synchrotron* DESY*);*
1969 in Darmstadt (*Gesellschaft für Schwerionenforschung* GSI).

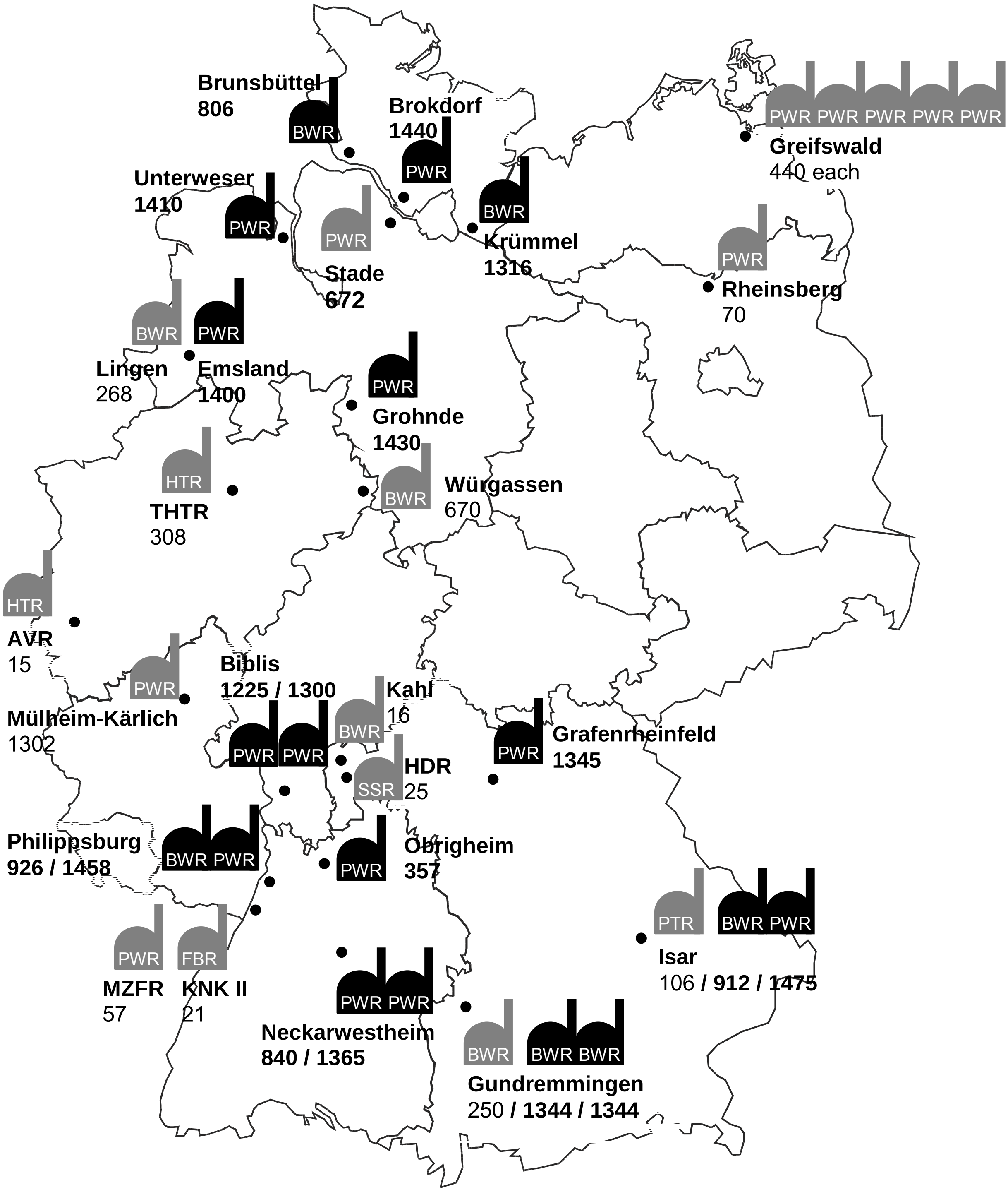
Many universities were equipped with research reactors.

In 1958, the first German nuclear power plant, the 16 MWe experimental nuclear power plant (VAK) in Kahl, was ordered from General Electric and AEG, which entered operation in 1960. The development of reactors in Germany began in 1961 with the order to BBK/BBC for the 16-MWe high-temperature pebble-bed reactor (Arbeitsgemeinschaft Versuchsreaktor (AVR)) in Jülich. It reached criticality in 1966 and has been in operation until 1988. Since then, it is finally shut down. Power reactors with 250-350 MWe and 600-700 MWe were ordered between 1965 and 1970.

After 15 years of German nuclear technology, the German industry received first orders from other countries, the Netherlands (Borssele) and Argentina (Atucha). In 1972, the construction of the pressurised water reactor with the largest capacity world-wide (at that time) was begun (Biblis A, 1,200 MWe) which reached first criticality in 1974. Between 1970 and 1975, three units were ordered per year on the average (Appendix 1). Since then, the share of nuclear energy in the electricity production in Germany is about 30 %.

In 1969, Siemens and AEG founded the Kraftwerk Union (KWU) by merging their respective nuclear activities. Here, the development of German pressurised water reactors began, and it ended after several steps with the standardised 1,300-MWe PWR, the Konvoi. The last nuclear power plants built in Germany were three of these Konvoi plants, which have been commissioned in 1988.

In the Federal Republic of Germany, one prototype was built each for the high-temperature reactor as pebble-bed reactor on the basis of thorium (Thorium High Temperature Reactor – THTR-300) and the fast breeder (SNR-300) with a capacity of 300 MWe each. The THTR-300 in Hamm-Uentrop reached criticality in 1983, and was shut down for decommissioning



12/2003

Legend

PWR	Pressurised Water Reactor
BWR	Boiling Water Reactor
FBR	Fast Breeder Reactor
HTR	High Temperature Reactor
PTR	Pressure Tube Reactor
SSR	Superheated Steam-Cooled Reactor

in operation

shut down

Numbers indicate gross capacity [MWe]

Figure 6-1 **Nuclear Power Plants in Germany**

after only five years of operation. The SNR-300 project in Kalkar was stopped in 1991 without having reached criticality.

The other part of Germany, the former German Democratic Republic (GDR), also began to develop a nuclear programme for the peaceful use of nuclear energy in 1955 and was supported by the Soviet Union. In 1956, the Central Institute for Nuclear Research (ZfK) was founded in Rossendorf near Dresden. There, a research reactor delivered by the Soviet Union was taken into operation in 1957. The first commercial reactor – a 70-MWe pressurised water reactor of Soviet design – was built in Rheinsberg and reached criticality in 1966.

From 1973 to 1979, four pressurised water reactors of the Soviet WWER-440/W-230 type were taken into operation in Greifswald. In 1989, Unit 5 (WWER-440/213) was commissioned. In the course of the German reunification, in-depth safety analyses were performed for the Soviet-type nuclear power plants which showed safety deficiencies compared to the West German regulations. Due to technical and above all economical reasons – mainly the imponderabilities in the licensing procedures for backfitting measures and a decreasing electricity consumption at the same time – no investor was found for the backfitting of the reactors. They were decommissioned. The construction of the Units 6, 7 and 8 (WWER-440/W-213) in Greifswald and the works at the two WWER-1000 units in Stendal were also stopped.

Soon after the euphoria of the fifties and sixties, scepticism towards nuclear energy grew in Germany. More and more citizens offered resistance against the risks of nuclear energy, especially against the further construction of nuclear power plants. Names like Wyhl, Brokdorf, Gorleben, Wackersdorf or Kalkar are synonyms for this protest. At the latest after the Harrisburg accident in 1979 and then finally after the disaster of Chernobyl in 1986, it had become clear that the risks associated with the use of nuclear energy are not only of theoretical nature. Following the declared will of the Federal Government to phase-out nuclear energy, leading to the agreement between the Federal Government and the power utilities of 14 June 2000 (signed on 11 June 2001), the orderly procedure for ending the use of nuclear energy in Germany has begun. The orderly procedure for ending the use of nuclear energy is now the specified object of the meanwhile amended Atomic Energy Act.

Nuclear installations as defined by the Convention

Currently, 18 nuclear power plant units are in operation at 13 different sites producing a total of 21,693 MWe. Appendix 1-1 presents an overview of the nuclear power plants and Figure 6-1 shows the geographical location of the individual sites.

According to the time of their construction, the nuclear power plants with pressurised water reactors reflect four design generations, whereas those with boiling water reactors belong to two different construction lines. The design generations and construction lines of the plants are noted in Appendix 1-1 and will be used throughout the report in the results presented. Several of the basic plant characteristics important to safety and with respect to this classification are presented in Appendix 4. These also illustrate the continuous development in safety technology.

There are two other nuclear power plants being nuclear installations as defined by the Convention. They have been shut down for decommissioning, but at one plant, the nuclear fuel elements have not been removed yet and for both, the authority has not yet given its approval to a decommissioning programme. For the Mülheim-Kärlich nuclear power plant with 1,302 MWe, shut down by court order since 9 September 1988, the plant operator filed an application for decommissioning and dismantling of the plant on 12 June 2001. The fuel elements have completely been removed from the plant. The Stade nuclear power plant with 672 MWe was shut down for decommissioning on 14 November 2003. An application for decommissioning and dismantling of the plant was filed on 23 July 2001. The fuel elements are expected to be removed until the middle of 2005.

Since 1988, nuclear energy covers about one third of the public electricity supply and about 12 % of the entire primary power supply in Germany. In 2003 (2002), the electricity generated by German nuclear power plants amounted to 165.0 (165.0) TWh.

Table 6-1 Average Availability of German Nuclear Power Plants

Year	Time availability %	Energy availability %	Energy utilisation %
1999	91,1	90,2	87,0
2000	91.0	90.6	85.9
2001	91,7	91,4	87,1
2002	85,7	86,0	83,8
2003	87,7	87,0	84,3

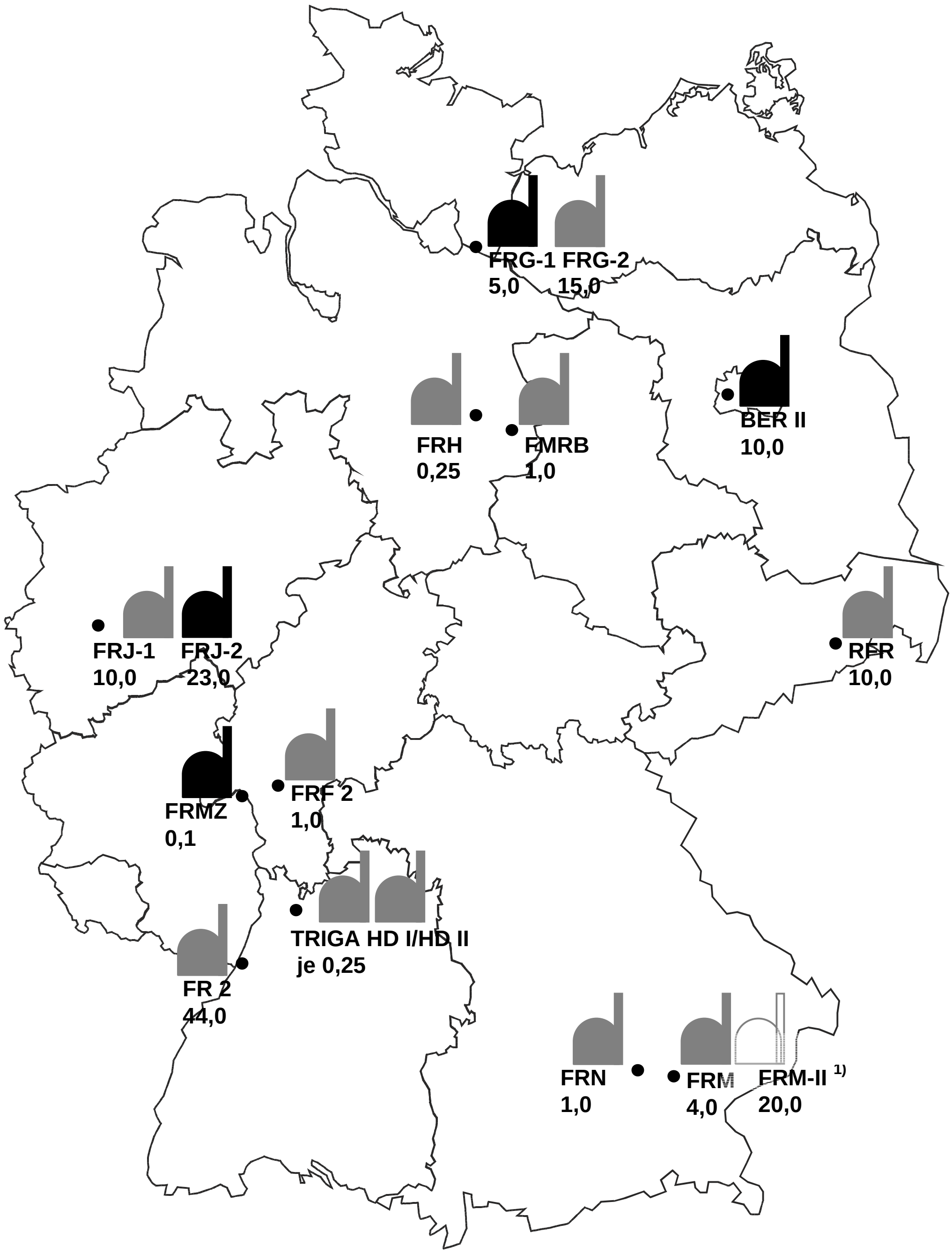
time availability = available operating time / calendar time
energy availability = available energy / nominal energy
energy utilisation = energy generated / nominal energy

In the Federal Republic of Germany, experience was also gained in the field of plutonium recycling in light-water reactors by the use of mixed oxide (MOX) fuel elements. The competent authorities of the *Länder* (federal states) have issued licence permits for the use of MOX fuel elements in ten pressurised water reactors. The individually licensed deployable amounts lie between 9 % and 50 % of the total core inventory. In the case of boiling water reactors, for the two units at Gundremmingen (KRB B and C) licences have been issued to deploy up to 38 % of the core inventory. Further licences have been applied for. To date, MOX fuel elements have been deployed up to 33 % of the core inventory at pressurised water reactors and up to 24 % at boiling water reactors.

Currently, the achieved or targeted discharge burn-ups lie in the order of 40-50 GWd per ton of heavy metal. A number of licensees are either planning, have applied for or have already been issued licence permits to increase the initial enrichment of U-235 and fissile plutonium in MOX fuel elements. It will then be possible to achieve a burn-up of more than 55 GWd per ton of heavy metal. In pressurised water reactors, this may require the use of boric acid enriched in B-10.

Research reactors

As already mentioned in the introduction, account shall also be given here on research reactors, which do not represent nuclear installations as defined by the Convention. This is to comply with the recommendations stated in the "Code of Conduct on the Safety of Research Reactors" of 2004.



Legend

In operation | In decommissioning or decommissioning decided | In construction

Numbers indicate thermal power [MW]

1) First criticality 2 March 2004, under commissioning

Figure 6-2 Research Reactors > 50 kW in Germany

In Germany, four research reactors with a capacity of more than 50 kW and nine small training reactors are in operation (Appendix 2-1). An additional research reactor has been completed and is in the commissioning phase since the beginning of 2004. Eleven research reactors have been decommissioned and are being dismantled, another 21 have already been dismantled completely (Appendix 2-2 and 2-3). The sites are presented in Figure 6-2. The research reactors in operation have a capacity of up to 23 MW, and the one to be commissioned will reach a maximum thermal neutron flux density of $8x10^{14}$ per cm² and s with a capacity of 20 MW. In Germany, research reactors are primarily licensed pursuant to the regulations for power reactors with application limitations according to the physical characteristics, but they are, as the power reactors, also subject to, e.g. the obligations to report in case of reportable events (→ Chapter 19 (vi)). The operators of research reactors are universities and research centres which are financed by the Federal Government, thus being the owner of the research reactors. In so far, costs of operation and decommissioning of research reactors fall within the government's responsibility.

Other nuclear installations

To complete the picture of the utilisation of nuclear energy in Germany, a short survey of the other nuclear installations also outside the scope of the Convention will be presented. Some of these installations have been dealt with in the Report under the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

Altogether, 22 nuclear power plants have been decommissioned or abandoned as project during the construction phase (Appendix 1-2). From these, 12 nuclear power plants are currently being dismantled, two nuclear power plants are in safe enclosure and two plants have already been completely removed. The two nuclear power plants Mülheim-Kärlich and Stade have been shut down for decommissioning, but they are still nuclear installations as defined by the Convention (see above). Six nuclear power plants did never start operation since the projects were abandoned during the construction phase.

The other nuclear installations are facilities of the nuclear fuel cycle and for the treatment and final disposal of radioactive waste. An uranium enrichment plant at Gronau and a fuel element fabrication plant at Lingen are in operation. The pilot reprocessing plant at Karlsruhe has been decommissioned and is in the process of being dismantled. It is intended to vitrify the highly radioactive solutions of fission products still present at this plant at the on-site vitrification plant and, thus, prepare them for final disposal. Further, the following facilities of the nuclear fuel cycle are in the decommissioning phase: the fuel fabrication plant NUKEM in Hanau, the two Siemens fuel fabrication plants with MOX and uranium sections – also in Hanau – and the facility for molybdenum production at the Rossendorf research site. A number of facilities in operation serve the purpose of interim storage of fuel elements as well as the treatment, conditioning and interim storage of radioactive waste. The licensing procedure for the pilot spent fuel conditioning plant was completed in December 2000 with the granting of the third partial construction licence including the operation licence. According to the agreement between the Federal Government and the power utilities of 11 June 2001, the use of the plant shall be limited to the repair of defective containers.

For the final storage of radioactive waste (except nuclear fuels), the Morsleben repository had been operated until September 1998. The plan approval procedure for the Konrad mine repository has been completed after 20 years with the plant approval decision. The exploration works in the Gorleben mine were interrupted in 2000 for at least 3, and at most 10 years.

Results from the safety review of nuclear installations

All currently operated nuclear power plants, as listed in Appendix 1-1, have an unlimited operating licence, but the amended Atomic Energy Act limits the operating life of the plants according to the remaining electricity output still to be generated. The licences for the nuclear power plants were only granted if the applicant had proven to the nuclear licensing authority that the required protection against according to the state of the art in science and technology at that time was achieved by the plant design and construction and the on-site provisions applied for.

Within the framework of the regulatory system for the utilisation of nuclear energy and, especially, of the regulatory supervision, safety assessments are performed both, continuously and on special occasions, as well as safety reviews as a supplement. Whenever new safety-relevant findings are available, the necessity and adequacy of possible improvements are checked. By this approach, a progressive improvement of plant safety is achieved. Deficiencies identified during safety reviews are eliminated in accordance with the regulations within the frame of regulatory supervision (→ Chapter 14). The safety assessments within the frame of regulatory supervision represent reviews according to Article 6 of the Convention.

Over the past years, numerous improvements have been realised (→ Chapter 14 (ii)), in particular in the area of beyond-design basis accidents (→ Chapter 18 (1)). As a result, the safety level has been improved also at older nuclear power plants.

In summary, the German Federal Government ascertains that the prerequisites for a safe operation of the German nuclear power plants for their remaining operating times until ending the use of nuclear energy in Germany are given.

Chapter 6: Progress and changes since 2001

Research reactors were included in the report in order to comply with the recommendations stated in the "Code of Conduct on the Safety of Research Reactors".

Chapter 6: Future activities

Continuation of the safety assessments as it is common practice within the framework of supervision and the comprehensive safety reviews to be performed on the statutory dates.

7 Legislative and Regulatory Framework

7 (1) Legislative and Regulatory Framework

In accordance with the federal structure of the Federal Republic of Germany, its Constitution (Article 74 (1) 11a of the Basic Law [1A-1]) bestows upon the Federal Government the responsibility for legislation and regulation regarding "production and utilisation of nuclear energy for peaceful purposes, construction and operation of facilities serving such purposes, protection against hazards arising from the release of nuclear energy or ionising radiation and disposal of radioactive substances."

The Atomic Energy Act [1A-3] was promulgated December 23, 1959, right after the Federal Republic of Germany had officially renounced any use of atomic weapons. Since then, it has been updated and amended several times.

In Germany, the legislation and its execution must also take into account any binding requirement from regulations of the European Union. With respect to radiation protection there are, e.g. the EURATOM Basic Safety Standards [1F-18] for the protection of the health of workers and the general public against the dangers arising from ionising radiation. These were issued on the basis of Article 30 ff. of the EURATOM Treaty [1F-1]. In accordance with Article 77 ff. of the EURATOM Treaty, any utilisation of ores, source material and special fissile material is subject to surveillance by the European Atomic Energy Community.

7 (2i) Nuclear Safety Regulations

Acts and ordinances, in particular, the Atomic Energy Act

The Atomic Energy Act comprises the general national regulations for the safety of nuclear installations in Germany and constitutes the basis for the associated ordinances. Its purpose after the amendment of 2002 is to end the use of nuclear energy in a structured manner and to ensure on-going operation up until the date of discontinuation, as well as to protect life, health and property against the hazards of nuclear energy and the detrimental effects of ionising radiation and, furthermore, to provide for the compensation for any damage and injuries incurred. It also has the purpose of preventing the internal or external security of the Federal Republic of Germany from being endangered by the utilisation of nuclear energy. Another purpose of the Atomic Energy Act is to ensure that the Federal Republic of Germany meets its international obligations in the field of nuclear energy and radiation protection.

With respect to the protection against the hazards from radioactive materials and to the supervision of their utilisation, the Atomic Energy Act requires that the construction and operation of nuclear installations is subject to regulatory licensing.

Licences for new facilities fission of nuclear fuel for the commercial production of electricity will no longer be granted. A prerequisite for the licensing of the existing plants was, above all, that the measures regarding precaution against damage must comply with the state of the art in science and technology. This was a tightening of the requirement to comply with the state of the art applied in the German technical safety regulations or the even less stringent generally acknowledged technical standards. Thus, the licensing of a nuclear installation required a degree of precaution against damages that is considered necessary with regard to safety on the basis of latest assured scientific findings. Today, these requirements for the licensing of nuclear power plants are only significant for modifications of existing plants. According to a regulation introduced into the Atomic Energy Act in 2002, the operators of nuclear power plants have to perform special safety reviews of the plants and to submit their results to the supervisory authority.

A number of ordinances in the field of nuclear energy have been promulgated on the basis of the Atomic Energy Act. The most important pertain to:

- radiation protection [1A-8],
- the licensing procedure [1A-10],
- the checking of personnel reliability, and
- the reporting of reportable events [1A-17].

The safety provisions and regulations of the Atomic Energy Act and of the associated ordinances are put into concrete terms by general administrative provisions, by regulatory guidelines, by safety standards of the Nuclear Safety Standards Commission (KTA), by recommendations from the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK), and by conventional technical standards.

The Atomic Energy Act, which regulates the safety of the installations, is supplemented by the Precautionary Radiation Protection Act of 1986 [1A-5], which came about in the wake of the reactor accident at Chernobyl and specifies the tasks of environmental monitoring also in the case of events with significant radiological effects (→ Chapter 15 and 16).

General administrative provisions

At a legal level just below that of acts and ordinances, general administrative provisions present binding regulations for the actions of the authorities. The following provisions are relevant with respect to nuclear technology and pertain, specifically, to:

- the calculation of radiation exposure during operating conditions of nuclear power plants [2-1],
- the radiation passport [2-2],
- the environmental impact assessment [2-3], and
- the environmental monitoring [2-4].

Regulatory guidelines

After having consulted the *Länder* and generally with their consent, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), issues guidelines. These guidelines serve the specification of technical and administrative questions arising from the licensing and supervisory procedure in detail (→ Chapter 8 (1)). They describe the view of the BMU on general questions related to nuclear safety and the administrative practice, and serve as orientation for the *Länder* authorities regarding the execution of the Atomic Energy Act. However, these guidelines are not binding for the *Länder* authorities in contrast to the general administrative provision. Currently, about 50 guidelines exist in the field of nuclear technology (see Appendix 5 under *Bekanntmachungen* [3-...]). These guidelines pertain to:

- general safety requirements ("Safety Criteria"),
- details on the design basis accidents to be considered in the design,
- dispersion calculations,
- accident management measures to be planned by the licensee with regard to postulated severe accidents,
- measures regarding disaster control in the vicinity of nuclear installations,
- measures against malevolent acts or other illegal interference by third parties,

- radiation protection during maintenance work,
- general documentation,
- documents to be supplied with the application for a licence, and
- qualification of the personnel in nuclear installations.

Recommendations of the RSK and SSK, RSK-Guidelines

Regarding licensing and supervision procedures, the recommendations of the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK) play an important role. Both of these expert commissions advise the Federal Ministry for the Environment in questions related to nuclear safety and radiation protection (→ Chapter 8 (1)).

In the last version of the RSK-Guidelines of 1996 [4-1], the Reactor Safety Commission summarised the safety requirements to be fulfilled regarding the design, construction and operation of a nuclear power plant. The RSK uses these guidelines as a basis of its consultations and recommendations. The RSK deviates from them if the state of the art in science and technology has meanwhile changed in specific areas.

KTA Safety Standards

Detailed and concrete technical requirements are contained in the safety standards of the Nuclear Safety Standards Commission (KTA), (→ Chapter 8 (1)). In accordance with its statutes, the KTA specifies requirements wherever "experience leads to a uniform opinion of the experts within the groups of manufacturers, construction companies, and licensees of nuclear installations, and of the expert organisations and the authorities." On the basis of the regular reviews and eventual amendment of the issued safety standards at intervals of no more than five years, the standards are adjusted to the state of the art in science and technology. In themselves, KTA safety standards are not legally binding. However, due to the nature of their origin and their high degree of detail, they have a far-reaching practical effect. Until today (12/03), the KTA has issued a total of 91 safety standards and 2 draft standards; additional 9 draft standards are in preparation and 22 safety standards are in the process of being revised.

The KTA safety standards pertain to

- administrative provisions,
- industrial safety (specific additional requirements within the field of nuclear technology),
- civil engineering,
- nuclear and thermal-hydraulic design,
- issues regarding materials,
- instrumentation and control,
- monitoring of radioactivity, and
- other provisions.

Quality assurance occupies a major part in this endeavour; this aspect is treated in most of the safety standards. The term quality assurance as used in the KTA safety standards also comprises the area of ageing which, today, is internationally treated as a separate issue (→ Chapter 13).

Historically, the KTA safety standards have been developed on the basis of applicable German technical standards and regulations and on the American nuclear safety standards.

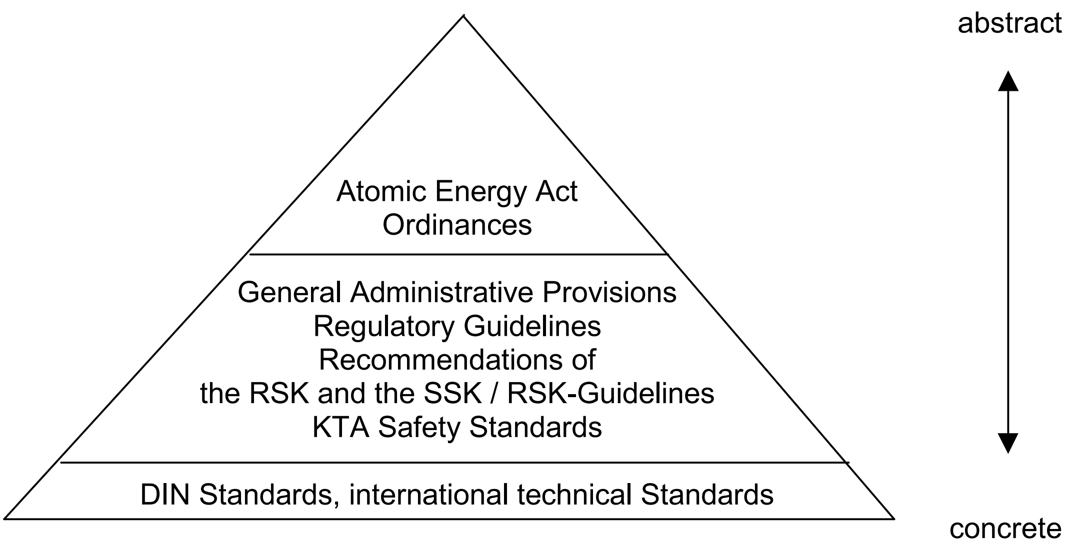
The ASME-Code (Section III) was used as a model for specifying the requirements regarding the design and construction of components.

Conventional technical standards

Furthermore, conventional technical standards, in particular the national standards of the German Institute for Standardisation (DIN) and also the international standards of ISO and IEC, are applied just as they are in the design and operation of all technical installation, as far as the conventional standards correspond to the state of the art in science and technology.

Overall picture of nuclear rules and regulations

The German nuclear rules and regulations may be seen as hierarchically structured in the form of a pyramid.



Nuclear regulations, except laws, ordinances and general administrative provisions, only have regulatory relevance due to the legal requirement regarding the state of the art in science and technology. According to legal practice, it can be presumed that the nuclear rules and regulations reflect this state appropriately. Therefore, an acknowledged further development of the state of the art in science and technology pushes aside the application of a standard which has become obsolete by this development without the necessity to suspending this standard. Thus, the dynamic improvement of the safety requirements requested by law is not bound to the formal development of standards.

In this report, reference will be made to the contents of the individual regulations as the corresponding articles of the Convention are dealt with. Appendix 5 "Reference List of Nuclear Rules and Regulations" lists the current regulations applicable to nuclear installations in the mentioned hierarchical order. Apart from some texts on physical protection, all of the listed regulations are accessible to the public. The regulations are published in official publications of the Federal Government. Since then, they have been applied in all nuclear regulatory licensing and supervisory procedures and have been further developed, where required, in accordance with the state of the art in science and technology

Revision of the nuclear rules and regulations

Parts of the German rules and regulations date back to the seventies and eighties of the 20th century, whereas at the international level, the rules and regulations have continuously been further developed. Since then, the German rules and regulations have not completely been reviewed for differences to the international rules and regulations.

In order to facilitate the assignment and integration of international rules and specifications into the national regulations, the KTA had initiated the "KTA 2000" work programme. The main objective of this project was to present the requirements of the rules and regulations with regard to nuclear safety (design, construction and operation of nuclear power plants) consistently and hierarchically structured in form of a regulatory pyramid as Basic KTA-Guidelines, Basic Standards of the KTA and Technical KTA Safety Standards. First drafts were passed at the KTA meeting in June 2001. The BMU had set concrete standards for the KTA 2000 project, particularly with regard to the implementation of safety requirements according to the state of the art in science and technology including their further development. According to the conclusion of the BMU, these requirements had not been fulfilled so that it was not possible to complete this project successfully.

For this reason, the international nuclear rules and regulations are being compared with the current rules and regulations in Germany to be able to assess the significance of the differences between these rules and regulations, particularly with regard to the damage precaution required according to the state of the art in science and technology during the remaining operating lives of the nuclear power plants in Germany. The comparison will also show which revisions and amendments of the German nuclear rules and regulations are necessary. In this regard, the decisive benchmark rules and regulations are those of the IAEA. The IAEA Safety Standards represent the basically internationally acknowledged reference.

Moreover, the BMU participates in the working group on harmonisation of the Western European Nuclear Regulators' Association (WENRA). The objective is the establishment of uniform and high reference levels for the safety of the nuclear power plants currently operated in the European Union. These reference levels are currently being established on the basis of the IAEA Safety Standards and will be supplemented by more stringent European regulatory requirements or best practice. After that, compliance with the reference levels shall be determined by each member state at the national level. Deviations shall be assessed and serve as a basis for the development of corresponding improvement measures.

The overall objective of the BMU is to update the higher-level rules and regulations according to the state of the art in science and technology. For this purpose, reference is made, among other things, to the international rules and regulations, practical experiences from the application of the existing German nuclear rules and regulations, findings from the safety-related assessment of events and further operating experiences, as well as to the licensing and supervisory practice. In this respect, results from the KTA 2000 project shall also be taken into account. Further, the hierarchical interaction of all relevant safety-related requirements of the nuclear rules and regulations developed over many years shall be presented in a basic paper. The modular structure of the draft wording of rules and regulations enables a dynamic process regarding the revision of the rules and regulations. In this respect, the defence-in-depth concept is generally referred to as safety (→ Chapter 18 (i)). The basic paper and the updated general safety requirements will be available by the middle of 2005.

The BMU will ensure that all relevant parties will be involved in the process of the revision of the rules and regulations.

7 (2ii) System of Licensing

General provisions

According to the applicable law, licences for the construction of nuclear power plants are no longer issued. Licensing procedures are therefore only performed for the modification of existing installations and for decommissioning. Thus, the procedure described below represents - as far as the construction of installations is concerned - the legal regulations applicable to the construction of installations still in operation today. For research reactors, these are applicable without any restrictions.

The licensing of nuclear installations is regulated in the Atomic Energy Act [1A-3]. According to Section 7 of this Act, a licence is required for the construction, operation or any other holding of a stationary installation for the production, treatment, processing or fission of nuclear fuel, or for essentially modifying such installation or its operation. Such a licence may only be granted if the licence prerequisites stated in Section 7 of the Atomic Energy Act are fulfilled by the applicant:

- necessary precautions against damage according to the state of the art in science and technology,
- trustworthiness and technical qualification of the responsible personnel,
- necessary knowledge of the otherwise engaged personnel regarding safe operation of the installation,
- protection against malevolent acts or other illegal interference by third parties,
- necessary financial security with respect to legal liability for paying damage compensation,
- consideration of public interests with respect to environmental impacts.

It must also be considered that any handling of radioactive material - and this includes the construction and operation of nuclear power plants - is subject to the requirements regarding supervision and protection that are specified in a legally binding way in the Radiation Protection Ordinance [1A-8]. The Radiation Protection Ordinance regulates, among others, the reporting by name of the responsible persons of the licensee and the dose limits of radiation exposure during operating conditions for the personnel engaged at the plant and for the general public. Furthermore, it contains planning values for the design of nuclear power plants against design basis accidents.

When issuing a licence for a nuclear power plant, obligations may be imposed if it seems necessary to ensure safety. Any act of operating, otherwise holding, essentially modifying or decommissioning a nuclear installation without the required corresponding licence permit is punishable by law [1B-11].

The licensing of nuclear installations lies within the responsibility of the individual *Länder*. The *Länder* have ministries that are responsible for licensing of construction, operation, essential modification and decommissioning of nuclear power plants (Table 8-1). The Federal Government (represented by the BMU) exercises its supervision on the implementation of the Atomic Energy Act and Radiation Protection Regulations regarding lawfulness and expediency via the *Länder* (Federal Regulator). This also includes the right to issue binding directives on factual and legal issues in each individual case.

The actual details and procedure of licensing in accordance with the Atomic Energy Act are specified in the Nuclear Licensing Procedure Ordinance [1A-10]. It deals specifically with the application procedure, with the submittal of supporting documents, with the participation of the general public and with the possibility to split the procedure into several licensing steps (partial licences). It deals, furthermore, with the assessment of environmental impacts [1F-13] and with the consideration of other licensing requirements (e.g. regarding the possible release or discharge of non-radioactive pollutants into air or water (→ Chapter 17 (ii)).

In accordance with Section 20 of the Atomic Energy Act, the competent authorities may involve authorised experts in technical or scientific questions related to regulatory licensing and supervision, who have, similar to the authorities, the right of inspections and requesting information. However, the authority is not bound by the assessments of their authorised experts (→ Chapter 8 (1)).

The interaction of the different authorities and organisations involved in the nuclear licensing procedure and the participation of the general public is shown in Figure 7-1. This creates a broad and differentiated base for making decisions accounting for the considerations of all matters concerned.

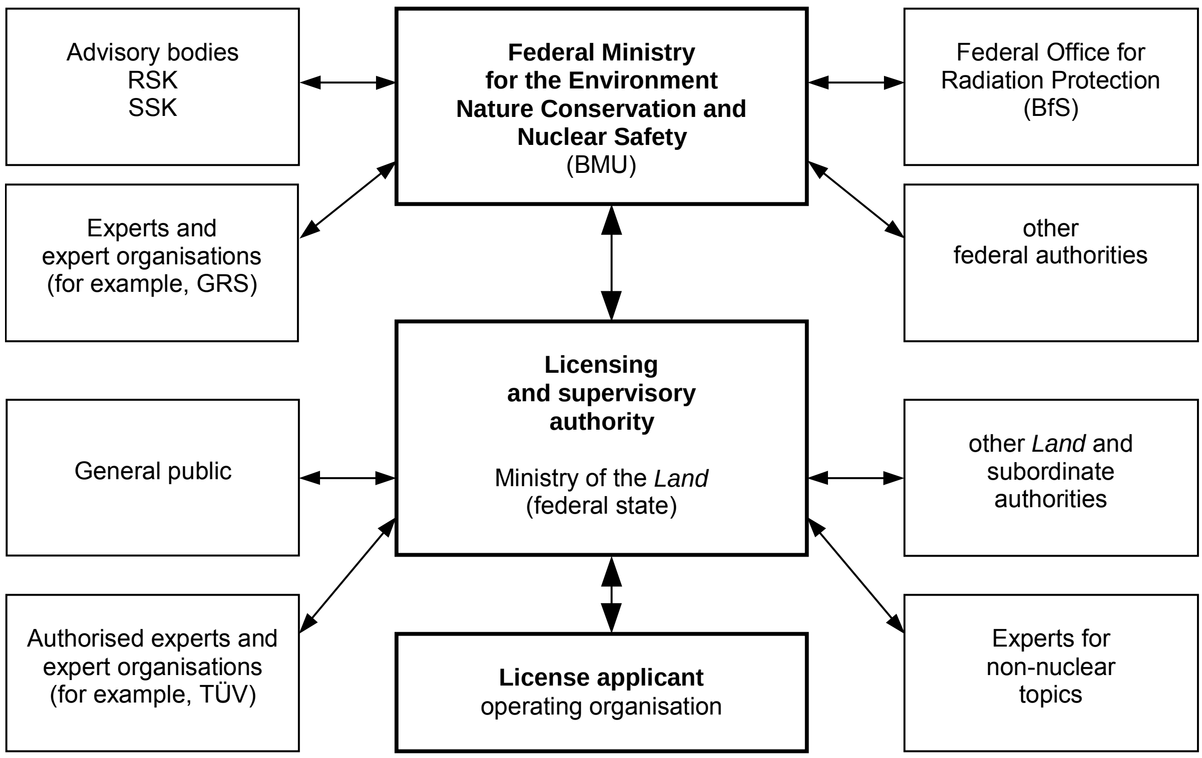


Figure 7-1 Participants in the Nuclear Licensing Procedure

The Paris Convention on Third Party Liability in the Field of Nuclear Energy [1E-11] and the Joint Protocol [1E-12] have been implemented into national nuclear liability legislation with direct applicability (self-executing) and are supplemented by it. For damages due to a nuclear event caused by a nuclear installation, the operator generally has unlimited liability. In order to fulfil the obligation to pay any damages, the operator has to provide financial security which may amount – according to the Atomic Energy Act as amended in 2002 – to € 2.5 billions; details on this issue are regulated by an ordinance [1A-11]. Financial security may be ensured by liability insurance or other financial means – e.g. private warranty. Where the legal liability to pay damages is not covered by the financial security provided or cannot be fulfilled with it, the Atomic Energy Act grants the operator the right against the Federal

Government and the *Land* issuing the licence to be exempted from this liability to pay damages. The maximum indemnity carried by the Federal Government amounts to € 2.5 billions.

Details of the nuclear licensing procedure

Licence application

The individual power utilities or their subsidiaries are the licence applicants for the construction and operation of a nuclear power plant. They submit a written licence application to the competent licensing authority of that *Land* in which they intend to erect the nuclear installation. The licence application is accompanied by documents that are stated in the Nuclear Licensing Procedure Ordinance [1A-10] and specified in guidelines. An important document is the safety analysis report (→ Chapter 14 (i)) which describes the plant, its operation and the related effects, including the effects of design basis accidents as well as the associated precautionary measures. It contains site plans and assembly drawings. In fulfilment of the licence prerequisites, further documents are to be submitted, e.g. supplementary plans, drawings, descriptions as well as information regarding

- the protection of the plant against malevolent acts or other illegal interference by third parties,
- the applicant and those holding responsible positions, including their qualification and trustworthiness,
- the necessary knowledge of the personnel otherwise engaged in the operation of the plant,
- the safety specification,
- the financial security,
- the type of residual radioactive material and its disposal,
- the intended environmental protection measures.

In addition, with respect to public participation, a brief description of the planned installation is to be submitted with the application that includes information on the probable effects on the general public and environment in the vicinity of the installation.

Examination of the application

On the basis of the submitted documents, the licensing authority examines whether or not the licence prerequisites have been met. All federal, *Länder*, local and other regional authorities whose jurisdiction is involved shall take part in the licensing procedure. These are, in particular, authorities responsible under the building code, the water code, for regional planning and for off-site disaster control. Due to the large scope of the safety issues to be examined, it is common practice to engage expert organisations to support the licensing authority in the evaluation and examination of the application documents. In their expert analysis reports they explain whether or not the requirements regarding nuclear safety and radiation protection have been met. The role of the expert organisation is strictly advisory in nature.

Within the frame of federal executive administration, the licensing authority of the individual *Land* also involves the BMU. In performing its function of federal supervision, the BMU consults the Reactor Safety Commission, the Commission on Radiological Protection and in many cases the Gesellschaft für Anlagen- und Reaktorsicherheit for advice and technical support. The BMU states its position to the competent licensing authority. This federal position is binding for the decision of the licensing authority.

Participation of the general public

The licensing authority also involves the general public in the licensing procedures, above all for direct protection of the citizens who might be affected by the planned installation. The Nuclear Licensing Procedure Ordinance [1A-10] includes regulations concerning:

- the public announcement of the project and public disclosure of the application documents at a suitable location near the site for a period of two months, including the request for raising any objections within the presentation period.
- the holding of a public hearing where the objections are discussed between licensing authority, licence applicant and the persons who have raised the objections.

The licensing authority acknowledges all of the objections in its decision making process and states the reasons for the decision.

Environmental impact assessment

The Act on the Assessment of Environmental Impacts [1F-12] in conjunction with the Nuclear Licensing Procedure Ordinance specify the requirement for an environmental impact assessment and its procedure within the nuclear licensing procedure for the construction, operation and decommissioning of a nuclear power plant or for an essential modification of the plant or its operation. The competent authority performs a final evaluation of the environmental impacts on the basis of the requirements in nuclear and radiation protection regulations. This final evaluation is the basis for the decision about the permissibility of the project with regard to achieving an effective environmental protection.

Licensing decision

The final decision of the licensing authority is based on the entirety of application documents, evaluation reports by the authorised experts, the statement by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, the statements by the authorities involved and the findings from objections raised in the public hearing. Prerequisite for the legality of this decision is that all procedural requirements of the Nuclear Licensing Procedure Ordinance are fulfilled. Action can be brought against the decision of the licensing authority before the administrative courts.

7 (2iii) Regulatory Inspection and Assessment (Supervision)

Over their entire lifetime, from the start of construction to the end of decommissioning with the corresponding licences, nuclear installations are subject to continuous regulatory supervision in accordance with the Atomic Energy Act and accessory nuclear ordinances. Also regarding the supervisory procedure, the *Länder* act on behalf of the Federal Government (→ Chapter 7 (2ii), i.e. the Federal Government again has the right to issue binding directives on factual and legal issues in each individual case. Just as in the licensing procedure, the *Länder* are assisted by independent authorised experts.

As in licensing, the supreme objective of the regulatory supervision of nuclear installations is to protect the general public and the people engaged in these installations against the hazards connected with the operation of the installation.

The supervisory authority pays particular attention to

- the fulfilment of the provisions, obligations and ancillary provisions imposed by the licensing notices,

- the fulfilment of the requirements of the Atomic Energy Act, the nuclear ordinances and the other nuclear safety standards and guidelines, and
- the fulfilment of any supervisory order.

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

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

The operators of nuclear power plants have to supply written operating reports to the supervisory authorities at regular intervals. These include data on the operating history, on maintenance measures and inspections, on radiation protection and on radioactive waste material. Any events that are relevant to safety and physical protection must be reported to the authorities [1A-17]. The regulations and procedures regarding reportable events and their evaluation are described in Chapter 19 (vi)-(vii).

7 (2iv) Enforcement of Regulations and Provisions

The enforcement of applicable regulations in the nuclear field is supported by certain measures contained in the Penal Code [1B-11], in the Atomic Energy Act [1A-3] and the nuclear regulatory ordinances in case of any violations.

Criminal offences

Any violation that must be considered as a criminal offence is dealt with in the Penal Code. Imprisonment or fines are imposed on anyone who, for example:
- operates, otherwise holds, changes or decommissions a nuclear installation without the required licence,
- knowingly constructs a defective nuclear installation,
- handles nuclear fuel without the required licence,

- releases ionising radiation or causes nuclear fission processes that can damage life and limb of other persons,
- procures or manufactures nuclear fuel, radioactive material or other equipment for himself with the intent of performing a criminal offence.

Administrative offences

The Atomic Energy Act and the accessory ordinances deal with administrative offences and provide for the imposition of fines on the acting persons. An administrative offence is committed by anyone who:
- erects a nuclear installation without a licence permit,
- acts in violation of a regulatory order or provision,
- handles radioactive material without a valid licence permit,
- as the ultimately responsible person fails to see to it that the protective and surveillance regulations of the Radiation Protection Ordinance are fulfilled.

The Atomic Energy Act and the accessory ordinances require that the persons are named who are ultimately responsible for the handling of radioactive material, for the operation of nuclear installations or for their supervision. A person committing an administrative offence is personally liable for a fine up to € 50,000. A legally effective fine against a person may put in question the personal trustworthiness that was a prerequisite for the licence and may, therefore, require the replacement of this person in his position of responsibility (→ Chapter 9).

Enforcement by regulatory order, particularly in urgent cases

In the case of non-compliance with respect to legal provisions or to requirements of the licence permit, and also in case of potential danger to life, health or property, the competent nuclear licensing and supervisory authority is authorised by Section 19 of the Atomic Energy Act to issue orders stating:
- that protective measures must be applied and, if so, which ones,
- that radioactive material must be stored at a place prescribed by the authority, and
- that the handling of radioactive material, the construction and operation of nuclear installations must be interrupted or temporarily - in case of lack or revocation of the licence permanently - be suspended .

Enforcement by modification or revocation of the licence

Under certain conditions, stipulated in Section 17 of the Atomic Energy Act, obligations for ensuring safety may be decreed by the nuclear licensing and supervisory authority even after a licence has been granted. In case a considerable hazard is suspected from the nuclear installation endangering the persons engaged at the plant or the general public, and cannot be removed within a reasonable time by appropriate measures, then the licensing authority has to revoke the issued licence. A revocation is also possible if prerequisites for the licence permit cease to be met at a later time or if the licensee violates legal regulations or decisions by the authorities.

Experience

As a result of the intense regulatory supervision carried out in Germany in the course of design, erection, commissioning, operation and decommissioning of nuclear installations (→ Chapter 7 (2iii)), any inadmissible condition is usually detected at an early stage before the possible legal actions such as imposed obligations, orders, administrative offence procedures and criminal proceedings have to be taken.

Chapter 7: Progress and changes since 2001

– The amendment of the Atomic Energy Act regarding its purpose entered into force in 2002.
– A revision of the higher-level nuclear rules and regulations by the BMU has been initiated.

Chapter 7: Future activities

– Within the framework of its revision of the rules and regulations, the BMU initiated a comparison of the national nuclear rules and regulations with IAEA standards, which is currently being performed, to identify potential needs for improvement of the current German rules and regulations or the national safety practice and to take them into account when revising the German rules and regulations.
– The participation in the WENRA working group for the establishment of common reference levels for the safety of the nuclear power plants operated in the EU will be continued.

8 Regulatory Body

8 (1) Authorities, Committees and Organisations

Germany is a federal republic. Unless otherwise specified, the execution of federal laws lies within the responsibility of the federal states, the *Länder*. In the case of the use of nuclear energy, where it is particularly important that laws are executed in a uniform manner across the Federation, the order for the *Länder* is that they execute the laws acting as agents of the Federation (federal executive administration). This means that in executing the Atomic Energy Act and its associated ordinances, the *Länder* are under the supervision of the Federation with regard to the lawfulness and expediency of their actions and are subject to the directives issued by the Federal Government (Article 85 Basic Law, Section 24 Atomic Energy Act).

The nuclear licensing and supervisory authorities are state ministries of those *Länder* in which the site of the nuclear installation is located (→ Chapter 7 (2ii) and (2iii)). The federal supervisory authority is the BMU. Table 8-1 lists the nuclear licensing and supervisory authorities of those *Länder* in which the sites of the nuclear installation as defined by the Convention are located.

Table 8-1 The *Länder* Licensing and Supervisory Authorities for Nuclear Installations According to the Convention

<table>
<tr><th>Land</th><th>Nuclear Installations</th><th>Licensing Authority</th><th>Supervisory Authority</th></tr>
<tr><td>Baden-Württemberg</td><td>Obrigheim
Neckarwestheim 1
Neckarwestheim 2
Philippsburg 1
Philippsburg 2</td><td>Wirtschaftsministerium after consultation with Ministerium für Umwelt und Verkehr und Innenministerium</td><td>Ministerium für Umwelt und Verkehr</td></tr>
<tr><td>Bayern</td><td>Isar 1
Isar 2
Grafenrheinfeld
Gundremmingen B
Gundremmingen C</td><td>Staatsministerium für Umwelt, Gesundheit und Verbraucherschutz
in agreement with Staatsministerium für Wirtschaft, Infrastruktur, Verkehr und Technologie</td><td>Staatsministerium für Umwelt, Gesundheit und Verbraucherschutz</td></tr>
<tr><td>Hessen</td><td>Biblis A
Biblis B</td><td colspan="2">Ministerium für Umwelt, ländlichen Raum und Verbraucherschutz</td></tr>
<tr><td>Niedersachsen</td><td>Stade *)
Unterweser
Grohnde
Emsland</td><td colspan="2">Umweltministerium</td></tr>
<tr><td>Rheinland-Pfalz</td><td>Mülheim- Kärlich *)</td><td colspan="2">Ministerium für Umwelt und Forsten</td></tr>
<tr><td>Schleswig-Holstein</td><td>Brunsbüttel
Krümmel
Brokdorf</td><td colspan="2">Ministerium für Soziales, Gesundheit und Verbraucherschutz</td></tr>
</table>

*) decommissioning applied for

Federal States Committee for Nuclear Energy

In the interest of a uniform execution in all areas of administrative execution of the Atomic Energy Act and for the preparation of amendments of legal and administrative provisions, the federal structure of the Federal Republic of Germany requires continuous co-ordination between the Federal Government and the *Länder*. This also applies – notwithstanding the right of the federal supervisory authority to issue directives in individual cases - to the Atomic Energy Act executed within the frame of federal executive administration.

In the field of nuclear law, a committee of the Federal Government and the *Länder*, the Federal States Committee for Nuclear Energy (LAA) was therefore founded for the general co-ordination needs. This committee is made up of representatives of the Federal Environment Ministry, which chairs the LAA, and of the competent *Länder* authorities. The LAA discusses in depth all relevant issues of legislation and general legal execution, especially safety issues. The committee reaches its decisions usually by mutual consent. In case of a technical or legal dissent, such cases are decided outside the LAA by the federal supervisory authority. The LAA consists of the General Committee and four subordinate Technical Committees on the issues of Law, Nuclear Safety, Radiation Protection, and Fuel Cycle. The Technical Committees dispose of permanent or ad hoc working groups according to requirements. The General Committee, Technical Committees and the permanent working groups meet at least twice per year, or more often if required.

In the area of legislation, the LAA is an important instrument of early and comprehensive involvement of the *Länder* which supplements the formal right of participation of the *Länder* in the legislative procedure of the German Federal Council (*Bundesrat*).

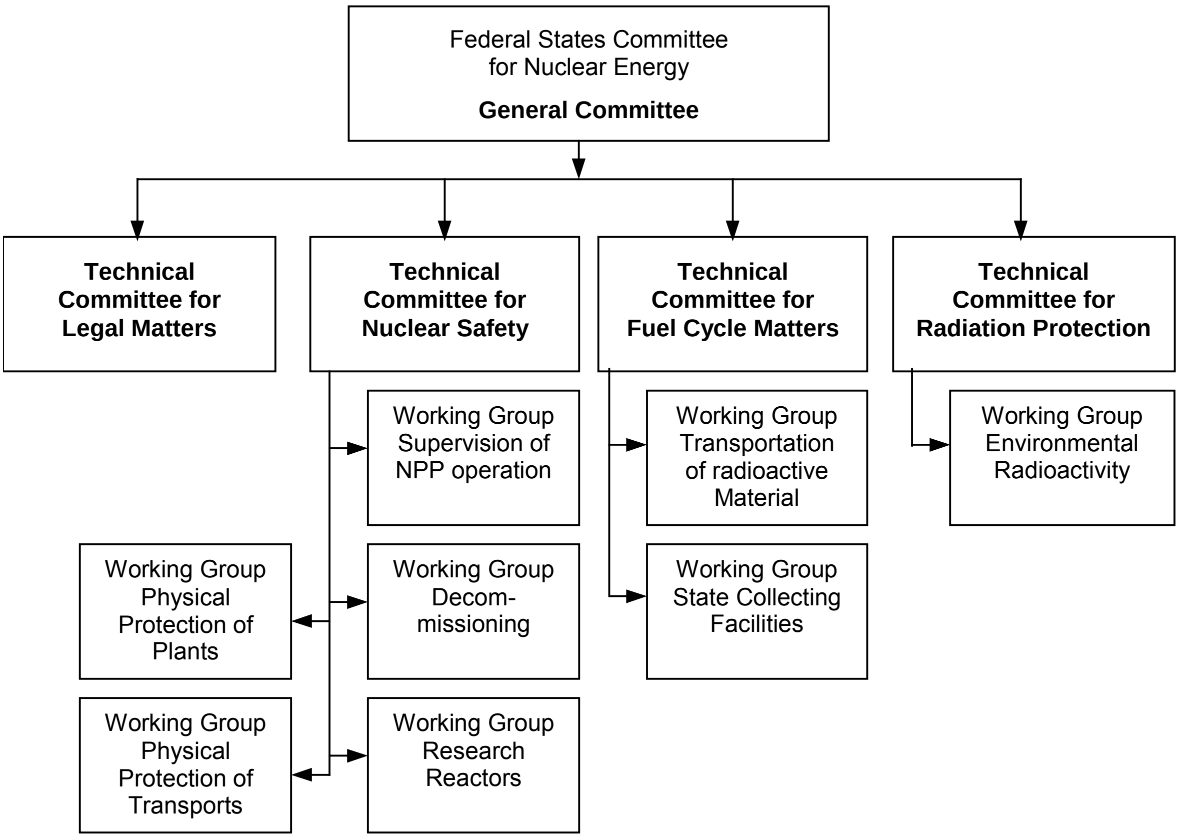


Figure 8-1 Federal States Committee for Nuclear Energy

Personnel and financing

No limits are specified for the number and costs of authority personnel involved in licensing and supervisory activities. These costs strongly depend on whether, and how many, nuclear power plants are subject to licensing and supervision in each individual *Land*. Regulatory supervision, including the activities of authorised experts, requires an annual personnel deployment of 30 to 40 person years per year and nuclear power plant unit. The funds available to the authorities for their own personnel and for the consultation of external experts are allotted by the *Bundestag* (the German Federal Parliament) and *Länder* parliaments in their respective annual budgets; the project-related costs of licensing and supervision are charged to the applicants and licensees.

The licensee of a nuclear power plant is liable for the costs of the licence permits issued and for the associated supervisory activities of the *Länder*. These costs are payable to the public treasury. The overall costs of the licences for construction and operation are set at 2 per mil of the construction costs. A modification requiring a licence permit will cost between € 500 and € 500,000. The costs for supervisory activities are charged on the basis of the individual activity and range between € 25 and € 250,000. The licence applicant or licensee also carries the costs charged as reimbursements for the authorised experts.

From the budget of the BMU, the Federal Government currently finances the federal supervisory activities in the field of nuclear safety to the amount of about € 23 millions each year. The activities include evaluation of operating experience, safety investigations, development of advanced requirements for nuclear installations, and handling of specific issues regarding the licensing and supervision of nuclear power plants. Another approximately € 9 millions are spent on studies in the field of radiation protection each year. These amounts do not include the cost of personnel of the BMU and the BfS.

The Federal Republic of Germany participates in the world-wide efforts to further develop the safety of nuclear power plants by performing independent safety research on a national level. The Federal Government currently provides approximately € 16 millions annually for reactor safety research. This research deals, among others, with experimental or analytical studies of the plant behaviour of light water reactors under accident conditions, the safety of pressure retaining components, core meltdown, human factors, non-destructive early detection of damage for materials difficult to inspect, and the development of probabilistic safety analysis methods.

Maintaining competence

Safe operation of the nuclear power plants has to be ensured for their remaining operating lives. An essential condition for that is an efficient and competent nuclear regulatory supervision. To ensure this, the government agencies responsible in Germany will guarantee the necessary financial resources, the technical competence of their personnel, the required number of personnel as well as an expedient and effective organisation.

Due to the danger of a loss of competence in the nuclear area, the Federal Regulator and the supervisory authorities of the *Länder* Federal Regulator combine their efforts (e.g. knowledge management, promotion of the coming generation of scientists) in order to maintain the necessary competence of the utilities, of the expert organisations and the licensing and supervisory authorities during the remaining operating lives of the German nuclear power plants.

Maintaining competence also includes measures for passing on the body of acquired know-how in an adequate form. For maintaining the safety-related knowledge base in the field of nuclear technology, the leading German institutions in this field founded the Nuclear Technology Competence Pool (4 research institutions and their partner universities).

For the relevant technical areas, the accrued experience and the knowledge from special research and development activities are put down as codified requirements in the nuclear rules and regulations (e.g. KTA safety standards). For maintaining competence it is necessary to explain context and background to the requirements of the rules and regulations. For this purpose, technical seminars were developed that are open to all who are interested. These seminars on the KTA safety standards shall be held regularly also in future. Furthermore, German organisations actively participate in the development of so-called EUROCOURSEs which impart specialised knowledge in the field of nuclear engineering with the support of the European Union, as, for example, the EUROCOURSEs on probabilistic safety analyses and on the integrity of pressure-retaining components.

Maintaining competence at authorities and their expert organisations

A large number of experienced staff of he nuclear licensing and supervisory authorities has already been retired from service in the last years because they reached the age limit or will retire in the next years. For the nuclear authorities, this generation change means a considerable loss of competent and experienced personnel in the field of nuclear safety and radiation protection. The situation is further aggravated by the fact that vacancies, especially at the federal supervisory authority, are either not refilled as a result of government economy measures or only filled partly. In these cases, junior staff cannot be recruited to the necessary extent and not be trained for their special duties in time either. However, some authorities of the *Länder* have increased their staff after having analysed their personnel organisation structures.

In the light of its overall responsibility for the ensurance of a functioning and qualified regulatory licensing and supervision structure for the execution of the Atomic Energy Act and the Radiological Protection Ordinance, the BMU started work on a concept for maintaining competence within the authority area. This concept is aimed at two objectives:

– Performance of a competence loss analysis, and
– definition and implementation of a concept for basic and advanced training.

With regard to the maintenance of competence, the situation at the expert organisations is similar to that at the authorities, i. e. that in the nuclear sector more experienced staff retires for age reasons than is replaced by new staff. Some problems can be managed by an appropriate personnel strategy, but profound training of the new staff is urgently required. For this reason, the VdTÜV (umbrella organisation of the technical inspection agencies including GRS) revised the expert training and set up a training centre for nuclear technology. Thus, training courses are offered that can be used within the framework of the training concepts of the authorised expert organisations. Further training offers are available at some expert organisations (e.g. GRS). At GRS, an own personnel development concept was implemented with targeted personnel increase.

Information and knowledge management system

The BMU is currently developing an information management system. In this respect, document collections and relevant technical information for nuclear authorities and the authorised expert organisations are surveyed and made available electronically structured, so that the user can evaluate them on his or her desktop PC almost without conventional filing. Another focal point will be the future international networking.

Authorised experts

The profession of the authorised expert has a long-standing tradition in Germany. Its beginnings lie in the private steam boiler inspection agencies of the 19th century which helped improve the quality, safety and reliability of such facilities by introducing independent supervision.

In performing their licensing and supervisory activities, the *Länder* ministries may engage expert organisations or individual experts. Section 12 of the Atomic Energy Act lists the following aspects which must be taken into consideration when engaging experts:

- vocational training,
- professional knowledge and skills,
- trustworthiness, and
- independence.

Details regarding these requirements are specified in corresponding regulatory guidelines [3-8, 3-34].

By involving authorised experts, an evaluation of the safety issues is performed that is independent of that of the licence applicant. The authorised experts perform their own tests and evaluations and their own calculations with preferably different methods and computer codes than those used by the licence applicant. The persons involved in preparing the expert analysis are not bound by any technical directives and are reported to the respective authority by name. In making their decision, the authorities are not bound by the evaluation results of the authorised experts.

For the federal supervisory activities the BMU equally will consult national and international experts, if necessary.

Federal Office for Radiation Protection

In performing its federal supervision of the respective *Länder* ministries, the BMU is supported by the Federal Office for Radiation Protection (BfS) in all matters concerning nuclear safety and radiation protection. The BfS was established in 1989 as subordinate authority of the BMU. Its functions are among other things:

- government custody of nuclear fuels,
- construction and operation of waste repositories,
- licensing of the storage of nuclear fuels,
- licensing of the shipment of nuclear fuels and large radiation sources,
- keeping of a register of the radiation exposure of occupationally exposed persons,
- determination of reference limits for diagnostics in medicine,
- support in technical and administrative matters concerning nuclear safety,
- documentation of reportable events from nuclear installations, and
- determination and publication of the remaining electricity output.

Reactor Safety Commission, Commission on Radiological Protection

The Federal Environment Ministry receives further advisory support from the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK). The Reactor Safety Commission was founded in 1958, the Commission on Radiological Protection in 1974. It has to be ensured that the commissions are independent and well qualified and that their members reflect the whole spectrum of scientific and technical opinions. The statues commit the members to voicing their opinion in an objective and scientifically sound manner. The two commissions currently consist of 15 and 16 members, respectively, who are experts in different specialist fields. The members are appointed by the BMU. Their main activity lies in advising the BMU on questions of fundamental importance, but they also initiate developments directed at furthering safety technology. The results of the discussions of the individual commissions are formulated as general recommendations and as statements on individual cases and published.

Gesellschaft für Anlagen- und Reaktorsicherheit

The Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) is a central expert organisation. GRS performs scientific research in the field of nuclear safety technology, predominantly under federal contracts, and supports the BMU in technical issues. A limited number of its tasks is also performed by order of the licensing and supervisory authorities of the *Länder*.

Nuclear Safety Standards Committee

The Nuclear Safety Standards Committee (KTA) was established in 1972 at the Federal Interior Ministry which was in charge of nuclear affairs at the time. It is made up of five interest groups of representatives of the manufacturers, the utilities, the federal and *Länder* authorities, the expert organisations and representatives of general concerns, e.g. unions, industrial safety, liability insurers. In accordance with its statute, the KTA formulates detailed safety standards (→ Chapter 7 (2i)) if "experience indicates that the experts representing the manufacturers and utilities of nuclear installations, the expert organisations and the federal and *Länder* authorities would reach a uniform opinion." The safety standards are prepared by experts meeting in sub-committees and working groups and are then passed on to the KTA for final approval. The five interest groups have an equal strength of ten representatives each. A safety standard requires a 5/6 majority to be passed. Therefore, no individual interest group voting unanimously can be outvoted by the others.

8 (2) Separation Between the Functions of Supervision and Utilisation of Nuclear Energy

Within the framework of the First Review Meeting under the Convention on Nuclear Safety, several contracting parties questioned on the subject of organisational separation of functions within German nuclear authorities as required by the Convention. In the centre was the question in how far compliance with Article 8 (2) of the Convention is affected by the fact that the functions of nuclear supervision and energy industry promotion rest within one single authority.

The Federal Government has taken up this question and in the following, presents the results in detail. In summary, there is confirmation that in Germany the governmental institutions concerned with the utilisation or promotion of nuclear energy are sufficiently separated, both legally and administratively, from those institutions that are responsible for the licensing and supervision of nuclear power plants.

Requirements of the Convention

Article 8 (2) of the Convention contains a substantive protective provision which stipulates the organisational-structural separation of the licensing and supervisory functions of the state from its promotion function. The resulting consequences for the state concerning the organisation of the fulfilment of its functions can be determined from the purpose of the provision of Article 8 (2) as well as from the fact that the principle of separation has been formulated to be unspecific due to the sometimes very differently structured national legal systems in the states of the contracting parties.

The Convention on Nuclear Safety serves for the preservation and further development of the safety level of nuclear installations. In this connection, the effective separation stipulated in Article 8 (2) is to ensure that the supervision of nuclear installations remains uninfluenced by any promotion interests.

The above mentioned fulfilment of the licensing and supervisory functions by state authorities necessarily makes use of sovereign powers towards the utilities. In a democratic state governed by the rule of the law, like the Federal Republic of Germany, the execution of state supremacy requires authorisation by the sovereign, i.e. the people. According to the constitutional provisions deriving from Article 20 (2) of the Basic Law, this authorisation is imparted by the ultimate responsibility of the respective political decision-makers.

Realisation in Germany

It has to be pointed out that legally, the licensing and supervisory authorities – both on federal and on *Länder* level – are administrative state authorities. Constitutional stipulations (Article 20 (3) of the Basic Law) require them to act according to the law. In this connection, emphasis is laid on the obligation pursuant to the Atomic Energy Act that the necessary precautions against damage resulting from the construction and operation of the installation have to be ensured on the basis of the state of the art in science and technology in the field of nuclear engineering. Further, the purpose of the promotion of the peaceful utilisation of nuclear power, formerly mentioned in Section 1 of the Atomic Energy Act, was deleted by the amendment of the Act.

Organisationally, a distinction has to be made between the activities of the competent licensing and supervisory authorities on *Länder* level and the powers of supervision and instruction held by the Federation. In some *Länder* – as it is also the case on federal level – different ministries are in charge of questions relating to the safety of nuclear installations on the one hand, and the promotion and use of nuclear power on the other hand. Where the fulfilment of the functions of nuclear supervision and energy industry promotion are accommodated within one single ministry, separation is ensured by a division of the responsibilities between different organisational units that are independent from each other. To support the administrative state authorities in technical matters, these can consult experts – acting under civil law – who in turn are obliged to deliver impartial and qualified statements (→ Chapter 7 (2ii) and (2iii) and Chapter 8 (1)).

The authority of the Federation to give orders concerning issues related to the licensing and supervision of nuclear installations - which is derived from Articles 85 (3) and 87 c of the Basic Law – lies with the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, which on its part does not fulfil any functions regarding the use and promotion of nuclear energy.

Also, other Federal Government agencies do not promote the utilisation of nuclear power, either. The policy of the Federal Government instead aims at phasing out the use of nuclear power in an orderly manner. In the area of reactor safety research, studies on new reactor designs were therefore terminated. The funds provided for the enhancement of safety-related knowledge and for the further development of safety assessment methods will be progressively relocated towards alternative energy research.

In relation to the above mentioned state agencies, the licensees of nuclear power plants – in their function as users and, may be, promoters of nuclear power – represent commercial enterprises under civil law. They are either power utilities themselves, or made up of shareholders from German power utilities. These power utilities are also commercial enterprises under civil law, usually stock corporations (→ Chapter 11 (1)) and have no influence on the safety-directed actions of the licensing and supervisory authorities.

In the negotiations between the Federal Government and the power utilities about the nuclear phase-out, from the very beginning the Federal Government also made clear that there will be no cut-back in safety.

As a result, it can therefore be concluded that the governmental organisation in Germany fulfils the requirements of Article 8 (2) of the Convention.

Independence of the nuclear authorities

According to Article 8 (2) of the Convention, the contracting parties are obliged to ensure an efficient separation of the tasks of the government agencies from those of other agencies or organisations which are concerned with the promotion or use of nuclear energy. As stated above, the current organisation of the German nuclear supervision (→ Chapter 7) in form of the federal executive administration complies with these requirements under international law.

In addition to this obligation under international law from Article 8 (2), the BMU investigated whether and, where applicable, to which extent the current nuclear administration can be optimised also with a view to independence.

According to this investigation, the independence of the federal supervisory authority is no end in itself, but only to be required to such a degree that the authorativeness of expert opinions is promoted, nuclear supervision becomes more effective, thus finally reducing the possibility of decisions that are not fully directed to safety.

Chapter 8: Progress and changes since 2001

– A work programme was agreed upon for the self assessment of the Federal Regulator by means of the IRRT questionnaire of IAEA. The work first focuses on the preparation of reference material and on answering the questionnaire within the framework of federal supervision. The results will be subject to internal assessment at the BMU. As a next step, it will be decided whether and to which extent the *Länder* shall be involved.
– The BMU dealt with the question whether and, where applicable, to which extent the current nuclear administration can be optimised also with a view to increase independence.
– The BMU is currently developing a quality management system. Here, emphasis is laid on the organisational recording, analysis and optimisation of the processes at the directorate competent for nuclear safety, radiation protection and the nuclear fuel cycle.

This will also provide a contribution to knowledge management regarding relevant work processes. At several nuclear authorities of the *Länder*, organisation reviews are being performed or in preparation. The BMU is currently developing a concept for maintaining competence within the authority area. This concept consists of the performance of a competence loss analysis and the definition and implementation of a concept for basic and advanced training, especially with regard to the limited operating lives of the German nuclear power plants.

Chapter 8: Future activities

– The results of the BMU studies on independence will be considered in the planning on a fundamental reform of the nuclear administration.

9 Responsibility of the Licence Holder

The licensee has the primary responsibility for the safety of a nuclear power plant. He may be issued a licence only if he fulfils all prerequisites for a licence as specified in Chapter 7 (2ii). One prerequisite is the trustworthiness of the persons responsible. They must also give certified proof that they possess the required technical qualification. These facts provide the basis for responsible performance under the licence.

In the case of companies with a number of board members authorised to represent, the ultimately responsible person is reported to the authority by name. This same person is also responsible for a functioning organisational structure and qualified personnel at the nuclear power plant. Other personnel with individual responsibilities are specified in the regulatory guideline on technical qualification [3-2] as follows:

– The plant manager is ultimately responsible for the safe operation of the entire plant and, especially, for the fulfilment of the provisions and requirements under the Atomic Energy Act and licence permits. He is authorised to give orders to the heads of the subordinate divisions and subdivisions.
– The division and subdivision heads are responsible for their technical areas and are authorised to give orders to their subordinate personnel.
– The responsible shift personnel - i.e. the shift supervisors and their deputies and the reactor operators - carry the responsibility that during operating conditions, the nuclear installation is operated in accordance with the written operating instructions, and with the prescribed operating schedule and that in case of accidents, appropriate actions are taken (immediate operating process).

The plant manager or the division and subdivision heads will only intervene with immediate operating processes in well-founded exceptional cases. Outside regular workday hours, the shift supervisor is the designated representative of the plant manager also with respect to his ultimate responsibility for the safe operation of the nuclear power plant. A technical qualification examination and a regulatory work licence are prescribed for the shift supervisors, their deputies and the reactor operators (→ Chapter 11 (2)).

The holder of a licence is, concurrently, the so-called radiation protection supervisor, and as such also responsible for the entire area of radiation protection (→ Chapter 15). He appoints the radiation protection commissioners to perform the corresponding tasks and to supervise operation. These commissioners, together with the radiation protection supervisor, must see to it that all protective and surveillance requirements specified by the Radiation Protection Ordinance are properly fulfilled (→ Chapter 15). The radiation protection commissioners must not be hindered in performing their duties and must not be put at a disadvantage due to their activities.

To better account for the particular issues of nuclear safety, the additional position of nuclear safety commissioner was created as part of the organisational structure of the plant [1A-17]. It is his responsibility to supervise the issues of nuclear safety in all areas of operation. With respect to this task he acts independently of the company interests of economic plant operation. He participates in all activities regarding modifications, assesses the reportable events (→ Chapter 19 (vi)) and the evaluation of operating data and has the right to report directly and at any time to the plant manager.

The radiation protection commissioners as well as the nuclear safety commissioner act independently from the company hierarchy in performing their tasks.

In accordance with the regulatory guideline on technical qualification [3-2], further persons in special positions with functions directly related to plant safety and who, therefore, also have the right to report directly to the plant manager are:

- the training manager,
- the head of the quality assurance division, and
- the physical protection commissioner.

The actual structure of the plant organisation is at the sole discretion of the licensee, provided it accounts for the requirements of the above-mentioned responsible persons as well as for the general requirements regarding quality assurance (→ Chapter 7 (2i), KTA safety standards). The organisation chart showing the task distribution and the names of the responsible persons must be submitted to the licensing and supervisory authority; modifications of the organisation chart and replacement of responsible personnel also require the approval of the competent nuclear authority.

Any enforcement measures by the competent authorities will always first be directed at the holder of the licence with the objective that the ultimately responsible persons will personally meet their obligations. If this is not the case, the authorities can question the trustworthiness of these persons, which is a prerequisite for granting the licence. It is only logical in this case that any procedures regarding an administrative or criminal offence will be directed at the individual persons (→ Chapter 7 (2iv)).

10 Priority to Safety

The protection of the general public and property from the hazards of nuclear energy and ionising radiation is the primary objective of the Atomic Energy Act. On this basis, the requirement is derived to guarantee nuclear safety that at all times has to be considered in the application. As early as 1972, the Federal Administrative Court, being the supreme administrative court of Germany, ruled that nuclear safety has priority over any of the other objectives of the Act. This ruling has always been upheld in later court decisions. This principle is put in concrete terms in Section 7 of the Atomic Energy Act according to which a licence to erect, operate or modify a nuclear power plant may only be granted if the necessary precautions against damage required in the light of the state of the art in science and technology have been taken by the design and operation of the installation (precautionary aspect).

In establishing the ordinances, the general administrative provisions, the nuclear safety standards and regulatory guidelines for design, construction and operation of nuclear power plants (→ Chapter 7 (1)), this precautionary aspect specified in the Atomic Energy Act has always been given special emphasis and has been viewed as the one requirement basic to all technical realisations. Also in future developing of requirements in nuclear safety this precautionary aspect will continue to be considered.

An important element of the implementation of the safety-first principle continues to be the licensee's primary responsibility for nuclear safety (→ Chapter 9). In this context, the licensee's safety management has to comprise all measures that are necessary to ensure a sufficient level of safety and has to anticipate all foreseeable new challenges.

The agreement between the Federal Government and the power utilities of 14 June 2000 (signed on 11 June 2001) specifies the general conditions for the implementation of the Federal Government's decision to phase-out the electricity production from nuclear energy in an orderly manner. The central point of the agreement is to limit the utilisation of the existing nuclear power plants by restricting the residual electricity output that may be produced, with the basic assumption of an overall operating lifetime of 32 years. This boundary condition was implemented with binding effect through the amendment of the Atomic Energy Act in 2002. This is accompanied with new challenges to maintain and enhance the technical safety and the safety culture. During the remaining operating lives, the legally required high level of safety has to be kept, and cut-backs in safety are not acceptable. In particular, this means:

– economic constraints must not lead to restrictions of safety precautions or to a renunciation of safety-related improvements,
– safety-related competence has to be maintained as long as necessary for safe operation during the remaining operating lives.

The Federal Government realises these challenges that have to be met in addition to challenges due to the deregulation of the electricity markets. In this context, the Federal Government will take care that no cut-backs in safety will happen and the "safety first" principle will continue to be vigorously enforced.

Organisation, safety culture and safety management

In all nuclear power plants, the tasks and responsibilities of the personnel are specified in the plant personnel organisation, which is part of the safety specification (→ Chapter 19 (iii)). The three functions operation, maintenance and radiation protection have separate organisational structures. The management concept builds on expertise, understanding of the safety-related context, creation of good working conditions, and responsibility for safety (→ Chapter 9). On behalf of the BMU, studies on core competences for tasks of safety-relevant work processes that are not delegable were conducted. The requirements regarding personnel capacity and personnel qualification to ensure safe operation of nuclear power plants were developed on the basis of work processes. As a result, it can be stated that no significant deficits were identified regarding personnel capacity and qualification.

In their policy statement on the safety culture in German nuclear power plants, the power utilities have described fundamental principles of safety-conscious thinking, acting and communication. It is to contribute to a common understanding of the term "safety culture" and contains at the same time a catalogue of characteristic features for the in-house assessment of safety culture.

In co-operation with an independent expert organisation, the utilities developed and introduced a self-assessment programme to determine the status of operational management and safety culture at the German plants. Improvement potentials and weak points in safety management and the general behaviour of the staff shall be identified. Regular performance of this self assessment shows trends. Moreover, one utility currently develops a process monitoring system for the identification and description of all safety-relevant processes on the one hand and for the derivation of indicators for the assessment of these processes on the other hand.

In response to several events at German nuclear power plants with deficiencies in the areas of personnel and organisation, the BMU developed a questionnaire to determine the status quo of the safety management systems of the utilities and the practice at the plants. This questionnaire also considered the relevant IAEA rules and regulations. The questions deal with the general tasks of

- safety policy and safety objectives,
- safety-directed organisation,
- regulations and instruments for the planning and performance of safety-relevant tasks,
- monitoring and check of performance of safety-related tasks,
- auditing, and
- reviews and experience feedback.

The answers of the utilities show that elements of the safety management system are existent at the German nuclear power plants, but that no generally applied safety management systems have been established.

On behalf of the BMU, bases for safety management at nuclear power plants have been developed that are based on requirements of the German nuclear rules and regulations, the ISO regulations on quality management and rules of the IAEA.

In consequence of the insistent requests of the supervisory authorities, the German utilities committed themselves to develop a concept for a comprehensive safety management system. This safety management system shall not only enable the early identification of deviations from limits or other specified safety-related licensing requirements, but also the information of the supervisory authorities in case of indications of doubts about the management and control of design basis accidents to order to take safety-directed countermeasures (→ Chapter 19 (vii)).

Chapter 10: Progress and changes since 2001

- The bases for safety management systems - “Grundlagen für Sicherheitsmanagement-systeme” - at nuclear power plants have been published. The VGB submitted a self-assessment system for the safety culture at nuclear power plants that is applied by the operators of the nuclear power plants.

Chapter 10: Future activities

- Full introduction of process-oriented safety management systems according to the BMU bases, including indicators.
- The BMU will oblige the utilities within the framework of the safety management system
  - to inform the competent supervisory authority without delay about findings indicating that the demonstration of the control of design basis accidents could be questioned,
  - to shut down the plant temporarily if the required control of design basis accidents cannot be demonstrated at short notice, and
  - to submit a work plan for analyses and backfitting measures.

11 Financial Means and Human Resources

11 (1) Financial Means

Expenditures by the licensees

All nuclear power plants in operation are run by private corporations. The necessary financial means are provided by the corporations out of their sales revenues from electricity production. Besides the adaptation to developments in plant safety, the above mentioned expenditures include investments in means for a more reliable and economical operation. In general, financing is carried out on the basis of economic plans which list the finances needed for the implementation of measures planned for the subsequent fiscal year. In the case of larger backfitting measures extending over several years, project related work schedules are prepared which include the specification of the required financial means and the time in the course of the project when they will be needed. An approval of projects by the top management or by the supervising bodies always also includes approval of the necessary financial means.

The VGB PowerTech, of which all German and several foreign licensees of nuclear power plants are members, annually spends approximately € 2.5 millions for the evaluation and feed-back of operating experience (→ Chapter 19 (vii)). In addition, VGB has financed more than 300 research projects over the past ten years, about three-quarters of which - for a total amount of about € 60 million - were directly aimed at improving the safety of the installations.

In order to be prepared for the follow-up costs connected with the operation of a nuclear power plant, the licensees are obliged pursuant to the commercial law to build up financial reserves for the decommissioning and dismantling of the installations, and the treatment and disposal of radioactive material including spent fuel elements. These reserves are tax-free. So far, reserves amounting to € 35 billion have been set aside. Due to the changes in taxation that came into force in 1999, part of these reserves had to be dissolved. This is mainly because the reserves now are subject to yield interest of 5.5 % until the time of probable utilisation. Type and amount of the reserves are controlled by independent accountants and by the financial authorities.

11 (2) Human Resources

German nuclear power plants currently in operation are staffed with personnel that has a long experience in the operation of nuclear power plants. Single-unit plants are staffed with about 300 people, double-unit plants with about 500. Additional personnel - partly at the headquarters of the utilities - is engaged in project planning, project management, licensing, and technical support. Supplementing its own personnel, the licensees of nuclear power plants extensively use personnel from contracted external firms, particularly for maintenance work during the annual inspection outages, during refuelling and plant modifications. This also includes personnel of the manufacturer of the nuclear power plants and other external specialists for specific tasks, e.g. contractually required maintenance and inspection of specific components.

Regulations regarding the qualification of personnel

Section 7 of the Atomic Energy Act [1A-3] specifies the prerequisite that a licence for the construction and operation of a nuclear power plant shall only be granted if the persons responsible for the construction and operation have the necessary qualification. Likewise, the personnel otherwise engaged during operation must have the necessary knowledge with respect to safe operation, possible risks, and relevant protection measures to be applied. Furthermore, there must be no doubts as to the trustworthiness of the personnel. Accordingly, proof of the qualification of the responsible personnel as well as of the necessary knowledge of the personnel otherwise engaged during operation must already be included in the licence application for construction, operation or essential modifications [1A-10]. The trustworthiness of the personnel is evaluated directly by the licensing authority according to the relevant regulatory ordinance [1A-19]. The qualification certificates and regular training measures to maintain qualification are checked within the framework of regulatory supervision (→ Chapter 7 (2iii)).

The requirements regarding qualification and technical know-how of the personnel are specified in the Regulatory Guidelines on Qualification, [3-2] and [3-27]. The specified requirements regarding initial qualification build on the training and skills received by technical personnel within the public vocational system.

The German public vocational training system ensures that the operators of nuclear power plants can find skilled workers, foremen, technicians, engineers and scientists who received relevant technical basic training within their schooling and vocational training that is documented by a state-approved certificate. Generally, craftsmen and engineers in the fields of mechanical and electrical engineering, process engineering, physics and chemistry are already qualified before they begin employment in a nuclear power plant. Engineers can specialise in nuclear engineering during their course of study. To supplement the public vocational training system, in 1970 the utilities founded a power plant school to correspond to the requirements regarding the specific skills of the nuclear power plant personnel. Courses are offered that lead to a degree as skilled power plant worker, as power plant foreman in the fields of mechanical and electrical engineering, instrumentation and control, and nuclear engineering.

The above-mentioned Regulatory Guidelines on Qualification [3-2; 3-27] are supplemented by regulatory guidelines [3-38; 3-39; 3-40; 3-61; 3-65] on the certification of the qualification of responsible shift personnel, on the maintenance of qualification, and on the specific qualification of personnel responsible for radiation protection. These guidelines specify the task-related initial qualification, additional training requirements, performance of training and the acquisition of practical experience required for the technical personnel, and furthermore, for the responsible shift personnel, the examinations and certification required in their respective responsibilities. In accordance with the safety relevance of their duties, the required qualification of the responsible shift personnel is specified in detail.

The nuclear safety regulations define the following as responsible personnel:

- the plant manager,
- the division and subdivision heads,
- the responsible shift personnel,
- the training manager,
- the head of the quality assurance division,
- the radiation protection commissioners,
- the nuclear safety commissioner, and
- the physical protection commissioner.

The regulatory guideline [3-27] requires for personnel not belonging to the group of responsible personnel (otherwise engaged personnel) specific knowledge related to safety, at least in the fields of radiation protection, fire protection, industrial safety, and plant organisational structure and procedures. The requirements specified in this regulatory guideline with respect to occupational qualification, practical experience and certification of knowledge differ in extent and depth in accordance with the respective duties. The otherwise engaged personnel comprises the following groups:

- supervising personnel,
- control panel operator,
- deployed personnel,
- assisting personnel,
- other personnel.

These groups also apply to personnel from external firms.

The training manager of the licensee of the nuclear power plant, together with the division and subdivision heads if so required, is responsible for the planning, performance, follow-up and documentation of the training activities. On the basis of the training objectives given in [3-27; 3-38; 3-39] he draws up a plant- and task-specific programme to acquire and maintain qualification and to maintain the necessary knowledge. The training of the responsible shift personnel is performed at a nuclear training facility, at the manufacturers, on-site at the nuclear power plant itself, and on a plant-specific full-scope training simulator.

Training of shift personnel

The qualification requirements for responsible shift personnel depend on the function to be fulfilled. Shift supervisor have to complete their education at a university, a university of applied sciences or a school of engineering in an appropriate branch of study. For their deputies and for reactor operators, a qualification as foremen, technician or skilled worker is sufficient as minimum requirement. The candidates for the shift supervisor or reactor operator function generally have many years of experience at NPPs before starting their specific training. The training of the shift supervisor candidates, however, already begins a few months after their recruitment, using this time for a comprehensive NPP introduction.

The responsible shift personnel first attends for nearly four months an external course on basic nuclear engineering which must be recognised by the competent authorities on the basis of standardised criteria [3-65]. At the end of this course there are examinations at the different levels of training. Within the framework of customer training, the manufacturer provides courses on specific topics (e.g. thermal-hydraulics, instrumentation and control, pumps) and a number of systems courses, each with a duration of several weeks, dealing with the functions and operation of all essential systems of the plant. The initial plant-specific training at the installation itself consists of theoretical instructions, on-the-job training in various divisions, and a longer term as shift member in the control room. A simulator training course of at least seven weeks (boiling water reactor) or eight weeks (pressurised water reactor) is mandatory. The initial simulator training covers all operating procedures from normal operation, abnormal operation and the control of design basis accidents up to beyond-design basis event sequences.

The qualification of the responsible shift personnel requires successful written and oral examinations on basic nuclear engineering and plant-specific issues. The oral examinations are taken in front of an examination board composed of representatives from the supervisory authority, independent experts, and representatives from the training institutions (only in case of test in basic nuclear engineering) and from the utility.

When all prerequisites are met, members of the responsible shift personnel receive a licence, unlimited in time, for their respective functions at the particular nuclear power plant. To maintain their licence they are required to participate in follow-up courses, in simulator training, and to work in the control room for at least two weeks within a six months period. If the licensed person moves on to another nuclear power plant or if he has not worked in the licensed function for a longer period of time (more than one year), he has to repeat the examinations regarding his qualification.

The physical aptitude of the responsible shift personnel for work in the control room must be checked by authorised physicians before they begin their duties. Their physical and psychological fitness is re-evaluated at annual intervals by medical check-ups as well as by continuous observation by their supervisors. This is carried out in direct responsibility by the licensee.

Training on simulators and models

Full-scope simulators are available for all nuclear power plants that are similar to a given plant, but the major part is plant-specific. Two simulators are located at the sites of the nuclear power plants (Stade, in use until the end of 2003, and Krümmel), the other 13 are located at the simulator centre of the Kraftwerks-Simulator-Gesellschaft mbH (KSG) in Essen. The courses are carried out by the Gesellschaft für Simulatorschulung mbH (GfS). Both companies, with an overall staff of 150, are joint subsidiaries of the licensees of German nuclear power plants. Their responsibility is the maintenance and updating of the simulators and the conduction of courses. Table 11-1 shows which simulator applies to which nuclear power plant.

The specifications by the utilities ensure a uniform minimum standard for the capabilities of the simulators which is continuously further developed on the basis of the practical experience feedback, and ensure the qualification of the instructors and an adequate course programme. With respect to maintaining qualification of the responsible shift personnel, the following courses must be attended within a three-year cycle: a minimum of 20 days of instructions with at least 80 hours of simulator training (pressurised water reactor) and 15 days of instructions with at least 60 hours of simulator training (boiling water reactor). A minimum of five days of simulator training per year is required. The training is focused on normal operation, abnormal operation, design basis accidents, and beyond-design basis accidents.

Since 1990, an additional simulator had been operated by the nuclear power plant manufacturer Siemens initially at Karlstein and from 1997 until 2001 in Offenbach. In 2002, it was shut down. This simulator was a nuclear function trainer at its company-owned training centre, capable of simulating the most important safety procedures in a pressurised water reactor of recent design (4^{th} design generation, Konvoi).

A glass model of the primary system of a PWR scaled 1:10 is located at the KSG in Essen. It allows the study and visual presentation of thermal-hydraulic phenomena occurring during design basis accidents. This glass model is used for initial and continual training of personnel from all nuclear power plants, including BWRs.

Table 11-1 Simulators for Nuclear Power Plants

4/2004

	Nuclear Power Plant	Type Gross capacity MWe	Identification and site of the simulator	a) Manufacturer of the simulator b) Number of signals sent to control room	Start of training
1	Obrigheim KWO	PWR 357	D56 KSG/GfS	a) Thomson b) 10 600	1997 until 1996 at D1
2	Stade KKS	PWR 672	D1 KSG/GfS	a) Singer b) 12 900	1977 until 2003
			KKS Simulator Stade	a) CAE b) 18 000	1998 until 2003
3	Biblis A KWB A	PWR 1225	D1 KSG/GfS	a) Singer b) 12 900	1977
4	Biblis B KWB B	PWR 1300	D1 KSG/GfS	a) Singer b) 12 900	1977
5	Neckarwestheim 1 GKN 1	PWR 840	D52 KSG/GfS	a) Thomson b) 11 100	1997 until 1996 at D1
6	Brunsbüttel KKB	BWR 806	S4 KSG/GfS	a) CAE b)	2001
7	Isar 1 KKI 1	BWR 912	S31 KSG/GfS	a) Atlas Elektronik b) 18 000	1997 until 1996 at S1
8	Unterweser KKU	PWR 1410	D51 KSG/GfS	a) Thomson b) 16 000	1997 until 1996 at D1
9	Philippsburg 1 KKP 1	BWR 926	S32 KSG/GfS	a) Atlas Elektronik b) 16 600	1997 until 1996 at S1
10	Grafenrheinfeld KKG	PWR 1345	D3 KSG/GfS	a) Krupp Atlas Elektronik b) 26 500	1988
11	Krümmel KKK	BWR 1316	Simulator KKK Krümmel	a) Siemens/S3T b) 27 000	1997 until 1997 at S1
12	Gundremmingen B KRB B	BWR 1344	S2 KSG/GfS	a) Siemens / Thomson b) 21 800	1993
13	Grohnde KWG	PWR 1430	D3 KSG/GfS	a) Krupp Atlas Elektronik b) 26 500	1988
14	Gundremmingen C KRB C	BWR 1344	S2 KSG/GfS	a) Siemens / Thomson b) 21 800	1993
15	Philippsburg 2 KKP 2	PWR 1458	D42 KSG/GfS	a) Siemens/S3T b) 26 700	1997, until 1997 at D1,D3
16	Brokdorf KBR	PWR 1440	D43 KSG/GfS	a) Siemens/S3T b) 28 700	1996 until 1997 at D3
17	Isar 2 KKI 2	PWR 1475	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 until 1995 at D3
18	Emsland KKE	PWR 1400	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 until 1995 at D3
19	Neckarwestheim 2 GKN 2	PWR 1365	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 until 1995 at D3

Maintaining qualification, advanced training

Three-year-programmes are planned and conducted for maintaining the qualification of responsible shift personnel. They are regularly adapted to new findings and technical facts. The minimum training duration is approximately 100 hours per year; if the mandatory simulator training is included, the average training sums up to about 150 hours. The training deals, among other things, with the modifications of the plant itself or of its mode of operation, new regulatory requirements or provisions, as well as with methods for coping with stress situations. Particular attention is paid to the feedback of operating experience. An important part of this training is repeated training at the plant-specific simulators (see above) which centres on coping with abnormal operation and design basis accidents. The regular emergency exercises (→ Chapter 16 (1)) also serve to maintain qualification and competence. In recent years, these increasingly employ the simulators to achieve training situations that are as close to reality as possible. For several years now, the plant simulators have been used to exercise the protection-goal-oriented actions which are necessary to cope with beyond-design basis accidents.

Each licensee of a nuclear power plant puts together a report for the competent supervisory authority describing in detail the overall concept of the three-year training programme and the contents and depth of treatment of the training measures as well as the experience gained by these measures. In a regular annual report, the supervisory authority receives certified proof with respect to the training measures actually performed and their success, as well as to the participation of the operating personnel.

Measures directed at maintaining the qualifications are also performed for the plant manager, the division and subdivision heads, the training manager, the nuclear safety commissioner and the head of the quality assurance division. Requirements for these measures are defined by specification of the qualification maintenance measures being essential for these functions and of the minimum times within a three-year period, for which proof has to be furnished. The time to be scheduled for the individual measures to maintain qualification may vary in dependence on the respective function; however, the total time for these measures for which proof has to be furnished must not be less than 300 or 240 hours, according to function, within a period of three years. Corresponding measures to maintain the qualification of the responsible nuclear power plant personnel at executive level are, in particular, simulator training, specialised basic and advanced training measures, technical or scientific conferences and job related training. The measures to maintain qualification performed for these persons have to be documented by the utility and submitted to the competent supervisory authority for review.

Likewise, the training programme for otherwise engaged personnel (persons not being part of the responsible personnel) is regularly updated with respect to the knowledge related to safety to be transmitted. The personal participation in the training courses is documented.

The operating personnel of all nuclear power plants generally has many years of practical experience in the operation of nuclear power plants. The technical personnel - during initial training and repeatedly during advanced training - is regularly made aware of the importance of safety-oriented actions. Here, the findings from the evaluation of operating experience and operational events are of particular importance.

Maintaining of competence at the utilities

Compared to the situation at the authorities (→ Chapter 8 (1)), the change of generations at the utilities generally takes place in a more balanced manner. In addition, established programmes are available for the training of new personnel. The procedures have largely been harmonised by a number of regulatory guidelines. Directives on further specification of the measures to maintain the qualification of responsible nuclear power plant personnel were defined by the competent authorities at the end of 2003.

The stress of competition initiated by the liberalisation of the electricity market increased the pressure at the utilities to reduce costs. Corresponding measures extended both to the technical area and the areas of organisation and administration. A cost reduction measure that may be taken is the lowering of personnel costs through reduction of plant personnel or outsourcing of certain activities from the plant to central units of the utility or to external contractors. As such measures may have a negative impact on the safe operation of a nuclear power plant, investigations were performed on the technical competence that has to be kept available on site. Further, procedures to be applied uniformly by the authorities of the *Länder* were agreed upon according to which essential changes in manpower are subject to licensing.

12 Human Factors

The safe operation of a nuclear power plant depends not only on the reliability of the technical and structural systems but also on the safety-oriented actions of the personnel in an environment of adequate plant organisation. In this respect, the ergonomic design of equipment and work procedures are just as important as the proper qualification of the personnel and the preservation of competence (→ Chapter 11 (2)). In the following, the status of German nuclear power plants with respect to the design of equipment and work procedures is summarised, initially under the aspect of the man-machine interface. After that, the administrative and organisational aspects are dealt with in the section 'Organisation and safety culture'.

Man-machine interface

German nuclear power plants are highly automated. In addition to the extensive instrumentation and control systems available for operation, many of the more complex procedures are activated by automatic controls. This relieves the personnel from many manual actions.

Of particular importance in this respect are the automatic limitation systems. They are to prevent any physical operating parameters from exceeding the set control range so that normally the reactor protection system is not actuated. The function of some of the limitation systems is also to ensure that the boundary conditions used in accident analyses are not exceeded. The reactor protection system is designed to automatically control design basis accidents for a period of at least 30 minutes without the need for any manual action. In the case of abnormal operation or design basis accidents, this is to ensure sufficient time to diagnose the situation and take appropriate actions. The actions initiated by the reactor protection system have absolute priority over manual actions and automatic operational controls. Should the control room lose its functional capability it is ensured that independent auxiliary emergency systems will take the plant - normally without the need for any manual actions - to a safe shutdown state in which it can remain for at least 10 hours.

German nuclear power plants are controlled and operated from a central control room. The control room is equipped with all the information, activation and communication systems that are necessary for normal operation and for coping with abnormal operation and design basis accidents. Should the control room not be available, the nuclear power plant can be taken to a permanent safe shutdown state from an emergency control room [KTA 3904] and can be kept in this state for unlimited time. Exceptions exist for Biblis A and B (double-unit plant).

The indicators and controls on the consoles and panels in the control room are arranged along process-related or electric flow charts which schematically represent the structures and interrelationships of the systems. Release buttons are to prevent inadvertent actuations. Computerised information systems support the operating personnel in all nuclear power plants.

With regard to maintenance, especially as concerns in-service inspections, extensive technical measures are provided to prevent human errors or to minimise their effects [KTA 3201.4; KTA 3211.4; 3-41; 3-43]. These measures range from permanently installed and unambiguously identifiable testing devices to testing computers and the automatic resetting of safety systems in the event of their inadvertent actuation by the reactor protection system in the course of an in-service inspection. The required positions of safety-relevant valves has to be ensured by two different measures. These are annunciation loops with associated alarm signals to quickly detect and rectify wrong positions, and key-switch systems.

Apart from the corresponding design of the technical systems, comprehensive operating instructions and a comprehensive and complete documentation of operation is necessary to assure reliable and safety-oriented actions of the operating personnel. The documentation [KTA 1404] contains lists of safety-relevant operational records, documents relating to the radiation protection of the personnel and the environment as well as certificates demonstrating the quality assurance of the plant and the fulfilment of instructions and requirements.

The plant operating procedures mainly include the operating manual, the testing manual, and the accident management manual (→ Chapter 19 (iii)).

The actions necessary during operating conditions as well as for coping with design basis accidents are laid down in the operating manual [KTA 1201] (→ Chapter 19 (ii)-(iv)). It comprises the plant regulations specifying tasks, authorisation and responsibilities of the personnel as well as relevant organisational processes, the limits and conditions for the safe operation and detailed instructions for the operation of the whole plant and the individual systems, as well as for the control of abnormal and design basis accident conditions. For example, the maintenance regulation specifies in detail the procedures for maintenance and modifications in accordance with the maintenance guideline [3-41]. Compliance with the safety specifications laid down in the operating procedures is binding. As the analysis of some reportable events of the last years (KKU, KKP 2) revealed deficiencies in the operational documentation, more detailed descriptions of the requirements on safety specifications [3-4] and on the other provisions of the operating manual are currently developed on behalf of the BMU.

The instructions for in-service inspections are laid down in the testing manual [KTA 1202].

The accident management manual comprises the procedures and measures to be taken to control severe accidents.

In addition to the documents in paper form, all nuclear power plants have an integrated operation management system. This enables the computerised specification and control of work sequences and, to a certain extent, also the automatic surveillance of boundary conditions to be fulfilled.

The operating experience is systematically evaluated with regard to human errors and possible improvements derived from this by the utilities and also by the authorities and their experts. The procedure for benefiting from operating experience is described in Chapter 19 (vii).

In addition, the licensees have installed their own human factors (HF) programme to optimise the man-machine interface. Apart from the reportable events, reports about other disturbances and voluntary reports made by staff members are also recorded and investigated. In the analysis and determination of the causes, generally accepted ergonomic methods are applied. Each German nuclear power plant has its own HF-officer in charge of the human factors programme. The results of the human factors programme and the measures implemented as a result are summarised by the utilities in an annual report to the regulatory authorities.

Chapter 12: Progress and changes since 2001

– The review of the requirements on the safety specifications is available as draft version.

Chapter 12: Future activities

– The requirements on the safety specifications stated in the draft version shall serve as a basis for the revision of KTA 1201.

13 Quality Assurance

All licensees of German nuclear power plants are obliged to perform comprehensive quality management. To this end, the licensees have installed quality assurance systems which are based on the provisions for quality assurance specified in the Safety Criteria [3-1] and in the KTA Nuclear Safety Standards. Their objective is to ensure the quality required for plant safety at all levels of the defence-in-depth concept (→ Chapter 18 (i)). By the high quality of plant operation systems a sound and environmentally compatible operation is established and accidents are prevented.

The general requirements regarding quality assurance are contained in [KTA 1401]:

– The objective of quality assurance is to ensure in a documented way that the quality requirements are specified for product forms, component parts, components, and systems, and are fulfilled during manufacture and installation and also during the erection of civil structures. Furthermore, it has to be ensured that the respective requirements continue to be fulfilled under the conditions of operation and maintenance up to the decommissioning of the nuclear power plant.
– The licensee is responsible for the planning, implementation and supervision of the effectiveness of his quality assurance system. It is, therefore, also within his responsibility to assure that his contractors and their sub-contractors plan and implement their quality assurance in accordance with the licensee’s quality assurance system.

On the basis of the requirements laid down in the nuclear safety regulations, the licensees develop a comprehensive quality assurance programme for each individual nuclear power plant. The related documents determine how and by whom the quality requirements necessary for safety have to be specified, how and by whom they have to be fulfilled, and how and by whom their fulfilment is to be certified. Furthermore, the quality assurance programme describes the structure of the quality assurance organisation and the work procedures to perform quality assurance. The programme is submitted to the licensing authority in the licensing procedure, and any changes to the programme are reported to the competent authority. Details of the quality assurance regarding pressurised components are presented in Chapters 18 (ii) and 18 (iii).

The essential requirements for a quality assurance system are summarised below [KTA 1401]:

– Prior to the erection of a nuclear power plant, but also prior to any material alterations or modifications, it has to be specified which component parts, components, systems and structures have an influence on plant safety and must, therefore, be classified as important to safety. In these individual cases quality characteristics must be specified and measures to assure that the quality characteristics are actually achieved.
– Persons charged with the task of implementing and auditing the quality assurance system must be authorised to have access to all relevant information, propose solutions to possible problems, and monitor compliance with the quality assurance measures. They must personally be independent from those persons or organisational units they monitor.
– All persons charged with the performance of certain duties are individually responsible for meeting the corresponding quality requirements.
– Those persons charged with the tasks of independent quality inspections must neither themselves have been responsible for, nor involved in, the manufacture of the product or the activity to be inspected.
– If it is essential for achieving the quality characteristics, the requirements for the qualification of the performing personnel have to be specified; the personnel qualification and its maintenance must be verifiable.

- All documents must be unambiguously marked according to the central plant system for filing, identification and revision. It must be ensured that only those documents are worked with that have been approved and cleared for application. All documents must be stored in their entirety and for a length of time as specified in [KTA 1404].
- Before placing an order, each ordering party is required to evaluate the contractor with regard to his ability to perform the tasks on the basis of his product-related description of the quality assurance system. This evaluation may only be omitted if the required quality of the product can be verified by product-related measures, e.g. by a receiving inspection.
- In the case of series-produced items which in most cases are not specifically designed and produced for the nuclear power plant, e.g. electronic modules, switches, cables, nuts and bolts, it is permissible that the verification of quality characteristics be performed in accordance with methods as specified in conventional or in nuclear safety regulations (i.e., type testing, factory tests, proven operational experience). In addition, it must be certified that the conditions of operation in a nuclear power plant do not exceed the service limits of the series-produced items.
- Any decisions important to safety may only be made, and measures may only be taken by those persons who are so authorised in accordance with their qualification and position within the plant structural organisation. The procedures to follow for meeting the quality requirements during plant operation are laid down in detail in the operating manual and the testing manual (→ Chapter 19 (iii)).
- The licensee and every one of his contractors have to assure themselves at regular intervals of the correct implementation and effectiveness of their respective quality assurance systems. In addition, each party has to assure itself of the effectiveness of the quality assurance systems used by the individual contractor before placing an order. The results of these examinations have to be documented in writing. Any detected gaps and weak points have to be remedied without delay. This must be proven by a corresponding re-examination.

Quality assurance is independently performed by the licensee within the framework of his responsibility for the safety of his plant. The supervisory authority performs corresponding audits to satisfy itself with regard to the correct implementation, appropriate execution, and overall effectiveness of the quality assurance system.

Ageing

It was already pointed out in Chapter 7 (2i) that measures for maintaining quality over a long period of time (ageing management) have been an integral part of the quality requirements specified in German nuclear safety regulations from the very beginning. In the German regulations, ageing phenomena are handled under the term operational influences (→ Chapter 14 (ii)).

Comprehensive measures are employed in German nuclear power plants to counter the inadmissible effects from ageing. These measures are, in particular:

- the consideration of current knowledge on ageing during design, manufacturing and inspection of technical systems (→ Chapter 14 (ii)),
- the monitoring of systems and operating conditions with respect to detecting any deterioration important to safety (→ Chapter 14 (ii)),
- the regular replacement of system component parts known to be susceptible to failure by preventive maintenance (→ Chapter 19 (iii)),
- an upgrading or replacement of technical systems in case weaknesses important to safety are found (→ Chapter 18 (ii)),

– the optimisation of technical systems and of operating conditions (→ Chapter 14 (ii)),
– continuous evaluation of operating experience, implementing findings of the back-flow of experience (→ Chapter 19 (vii)),
– acquisition and maintenance of qualification at a sufficiently high level (→ Chapter 11 (2)).

This practice is supplemented by appropriate research and development.

The evaluation of operating experience beyond a plant-specific level shows that the above-mentioned measures have largely been effective so far. The number of events with damages due to ageing phenomena at German plants is low. In this respect, all plants were affected by age-induced events, but to a different degree. Until now, a significant increase in age-induced events with increasing operating time has not been observed.

Chapter 13: Progress and changes since 2001

– For the assessment of ageing at nuclear power plants beyond a plant-specific level, activities have been started on the development of a computer-based knowledge basis on ageing-relevant damage mechanisms that can be used by all those involved in the licensing and supervisory procedure.
– The RSK submitted a recommendation on an ageing management system to be applied uniformly that considers all safety-relevant - not only the technical - ageing processes during the remaining operating lives of the German nuclear plants. The BMU decided to base its administrative actions on this recommendation.

Chapter 13: Future activities

– The BMU will initiate the submission of annual, plant-specific reports on ageing management by the licensees.

14 Assessment and Verification of Safety

14 (i) Assessment of Safety

The assessment of the safety during construction, commissioning and essential modifications of a nuclear power plant is performed within the licensing process. As stipulated in the amended Atomic Energy Act, licenses for new nuclear power plants will no longer be issued (→ Chapter 7 (2ii)). Continuous safety evaluation during operation is performed within the scope of regulatory supervision.

Safety assessment in the licensing procedure

To be granted a licence for the construction, operation, essential modifications, or decommissioning of a nuclear power plant, an application must be filed at the competent authority. This application has to include details about in how far the plant disposes of the requisite safety characteristics and fulfils the requirements of the current nuclear regulations. The safety assessment is then performed on the basis of the application and the documents to be submitted (→ Chapter 7 (2ii)).

Section 3 of the Nuclear Licensing Procedure Ordinance [1A-10] defines the type and extent of documents to be submitted with an application. These include:

- a safety analysis report which allows a conclusion as to whether the rights of third parties could be violated by the operation of the nuclear installation (see below),
- supplementing plans, technical drawings, and descriptions of the nuclear installation and its parts,
- details on protective measures against malevolent acts or other illegal interference by third parties,
- details on the trustworthiness and qualification of the personnel responsible for the construction and operation, and on the required knowledge of otherwise engaged personnel,
- a safety specification comprising all important details on the safety of the nuclear installation and its operation (see below),
- information on compliance with legal liability provisions,
- description of the accumulating radioactive residual substances and of the intended measures for their treatment,
- description of the antipollution measures regarding water, air and soil.

<u>Safety analysis report</u>

The safety analysis report describes and explains the concept, the safety-related design bases and the functions of the nuclear power plant as well as its operational and safety systems. The effects of the plant and its operation on the environment and of the design basis accidents taken into consideration are described. The precautionary measures to avoid damage caused by the construction and operation of the nuclear power plant are described.

Regulatory guideline [3-5] provides a standardised form for safety analysis reports of PWRs and BWRs specifying a detailed outline of the subjects and giving additional information on the contents. The safety analysis report is the basis for the safety assessment of the nuclear power plant. It contains information on:

- the site,

- the nuclear power plant itself,
- the organisational structure and responsibilities,
- the radioactive material existing at the plant and the corresponding physical protection measures taken,
- protection against internal and external impacts,
- the operation of the nuclear power plant,
- the analyses of design basis accidents.

Details on the future decommissioning of the nuclear power plant are also required. Details on the protection measures against malevolent acts or other illegal interference by third parties are required as part of a separate physical protection report which is classified as confidential.

<u>Additional information necessary for safety evaluation</u>

To standardise the licensing procedure and to facilitate evaluation, regulatory guideline [3-7-1] defines the point in time, extent, and detail for additional information which has to be submitted. It distinguishes between information required in advance of a licensing step, and those needed in accompaniment of construction, e.g. in fulfilment of imposed obligations. This information is the basis both for the competent authority to reach its decisions and for the authorised experts in their safety evaluation.

Details are given on the following subjects:

- siting,
- containment,
- reactor core and control rods,
- pressure boundary, including reactor pressure vessel,
- reactor pressure vessel internals,
- emergency core cooling and residual-heat removal systems,
- auxiliary systems of the reactor coolant system,
- equipment for handling and storing of fuel elements,
- systems for handling and storing of radioactive material,
- ventilation systems,
- steam power plant,
- turbine plant,
- cooling water systems,
- electric power supply of the safety system,
- alarm systems and communication equipment,
- instrumentation and control, main control room, local control stations,
- reactor protection system, and
- radiation protection and radioactivity monitoring.

On all of the above subjects, information is provided for the following procedural steps:

- concept,
- erection of civil structures,
- manufacturing of product forms,
- manufacturing of components,
- pressure test at the manufacturing plant,
- installation of components,

– pressure and leak rate tests at the construction site,
– commissioning of systems,
– delivery of fuel elements,
– initial core loading of the reactor,
– nuclear start-up of the facility, and
– refuelling.

The competent authorities under building legislation participate in the nuclear licensing procedure. Special documents are submitted for their review and assessment. The information required with respect to buildings and civil structures important to safety are specified in regulatory guideline [3-7-2]:

– safety analysis report,
– application for the construction permit,
– preparation of the construction site,
– carcass work,
– surveillance of construction,
– carcass work acceptance,
– inside finishing and corresponding quality control,
– final acceptance tests and inspections.

Safety specifications

The safety specifications to be submitted at the latest with the application for operation licence permit comprise all data, limits, and measures which are essential for a safe state of the nuclear power plant. They give an overview of the characteristics important to safety of the nuclear power plant and specify the conditions for safe operation. Measures to cope with abnormal operation and design basis accidents are also described. The schedule for the in-service inspections to be performed on those parts of the nuclear power plant which are important to safety is also part of the safety specifications (→ Chapter 19 (ii)).

The contents and structure of the safety specifications are laid down in a regulatory guideline [3-4]. According to this specification, the contents comprise information on:

– organisational structure of operation,
– provisions important to safety,
– safety system settings,
– technical drawings of important components including operating parameters, preceding limits, actuating limits, and design basis values,
– general in-service inspection plan for systems and components important to safety,
– handling of reportable events,
– description of the accident sequences.

Any changes with respect to the safety specifications require the approval of the licensing and supervisory authorities.

Involvement of authorised experts and subordinate authorities

The licensing authority normally consults external experts in accordance with Section 20 Atomic Energy Act for the assessment of specific technical aspects (→ Chapter 8 (1)). The general requirements for such expert assessments are specified in a special regulatory guideline [3-34].

The authorised experts carry out a detailed review and assessment of the documents submitted by the applicant. They perform independent analyses and calculations, preferably with analytical methods and computer codes different from those used by the applicant. The results are evaluated in the expert assessment, which also gives the criteria used in the assessment. The persons participating in the expert assessment are reported by name to the licensing authority. They are independent in their judgement and free of any directives regarding the results. The supervisory authorities themselves and subordinate authorities commissioned by them will also carry out own measurements and inspections.

Safety assessment in the supervisory procedure

After the respective licence has been granted, the safety assessment during construction, commissioning and subsequent power operation of the nuclear power plant is performed in accordance with Section 19 Atomic Energy Act (→ Chapter 7 (2iii)) by the nuclear supervisory authority. This authority verifies that the conditions and prerequisites on which the licence was based continue to be fulfilled during operation. The supervisory authority engages the services of authorised experts for these supervisory activities, too.

Supervision under nuclear legislation extends over the entire lifetime of a nuclear power plant and ends only after all radioactive substances have been removed from the site after decommissioning, or if radioactivity has dropped to a value below the limit set for mandatory surveillance. The supervisory authority may then release the nuclear power plant from supervision under nuclear legislation.

Accompanying inspections during construction and commissioning

In the course of the assessment of the documents submitted by the applicant, the authorised experts called in by the supervisory authority will also perform inspections during the construction and commissioning phase. These accompanying inspections are performed independent of those by the manufacturer. They are required to verify the values, dimensions, or functions specified in the submitted documents. This includes e.g. the verification of materials compositions, checking of the assembling of components, and the performance of functional tests at the manufacturing plant. Similar inspections are also carried out at the construction site. During commissioning, the provisions of the plant's safety specification as well as the applicability of the boundary conditions for the accident analysis are checked (→ Chapter 19 (i)).

On-site inspections during operation

On behalf of the supervisory authority, the authorised experts themselves carry out measurements, inspections and evaluations, or they participate in the measurements and inspections made by the licensee himself or on his behalf. This concerns the following areas:

- discharge of radioactive material,
- radiation monitoring of personnel and the environment,
- in-service inspections of systems, components and civil structures of the nuclear power plant.

In addition to the regular measurements and inspections, the supervisory authority and their authorised experts carry out plant walk-downs and inspections on specific aspects.

If deficiencies are found, the supervisory authority requests a corresponding correction by the licensee. In cases where it can be no longer guaranteed that there are sufficient safety precautions in place, the licensing and supervisory authority may demand that operation be suspended for the time being until the deficiencies have been rectified. This also applies if doubts about the validity of the licensing assumptions for the demonstration of accident control arise, as far as accident control is not only impaired to a minor extent. If accident control is no longer given without plant modification, this leads to a shutdown of the plant (→ Chapter 7 (2iv)). A potential plant modification will be subjected to a new licensing process. The specification of the obligations to give information in this procedure is currently being prepared by the Federal Regulator.

Evaluation of reportable events

The competent authority is notified by the licensee of any safety relevant event that occurs in his nuclear power plant. The reporting procedure and criteria are specified in the Nuclear Safety Commissioner and Reporting Ordinance [1A-17]. In addition, the events will also be classified in accordance with the International Nuclear Event Scale (INES). The supervisory authority informs the BMU and the BfS and usually consults authorised experts and requests them to assess the events as well as the remedial measures taken or planned by the licensee (→ Chapter 19 (vi)-(vii)).

14 (ii) Verification of Safety

Within his independent full responsibility for plant safety, each licensee has to adjust the safety level of the nuclear power plant to be in compliance with the state of the art in science and technology over the entire operating life of the plant. If new safety relevant findings come to light, the licensees have to assess the need for and appropriateness of improvements. The nuclear licensing and supervisory authority monitors and – if necessary – enforces the fulfilment of the licensee's obligations (→ Chapter 7 (iv)). The authority itself performs safety assessments continuously, on special occasions or periodically. In addition, there are international safety reviews.

These national and international safety reviews and their essential results will be looked at more closely in the following. The resulting safety-relevant modifications to German nuclear power plants are summarised below.

Routine verification of safety by the licensee

During operation, a regularly repeated verification on the basis of the licensing provisions is required to show that the system functions important to plant safety are executed properly, and also that the quality characteristics have not deteriorated below acceptable levels. To this end, the systems are subjected by the licensee to in-service inspections that are graded according to their individual safety relevance. These in-service inspections include functional tests performed to verify functional performance as well as non-destructive tests to verify faultless condition. Moreover, the licensee plans and performs regular and preventive maintenance of all plant systems during operation and evaluates the operational experience (→ Chapter 19 (vii)).

The in-service inspections of systems important to safety are performed in accordance with the requirements specified in the testing manual (→ Chapter 19 (iii)). The testing schedule contained therein specifies the test object, the nature, extent, and interval of the tests, the operating state of the nuclear installation at which they have to be performed, the identification and name of the test procedure, and which of the tests require the participation of authorised experts. The testing schedule is an integral part of the licensed safety specifications of the nuclear installation. The required verification is specified depending on the testability of a given system function. The objective is always to perform the test at realistic conditions representing the actual conditions at the time of required functional operation. If important system functions are not directly testable, e.g. integrity at higher levels of pressure and temperature, functional performance is verified indirectly. The specified required tests are reviewed regularly considering operating experience and new findings from safety research, and are adapted if necessary. Intended modifications of the testing manual are submitted to the supervisory authority for approval. Table 14-1 lists the nature and number of the mentioned in-service inspections, which is typical of a nuclear power plant with a pressurised water reactor.

Table 14-1 Annually Performed In-service Inspections, Typical for a PWR with one Major Refuelling Outage per Year

Items	during operation	during outage	total
Functional tests	2 780	330	3 110
Radioactivity monitoring system	395	15	410
Lifting gear	75	5	80
Non-destructive tests	-	40	40
Civil structures	50	10	60
Physical protection	150	-	150
Total	**3 450**	**400**	**3 850**

Apart from the mandatory in-service inspections of systems and components important to safety, the licensee performs additional inspections under his own responsibility, which primarily serve to increase plant availability.

In connection with the in-service inspections and the evaluation of operational experience, special attention is paid to the early detection of failure causes due to ageing. The causes of such failures can often be put down to systematic phenomena. There are specific regulatory requirements regarding ageing of certain plant components (e.g. fatigue analyses as part of component design, or type tests of instrumentation and control equipment in accordance with [KTA 3503] or [KTA 3504]). Due to the high frequency of inspections of the safety equipment in German nuclear power plants, ageing phenomena are usually detected at an early stage and counter-measures are taken. This is why failures due to ageing caused by systematic phenomena have so far been observed only rarely. A special case is the neutron embrittlement of the pressure-retaining boundary of the reactor pressure vessel. To be able to assess the change of the material properties due to neutron irradiation, suspended surveillance samples of the original material of the reactor pressure vessel have to be tested at several intervals over the entire operating lifetime of the nuclear power plant. The test results deliver fracture mechanical parameters on which an assessment of the integrity of the reactor pressure vessel can then be based.

Corresponding results are available for all plants and show sufficient fracture toughness until the end of the scheduled operating lives. In addition, the licensee performs the legally required tests and inspections on components in accordance with the conventional standards and regulations (e.g. the Operational Safety Ordinance).

Inspections under federal and *Länder* supervision

The continuous supervisory activities of the *Länder* require about 30-40 person-years (including authorised experts) per year and power plant unit. The function of federal supervision is to ensure that the *Länder* perform their supervision with consistent quality, especially when it comes to the consideration of recent safety-related findings on a national level. In this context, the federal supervisor seeks the advice of the RSK.

Reportable events, modifications of the plant or its operation, maintenance processes or new insights concerning the requisite safety level can lead to the supervisory authority demanding a safety review of certain systems, components or circumstances. Such safety reviews may also comprise probabilistic analyses. These reviews and analyses are usually carried out by the licensee and are assessed by the authorised experts involved.

There are numerous different plant-specific improvements which have resulted from plant-specific examinations during operation and from the evaluation of national and international operational experience, usually to the benefit of individual components and maintenance measures. These improvements at individual plants will not be dealt with here any further. In addition, there have also been modifications that have each affected a larger number of plants. They are presented in the compilation further below in tabular form, showing backfitting measures and safety-related improvements (Table 14-3).

Against the background of the Chernobyl accident, the Reactor Safety Commission performed a safety review of all German nuclear power plants. One particular focus of the investigations was on the area of beyond-design basis accidents; these investigations led to proposals regarding accident management measures (→ Chapter 18 (i)). It was recommended that periodic safety reviews should be performed every ten years.

Safety review

Since the beginning of the nineties, periodic safety reviews (PSRs) have been carried out according to standardised national criteria. They consist of a deterministic and a probabilistic part and supplement the continuous review process which is part of nuclear supervision. The PSR results have to be submitted to the supervisory authority and are usually assessed by independent experts who act by order of the supervisory authority. At that time, the licensees of the German nuclear power plants had committed themselves voluntarily to performing these PSRs at 10-year intervals. For seven nuclear power plants, such a PSR is a mandatory requirement that has been specified in the corresponding licensing decision.

The amended version of the Atomic Energy Act of April 2002 stipulates the performance of safety reviews every ten years. Due to the phase-out of the commercial use of nuclear energy, the safety review is no longer referred to as “periodic”. The dates for submission of the next SRs were included in the Atomic Energy Act (see Table 14-2). The obligation to present the SR results is lifted if the licensee makes the binding declaration to the licensing and supervisory authority that he is definitively going to terminate power operation at the plant no later than three years after the final date mentioned in the Atomic Energy Act.

The performance of the SR of nuclear power plants is based upon standardised national guidelines for the deterministic and probabilistic safety analysis. The existing guidelines [3-74] are currently being revised to adapt them to the state of the art in science and technology. The deterministic safety assessment of the nuclear power plants is to be based on accidents as compiled in Appendix 3 and furthermore on a spectrum of accident management measures (→ Chapter 18 (i)) to cope with beyond-design basis conditions. The compilation in Appendix 3 reflects the status of the guidelines as currently applicable.

Deterministic safety status analyses and probabilistic safety analyses were performed for all 18 operating nuclear power plants and the Stade nuclear power plant that meanwhile has been shut down.

Table 14-2 Comprehensive Safety Reviews of Nuclear Power Plants

	Nuclear Power Plant		Type	Submission of the SR to the authority
1	Obrigheim	KWO	PWR	31.12.1998
2	Stade *)	KKS	PWR	31.12.2000
3	Biblis A	KWB A	PWR	31.12.2001
4	Biblis B	KWB B	PWR	31.12.2000
5	Neckarwestheim 1	GKN 1	PWR	31.12.2007
6	Brunsbüttel	KKB	BWR	30.06.2001
7	Isar 1	KKI 1	BWR	31.12.2004
8	Unterweser	KKU	PWR	31.12.2001
9	Philippsburg 1	KKP 1	BWR	31.08.2005
10	Grafenrheinfeld	KKG	PWR	31.10.2008
11	Krümmel	KKK	BWR	30.06.2008
12	Gundremmingen B	KRB B	BWR	31.12.2007
13	Grohnde	KWG	PWR	31.12.2000
14	Gundremmingen C	KRB C	BWR	31.12.2007
15	Philippsburg 2	KKP 2	PWR	31.10.2008
16	Brokdorf	KBR	PWR	31.10.2006
17	Isar 2	KKI 2	PWR	31.12.2009
18	Emsland	KKE	PWR	31.12.2009
19	Neckarwestheim 2	GKN 2	PWR	31.12.2009
	Mülheim-Kärlich **)	KMK	PWR	-----

*) end of power operation 14 November 2003
**) decommissioning applied for

Probabilistic safety analyses

In the mid-seventies, Germany began to use probabilistic safety analyses in supplement to the deterministic safety assessments.

Methodically, the German Risk Study Nuclear Power Plants, Phase A (1979) largely followed the lines of the US American “Reactor Safety Study“ WASH 1400 and also investigated the accident consequences resulting from the release of radionuclides in accordance with the state of knowledge at the time this study was prepared. In Phase B of the German Risk Study Nuclear Power Plants (1985 to 1989), advanced methods were applied, but the investigations were restricted to the identification of system damage and core damage frequencies and to investigations on the processes in the containment after core meltdown without determination of release frequencies. For the purpose of further developing these methods of the probabilistic safety analysis (PSA) and to try them out in practice, PSAs have been performed since then for various nuclear power plants as research and development projects outside the regular nuclear licensing and supervisory procedure.

In the years 1990 to 2000, probabilistic safety analyses have been performed for all German nuclear power plants as part of the periodic safety review. The methods and data to be applied for a probabilistic safety analysis are described in supplementary documents to the regulatory guidelines [3-74]. In view of the obligatory performance of PSAs with extended analysis scope within the framework of the Safety Review (SR) required by law, the PSA guideline is currently being revised.

Probabilistic safety analyses now exist for all German nuclear power plants [3-74]; they have led to numerous new insights and system modifications.

Probabilistic safety analysis methods and data bases undergo constant development. Clear progress has been made concerning the expansion of the analysis depth to PSA, Level 2, the inclusion of low-power and shutdown states, the closer evaluation of operator actions, and the consideration of the initiation of fires and their consequences. At the end of 2000, a PSA for a Konvoi-type plant was completed in which the advanced methods available were used. Within this PSA, Level 2 analyses were performed for events during power operation, while Level 1 analyses were carried out for events during low-power and shutdown operation. In future, all safety reviews are to include a Level 2 PSA. As concerns the highly sensitive issue of common-cause failures, Germany participates actively in the international exchange of experiences and data (ICDE-International Common Cause Failure Data Exchange (OECD/NEA)). Germany participates intensively in the international exchange of experience also in other fields of the PSA.

Analyses for BWR plants of construction line 69 are under way and shall be completed at the end of 2004. Objective is the authorities' approval of the PSA methods of Level 2 for power operation.

International safety evaluations, OSART missions

Upon invitation, the IAEA has so far conducted four OSART missions at the following German nuclear power plants: Biblis A (PWR) in 1986, Krümmel (BWR) in 1987, Philippsburg 2 (PWR) in 1987 and Grafenrheinfeld (PWR) in 1991 (mission) and 1993 (follow-up visit).

For the areas of plant operation that were analysed in these missions, namely

- operational management, organisation and administration,
- personnel training and qualification,
- plant operation,
- maintenance,
- technical support,
- radiation protection,
- chemistry, and
- emergency preparedness and prevention,

no major deficiencies were identified. The improvements proposed for individual items were implemented on plant level, as confirmed by the follow-up inspection of the Grafenrheinfeld nuclear power plant. At that time, proposals had not been implemented yet only in those cases where the licensing process had not been finally approved by the licensing authority or where a standard national regulation was to be drawn up first.

For the Philippsburg 2 nuclear power plant, a further OSART mission is scheduled for October 2004.

Backfitting and safety improvements

The safety assessments performed during the operating times of the nuclear power plants have led to numerous backfitting measures and safety-related improvements of the plants. In many cases, results from PSAs were required for it, even if these were performed voluntarily or as research project. Table 14-3 presented below contains the major backfitting measures and safety improvements. Information on the individual backfitting measures is contained in the first and second national report for the corresponding Review Meetings under the Convention on Nuclear Safety.

Conclusion of the safety assessments performed

As result of the safety assessments performed and the resulting backfitting measures and safety-related improvements, it can be stated that the licensed safety status of the plants has not only been maintained but, also, that newer safety findings were given appropriate consideration during the time of operation. Thus, the safety of nuclear power plants has been adapted to the state of the art in science and technology widely and as far as the plant design allows. For the Biblis A plant, the supervisory authority imposed additional upgrading based on the safety analysis for this plant of 1991, which resulted in the licensee's applying for numerous modifications. Parts of the backfitting measures applied for have already been realised. However, other parts of the modifications applied for are still dealt within the licensing procedure.

Table 14-3 gives a survey of the major safety-related improvements that have been implemented for the four design generations of pressurised water reactors and the two construction lines of boiling water reactors.

Solved issues

The following issues, presented in the previous report as pending, can be regarded as solved:

<u>Cracks in weld seams of pipes – non-destructive tests</u>

Cracks in weld seams of the pressure boundaries at the Stade nuclear power plant (chlorine-induced stress-corrosion cracking in the base material) and Biblis A (cracks in the buffer zone of bi-metallic weldings) caused the Federal Regulator to investigate whether similar components at other plants are also affected. On behalf of the Federal Government, GRS investigates unexpected crack indications identified during in-service inspections (as far as reportable). Comprehensive additional tests were performed by the German plant operators and existing documentations on tests re-evaluated. This did not reveal any indications to systematic causes. Apart from a few minor indications there were no others from tests that required technical discussion.

Table 14-3 Major Backfitting Measures in Nuclear Power Plants According to Design Generation and Construction Line

	Objective of improvement / Improvement measures	PWR-design generation				BWR constr. line	
		1	2	3	4	69	72
1.	**Enhanced reliability of specified normal operation**						
	- Additional off-site power supplies	X	X	●	●	X	●
2.	**Enhanced effectiveness and reliability of safety systems and equipment**						
	- Additional emergency diesel generators	X	X	●	●	X	●
	- Additional high-pressure and low-pressure emergency core cooling systems (PWR)	X	●	●	●		
	- Extension of emergency core cooling systems / additional injection lines (PWR)	X	X	●	●		
	- Technical improvement of the high-pressure/low-pressure interfaces	X	X	X	X	X	X
	- Self-supporting emergency core cooling systems / new diversified emergency core cooling system (BWR)					X	X
	- Additional emergency feedwater systems	X	X	●	●	●	●
	- Technical improvement of components important to safety to withstand design basis accidents	X	X	●	●	X	●
	- Additional valves for containment isolation (BWR)					X	●
	- Diversified pilot valves for safety and pressure relief valves (BWR)					X	●
	- Diversified pressure relief valves (BWR)					X	X
3.	**Control of specific emergency situations**						
	- Emergency systems	X	X	●	●	X	●
4.	**Mitigation of fire consequences**						
	- Physical separation by installing new systems in separate buildings	X	●	●	●	X	●
	- Additional fire fighting systems	X	●	●	●	●	●
	- Backfitting of fire fighting systems	X	●	●	●	●	●
	- Technical improvement of fire dampers and fire partitions	X	X	●	●	●	●
	- Additional fire dampers	X	●	●	●	X	●
5.	**Improvement of barriers**						
	- New pipes of improved material for main steam, feedwater and nuclear auxiliary systems (BWR)					X	●
	- Optimised materials for steam generators (PWR)	X	●	●	●		
	- Removal of the former pressurised bearing water system with its connections outside of the containment (BWR)					X	●
6.	**Accident management**						
	- Improvement of technical equipment for damage prevention	X	X	X	X	X	X
	- Improvement of technical equipment for damage mitigation	X	X	X	X	X	X

X improvement through backfitting measures ● already covered by the design

Operating force margins of gate valves important to safety

At the nuclear power plants, there is a multitude of different types of gate valves with different design characteristics, operation conditions and requirements, whose function has to be ensured even in case of an accident. Since testing of the valves under accident conditions (e.g. main steam line break) is not feasible, their operability mainly was demonstrated by analytical calculations.

Since 1990, problems are known related to safety-relevant valves due to increased friction coefficients in the connection between stem and stem nut. Since then, various efforts have been taken to develop a generally accepted status of the requirements for the assessment and verification of operability of safety-relevant valves. By means of analyses and evaluations of individual effects, such as the friction behaviour of gland packings or at valve faces, and of comparisons between functional tests under differential pressure and the results of different calculation methods, the procedure for the determination of the necessary operating torques and for the strength of the components within the load path has been further developed since then. In parallel, assessments of the motor-operated gate and isolation valves installed in nuclear installations were performed on the basis of the respective state of knowledge reached.

To ensure the function of safety-relevant valves, the VGB developed the following concept:

- Demonstration of adequate operating forces as well as of the integrity and operability according to the safety-relevant requirements,
- construction criteria to ensure operability and for the assessment of the applicability of the calculation guideline, and
- maintenance criteria for the long-term maintenance of the function.

After consultations between the Federal Government and the *Länder*, this concept has been implemented.

Quality assurance for the fabrication of fuel elements

Problems related to quality assurance documentation at a foreign fuel element fabrication facility caused the Federal Regulator to require measures for improved quality assurance. This issue was examined by the supervisory authorities and their authorised experts, resulting in the review and improvement of regulations. Within the standard revision cycle, KTA will check how far KTA 1401 has to be revised.

Behaviour of fuel elements with cladding tubes made of new material types

At some German PWR plants, fuel elements are used with cladding tubes containing niobium which are expected to have a higher operational corrosion resistance. The Federal Regulator initiated the comprehensive testing of the new material with regard to its accident resistance. In this respect, the behaviour at high burn-up was particularly taken into consideration. As a result, it can be stated that there is no reason to change the criteria for emergency core cooling.

Further necessary activities

High burn-up of fuel elements

The licensees of the nuclear power plants intend to further increase the target burn-up of the fuel elements. The conservative accident analyses and analyses on damage extent with comprehensive consideration of high burn-up effects, which are required for safety assessment, are only available in part. In this respect, best-estimate analyses including uncertainty analyses will also be considered.

The Federal Regulator regards further experimental analyses on fuel behaviour both under operating and accident conditions as necessary and will monitor progress and result of the investigations started within the framework of OECD research programmes which are also supported by the German plant operators. Furthermore, the calculation methods for the assessment of the fuel and fuel rod behaviour will be examined with regard to the high burn-up effects expected.

The burn-up values reached in the nuclear reactor facilities were recorded. The licensees gave reports on operating experience and the available experimental database for the fuel rod behaviour in case of power ramps or during reactivity accidents. For reactivity accidents, the verifications are based on Japanese experiments at the Nuclear Safety Research Reactor (NSRR) and on the French CABRI-REP-Na test programme. The proof that the maximum permissible fuel rod loads are not exceeded is furnished with three-dimensional core models under realistic and also conservative assumptions. The German utilities participate in the OECD CABRI Water Loop Project in France aiming at the completion of the experimental database for higher burn-up and for representative cooling conditions of the fuel rods. The calculation methods to estimate the fuel rod behaviour in case of LOCAs are reviewed and further developed with regard to higher burn-up and cladding tube materials containing niobium.

ATWS events

The plans of the plant operators to increase the target burn-up of fuel elements and to increase the use of MOX fuel elements have caused the Federal Regulator to examine the safety margins with regard to the control of ATWS events. In Germany, the control of ATWS-events for PWRs is checked for each refuelling. The corresponding requirements are laid down in the RSK-Guidelines [4-1]. In addition, the RSK requires the control of such an event also without consideration of switch-off of the main coolant pumps. In additional analyses, the influence of the core design on the void reactivity curve and the sensitivity of the maximum pressure dependent of the effectiveness of different system functions will be determined.

Boron dilution

Thermal-hydraulic calculations showed that the necessary in-core boron concentration in case of small leak event might not be ensured continuously, thus jeopardizing subcriticality.

New test results from PKL and ROCOM and technical reports on the applicability of the test results to a nuclear reactor facility have been submitted. For the refuellings performed until now, qualified proof on the minimum boron concentration at the core entrance has been furnished. The activities on the validation of the analysis methods for the determination of the condensate quantity produced and accumulated, on the transport of the condensate to the core and on the mixing of the condensate with highly borated coolant, particularly in the lower plenum, are continued.

Digital instrumentation and control (I&C)

At present, digital I&C is already used at some German nuclear power plants for safety-relevant function. In the coming years, modification and backfitting measures of safety I&C at German nuclear power plants on the basis of computer-based systems are expected to be taken increasingly because analogue hard-wired systems are no longer produced and spare parts will not be available in future. Requirements for computer-bases systems with safety relevance currently exist in the nuclear rules and regulations as basic approach. The guidelines of the RSK only include general requirements for software-based safety I&C. However, for the practical examination and evaluation in the nuclear licensing procedure, they are insufficient. For the drafting of the necessary detailed requirements, the Federal Regulator will participate in the development of international standards to an increasing degree and will ensure the transferability to and compatibility with the safety requirements in Germany. This applies especially to the use of prefabricated hard- and software in the safety system.

Hydrogen depletion in case of core melt accidents

At present, catalytic recombiners for hydrogen depletion after beyond-design basis accidents with core melt in the containment are being installed at all German PWRs, except for the Obrigheim and Biblis-A nuclear power plants. However, there are questions to be solved regarding the inadvertent ignition of hydrogen by the passive recombiners (→ Chapter 18 (i)).

Impairment of water suction from the containment sump

Findings obtained by tests performed in the USA caused the Federal Regulator to check the measures initiated in German nuclear power plants due to the event at the Swedish nuclear power plant Barsebäck once again. These measures are to ensure that in case of LOCAs, during which the core has to be cooled with water from the containment sump, the water suction will not be impaired severely by fibres of pipe insulation material or other material. After the re-examination it has to be ensured that the necessary conservatism is given. Experiments on individual aspects are currently being conducted and partly have already been completed. The RSK developed an evaluation basis considering national and international findings.

Chapter 14: Progress and changes since 2001

See paragraph “Issues solved“

Chapter 14: Future activities

See paragraph “Further necessary activities“

15 Radiation Protection

Fundamentals

The Radiation Protection Ordinance [1A-8] is the legal basis for the handling of radioactive material. It includes provisions by which man and the environment are protected from damage due to natural and man-induced ionising radiation. In the Radiation Protection Ordinance, requirements and limits are laid down to be observed when using radioactive material. This also covers the handling of nuclear fuel, as well as construction, operation and decommissioning of nuclear installations in accordance with Section 7 of the Atomic Energy Act.

The ordinance has been amended and revised a number of times over the years, and has been adapted to the respective EURATOM Basic Safety Standards [1F-18]. These are the framework for radiation protection in the European Union. The amendment of the Radiation Protection Ordinance for adaptation to the EURATOM Basic Safety Standards of 1996 entered into force on 1 August 2001. The legal requirements for this amendment have been established before by amending the Atomic Energy Act on 3 May 2000.

The Radiation Protection Ordinance also specifies the requirements regarding licensing and reporting obligations for the handling of radioactive material, for their import, export and their transport, and it specifies requirements for administrative and technical protective measures and for medical surveillance. The scope of application also covers the handling of natural radioactive material.

The following principles of radiation protection specified in Section 6 of the Radiation protection Ordinance are decisive for any activity involving radiation protection:

– Any unnecessary radiation exposure or contamination of man and the environment shall be prevented.
– Any radiation exposure or contamination of man and environment shall be kept as low as practicable even where values are below the authorised limits, taking due account of the state of the art in science and technology and of the conditions of each individual case.

Together with the principle of proportionality - a constitutional principle to be accounted for in all cases - these principles lead to an obligation to minimise radiation exposure.

The essential dose limits specified in the Radiation Protection Ordinance are addressed in the following and listed in Table 15-1. For the first time, comprehensive regulations on the clearance of radioactive material (→ Chapter 19 (viii)) have been stipulated in the Radiation Protection Ordinance.

Occupationally exposed persons

The prescribed limit for the body dose of occupationally exposed persons is a maximum effective dose of 20 mSv per calendar year. Other limits are stipulated for organs and tissues. Stricter limits apply to persons under 18 years and women of childbearing potential. An unborn child must not receive more than 1 mSv due to the occupational exposure of the mother. The sum of effective doses of occupationally exposed persons added in all calendar years must not exceed the life time dose of 400 mSv.

Table 15-1 Dose Limits According to the Radiation Protection Ordinance

Sec-tion	Scope of Applicability	time period	limit value [mSv]
	Design and operation of nuclear installations		
46	**Environment of nuclear facilities**		
	Effective dose: direct radiation including radiation exposure from discharges	calendar year	1.0
	Partial body dose: eye lens	calendar year	15
	Partial body dose: skin	calendar year	50
47	**Limit values for the discharges with exhaust air or waste water during operating conditions**		
	Effective dose	calendar year	0.3
	Partial body dose: gonads, uterus, red bone marrow	calendar year	0.3
	Partial body dose: great gut, lung, stomach, bladder, breast, liver, gullet, other organs and tissues unless specified above	calendar year	0.9
	Partial body dose: bone surface, skin	calendar year	1.8
49	**Design basis accident limit values**		
	Effective dose	event	50
	Partial body dose: thyroid and eye lens	event	150
	Partial body dose: skin, hands, forearms, feet, ankles	event	500
	Partial body dose: gonads, uterus, red bone marrow	event	50
	Partial body dose: bone surface	event	300
	Partial body dose: great gut, lung, stomach, bladder, breast, liver, gullet, other organs and tissues unless specified above	event	150
	Dose limit values for occupationally exposed persons		
55	Occupationally exposed persons in **Category A**		
	Effective dose	calendar year	20
	Partial body dose: eye lens	calendar year	150
	Partial body dose: skin, hands, forearms, feet, ankles	calendar year	500
	Partial body dose: gonads, uterus, red bone marrow	calendar year	50
	Partial body dose: thyroid, bone surface	calendar year	300
	Partial body dose: great gut, lung, stomach, bladder, breast, liver, gullet, other organs and tissues unless specified above	calendar year	150
	Occupationally exposed persons in **Category B**		
	Effective dose	calendar year	6
	Partial body dose: eye lens	calendar year	45
	Partial body dose: skin, hands, forearms, feet, ankles	calendar year	150
	Effective dose for **persons under age 18**	calendar year	1
	Effective dose for trainees and students age 16-18 with agreement by the supervisory authority	calendar year	6
	Partial body dose: uterus of women of childbearing age	month	2
	Dose for the **unborn child**	time of pregnancy	1
56	Effective dose	entire life	400
58	Removal of **consequences of accidents** (only Cat. A, after approval by the supervisory authority)		
	Effective dose	entire life	100
	Partial body dose: eye lens	entire life	300
	Partial body dose: skin, hands, forearms, feet, ankles	entire life	1000
59	Removal of pending danger to persons	calendar year	100
	(only above 18 years, no pregnant women)	once in life	250

Exposures to radiation exceeding these limits may be allowed in order to defend danger to persons. The body doses received in these cases may not exceed 100 mSv in a given calendar year and 250 mSv only once in the lifetime.

For the period up to the year 2005, the amended Radiation Protection Ordinance contains limits which follow the limits of the previous Radiation Protection Ordinance, but still comply with the EURATOM Basic Standards. Table 15-1 presents the values of the new Radiation Protection Ordinance.

The body doses are to be determined for all persons spending any time in the controlled access area. This is usually done by measuring the personal dose. Occupationally exposed persons are examined by authorised physicians according to the exposure categories (A or B).

The Radiation Protection Ordinance also regulates the required documentation of personnel doses and of the results of the medical examination as well as the obligations with regard to reports to the supervisory authority. Data on the radiation exposure of occupationally exposed persons are recorded by the Federal Office for Radiation Protection in a central register.

A nuclear power plant must be designed in such a way that the protective provisions of the Radiation Protection Ordinance are met for the occupationally exposed persons working in the plant. As early as the design stage, the constructor has to consider aspects that are important to radiation protection [3-43], [KTA 1301]. The administrative and technical measures for the radiation protection of workers in nuclear power plants during plant operation are also laid down in [KTA 1301].

Radiation exposure of the general public during operating conditions

The dose limits and requirements applying to the radiation exposure of the general public during operating conditions of nuclear installations are laid down in Sections 46 and 47 of the Radiation Protection Ordinance.

A limit of 1 mSv per calendar year is specified for the effective dose by direct radiation, including the radiation exposure from discharges. In addition, there are limits for specified organs and tissues.

The technical design and operation of a plant or installation has to be planned in such a way that the radiation exposure of the general public caused by discharge of radioactive material with air or water from these plants or installations will not exceed the effective dose limit of 0.3 mSv per calendar year. Further limits are applicable to specified organs and tissues.

Any radioactive discharge is recorded in the nuclide-specific balance sheets. These allow calculating the radiation exposure within the vicinity of the nuclear installation. The analytical models and parameters used in these calculations are specified in the Radiation Protection Ordinance and in a general administrative provision [2-1]. Accordingly, the radiation exposure must be calculated for a reference person at the most unfavourable receiving points. These are locations in the vicinity of the nuclear installation where the highest radiation exposure from the discharge of radioactive material and from the consumption of food produced at these locations would be expected for the reference person. Unfavourable nutritional habits and durations of stay are assumed for the reference person to ensure that the radiation exposure will by no means be underestimated.

Radiation exposure of the general public during design basis accidents

Central issues evaluated during the licensing procedure of a nuclear power plant are the planned structural and other technical measures to protect against design basis accidents (→ Chapter 18 (i)). In accordance with Section 49 of the Radiation Protection Ordinance it has to be shown - notwithstanding the obligation to minimise radiation exposure - that the effective dose in the vicinity of the nuclear installation will not exceed the planning value of 50 mSv in a design basis accident (integrated over all exposure paths as 50-year dose commitment). Further planning values apply to specified organs and tissues. Regulatory guideline [3-33] specifies the analytical models and assumptions to be applied for these verifications. For other nuclear installations, the competent authorities specify in accordance with Section 50 of the Radiation Protection Ordinance the kind and scope of the protective measures taking into account the individual case, especially the hazard potential of the installation and the probability of the occurrence of an accident.

Emission monitoring

The discharges from nuclear installations have to be monitored according to Section 48 of the Radiation Protection Ordinance, specified according to type and activity and reported to the competent authority at least once a year.

The basic requirement for emission monitoring is converted into concrete measurement programmes. These are specified in the regulatory guideline on emission and immission monitoring [3-23]. In the general part of the guideline, the objectives and the basics of emission and immission monitoring are stated, and the requirements applicable to all nuclear installations are explained. In the appendices, the different measurement programmes are listed according to the type of nuclear installation.

Appendix A of regulatory guideline [3-23] pertains to nuclear power plants and refers to the corresponding KTA safety standards with respect to the monitoring of emissions. Safety standard [KTA 1503.1] deals with monitoring the discharge of radioactive material through the vent stack of nuclear power plants during operating conditions and [KTA 1503.2] in the case of design basis accidents. The corresponding requirements for measurements regarding the monitoring of discharge with water are specified in [KTA 1504].

The emission surveillance programme specified in regulatory guideline [3-23] is carried out by the licensee of the nuclear installation under his own responsibility. The measurement results are then submitted to the supervisory authority.

In order to be able to evaluate the radiological effects of emissions during operating conditions as well as in the case of design basis or severe beyond-design basis accidents, the licensee records the site-specific meteorological and hydrological parameters important to the dispersion and deposition of radioactive material. Generally, a meteorological instrumentation is installed for this purpose which continuously records all meteorological parameters that are required in the dispersion calculations [KTA 1508]. The major parameters influencing dispersion and deposition in the receiving water are also determined; these are the average water runoffs of the river over the full length of the year and over the six-months summer period.

The dose from direct irradiation is determined directly by monitoring the local dose at the fence surrounding the nuclear installation.

In addition to the monitoring equipment of the licensee there are also measuring instruments belonging to the competent authority, e.g. within the exhaust stack, that enable a remote and independent monitoring by the supervisory authority via the KFÜ data network (see below). The balancing measurements by the licensee of the weekly, monthly, quarterly and yearly water and air samples are regularly checked by an independent laboratory [3-44].

Immission monitoring

According to Section 48 of the Radiation Protection Ordinance, the licensees of the nuclear power plants perform a programme on immission monitoring in the vicinity of the plant as ordered by the authorities. In addition to the immission measurements by the licensee, the competent authority employs an independent organisation to monitor the environment for immissions routinely. These measurements are performed to supplement the emission monitoring with the purpose:

– to perform an additional check of the activity release,
– to verify that the dose limits in the environment are not exceeded, and
– to detect any long-term accumulation in the environment.

Administrative authorities of the Federal Government perform comparative measurements and analyses for quality assurance.

The programmes for immission monitoring in the environment prior to commissioning and during operating conditions for nuclear power plants, nuclear fuel fabrication facilities and facilities for interim storage and final disposal of nuclear fuel are specified in the appendices of the Regulatory Guideline on Emission and Immission Monitoring [3-23] for the licensee and for the independent institution, respectively.

In order to be able to detect an increase of radioactivity with respect to the natural background in the vicinity of nuclear installations, a measurement of background radiation is required for a two-year period prior to commissioning. The extent of this programme is oriented towards the measurements to be performed during operating conditions. In addition, it is specified at what time and to what extent which monitoring measures can be discontinued and which must continue to be performed after final shutdown of the plant and its long-term safe enclosure.

In accordance with regulatory guideline [3-23] it is, furthermore, required that the licensee and the independent institutions keep in readiness, and test, sufficient quantities of equipment for taking samples and for the measurements and evaluation in the event of a design basis accident or severe accident. The corresponding accident measurement programmes are specified in the appendices of the guideline both for the licensee and for the independent institution. These programmes are intended for the first measurements after the occurrence of an event. The programmes are conceived in such a way that the radiological situation can quickly be determined and evaluated by means of the specified measurements.

The required surveillance programmes must take every exposure path into consideration that could lead to a radiation exposure of a human being. The samples and measurements are defined in such a way that all relevant dose contributions from direct irradiation, inhalation and ingestion can be determined during operating conditions and in the case of design basis accidents or severe accidents.

Remote monitoring of nuclear power plants

As mentioned before, the radiation measurement programme performed under the responsibility of the licensee is subject to an independent surveillance where different measurement programmes are applied. The majority of these are discontinuous measurements on samples taken over more or less extended time periods. A continuous surveillance of actual plant parameters is performed with the remote monitoring system for nuclear power plants (KFÜ) [3-54]. A selection of measured variables from

– operation,
– monitoring of emissions,
– monitoring of immissions, and
– meteorology.

is transmitted online directly to the competent supervisory authority of the *Land*. This system is in operation at all times during operating conditions and in case of incidents or accidents, as far as the corresponding instruments are suited for, and still available under these conditions.

Integrated measurement and information system

In addition to the site-oriented surveillance of nuclear power plants described above, extensive measurements of the radiation level are performed in the entire region of the Federal Republic of Germany by means of the Integrated Measurement and Information System for the Monitoring of Environmental Radiation (IMIS) in accordance with the Precautionary Radiation Protection Act [1A-5]. Even slight changes in environmental radiation are quickly and reliably detected and evaluated by this system, making it possible to give early warnings to the public, if so required. IMIS is permanently in operation. In the event of increased values, the BMU will cause IMIS to switch from routine to intense operation which, essentially, means that measurements and samples will be taken more frequently.

The nation-wide measurement network is comprised of more than 2000 measurement locations, the data of which are continuously transmitted to the Central Federal Agency for the Monitoring of Environmental Radioactivity operated by the Federal Office for Radiation Protection and from there on to the BMU. The extent and procedures for the corresponding measurements are specified in the general administrative provision [2-4] and in the regulatory guidelines for routine operation [3-69] and for intense operation [3-69-2]. The results from these measurements are also used within the framework of international information exchange (→ Chapter 16 (2)). Two maps on environmental radiation have been placed in the Internet and are updated on a weekly basis.

Results of radiation protection in nuclear power plants

The data on the discharge of radioactive material with air and water from German nuclear installations and on the resulting radiation exposure are published by the Federal Government in its annual report “Environmental Radioactivity and Radiation Exposure“ to the *Bundestag* (the German Federal Parliament), and in a further more detailed annual report with the same name issued by the BMU. Generally, the nuclide-specific annual discharges stay considerably below the emission limits individually licensed for each nuclear power plant. For the most part, they reach just a few percentage points of these limits. In turn, the values calculated for the radiation exposure of the general public is far less than 5 % of the maximum allowed dose limits specified in the Radiation Protection Ordinance until now.

The nuclide-specific discharges of radioactive material from German nuclear power plants in 2002 are listed in Tables 15-2 and 15-3. With respect to tritium, there are technical reasons for the fact that its discharge with the waste water from PWR plants reaches a level up to 35 % of the licensed limits.

Table 15-2 Discharge of Radioactive Materials with Exhaust Air from Nuclear Power Plants in 2002

	Noble Gases [Bq]	Aerosol [Bq]	Iodine 131 [Bq]	Tritium [Bq]	Carbon-14 [Bq]
PWRs					
Obrigheim	1.3 E+12	2.1 E+05	2.2 E+04	9.8 E+10	6.2 E+10 [1)]
Stade	1.7 E+12	1.1 E+05	2.4 E+06	6.5 E+11	9.5 E+10 [2)]
Biblis A	3.8 E+11	1.5 E+06	2.3 E+04	4.8 E+11	3.1 E+11 [3)]
Biblis B	4.4 E+11	2.2 E+05	1.9 E+05	1.9 E+11	2.0 E+11 [4)]
Neckarwestheim 1	4.4 E+11	3.0 E+05	7.4 E+04	1.2 E+11	2.3 E+11 [5)]
Unterweser	3.0 E+12	7.3 E+05	< MDL *)	4.2 E+11	6.9 E+10
Grafenrheinfeld	7.6 E+10	1.7 E+06	< MDL	2.5 E+11	2.6 E+11
Grohnde	2.8 E+11	1.1 E+05	8.6 E+06	5.8 E+11	3.6 E+11 [6)]
Philippsburg 2	3.2 E+12	1.8 E+05	3.9 E+05	2.9 E+11	2.2.E+11 [7)]
Mülheim-Kärlich	< MDL	< MDL	< MDL	< MDL	3.0 E+10
Brokdorf	1.7 E+12	< MDL	1.5 E+06	2.5 E+11	3.3 E+11 [8)]
Isar 2	2.8 E+11	< MDL	< MDL	4.8 E+11	5.4 E+11
Emsland	1.5 E+11	2.3 E+04	< MDL	1.4 E+12	4.0 E+11 [10)]
Neckarwestheim 2	3.5 E+11	5.3 E+04	< MDL	2.0 E+11	2.2 E+11 [11)]
BWRs					
Brunsbüttel	7.4 E+11	4.8 E+06	1.8 E+06	4.4 E+10	1.7 E+11
Isar 1	9.8 E+11	< MDL	6.9 E+06	6.7 E+10	3.1 E+11
Philippsburg 1	6.6 E+10	3.5 E+06	6.1 E+06	3.5 E+10	5.5 E+11
Krümmel	1.2 E+12	7.5 E+06	2.6 E+08	3.8 E+10	9.8 E+10
Gundremmingen B+C	1.4 E+12	4.3 E+04	4.6 E+07	1.2 E+12	9.8 E+11

*) < MDL: less than minimum detectable limit
1) of the total value 1.7 E+10 Bq are due to CO_2
2) of the total value 1.4 E+10 Bq are due to CO_2
3) of the total value 3.2 E+10 Bq are due to CO_2
4) of the total value 3.7 E+10 Bq are due to CO_2
5) of the total value 4.9 E+09 Bq are due to CO_2
6) of the total value 3.9 E+10 Bq are due to CO_2
7) of the total value 8.5 E+10 Bq are due to CO_2
8) of the total value 1.0 E+11 Bq are due to CO_2
9) of the total value 1.7 E+11 Bq are due to CO_2
10) of the total value 2.0 E+11 Bq are due to CO_2
11) of the total value 1.0 E+11 Bq are due to CO_2

The personal dose of the workers in German nuclear power plants has decreased continuously over the past years. Figure 15-1 shows the annual average collective dose in nuclear power plants of different design generations and construction lines. The peaks shown for the nuclear power plants with BWRs in the early eighties was caused by extensive backfitting measures in the nuclear sector performed in those years.

The decrease in personal doses is a direct result of the continuous improvement in the fields of radiation protection and maintenance and in personnel management. In this context, the technical design of the three nuclear power plants with PWRs that most recently were taken into operation is of particular interest. The uncompromising abstention from using any materials containing cobalt in almost all components of the primary coolant boundary has led to a major reduction of the amount of Co-60 among the corrosion products in the coolant water. This, in turn, has noticeably reduced the dose rate at the corresponding components in comparison to older plants with PWRs. This is reflected in the low accumulated personal doses for the pressurised water reactors of the 4th design generation, especially during the annual outage (Figure 15-2).

Table 15-3 Discharge of Radioactive Materials with Waste Water from Nuclear Power Plants in 2002

	Fission and activation products (without Tritium) [Bq]	**Tritium** [Bq]	**α-emitters** [Bq]
PWRs			
Obrigheim	6.0E+07	5.9E+12	< MDL *)
Stade	1.4E+07	3.3E+12	< MDL
Biblis A	3.0E+08	1.7E+13	< MDL
Biblis B	2.2E+08	1.5E+13	< MDL
Neckarwestheim 1	4.6E+05	1.2E+13	< MDL
Unterweser	3.9E+08	1.2E+13	< MDL
Grafenrheinfeld	2.3E+07	2.1E+13	< MDL
Grohnde	2.4E+07	1.8E+13	< MDL
Philippsburg 2	3.9E+08	1.6E+13	< MDL
Mülheim-Kärlich	3.0E+07	1.4E+10	< MDL
Brokdorf	4.8E+06	1.8E+13	< MDL
Isar 2	8.3E+04	1.9E+13	< MDL
Emsland	1.8E+04	1.5E+13	< MDL
Neckarwestheim 2	1.7E+08	1.7E+13	< MDL
BWRs			
Brunsbüttel	3.4E+08	1.3E+11	< MDL
Isar 1	6.1E+07	3.5E+11	< MDL
Philippsburg 1	2.0E+08	4.6E+11	< MDL
Krümmel	9.9E+06	6.1E+11	< MDL
Gundremmingen B+C	7.3E+08	5.9E+12	< MDL

*) < MDL : less than minimum detectable limit

With respect to BWRs, a reduction of the personal dose of plant and external personnel was reached in the nineties in particular by two measures:

– omission of the pressurised bearing water system due to the reconstruction of the forced circulation pumps in BWR line 69, and
– reduction of the number of weld seams to be inspected due to replacement of pipes.

Primarily, this led to a considerable reduction of the duration of stay of the personnel in the radiation-monitored area, which is reflected in the reduced personal dose. As it is the case for PWRs, the more favourable features with regard to radiation protection also show up in BWRs of the newer line.

In the ten-year period from 1994 to 2003, up to 20 nuclear power plants (without the Mülheim-Kärlich plant) have been in operation, which corresponds to a total of 191 reactor operating years. During operation of these nuclear power plants, i. e. without the times after final shutdown of two of these plants, a total of 1191 events was reported which were reportable according to the nuclear reporting ordinance [1A-17] (→ Chapter 19 (vi)). 28 of these events led to radiological impacts which all remained limited to the plant itself. In none of these cases, an excess of the limits regarding discharges with exhaust air or waste water, or measurable releases of radioactive material outside the enclosed site of the plants occurred. Further, these events did not cause any excess of permissible personal doses, either.

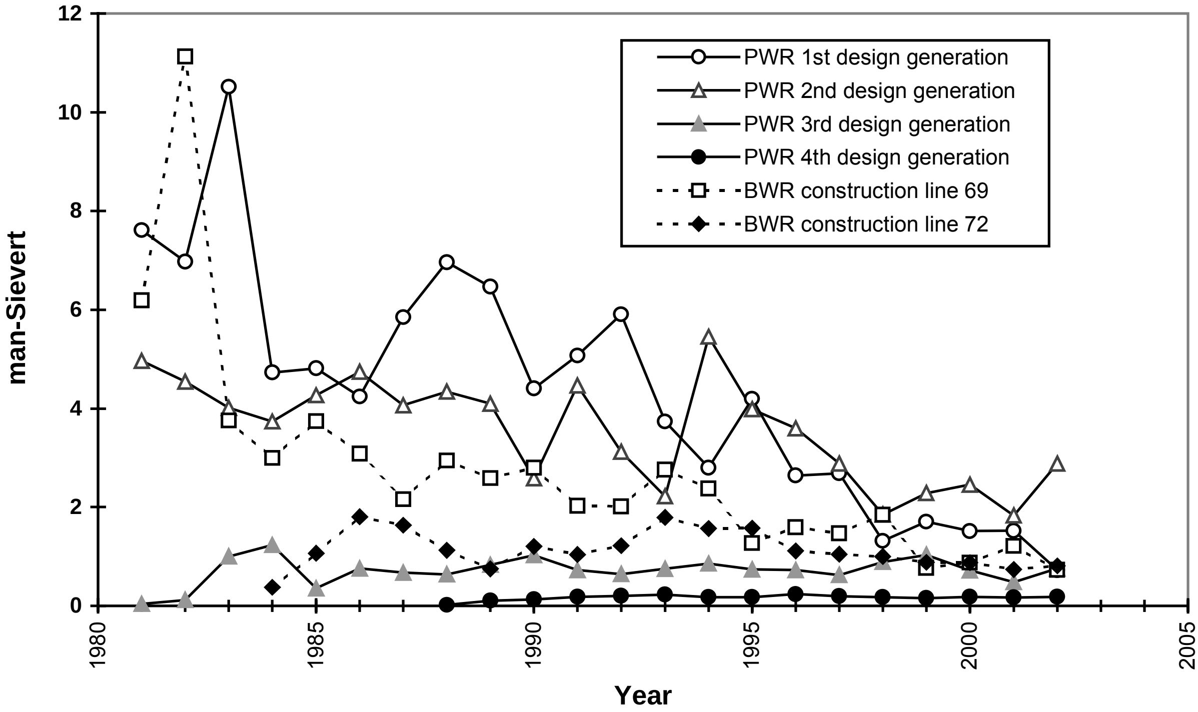
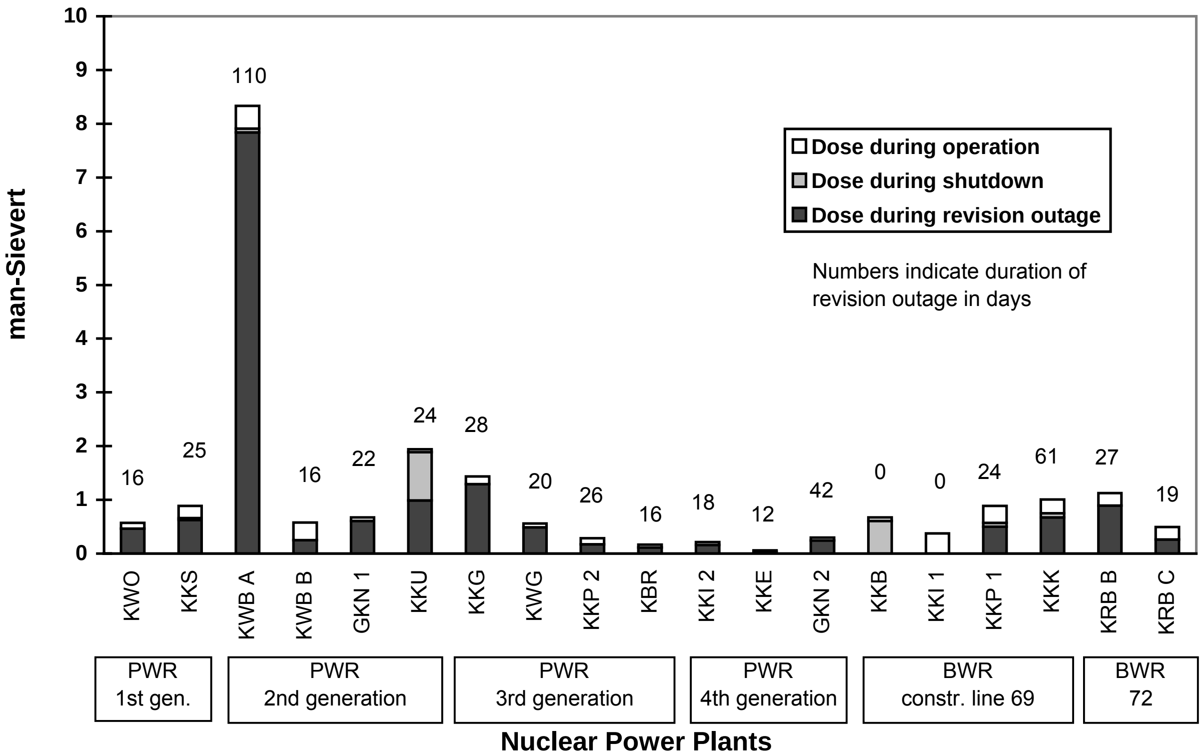


Figure 15-1 **Average Annual Collective Dose in Nuclear Power Plants**



Dose during operation:	Collective dose during power operation
Dose during shutdown:	Collective dose during plant shutdowns other than that for plant outage
Dose during revision outage:	Collective dose during plant outage (scheduled plant revision and refuelling)

Figure 15-2 **Annual Collective Dose in Nuclear Power Plants in 2002 According to Mode of Operation**

16 Emergency Preparedness

16 (1) Emergency Preparedness, Emergency Plans

Taking the federal structure of Germany and the resulting responsibilities at national and regional level into account, a nuclear emergency preparedness concept has been established in Germany which in case of an event can take effect at a very early stage, i.e. already if the release limits of operating conditions are exceeded or in the event of an accident.

Emergency preparedness supplements the measures implemented by the licensees for design basis accident control (→ Chapter 19 (iv)) and on-site accident management (→ Chapter 18) by regulatory off-site measures concerning precautionary radiation protection and disaster control.

Precautionary radiation protection measures serve for preventive health protection of the population also below the intervention levels of disaster control. They mainly lie within the responsibility of the Federal Government and comprise recommendations for measures especially in the area of agriculture. The measures are agreed with the competent federal ministries and the supreme *Länder* authorities.

Disaster control measures serve for immediate danger defence. They are regionally limited and of temporary nature. Responsibility for them lies with the authorities of the interior of the *Länder*, and the regional or local administrative authorities. In the event of a disaster, a disaster control task force is formed which – depending on the practice in each *Land* – is set up at the government of the *Land* or at one of the regional governmental agencies.

The task of the Federal Government within the framework of disaster control is the support and harmonisation of the associated measures. For this purpose, the Federal Government and the *Länder* have jointly prepared the "Basic Recommendations for Emergency Preparedness in the Environment of Nuclear installations" and the "Radiological Bases for Decisions on the Protection of the Population against Accidental Releases of Radionuclides" [3-15]. Apart from that, the Federal Government has the duty to inform foreign authorities. The language-independent ECURIE system facilitates the information exchange with foreign authorities in the event of a nuclear accident within the European Union.

If the authorised release limits for operating conditions are exceeded or if a design basis accident occurs, the licensee is obliged by the Guideline on Emission and Immission Monitoring [3-23] to measure the effects on the environment and to communicate the findings to the competent authority. Independent of these activities, the above-mentioned situation may also be detected by the Remote Monitoring System for Nuclear Power Plants (KFÜ) (→ Chapter 15) and, in the case of releases, by Integrated Measurement and Information System (IMIS) (also → Chapter 15).

In case of any threats caused by larger releases, e.g. in a nuclear accident, the competent authorities will take measures for the protection of the population within the vicinity of the plant. After the licensee has alerted the competent authorities, he supports them by way of measurements, information and data transmission, and by giving advisory information about his assessment of the situation within the plant. This support is independent of his own plant-internal measures to control the situation or to avoid or limit the release (→ Chapter 19). In the adjacent area outside where disaster control measures are no longer justified, precautionary radiation protection measures are taken to reduce radiation exposure of the population. As long as the events only have regional consequences, the competent radiation protection authority of the *Land* may also take precautionary measures, depending on the situation, to protect public health according to the Precautionary Radiation Protection Act [1A-5] even if the intervention levels of disaster control are not exceeded. These measures may take the form of e.g. bans on the consumption of certain foodstuffs or of behavioural instructions. If several *Länder* are affected by such a release, the responsibility to take measures according to the Precautionary Radiation Protection Act lies with the federal authorities.

The reporting channels provided and the specified obligation of the licensee to report to the competent *Land* authority enable the fulfilment of the duty to inform the public [1F-28] and the involved authorities and organisations at the international level [3-23, 1E-5].

Organisation of emergency preparedness

Due to the federal structure of Germany, authorities and organisations on different levels co-operate in case of an emergency in order to ensure – in the case of an event in Germany together with the licensee – the protection of the population by taking precautionary radiation protection or disaster control measures, depending on the situation (Figure 16-1).

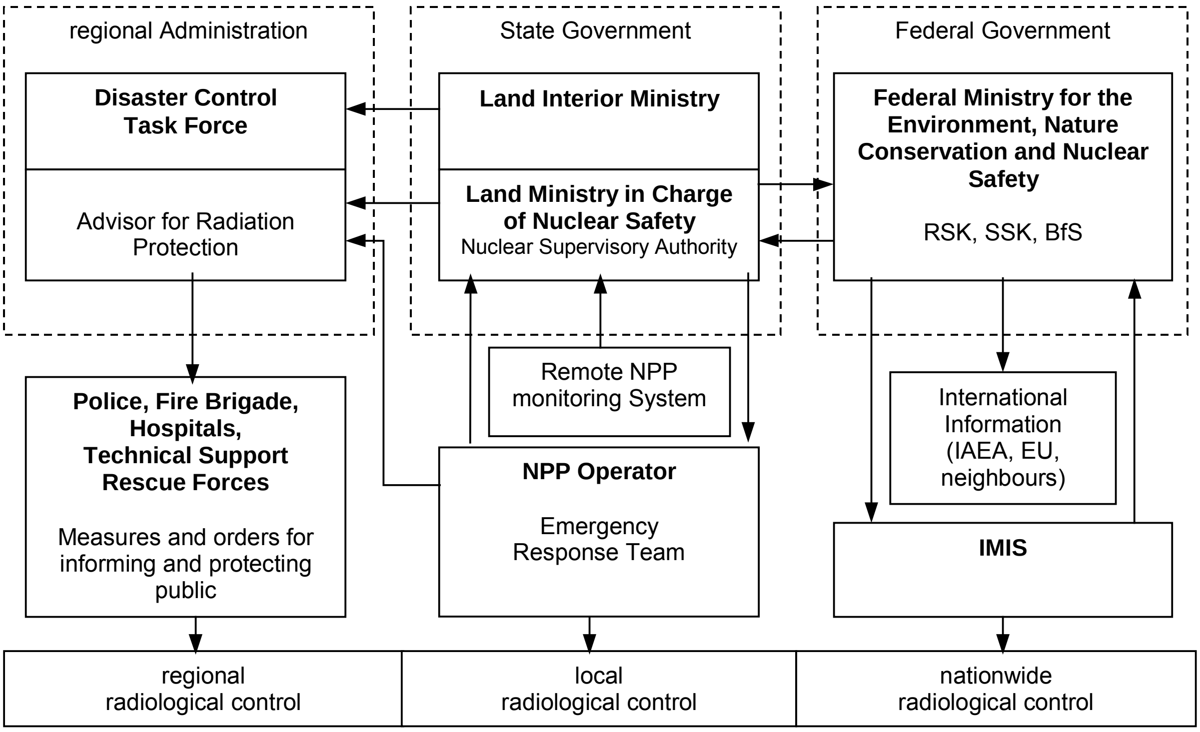


Figure 16-1 Organisational Diagram Disaster Control

Laws, ordinances, guidelines and recommendations

The Precautionary Radiation Protection Act [1A-5] specifies the responsibilities in the event of a not insignificant release of radioactive substances and contains regulations concerning

- measuring tasks of federal and *Länder* authorities to monitor radioactivity in the environment,
- establishment of an integrated measuring and information system (IMIS) including a central federal office for monitoring radioactivity in the environment,
- authorisation to define dose and contamination limits,
- authorisation to ban or restrict the use of foodstuffs, feedstuffs, drugs or other substances,
- authorisations concerning cross-border traffic.

The *Länder* have defined the tasks of disaster control by law. The different structure of the *Länder* as well as the time when these laws were developed contributed to the fact that there are different regulations, as in some cases different responsibilities of the authorities.

The Basic Recommendations [3-15] form the basis for the development of special emergency plans in the vicinity of the nuclear installations by the *Länder* authorities. The Basic Recommendations deal with:

- the obligation of a licensee of a nuclear installation to co-operate with the disaster control authority,
- the principles for the contents of the special emergency plans for the vicinity of nuclear installations, and
- explanations and notes concerning the alert measures provided.

The intervention levels for the initiation of measures, the issue of iodine prophylaxis and the setting-up and running of emergency centres are all explained in detail in several appendices.

The Radiological Bases within [3-15] describe the bases, such as exposure paths, and the consequences of radiation exposure to health. The measures to protect the population are described and the intervention levels derived for all measures. General references for decision making after occurrence of an event and the radiation protection for special occupational groups supplement the Radiological Bases.

The alarm criteria themselves are contained in the operating manuals of the nuclear power plants; they regulate the early alerting of the disaster control authority. Apart from the immission and emission criteria, which are oriented towards the intervention levels for disaster control measures, there are preliminary technical criteria that have been differently realised from plant to plant. They allow a much earlier judgement of the plant condition than only emission and immission criteria or just immission criteria alone would allow.

Responsibilities on the federal level

The Precautionary Radiation Protection Act [1A-5] stipulates that on federal level, the BMU is responsible for the measures in the area of precautionary radiation protection. Decisions are taken in co-ordination with other federal ministries. Here, other agencies subordinate to the BMU become active.

In the field of disaster control, in peacetime responsibility lies with the *Länder*. In case of a nuclear accident, the BMU assists and advises the *Länder* and co-ordinates if necessary the measures taken by the *Länder*.

Additional to these tasks, another function of the BMU is to fulfil its international and bilateral information obligations. As part of these obligations, the BMU maintains contact with the European Union, the IAEA and with other nations. In this context, corresponding bilateral agreements have been concluded with a large number of states (→ Chapter 16 (2) and 17 (iv)).

Within the framework of federal supervision of the execution of the Atomic Energy Act, the BMU has to ensure that it can become active in certain dangerous situations without delay, be it for commenting from the point of view of the federal supervision or for the supervisory correction of decisions taken by the *Länder*. Further, the BMU then has the function of a point of contact, information and – for certain tasks – co-ordination, focussing on the following issues:

- Decision about protection measures for the population within the scope of the Precautionary Radiation Protection Act,
- co-ordination between the different technically competent federal ministries,
- requests for assistance from other ministries,
- measures to co-ordinate activities of federal and *Länder* authorities,
- recommendations to *Länder* authorities,
- consultation/involvement of other organisations for assistance within the scope of the Precautionary Radiation Protection Act,
- information of the public,
- provision of information for external national and international organisations as part of the reporting obligations of the BMU.

Responsibilities on the *Länder* level

The competent *Länder* authorities draw up special emergency plans for nuclear installations and also perform, as part of their federal executive administration, tasks of precautionary radiation protection falling into their competence. These include measurements of environmental radioactivity, e.g. the determination of radioactivity levels in foodstuffs, feedstuffs, drinking water, ground water and surface water as well as in the soil and in plants. Other responsibilities according to the Precautionary Radiation Protection Act remain with the *Land* as far as the event is restricted to the territory of that particular *Land*.

On *Länder* level, several ministries of the *Land* government are responsible as technical authorities for emergency preparedness issues. They make independent decisions within the scope of their functions or advise other ministries within the *Land* government. The following distinction has to be made between

- disaster control authorities,
- the nuclear supervisory authorities,
- the precautionary radiation protection authorities, and
- the departments subordinated to these authorities.

Depending on the *Land*, several of these tasks may be within one single ministry.

Disaster control authorities on the *Land* level

The tasks in connection with disaster control on *Land* level are generally performed by the interior authorities in their function as supreme disaster control authority. Their task is to take short-term measures to protect the population in the vicinity of a facility affected by an accident. These tasks are distributed over the various different administrative levels of the authorities responsible for disaster control within the *Land.* Generally, the *Land* authorities are responsible for *Land*-wide co-ordination of the general planning activities whereas the subordinate regional disaster control authorities are responsible for drawing up the special emergency plans for the respective facilities, for the preparation and execution of measures, and for the qualification and the training of the task forces.

Nuclear supervisory authorities

In an emergency, the competent nuclear supervisory authority represents - in addition to its supervisory functions – the link in the communication chain between the licensee of the nuclear installations, the ministries concerned of the *Land* government, and the BMU with regard to information about the technical condition of the installation and the expected further sequence of the event. Supervisory authority personnel co-operate with the disaster control authority personnel of the *Land* and also assumes advisory functions.

Authorities responsible for precautionary radiation protection

The tasks of this authority of a *Land* are to perform requisite measurements according to the Precautionary Radiation Protection Act and to deal with technical issues of radiation protection. The radiation protection authority is integrated in the co-ordination process with the BMU about measures to be taken and implements the decisions that have been made on *Land* level. Together with the nuclear supervisory authority, it supports and advises the disaster control authority by providing assessments of the radiological situation. Certain tasks of the radiation protection authority may be delegated to subordinate authorities.

Responsibilities of the licensees

The licensee is responsible for the preparation and performance of the on-site accident management to prevent or reduce the consequences of an event sequence for the environment (→ Chapter 18). Part of the organisational prerequisites established in all nuclear power plants to control emergencies is an emergency response team that is supported by personnel from the operating staff. The emergency response team should be able to take up work within an hour. Suitable rooms, working appliances and means of communication are provided. Contractual agreements exist about co-operation with external institutions such as the Nuclear Emergency Service Company, a joint service set up by all nuclear power plant licensees to cope with emergencies and eliminate possible consequences.

The licensee takes care of the necessary qualification of the personnel as well as of the exercises necessary to preserve the knowledge and skills. Outside the facility, in case of an event involving a radioactive release, he is obliged to carry out measurements and take samples in the vicinity of the plant and in the most affected sector and to pass on the results of the measurements and evaluations to the authority [3-23]. In case of an event, the licensee alerts the competent authorities. Alarm procedures and organisational structures are specified in the operating manual, and the individual technical measures to be taken are described in a separate document, the accident management manual. A description of the transition from the operating manual to the accident management manual is contained in Chapter 19 (iv).

Measuring tasks to determine the radiological situation

Apart from the licensee's measuring programmes to monitor the environment during operation and in an emergency - the scope and individual features of which have been specified in the licence and in [3-23, 3-23-2] – *Land* as well as federal authorities also carry out independent measurements. The scope, kind and frequency of these measurements is oriented towards the respective requirements. In addition to the prognoses according to the plant's condition, the results of the emission and immission measurements in the vicinity of the nuclear power plant are decisive for the initiation of disaster control measures (→ Chapter 15). Depending on the situation, the remote monitoring system for nuclear power plants KFÜ (→ Chapter 15) may also be used by the supervisory authority for further assessing the plant's condition and the radiological situation.

In Germany, the integrated measuring and information system IMIS is used for an extensive and plant-independent monitoring of environmental radioactivity (→ Chapter 15).

Classification of emergency situations

In an emergency situation, the licensee alerts the disaster control authorities [3-15]. The licensee decides in accordance with the alarm criteria in the operating manual whether an early warning or an emergency alert has to be triggered in case of an event.

To allow a comparable assessment of events to the general public, the international INES scale - developed by IAEA - is applied. For the different event groups (Table 16-1) also given are the respective INES levels and the associated areas of national emergency preparedness. Other events, such as satellite crash, transport accidents during national and international transports involving large activities, danger situations resulting from threats to abuse radioactive substances (nuclear crime) or accidents with large stationary sources are covered by the categories indicated.

Table 16-1: **Grouping of Events for Planning of Emergency Preparedness Measures**

	Event	Categorisation according to INES-Scale	Categorisation Disaster Control vs. Precautionary Radiation Protection
at home	Design basis accident	3	Precautionary Radiation Protection
	Serious nuclear accident	4 to 7	Disaster Control (local site area) Precautionary Radiation Protection
abroad	Design basis accident (neighbouring foreign country)	3	Precautionary Radiation Protection
	Serious nuclear accident (neighbouring foreign country)	4 to 7	Disaster Control (local site area) Precautionary Radiation Protection
	Serious nuclear accident (far away foreign country)	4 to 7	Precautionary Radiation Protection

Emergency plans of the *Länder*

Special emergency plans are drawn up according to [3-15] as part of the precautions taken by the authorities to protect the population in the vicinity of nuclear installations. These plans document the competencies and responsibilities with regard to the disaster control task force and to the resources available for disaster control measures.

For nuclear power plants, these plans are drawn up for a radius of about 25 km around the plant. Plans for the performance of disaster control measures for the population are made for a radius of 10 km, beyond this radius, points for measurements and sampling measuring are defined and alerts prepared. The fact that plans are only provided for a certain area/radius does not mean that it is impossible to perform any measures to protect the population beyond this radius. The protection concept is based on the assumption that in the vicinity of the plant, disaster control measures may have to be performed at short notice and that therefore the measures for this area should be planned in advance. The decision about for which area an emergency alert is triggered in an acute case is solely orientated on the radiological intervention levels for disaster control.

Emergency plans of the licensees

The measures provided by the licensee to protect the population and the operating personnel are defined in the operating manual and the accident management manual. The organisational requirements applying to plant-internal events are laid down in the alarm regulation contained in the operating manual. This includes criteria for plant-internal measures, for convening the plant-internal emergency response team, and for alerting the disaster control authorities; furthermore, instructions for measures to be taken by the licensee's emergency response team within the plant itself and in support of the disaster control authorities (e.g. measuring teams) are also provided. Accident management measures (→ Chapter 18) are described in the accident management manual.

The licensee is obliged by the Basic Recommendations to assist the authority in drawing up the emergency plan for the plant.

Protection measures for the population

Disaster control measures taken in the event of an accident in a nuclear installation serve for immediate danger defence in the vicinity and are generally restricted in time. They are initiated as soon as a hazardous release of radioactive substances into the environment has been detected or is about to take place that may lead to the intervention levels for disaster control measures being reached or exceeded. In accordance with the principle that danger defence has greater priority than measures of precautionary radiation protection, disaster control measures are carried out with priority in an area affected by a nuclear accident. These disaster control measures are planned in advance to ensure that they can be performed effectively. The following measures to protect the population are provided as part of disaster control planning:

- sheltering,
- distribution of iodine tablets,
- evacuation,
- bans on the consumption of fresh, locally produced foodstuffs.

The decision about the first three measures is taken on the basis of radiological intervention levels; banning the consumption of foodstuffs is ordered as a precautionary measure which will either be confirmed or lifted afterwards when measuring results are available. The radiological intervention levels pertaining to the disaster control measures are laid down in [3-15] (Table 16-2). They are defined as so-called "starting levels", which means that any measures taken in case of dose levels below the intervention levels are not justified from a radiological point of view. Relocation as subsequent and supplementary measure outside an evacuation measure is not carried out on the basis of previously prepared plans but in dependence of the prevailing radiological situation determined from measurements.

Table 16-2: **Intervention Levels for Protection Measures [3-15]**

Measures	Intervention level		
	Thyroid dose	**Effective dose**	**Integration period, Exposure paths**
Staying indoors		10 mSv	exterior exposure within 7 days and effective dose caused by inhaled nuclides within this period
Taking iodine tablets	50 mSv children and teenagers up to age 12 and pregnant women *) 250 mSv persons of age 18 to 45		inhaled radioactive iodine within 7 days including effective follow-up dose
Evacuation		100 mSv	exterior exposure within 7 days and effective dose caused by inhaled nuclides within this period
Long- term resettlement		100 mSv	exterior exposure within 1 year caused by deposited nuclides
Short-term resettlement		30 mSv	exterior exposure within 1 month

*) the age limitation according to [3-15] with 12 years was raised to 18 years following recent considerations

The distribution of potassium iodide tablets was regulated by an ordinance [1A-20]. The *Länder* hold stocks of iodide tablets for the 25-km zone so that they can quickly be distributed when they are needed. In addition, potassium iodide tablets are stored at several central storage facilities and distributed by the Länder authorities in the 25-100-km in case of need.

Measures as well as recommendations according to the Precautionary Radiation Protection Act about which a decision is taken on the basis of acquired or, where necessary, predicted data are provided in the following areas:

– Measures in the field of agriculture.
  These comprise measures to prevent contamination of agricultural products, to reduce contamination during production or processing, and to improve soil quality in the long run or change the way agricultural surfaces are used.
– Measures to protect the population in the area of every-day life and work.
  These comprise e.g. measures in connection with the replacing of filters, measurements and decontamination in cross-border traffic, and behavioural measures and recommendations for the population with a view to radiation protection.

Of particular relevance is the fact that on the basis of the Precautionary Radiation Protection Act [1A-5] it is possible to define national dose and contamination limits for the initiation of measures and to ban or restrict the consumption of foodstuffs and feedstuffs.

Further, the measures of disaster control and of precautionary radiation protection are compiled in a catalogue of measures. It contains reference levels as a basis for decisions about the initiation of the respective measures. These reference values are oriented towards the intervention levels given in [3-15], a reference level of 1 mSv, and on the maximum limits of the EU concerning radioactivity levels in foodstuffs and feedstuffs [1F-30].

Training exercises

Emergency preparedness measures can only be performed effectively if the participating personnel and parties are properly qualified and prepared for this task. Therefore, corresponding training exercises are of particular importance.

In accordance with [3-2] and [3-38], the basic and advanced training of the responsible shift personnel also includes auxiliary measures required in the case of unforeseen events.

The emergency preparedness measures of the licensee are practised regularly at the plant, in particular also the co-ordination with the on-site emergency response team. In recent years, exercises close to reality have increasingly been carried out using plant simulators. Training exercises involving the plant manufacturer are performed every three years. They are part of the contractual agreements between the licensee and the manufacturer which also extend to the setting-up of emergency response centres on the manufacturer's premises with technical equipment as well as to other supporting measures.

The disaster control authorities at *Land* and regional level also perform large-scale emergency exercises at the nuclear power plant sites, albeit at intervals of several years due to the considerable efforts and expenditure required. In these exercises, the interaction between the different organisations and authorities involved is practised. The licensee also takes part in these exercises. Active involvement of the population potentially affected only takes place to a very small extent. The exercise scenario is worked out by the authority. Generally, this involves the assumption of a release into the environment but no reference to a specific accident sequence within the plant.

As part of international co-operation and on the basis of bilateral contracts, representatives from authorities of neighbouring countries are also involved – at least as observers, but usually also in an active manner – in exercises concerning plants near the border. BMU representatives take part – in line with their respective responsibilities - in the regular exercises of the European Union (ECURIE exercises) and the OECD/NEA (INEX exercises), in which supporting agencies, other federal ministries and the relevant *Länder* authorities also participate depending on the situation. Recently, a harmonisation of disaster control measures takes place within the framework of these bilateral contacts by the countries involved in case of nuclear installations in border regions.

16 (2) Informing the General Public and Neighbouring Countries

The requirements of the EURATOM directive regarding the information of the public in case of a radiation emergency [1F-29] have been incorporated in the Radiation Protection Ordinance according to which the general public has to be informed at least every five years of the planned safety measures and of appropriate behaviour in such a situation. The most important facts to be included in this information are:

– basic terminology on radioactivity and its impacts on humans and the environment,
– radiation emergencies and their consequences for the population and the environment,
– information on how the affected persons will be alerted and how they will be continually updated on the development of the situation, and
– information on how the affected persons should behave and what they should do.

This is realised by means of a brochure – financed by the licensees – which is posted to the population living in the vicinity of a nuclear installation.

If an event occurs in a nuclear installation, the population is informed by the licensee and the competent authorities and, if need be, alerted by the authority. While the licensee provides information about the plant state, the competent authority will issue instructions and information concerning the performance of disaster control measures or call attention to behavioural recommendations for precautionary radiation protection if the situation calls for such measures, in addition to providing information about the technical and radiological situation. The plans also provide for the co-ordination of public announcements or press releases among those involved.

In the event of an emergency, the measuring data acquired within of the above-mentioned monitoring programmes will be the basis for the reports required in accordance with the EU agreement on rapid information exchange [1F-28], the corresponding agreement with IAEA [1E-6] which was signed by Germany in 1989, and for fulfilling any of the bilateral agreements. This ensures that Germany's neighbouring countries will receive timely information. The measurements routinely performed in accordance with [3-69] are also used for the reports to the EU in accordance with Article 36 of the EURATOM Treaty.

Germany has signed bilateral agreements regarding mutual assistance in the case of an emergency with all of the nine neighbouring countries. This has also led to co-operation on a local level wherever sites are located close to the borders. In addition, corresponding assistance agreements have been concluded with Lithuania and the Russian Federation; similar agreements with Italy and Bulgaria have been initialled or are in preparation.

Furthermore, agreements regarding information on nuclear safety and radiation protection have been signed with the following 14 states: Argentina, Brazil, Bulgaria, China, Finland, United Kingdom, Japan, Canada, Norway, Sweden, Spain, Ukraine, Hungary and the USA.

16 (3) Emergency Preparedness of Contracting Parties without Nuclear Installations

not applicable to Germany

Chapter 16: Progress and changes since 2001

– The ordinance concerning potassium iodide tablets (Kaliumiodidverordnung) now forms the legal basis for stockpiling the tablets.
– The criteria for alerting the disaster control authority by the operator of a nuclear installation were revised on the basis of a common recommendation of RSK and SSK.

Chapter 16: Future activities

– The revision of the Basic Recommendations for Emergency Preparedness in the Environment of Nuclear Facilities [3-15] will start in 2004.
– In 2004, the potassium iodide tablets required as thyroid blocker will be produced and distributed to the *Länder* and then stored at central facilities.

17 Siting

In Germany, no licences are granted for the construction of new nuclear power plants. Therefore, the following presentation is limited to procedures regarding the siting of the existing plants in operation, the design against external impact and their current evaluation and the potential siting of research reactors.

17 (i) Evaluation Criteria for Site Selection

Uniform criteria for the evaluation of sites for nuclear power plants are specified in regulatory guideline [3-12] and are applicable in all *Länder*. This guideline contains, in particular, the site-specific criteria important to the selection of the site by the licensee and to the nuclear licensing procedure and, in addition, those criteria pertaining to the suitability of the site with respect to regional planning as well as to nature conservation and care for the countryside. With respect to nuclear safety, the following points must be taken into account:

- meteorology with regard to atmospheric dispersion conditions,
- hydrology with regard to cooling water supply, the discharge of radioactive material via the water path, and the protection of drinking water supplies,
- population distribution at the site and in the vicinity,
- geological condition of the building ground and seismics,
- external impact through flooding, from aircraft crash or pressure waves of possible explosions off-site,
- roads and transportation with regard to availability and site accessibility,
- distance to military installations.

Proceeding within the licensing procedure

After the applicant has pre-selected a site, a regional planning procedure is initiated which precedes the nuclear licensing procedure. This takes into account all impacts of the individual project on the public, on traffic ways, regional development, protection of the countryside and nature conservation. Besides the site characteristics, the design of the plant against external impact is checked in the nuclear licensing procedure (→ Chapter 7 (2ii)). Further, investigations are carried out as to whether general public interests oppose the selection of the site. Within the nuclear licensing procedure, the respective competent authorities analyse if the requirements regarding water utilisation, immission control and nature conservation are met. The licenses of the German nuclear power plant have all been granted before the European Directive on Environmental Impact Assessment [1F-12] entered into force; assessments of environmental impacts were exclusively performed according to national law.

Design against external impact

The safety criteria [3-1] require that all plant components necessary to safely shut down the reactor, to remove residual heat or to prevent uncontrolled release of radioactive material shall be designed to be able to perform their function even in the case of nature or man-induced external impact. In this respect, the following has to be considered:

- external impact caused by nature, such as earthquakes, flooding, extreme weather conditions, and
- man-induced external impact, such as aircraft crash, impact of dangerous and explosive substances, and
- malevolent acts or other illegal interference by third parties.

The design requirements specified in [3-33] for these external impacts distinguish between design basis accidents and other impacts, which, on account of their low risk, are not considered as design basis accidents, and for which measures must be taken to minimise the risk . The external impacts caused by nature (earthquake, floods, external fire, lightning) are considered as design basis accidents, whereas external impacts such as aircraft crash and pressure waves or impacts from dangerous materials from outside of the nuclear power plant require risk minimisation.

The requirements for the design and for protective measures in the event of external impact for construction of the existing German nuclear power plants followed the then current nuclear safety regulations. In cases where detailed requirements were not yet formulated in the regulations, the concrete requirements were specified in the respective licensing procedure. The steps in developing these requirements are described below. The corresponding re-evaluation of nuclear installations is dealt with in Chapter 17 (iii).

All nuclear installations have not only been designed taking into account natural external impacts, such as wind and snow, but also floods and - where there is a risk of this kind - against earthquakes. In this context, both, nuclear safety standards and conventional civil engineering standards were applied. Depending on the overall cooling concept for the nuclear installation, the system design results also in requirements important to safety for the cooling water supply. It has to be verified for the individual site conditions that the cooling water supply will function even under unfavourable conditions, e.g. low water in the receiving water or failure of a river barrage.

<u>Design against floods</u>

The requirements for flood protection measures are included in the nuclear safety standard [KTA 2207]. The design of the plants was based on the respective applicable version of this standard. The sites of the nuclear power plants mostly are located inland at rivers and, in some cases, at estuaries with tidal influences. In most of the cases, sites have been selected which are located sufficiently high. In other cases, the civil structures important to safety were insulated for water tightness and were built with waterproof concrete. Furthermore, the openings (e.g. doors) are located above the level of the highest expected flood. If these measures should not be sufficient, mobile barriers are available to close off openings. KTA 2207 is available as updated draft, the modifications concern the design basis flood.

<u>Design against earthquakes</u>

Since 1990, the design against earthquakes is based on a design basis earthquake (formerly called safe shut-down earthquake) in accordance with safety standard [KTA 2201.1]. The so-called operating basis earthquake, formerly considered additionally, was replaced by an inspection earthquake where only the plant condition has to be checked. This design basis earthquake has the largest intensity that, under consideration of scientific findings, could occur in a wider vicinity of the site (up to a radius of about 200 km). Depending on the site, the intensity of the design basis earthquake varies between less than 6 and a maximum of 8 on the MSK scale. In the older nuclear installations, the seismic qualification of civil structures, components and plant equipment was partly based on simplified (quasi-static) procedures which delivered the basic values for the corresponding design specifications. In more recent nuclear installations the newly developed dynamic analyses were also applied. It is planned to adapt the definition of the design basis earthquake, the requirements for the determination of the seismic impact on civil structures and plant components as well as the methods for the verification of an adequate design to the state of the art in science and technology.

Protection against aircraft crashes

Protection against aircraft concerns the accidental crash of an aircraft onto safety-relevant plant components. The protection measures were taken against the background of the increasing number of nuclear power plants in Germany in the seventies and a high crash rate of military aircrafts in those years. The general basis was the analysis of the crash frequency (the theoretical impact frequency for the reactor building averaged over all sites amounted to about 10^{-6} per year and plant) and of the loads on the reactor building that would be caused by such a crash. From the mid-seventies onwards, load assumptions were developed for the event of an aircraft crash which were then applied to the design of preventive measures in the nuclear power plants built in the following years. In accordance with [4-1], a site-independent impact-load-over-time diagram is used for the design with an impact time of 70 ms and a maximum impact load of 110 MN. Since the end of the eighties the crash rate of military aircraft has decreased considerably. This has the effect that the crash frequency today can be assumed to be smaller by about one order of magnitude.

For the more recent nuclear installations, the design against aircraft crash also covered aside from the reactor building further civil structures containing systems required for the control of this external impact (e.g. the emergency feed-water building in more recent PWR). Furthermore, protective measures were taken to account for pressure waves from aircraft crash, e.g. by uncoupling the ceilings and inner walls from the outer wall or by a special design, so that no vibrations would be induced in components and internals.

Protection against pressure waves from explosion

The requirements for protecting nuclear power plants against pressure waves from chemical reactions in case of an accident outside of the plant were developed in the seventies due to the specific situation of nuclear power plants located on rivers with corresponding ship traffic and transport of explosive goods. The load assumptions - based on a maximum overpressure of 0.45 bar - are specified in regulatory guideline [3-6] and are being applied since its publication independently of the individual site. Furthermore, with respect to possibly larger peak pressure at the accident location itself, a sufficient safety distance is kept from potential sources of explosions (e.g. traffic routes, industrial complexes).

17 (ii) Evaluating of Impacts

With the impact that an operating nuclear power plant has or could have on the environment and on the people living in its vicinity, distinction is to be made between conventional impacts which would also emanate from other industrial facilities and radiological impacts both during operating conditions of the plant and in case of design basis accidents.

Conventional impacts of the nuclear installation on the environment

The construction or essential modification of nuclear installations must also fulfil special requirements under the laws on protection against dangerous conventional environmental effects, e.g. air pollution with toxic or corrosive materials, and noise pollution. Since the early nineties, these requirements are assessed explicitly on the basis of the Act on the Assessment of Environmental Impacts [1F-12] (→ Chapter 7 (2ii)). The impact of the nuclear installation on the environment are comprehensively determined, described and evaluated by this assessment. The objective is to keep any detrimental environmental impact during operation of a nuclear installation as low as possible. In this respect, the provisions of the Federal Immission Control Act [1B-3] must be observed together with its individual ordinances.

The heat input to rivers or water bodies from discharged cooling water during power operation (either from fresh water cooling systems, or from direct-contact cooling systems with wet cooling towers) is not permitted to exceed the limits specified in the licensing procedure. If, under extreme weather conditions, it is foreseeable that these limits would be exceeded, the respective nuclear installation must reduce its power accordingly. The heat input should not cause a temperature increase, ΔT, in excess of 3 to 5 K. In the past, weather conditions have caused the power to be reduced at some German sites.

An individual licensing procedure according to the water law is required with respect to the utilisation of water and to the discharge of cooling water and waste water. This is performed in close co-ordination with the nuclear licensing procedure.

Radiological impacts during operation and design basis accidents

The Radiation Protection Ordinance [1A-8] specifies dose limits for the radiation exposure of the general public to be adhered to during operating conditions and planning values for the radiation exposure during design basis accidents. These are dealt with in Chapter 15.

17 (iii) Re-evaluation to Ensure Continued Safety Acceptability

Chapter 17 (i) describes the current design of German nuclear power plants against external impact. Within the safety reviews, extensive analyses and evaluations have been performed with regard to the actual protective condition of the nuclear installations. Recent findings concerning safety and developments of the nuclear safety regulations have also been included. The safety reviews which are scheduled to be repeated every ten years (→ Chapter 14 (ii)) also include a re-evaluation of the protective measures of the particular nuclear installation with regard to external impacts. These take site-specific factors into consideration and the development in the state of the art regarding the load assumptions, their effects, and the behaviour of the civil structures and components under the assumed loads resulting from individual external impacts. As a result of the reviews, measures have been taken or planned as far as necessary.

Essential developments and more recent evaluations with regard to the external impacts of floods, earthquakes, aircraft crash, and pressure waves from explosion are described below.

<u>Floods</u>

The re-examinations on flood protection in the years 2000 to 2002, initiated by the federal supervisory authority, showed that the plant-specific specifications on the design basis flood as well as on the technical and administrative protection measures generally are in compliance with the safety standard [KTA 2207] applicable at that time. However, the results of the examinations also show that the approaches on the determination of the design basis flood as well as the maintenance of the flood protection measures are different. This raises further questions which, in the end, are due to the lack of specification in the nuclear rules and regulations. The draft of the correspondingly amended nuclear safety standard [KTA 2207] on flood protection was issued in 2003. The approach on the determination of the design basis flood was specified. This updated rule will be applied to all future modification licences where flood protection is concerned. Moreover, it has to be referred to as assessment criterion for safety reviews, e.g. the SR.

Earthquakes

For older nuclear installations, the ongoing development of methods to determine seismic load assumptions and to verify design specifications led to a re-evaluation of seismic safety. Regarding the approach for the determination of seismic load assumptions, different opinions showed up among the experts. The different opinions are characterised by different approaches in the seismogeographical and seismotectonical zoning, different approaches regarding the database for the determination of site-specific ground response spectra, and the different consideration of probabilistic methods. In general, the re-evaluations with regard to the design of components showed that, on the basis of more precise seismic parameters and modern verification methods, the technical equipment of the plants has considerable margins with respect to seismic loading. For some older plants instead (e.g. Philippsburg 1 and Biblis A), the re-evaluations also indicate the necessity for comprehensive backfitting of systems and components. Regarding the nuclear safety standards [KTA 2201.1], it was concluded from expert discussions that the basic seismic design requirements have to be revised, which particularly applies to the development of seismic load assumptions.

Aircraft crashes

As regards accidental aircraft crash, the older nuclear installations were re-evaluated with regard to the transfer of the respective loads in conjunction with the probabilistic safety assessments. The results from these assessments showed that even in the cases where the reactor building does not withstand the load assumptions as defined today, the contribution to damage states with considerable release is assessed to be sufficiently low. A further risk reduction was achieved by backfitting the older plants with physically separated auxiliary emergency systems that are completely independent from other systems (→ Chapter 14 (ii)). All in all, the risk contribution from accidental aircraft crash is considered as being negligible.

Pressure waves from explosion

In those cases where the design of nuclear installations did not already account for protective measures against pressure waves from explosion and where such an external impact cannot be precluded due to the site conditions, corresponding analyses were performed in the course of the safety reviews. The results show that in almost every case the actual structural design will withstand the specified assumed loads. In every case, however, the nuclear installations are sufficiently protected under general risk aspects. The certifications required in the licensing procedures for industrial complexes ensures that new industries settling in the vicinity of nuclear power plants will not entail any unconsidered events that could endanger the nuclear power plant.

17 (iv) Consultations with Neighbouring Countries

From a very early stage, Germany took up cross-border consultations in connection with the construction of nuclear installations in the border regions. Bilateral agreements regarding the exchange of information on those nuclear installations built in the border regions have been signed with six of Germany's nine neighbouring countries: the Netherlands, France, Switzerland, Austria, the Czech Republic and Denmark. The subject matters of these agreements comprise the following:

– taking the interests of the neighbouring country into consideration when selecting the site,
– accessibility of licensing documents,
– the area of obligatory mutual information, and
– the framework for meetings.

Joint commissions for regular consultations on questions of reactor safety and radiation protection were formed with the Netherlands, France, Switzerland, Austria and the Czech Republic. The information exchange on nuclear installations in the border region concerns the following:

– technical or other modifications relevant to licensing
– operating experience especially with regard to reportable events,
– general reports on developments in nuclear energy policy and in the field of radiation protection,
– regulatory development of the safety requirements especially with regard to accident management measures in the case of severe accidents.

The legal obligation in Europe for a cross-border participation of the competent authorities [1F-12] was transposed into German law by a corresponding amendment of the Nuclear Licensing Procedure Ordinance [1A-10]. Accordingly, the competent authorities of neighbouring countries participate in the licensing procedure if a project could considerably affect the other country.

Germany signed the Espoo Convention on Environmental Impact Assessment in a Transboundary Context [1E-1], the EU accessed to it bindingly.

In accordance with Article 37 of the EURATOM Treaty, the European Commission will be informed of any plan for discharging radioactive material of any sort. For this purpose, general information on the planned discharge, on the site and the essential characteristics of the nuclear installation are reported to the Commission six months before the competent authority issues a licence permit for the discharge in question. This serves to establish the possible impacts on the other member countries [see also 1F-14]. After a hearing with a group of experts, the Commission presents its position on the case of intended discharge.

Seen together, the German legal regulations, the bilateral agreements and the joint commissions put neighbouring countries in a good position to independently assess the impacts nuclear installations in border regions will have on the safety of their own country. Chapter 16 (2) has already dealt with the joint agreements with neighbouring countries regarding information exchange and mutual assistance in the case of emergencies, and with the further agreements entered into with other countries, the IAEA and the EU.

Chapter 17: Progress and changes since 2001

– The flood at the French nuclear power plant Blayais at the end of 1999 and the revision of the nuclear safety standard KTA 2207 on flood protection caused the Federal Regulator to re-examine the design against flood at all German nuclear power plants. The revision work is in a very advanced stage.

- In connection with the extremely low water levels of the rivers used for cooling water supply and high river water temperatures, which was observed at several German sites in 2003, safety-related reviews were performed.
- Buildings, components and plant equipment of the existing plants were designed against earthquakes according to the procedure established at that time. In the meantime, the methods to determine seismic load assumptions have undergone further development. It has meanwhile become international practice to use site-specific ground response spectra. The re-assessments of existing nuclear power plant sites have revealed relevant differences in the assessments of the experts.
- At the instigation of the Federal Regulator, the RSK submitted recommendations on the revision of KTA 2201.1 (Design of Nuclear Power Plants against Seismic Events).

Chapter 17: Future activities

- The revision of safety standard KTA 2207 has not been finalised yet because the approach for the determination of the design basis flood still has to be subjected to an in-depth verification. Accordingly, the planned revision of plant design on the basis of the updated KTA safety standard is also still in process.
- Further examinations to determine the safety margins in the plant design with regard to low water are intended.
- In addition, generic studies on the impact of climate changes with extreme weather conditions on the safety of the nuclear power plants are planed in the long term.

18 Design and Construction

18 (i) Safety Concept

In Germany, the main safety principle for the peaceful utilisation of nuclear energy is the protection of life, health and property against the hazards of nuclear energy and the detrimental effects of ionising radiation. This principle is established in Section 1 of the Atomic Energy Act and it governs the design and safety concept of the nuclear power plants. These must include effective safety precautions that will protect the plant personnel and the public as well as the environment from the radioactivity related to the operation of the nuclear installation.

In order to ensure this protection, the radioactive substances - essentially concentrated in the reactor core - are retained by a number of barriers. These are the fuel-rod cladding, the pressure-retaining boundary of the reactor coolant system and the containment. To ensure the integrity of these barriers, the changes of reactivity in the reactor core must be kept within narrow limits, and the fuel elements must always be sufficiently cooled. These three protection goals - in the IAEA standards referred to as fundamental safety functions, sometimes as basic safety functions - supplemented by the fourth protection goal, the limitation of radiation exposure, cover all requirements whose fulfilment ensure the protection of life, health and property. To achieve these protection goals, the following basic requirements must be met:

<u>Control of reactivity</u>

– Reactivity changes are kept limited to permissible values,
– it is possible to safely shut down the reactor core and keep it in a subcritical condition in the long term,
– the fuel elements always stay subcritical during handling both in the storage area for fresh fuel elements as well as in the spent fuel pool.

<u>Cooling of fuel elements</u>

– Both coolant and heat sinks are kept available,
– the transport of heat from the nuclear fuel to the heat sink is ensured,
– heat removal from the spent fuel pool is ensured.

<u>Confinement of radioactive material</u>

– The radioactivity contained in the reactor core is safely retained by several barriers which, together, will ensure sufficient leaktightness in the event of design basis accidents,
– it is ensured for the inventory of other radioactive material in the nuclear installation that an impermissible release of radioactive material into the environment is prevented even in case of leakages.

<u>Limitation of radiation exposure</u>

– Both the inventory and flow of radioactive material in the nuclear installation are monitored and kept below prescribed limits,
– the discharge of radioactive material is kept below prescribed limits,
– civil structures and technical equipment fulfil the requirements of radiation protection,
– radiation and radioactivity are monitored in the nuclear installation and in the environment.

The criteria for the fundamental safety functions and, in turn, the integrity of the barriers are ensured by applying a concept of progressive safety at several levels of safety (the "defence-in-depth concept“). This concept comprises a combination of measures for the prevention of abnormal occurrences and design basis accidents (Levels 1 and 2), of measures for their control (Level 3) and, finally, of measures for the mitigation of the consequences from any severe accident (Level 4) that might occur. The classification of the plant states is presented in Table 18-1.

At the first level of safety - the operating level - high quality of design and manufacturing and careful operational management shall ensure a high availability of the nuclear installation; at the same time, abnormal occurrences shall be prevented by it.

The following basic principles are to be observed:
- sufficient safety margins in the design of systems and plant components,
- careful choice of materials, comprehensive material testing,
- comprehensive quality assurance during manufacturing, construction and operation,
- independent examination of the quality achieved,
- quality surveillance by in-service inspections (corresponding with the operational loads),
- ease of maintenance of the systems taking the possible radiation exposure of the personnel into account,
- reliable monitoring of the operating conditions,
- taking operating experience into account,
- comprehensive training of the operating personnel, and
- prevention of inadvertent operation, e.g. by interlocks.

Despite of this, malfunction of systems or components may occur, which then might lead to abnormal operating conditions of the second safety level. For the control of these abnormal operating conditions, the systems are designed or operational measures have been provided in such a way that the plant stays within the design limits for specified normal operation so that dose limits for the population are not exceeded (Table 15.1). Such preventive measures at the second level are:
- the reactor core is designed such that even in case of failure of control equipment, the temperature and pressure in the core will reach stable conditions without need for manual actions,
- the control room is equipped with status indicators and failure alarms to inform the operating personnel and to enable any required manual actions,
- control and limitation equipment will keep the nuclear power plant within permissible design limits, and prevent abnormal occurrences from developing into accident situations.

At the third level of safety, the nuclear power plants are designed against postulated accidents, the design basis accidents. Within the licensing procedure it has to be demonstrated that these accidents can be kept under control by safety equipment in such a way that radiation exposure of the most affected individual of the population stays below the accident planning values specified in Table 15.1 even in case of radiologically relevant occurrences.

The demonstration of accident control is realised by means of a safety analysis, originally of deterministic nature. In the course of time, probabilistic investigations supplemented these safety analyses, e.g. reliability analyses of the reactor protection system and of the emergency core cooling systems.

Table 18-1 Safety Levels in the Defence-in-depth Concept

Safety Level			Measures	Objectives
1	operating conditions	normal operation	quality of the operating systems and procedures as well as safety consciousness at work	prevention of abnormal occurrences
2		abnormal operation	inherently safe plant behaviour; limitation systems	prevention of design basis accidents
3	design basis accidents		inherently safe plant behaviour; passive and active safety equipment	control of design basis accidents
4	beyond-design basis accidents	specific, very rare events	specific precautionary measures	control of specific very rare events
			on-site accident management measures	prevention of core damage
		severe accidents, emergencies	on-site and off-site accident management measures	limitation of the impacts to the environment in case of core damage

The deterministic safety analysis consists of a system analysis and an accident analysis. The system analysis is performed to demonstrate that the available systems for operation and surveillance are designed so that accidents due to abnormal operation are prevented with a high reliability. Furthermore, these analyses shall demonstrate that reliable technical safety equipment is available for the control of design basis accidents. Another important task of the system analysis is the demonstration that the safety equipment meets the following design principles in accordance with [3-1]:

- redundancy; generally, a single failure in conjunction with a maintenance case is postulated,
- diversity,
- largely unmeshed system trains,
- physical separation of redundant system trains,
- high degree of automation (the 30-minutes concept),
- safety margins, and
- fail-safe behaviour of the systems in the event of malfunction.

In an accident analysis, the behaviour of the nuclear power plant is analysed for all operational transients and design basis accidents to be postulated in accordance with the nuclear safety regulations (Appendix 3). Central issue of the analysis is to demonstrate the effectiveness of the safety equipment and systems. The analytical models applied take into account all important physical phenomena and have been verified by experiments as far as possible. The conservativity of the results is ensured by using unfavourable assumptions and boundary conditions. If calculation methods and input data have sufficient quality, nowadays best-estimate analyses are also performed and the uncertainties in the results are indicated.

The individual accidents for the design of the safety equipment (design basis accidents) are chosen to be representative for all events to be postulated. In this respect, adequate damage prevention can also be demonstrated by meeting preceding technical criteria, e.g. the

Table 18-2: Implementation of Accident Management Measures at PWRs

5/2004

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

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

regarding reactivity gradients and of temperature and pressure limits at which the integrity of barriers for the containment of radioactive material cannot be endangered.

With the fourth level of safety, events are taken into account in the defence-in-depth concept which originally have not been considered as design basis accidents due to their low probability of occurrence (Appendix 3). At this level, measures are provided against specific, very rare events such as aircraft crashes, external pressure waves and anticipated transients without scram (ATWS). To cope with these events, there are reduced requirements in comparison with the third level of safety regarding the observance of accident planning values, but the verification is similar. Moreover, accident management measures have been implemented at this level since the eighties in order to detect beyond-design basis accidents timely and reliably, to keep them under control and to bring them to an end with as little damage as possible. The preventive measures of accident management are to avoid serious core damage. Main goal is to maintain or restore cooling of the reactor core and to convey the nuclear power plant into a safe condition. The mitigating measures, on the other hand, are to reduce serious radiological impact on the plant site and the environment. Here, the main goal is maintaining the activity-retaining barriers still available and to ensure long-term controlled conditions of the plant for the protection of the environment.

This defence-in-depth concept with its four levels of safety has been implemented until today in all German nuclear power plants.

Accident measures management

The accident management measures are based on a flexible utilisation of available safety and operating systems even beyond design usage - also with the risk of their damage - and on the utilisation of external systems. Extensive technical and administrative precautions have been taken in the German nuclear power plants in order to be able to perform effective accident management measures should an event actually occur.

In addition to the introduction of emergency operating procedures (→ Chapter 19 (iii)) at all plants, the precautions in the case of pressurised water reactors for ensurance of core cooling concern the preventive measures:

– secondary side bleed and feed,
– primary side bleed and feed,

und for activity retention the mitigating measures:

– assured containment isolation,

Table 18-3: Implementation of Accident Management Measures at BWRs

1/2004

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

* wetwell inerted, drywell equipped with catalytic recombiners

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

- RPV primary side bleed,
- hydrogen countermeasures,
- supply-air filtering for the main control room.

In the case of boiling water reactors the preventive measures for ensurance of core cooling concern:
- an independent injection system,
- additional possibility for injection and refilling of the reactor pressure vessel,

and the mitigating measures for activity retention:
- assured containment isolation,
- pressure relief of the reactor pressure vessel,
- filtered containment venting,
- inertisation of the atmosphere of the containment (construction line 69) or of the pressure suppression pool air volume only, supplemented by H_2-countermeasures (construction line 72), and
- supply-air filtering for the main control room.

Auxiliary measures supporting the preventive and mitigating measures in both reactor types are:
- sufficient capacity of the batteries or emergency power supply from neighbouring plant unit (if existent),
- possibilities for a fast restoration of off-site power supply,
- an additional off-site power supply (underground cable),
- sampling system in the containment,
- emergency organisation with training and emergency exercises.

The installation of catalytic recombiners for the limitation of hydrogen accumulation in pressurised water reactors and the realisation of the sampling system for the control of the atmosphere in the containment have been completed for most of the plants. Apart from a few exceptions, all of the other on-site accident management measures have been realised in all nuclear power plants (Tables 18-2 and 18-3). The functional efficiency of the accident management measures is demonstrated on the basis of representative estimations and plausibility considerations. The accident management measures have generally to be feasible, appropriate and effective, as well as compliant with the safety concept of the respective plant. The fulfilment of these requirements has to be verified in the corresponding licensing and supervisory procedures.

18 (ii) Qualification and Proof of Incorporated Technologies

The requirements with respect to qualification and proof of the incorporated technologies and to the reliability of the structures, components and systems important to safety are in accordance with the principles of the defence-in-depth concept. In a general form, these requirements are laid down in the Safety Criteria [3-1]. In detail, the requirements are derived from safety analyses. Details regarding the technical realisation are specified in the nuclear safety regulations. The corresponding KTA safety standards are listed in Appendix 5, in particular the series 1400, 3200, 3400, 3500, 3700 and 3900. In these standards, reference is always made to the employment of proven technologies.

Passive systems

Passive systems are systems not requiring activation with regard to their function (e.g. pipes, vessels).

General requirements apply to the qualification of the materials used. The qualification tests closely follow the practice from engineering experience with industrial installations requiring supervision and from construction regulations. In the case of nuclear power plants, both type and extent of the required certification are expanded in accordance with the safety relevance of the components.

With respect to the structural design, the requirements specify a design optimised with respect to stress and strain and to ease of inspection. In as far as nuclear influences are expected, e.g. by radiation, this is accounted for in the corresponding requirements regarding materials and qualification certifications. The influence of identified quality reducing factors on the safety margins regarding the manufacturing of components with barrier functions was examined, and proof has been delivered that the requirements contained in the standards ensure sufficient margins.

The detailed requirements for a qualification proof of the manufacturing process used are specified in safety standards. Different standards apply, depending on the materials, product forms, or the scope of application, e.g. pressure-retaining boundary, secondary systems, containment and lifting equipment. The qualification proof of a manufacturing process is carried out for each manufacturer individually and is repeated at specified time intervals. An independent authorised expert will participate in certain manufacturing steps that are important with respect to the qualification of the materials, the manufacturing process and the components. The results of the tests are documented. and the evaluations of the authorised experts are submitted to the licensing authority.

Active systems

Active systems are systems activated and controlled by I&C systems, as well as manually operated systems.

The majority of active components and their operating hardware are series-produced items for which extensive industrial experience is available. This applies in particular to the instrumentation and control equipment, such as electric motors, controller drives, switch gears, electronic measuring instruments, data processing equipment and cables. However, components used in mechanical engineering may also be series-produced items. Typical examples are the valves and pumps, as far as they do not belong to the pressure-retaining boundary, but, e.g., those used in cooling water and auxiliary systems and within the range of the turbine. Such equipment is deployed in conventional power producing facilities and in the chemical industry. The same applies to the consumable operating media, like oils, lubricants, fuels, gases and chemicals, e.g. for water conditioning.

The requirements pertaining to the qualification proof of active components of the safety system concentrate on the series production, more than in the case of passive components (→ Chapter 13). Type and extent of the qualification proof are specified both in nuclear and in conventional standards in accordance with the individual safety significance. Wherever specific nuclear influences are expected, e.g. by the ambient conditions, the qualification is proven with supplementary certificates. In those particular cases where no industrial experience is available for individual components, e.g. like the control rod drives or the internal axial pumps for boiling water reactors, the qualification of the technology involved is verified in extensive series of tests. The results of these tests are then submitted to the licensing authority for review.

Extensive cold and warm test runs are performed during plant commissioning in order to verify the proper functioning of the systems, the interaction of components and the effectiveness of the safety equipment (→ Chapter 19 (i)).

Proof of qualification

The qualification of the installed techniques are proven in various ways. These are:
- practical experience with long-term use under comparable operating conditions,
- experimental investigations on the behaviour of the materials and components used under operating and accident conditions,
- proof on the basis of verified models,
- reliability data or service life certificates in the case of the components of the I&C equipment,
- critical load analyses.

The qualification of the computer codes used in the design is also subject to proof.

The test programmes are submitted to the licensing and supervisory authority and are checked by the authorised expert consulted. The authorised expert, furthermore, participates in the tests. With regard to questions important to safety, the authorised expert performs additional controlling calculations preferably with independent analytical models. The authorised expert reviews all aspects subject to the licensing and supervisory procedure with regard as to whether additional requirements are necessary beyond those specified in applicable standards and guidelines.

The feedback of experience from manufacturing and operation are of great significance to the evaluation of qualification proof of the installed techniques (→ Chapter 19 (vi) and (vii)).

Experience feedback from individual failures or from general technical findings has shown in a number of cases that certain technical equipment is or would seem to be ill-suited to long-term operation. It is part of the safety culture in Germany, and has proven very effective, that all parties involved look for a technical solution in consensus together that would not only solve the immediate safety problem but would also bring about long-term improvements. Typical examples for such cases are the replacement of pipes in the main steam and feed-water systems of boiling water reactors both inside and outside of the containment, or the backfitting of diverse pilot valves in the overpressure protection system of boiling water reactors. Other examples are the conversion of all pressurised water reactors to a high-AVT-operation with respect to the secondary-loop water chemistry, or the fabrication of weld seams for better testability with ultrasonic procedures either by machining the weld surfaces or by re-welding the seams on components and pipes in pressurised and boiling water reactors. Furthermore, the instrumentation needed for a more exact determination of local loading, e.g. due to thermal stratifications and fluctuations, was increased in all nuclear installations. The results from these measurements are used both for optimising operating procedures as well as in ageing assessments for a more reliable determination of the utilisation factor of components.

18 (iii) Design for Reliable and Easily Manageable Operation

General requirements with the objective of ensuring a reliable and sound operation of the nuclear power plant are already specified in the Safety Criteria [3-1, 3-51]. These pertain to simplicity of system design, physical separation, as well as to the accessibility for inspections, maintenance and repairs. They also contain further requirements regarding workplace design, work sequences and working environment. Detailed requirements both with regard to technical measures and to the administrative procedures of work tasks are specified in safety standards [KTA safety standard series 1200 and 3200].

The implementation of these requirements led to concepts characterised by the following features (→ Chapter 18 (i), 19 (iii)):

– a design of the barriers meeting all requirements with respect to loads, fabrication and ease of inspection,
– easy and spacious accessibility of components with respect to maintenance, inspections and repair,
– physical separation of systems to avoid interactions between redundant systems,
– high degree of redundancy of the safety system,
– consideration of the possibility for a common-mode failure,
– limitation equipment preceding the safety system,
– high degree of automation of the safety system,
– independent emergency systems,
– ergonomic design of the control room,
– ergonomically designed and self-contained operating manual covering all plant conditions,
– appropriate processing of the alarms to properly represent the situation in the case of abnormal operation, design basis accidents and in-service inspections.

In addition to the general requirements mentioned for all barriers, the concept of "General Specification of Basic Safety" was developed for the pressure-retaining boundary and other pressure-retaining components in the late seventies. It comprises a catalogue of detailed technical requirements with the special objective of preventing catastrophic failure of plant components due to manufacturing defects. These requirements were included in the corresponding KTA safety standards. The basic safety of a plant component is characterised by the following principles:

– high-quality materials, especially with respect to fracture toughness,
– conservative stress limits,
– avoidance of peak stresses by optimisation of the design,
– application of optimised fabrication and test technologies,
– awareness of any possible fault conditions and their evaluation,
– accounting for the operating medium.

These principles were immediately applied in the newer nuclear power plants and have led to post-qualifications in the older plants to comply with these principles or for the assessment of identified non-compliances. The assessments showed a need for extended safety demonstrations and measures which have not been implemented at all plants affected by now (PWRs of the second design generation, → Appendix 4).

The development of the materials for the containments of pressurised water reactors was realised by means of different high-strength steels, as they were used in PWRs of the first to

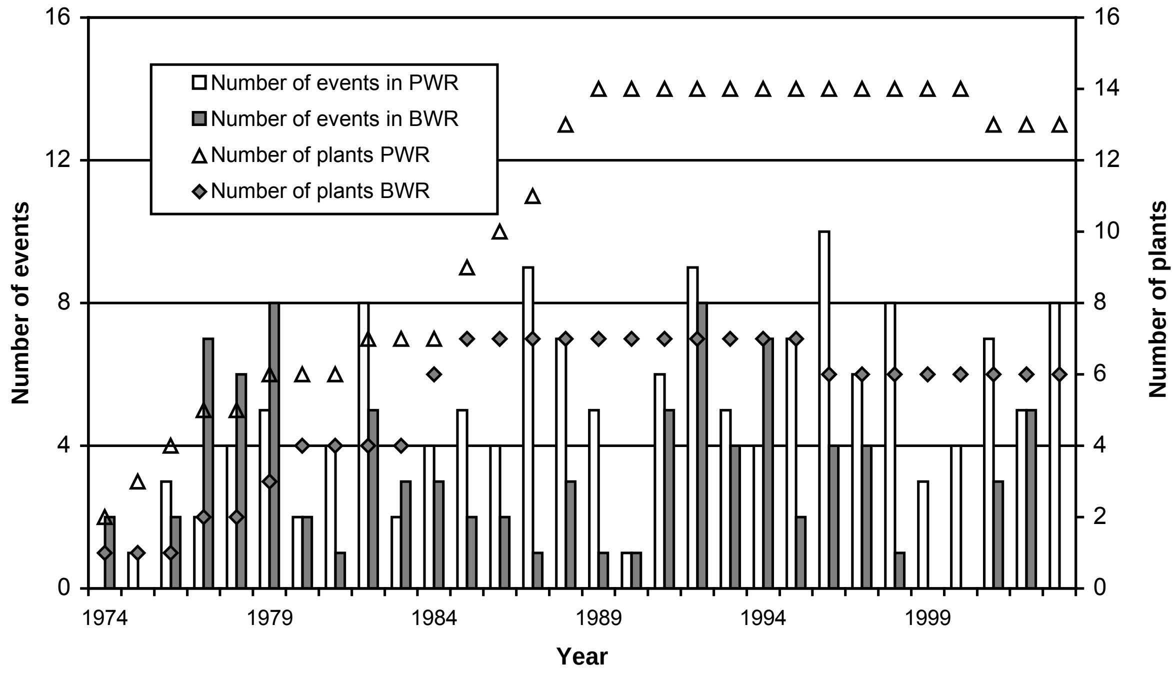


Figure 18-1 **Reported Pipe Damage Events in Nuclear Heat Generation Systems and in Nuclear Auxiliary Systems**

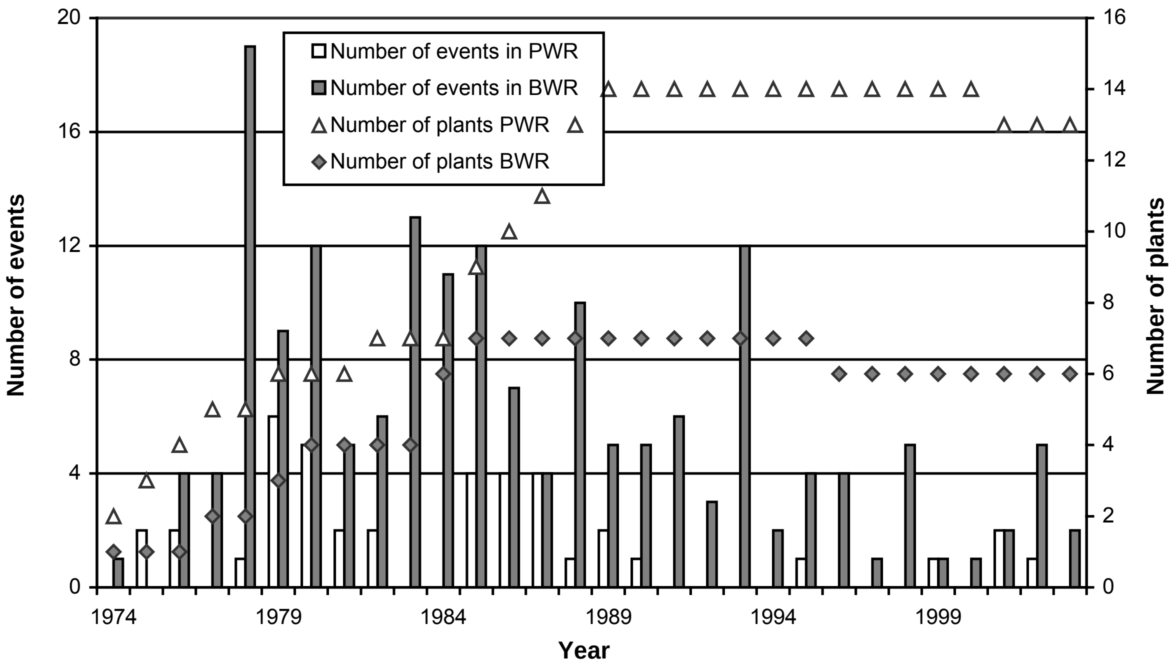


Figure 18-2 **Reported Pipe Damage Events in main-steam and feed-water systems**

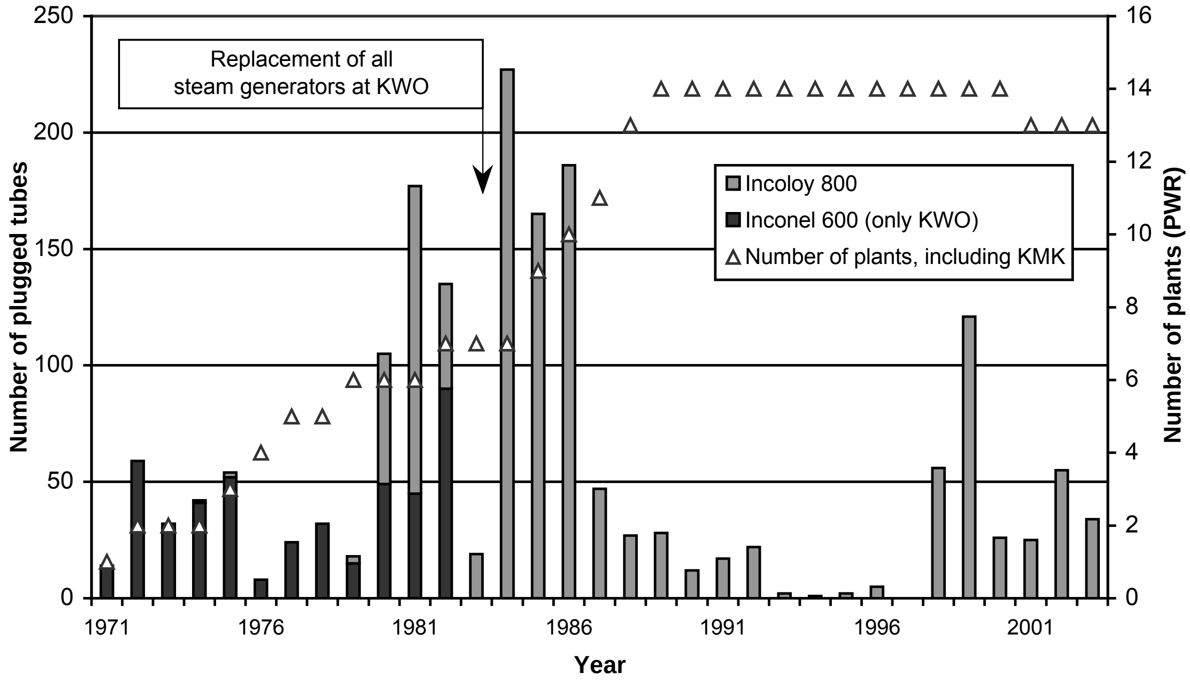


Figure 18-3 Number of Steam Generator Tubes Annually Plugged in PWR

the third design generation, which led to an optimised steel type characterised by a lower strength but a higher fracture toughness and better workability (15 MnNi 6 3).

According to operating experience, the results achieved with respect to a reliable operation can be characterised as follows:

– The number of defects of the fuel element cladding that have led to leakages, averages to 1 to 2 per year and plant, for the fuel assembly burn-ups reached at present (approx. 60 MWd/kg for PWRs and 55 MWd/kg for BWRs) .
– The frequency of occurrences of minute leakages of the pressure boundary averages to 10^{-1} per year and plant. In the last twenty years, not a single leakage occurred at this barrier in German nuclear power plants that would have led to an actuation of the safety system. In all cases where general indications for cracks due to reduced quality and to operating influences were found, the respective components were repaired or exchanged. The countermeasures applied were effective by now, as the continuously low number of damage reports on pipe failures in the nuclear auxiliary systems, the reactor coolant system and the water-steam cycle show (Figures 18-1 and 18-2). These reports comprise both the through-wall cracks with leakages as well as the detected incipient cracks of the tube wall without leakages.
– The fulfilment of the requirements regarding leak-tightness of the containment has been demonstrated in corresponding tests. Functional restrictions occurred only in isolated cases, e.g. in the case of a rupture of an instrumentation line without stop valve.
– The frequency of leakages between the reactor coolant pressure boundary and the connected systems is very low. In the case of pressurised water reactors, the measures finalised in 1987 on optimisation of the water chemistry with regard to the materials being employed in the steam generator tubes that are insensitive to stress-corrosion cracking have been effective (Figure 18-3). Ever since, the number of steam generator tubes that have to be plugged due to a wall thickness reduction has been reduced to a few tubes per year, all pressurised water reactors in operation taken together. The increase of damages in the years 1998 and 1999 can be ascribed to an unsuitable procedure chosen

for cleaning of the SG tube support plates or damages resulting from fretting, caused by loose parts, were detected. After detection of these damages, the scope of inspection was extended considerably. The additional tests revealed indications, not detected before, caused by fretting with the support structures and further indications. As a consequence of these test results, tubes with indications were plugged even with wall thickness reductions far below the permissible limit value of 40%. However, the damages described only occurred at some and not at all PWRs.

– The functional tests performed according to the respective test programme (→ Chapter 14 (ii) and 19 (iii)) show that the safety system is reliable under test conditions.

The continuous feedback of experience (→ Chapter 19 (vi) and (vii)) ensures that current data are available for all systems important to safety regarding the quality of manufacturing and the reliability of operation. This ensures an early detection of any deviation from expected behaviour in these systems.

Chapter 18: Progress and changes in 2001

Completion of the implementation of the preventive and mitigative accident management measures at the nuclear power plants.

19 Operation

19 (i) Technical Basis for the Initial Permit to Operate

The following description in this Chapter 19 (i) explains the proceeding according to the rules and regulations, as it has been performed in the past. The construction of new nuclear power plants in Germany is not provided.

Construction, commissioning and commercial initial trial operation of the nuclear power plants are usually performed by a general contractor who will be the licensee together with the future plant operator. After a successful initial operation, the turnkey plant is turned over to the plant operator by the general contractor. The responsibility for the safety of the plant stays with the general contractor until his official handover to the plant operator. The personnel required for commissioning is supplied by the manufacturer. It has to demonstrate the required qualification according to [3-2]. The personnel of the future plant operator participates in the commissioning activities and successively takes over the surveillance of those parts of the plant that are completed and ready for operation.

The granting of permits for the initial operation of the existing nuclear power plant is based, firstly, on the results of a safety analysis and its detailed evaluation by an authorised expert organisation called in by the competent authority (→ Chapter 14 (i)), secondly, on accompanying inspections during erection and, thirdly, on the results of a comprehensive commissioning programme, subject to approval by the competent authority. Special emphasis is put on verifying that all applicable safety requirements specified in the nuclear safety regulations are fulfilled at the time the permit for initial operation is granted. It is generally ascertained that the nuclear installation in its as-built condition meets all applicable design and safety requirements at this time.

The tests and inspections performed by the manufacturer on the construction site as well as the commissioning tests are monitored by independent expert organisations (e.g. the Technical Inspection Agencies) called in by the competent authority. These experts also perform supplementary tests.

Safety analysis

The safety analysis covers a systems analysis and an accident analysis. Originally, this safety analysis was merely performed deterministically and has later been supplemented by probabilistic assessments to an increasing extent. Scope and performance of the safety analysis are described in Chapter 18 (i). The safety analysis is submitted to the competent authority and is subject to a detailed review by the authorised experts. In his review the authorised expert, to a considerable extent, uses independently developed computer codes or generally accepted alternate analytical methods.

Accompanying inspection during erection

The accompanying inspection during the entire manufacturing process ensures that the actual design of the systems and components important to safety meets the requisite requirements. The accompanying inspection is subdivided into the design review, materials testing, construction and assembly tests, pressure tests, and acceptance and functional tests. The test results are recorded and documented in reports, attestations and certificates. The design review is an evaluation performed on the basis of plans and technical drawings. It concentrates on the design, dimensioning, materials used, the manufacturing and assembling procedures, the ease of inspection, accessibility for maintenance and repair, and on instrumentation and control. The material, construction and pressure tests are carried out to ascertain that the actual realisation is in conformance with the approval documents. The acceptance and functional tests ensure that the components and systems have been properly assembled and are in proper functioning order. For special components they are performed on test stands, otherwise during commissioning.

Commissioning programme

The tests and inspections carried out within the commissioning programme certify that the individual components and systems and the plant as a whole are as planned and designed and are in safe, functioning order. In general, the commissioning is carried out in four steps:

– commissioning of the systems,
– hot functional run, phase 1,
– hot functional run, phase 2, and
– zero-load and power tests.

In the pre-operational tests (commissioning of the systems), all necessary functional and operational tests are performed to ensure that the individual components and systems are in proper functioning order. In the hot functional run, Phase 1, the reactor coolant system is operated for the first time together with the reactor auxiliary and other systems to ensure proper functioning of the plant as a whole, as far as this is possible without fuel loading and nuclear steam generation. Hot functional run, Phase 2 is performed after initial fuel loading of the reactor. It covers those commissioning activities which are not feasible or not sensible to perform before the core is loaded. Its objective is to verify the functionality and the safety of the plant as a whole before starting nuclear operation. The final step of commissioning begins after first criticality and covers comprehensive tests at zero- and partial-load levels. The levels are chosen to be most suitable for the technical or physical verification of satisfactory functioning.

The whole commissioning process is reviewed by an authorised expert organisation called in by the supervisory authority. The authorised expert examines the commissioning programme and participates in tests and examinations chosen by him. The approval of the different load levels is given by the supervisory authority in the final step of commissioning (zero-load and power tests).

19 (ii) Operational Limits and Conditions for Safe Operation

All data, limits and measures important with respect to safe operation and to the controlling of design basis accidents are collated as the so-called safety specifications in accordance with the Nuclear Licensing Procedure Ordinance [1A-10] and with the Regulatory Guideline on the Requirements for Safety Specifications of Nuclear Power Plants [3-4] (→ Chapter 14 (i)). They give a quick and comprehensive survey of all limits, conditions, requirements and measures that determine the safety of the nuclear installation. The safety specifications are a constituent part of the operating manual. Part 2 of the operating manual contains the following chapters as safety specifications:

– prerequisites and conditions for power operation, for start-up and plant shutdown and for refuelling,
– safety system settings,

- specified actions with respect to abnormal operation (e.g., load rejection to auxiliary station supply, turbine trip, failure of a main coolant pump), and
- reporting procedure and criteria for reportable events.

The limits and conditions for safe operation prescribed by the licensing authority must be met at all times. Any modifications of the prescribed limits and conditions require approval by the licensing or supervisory authority.

The specified values in the safety specifications are regularly checked by the licensee and by the supervisory authorities and their authorised experts to see whether any modifications are required in the light of plant-specific or national and international operating experience as well as of recent findings from safety research. For example, as a direct result of the probabilistic safety assessment of German PWRs, the two limit values for "steam generator water level - high" and "reactor pressure vessel water level - low“ were added to the safety specifications in order to optimise the procedures in the event of leakage in the pressuriser or in the steam generator tubes.

The prerequisites and conditions for operation are derived from the provisions specified in the licence permits, from the boundary conditions specified in the licensing documents, from the technical standards and guidelines, and from the general responsibility of the licensee for safe operation of his nuclear installation. They comprise:

- the prerequisites and conditions for plant operation, e.g. definition of the permitted plant conditions, reference to the regulations and on-site provisions for reports to the authorities, documentation and the retention periods of documents, procedures for technical modifications to the plant and for changes to operating procedures, as well as conditions regarding the discharge of radioactive material with exhaust air or waste water;
- prerequisites and conditions for start-up, power operation, plant shutdown and refuelling;
- conditions for maintenance during power operation including the specification of permissible maximum unavailability times of safety equipment; provisions for preventive maintenance.

The limits for safe operation comprise all protection and hazard limit values, including the limit values regarding reactor protection and alarm indications, which

- necessitate power reductions for safety reasons, or
- serve the protection of the operating staff, or
- indicate an impermissible environmental impact.

In addition to these limits important to safety, this chapter of the operating manual covers further values important to safety and a compilation of important alarms:

- measured values on the effectiveness of the reactor scram system, of the residual heat removal, of the overpressure protections, and of the activity confinement,
- measures values on the conditions inside the containment, inside the reactor building and the turbine building,
- measured values of emission monitoring,
- alarms important to safety and short descriptions of the actions to be taken, e.g. in the event of switch-over to residual heat removal mode, failure of the operational feed-water supply, or steam generator tube leakage (this latter alarm requires short-term manual actions by the operating staff),
- alarms important to safety of the conventional alarm system including the respective limits,

– compilation of the accident monitoring instrumentation at the control room and the emergency control room in tabular form.

The scope of the specifications concerning the limits and conditions for safe operation and their compilation in marked sections of the operating manual are currently being revised on behalf of the BMU with the aim to realise a standard national regulation (→ Chapter 12).

19 (iii) Compliance with Approved Procedures during Operation, Maintenance, Inspection and Testing

Compliance with the approved procedures during operation, but also for the control of abnormal occurrences and accidents described in Chapter 19 (iv), is essentially ensured by the organisational structure in the nuclear power plant. This structure is laid down in detail in the operating manual. For the organisation, the following principles are of importance, among others:

– The plant manager or his deputy is responsible for safe operation. In the event of their absence, this responsibility is transferred to the shift supervisor on duty. In addition, stand-by services are available.
– Instructions to the shift supervisor significant to the safety of the plant, may only be given by the plant manager or the immediate superior of the shift supervisor. However, these will only intervene with immediate operating procedures in well-founded exceptional cases.
– The tasks of the managing personnel are clearly and completely specified, so that concurrent instructions of more than one person are avoided.
– To avoid any conflict of interests, the organisational units for quality assurance and for radiation protection are independent of the division responsible for operation.

The organisational procedures required for a safe and licence-conform operation of the plant are laid down in the operating manual and the testing manual.

Operating manual

Structure and contents of the operating manual of a nuclear power plant are laid down in KTA safety standard [KTA 1201]. The operating manual covers the plant regulations valid throughout the plant, as well as instructions for operating and accident conditions, such as detailed instructions for the shift personnel with additional information regarding the particular plant conditions involved. All parts of the operating manual that belong to the safety specifications are marked accordingly. The operating manual consists of the following parts:

– Plant regulations
  These comprise the personnel organisation (tasks, responsibilities, subordination, etc.), the control room and shift regulation, maintenance regulation, radiation protection regulation, guard and access regulation, alarm regulation, fire protection regulation and first aid regulation. All plant regulations are part of the safety specifications.
– Plant operation
  This part contains the prerequisites and conditions for operation and the safety system settings (→ Chapter 19 (ii)), the criteria for the reporting of events to the supervisory authority and detailed instructions for normal and abnormal operation of the plant.
– Design basis accidents
  This part of the operating manual includes the design basis accidents with and without

loss of coolant and originating from external impacts. The procedures regarding design basis accidents are treated in Chapter 19 (iv).

– Systems Operation
  This part covers the initial conditions for the different operating modes for all systems and the actions to be taken by the shift personnel as step programmes. In addition, it contains supplemental information, technical drawings and remarks.
– Alarms
  This is a complete listing by systems of all alarm signals from failures or dangerous conditions together with corresponding instructions on counteractions and possible alternatives.

Alarm plans and organisational structures for the control of possible emergencies are also specified in the operating manual.

Emergency management manual

For beyond-design basis accidents, the technical measures to be taken at the plant, the accident management measures and auxiliary means required are contained in a separate document, the emergency management manual.

Testing manual

Structure and contents of the testing manual are laid down in KTA safety standard [KTA 1202]. The testing manual comprises general instructions, the testing schedule and corresponding testing instructions for all in-service inspections.

The general instructions deal with the application and handling of the testing manual and the corresponding preconditions, e.g. the administrative procedures regarding test performance and result evaluation, permissible deviation from test intervals, participation of authorised experts in the test performance and in the case of modifications of the testing manual.

The testing schedule contains a list of all in-service inspections important to safety. It covers the test object, extent of test, test interval, required plant conditions under which the test is performed and a clear notation of the testing instruction. The testing schedule is part of the safety specifications.

The testing instructions identify the test object and the reason for performing the test (e.g. licensing provision), the testing method, the target and the extent of the test. It also lists the supporting measures and documents, and describes the prerequisites, the performance (in case of functional tests e.g. switching sequence programme) and documentation of the test as well as the procedure for establishing a defined final condition after the test.

Specification of procedures and intervals for tests, inspections and maintenance

The procedure employed by the licensee to verify that the essential functions for the safety of the plant are ensured and that the corresponding quality characteristics remain within the permissible limits is described in Chapter 14 (ii). The essential system functions, systems and components important to safety have to be stated initially during construction and recurrently with every plant modification. In addition, the required qualification proofs, the recurrent tests and inspections, the preventive maintenance and repair measures as well as the permissible operating procedures for systems and components have to be specified according to their safety relevance. The basis for it is the regulatory guideline on maintenance [3-41]. The following measures are performed during operation on the basis of these specifications:

- in-service inspections in accordance with the testing manual; the tests should be as comprehensive as possible for the respective requirement. If this should not be possible, the principle of overlapping partial tests will be applied,
- preventive maintenance regularly scheduled and performed under the independent responsibility of the licensee,
- functional tests of systems and components after maintenance and repair,
- periodic evaluation of the documentation from operation and testing,
- feedback of operating experience to operational practice.

Since the erection of the nuclear power plants, the test and maintenance concepts have been developed against the background of operating experience and of findings from safety research. Deterministic verifications were dominant at the time of the construction of the plants (1964 to 1989). Thus, the classification of systems important to safety, components and other plant equipment as well as the specification of the scope and intervals of the tests were essentially based on straightforward engineering judgement. Technical drawings and documents were evaluated with respect to identifying those components required for the safety functions of the nuclear power plant. The concept of in-service inspections was, then, developed on the basis of operating experience, of knowledge regarding component reliability and of recommendations by the component manufacturers. During implementation of this in-service inspection concept, a number of shortcomings caused by inaccessibility, technical restrictions, or an insufficient validity of the tests regarding activation of the a component in case of demand were revealed, which have been overcome to a large extent by appropriate modifications of the components, of the testing techniques, or of the testing procedures. With respect to the component reliability achieved, also see Chapter 18 (iii).

In recent years, probabilistic safety assessments are increasingly used to supplement the engineering judgements. In the past, in individual cases, the results from the deterministic approach were already checked by probabilistic means (e.g. with regard to the reactor protection system and the emergency core cooling systems).

19 (iv) Procedures for Responding to Abnormal Occurrences, Accidents and Emergencies

Although abnormal occurrences during operating conditions will cause operational restrictions (e.g. reduction of reactor power in case of a failure of one main coolant pump) there will be no safety reasons to discontinue operation. In the case of accidents, on the other hand, plant operation must be discontinued for safety reasons. Detailed procedural instructions are specified for the shift personnel covering the individual operating modes for each of the abnormal occurrences or design basis accidents dealt with in the licensing procedure. These are contained in Part 2 and 3 of the operating manual.

Design basis accidents

The procedures for control of design basis accidents are a combination of procedures either oriented on the (known) events or on the fundamental safety functions. The procedures for control of design basis accidents are based on the following types of written instructions and visual aids:

- accident sequence diagram,
- check of the fundamental safety functions criteria,
- accident decision tree,
- fundamental-safety-functions-oriented handling of accidents,
- event-oriented handling of accidents.

In case of an event leading to a reactor scram, the proceeding of the shift personnel follows the accident sequence diagram (Figure 19-1). In a first step, the shift personnel should control the fundamental safety functions criteria to determine whether or not

- control of reactivity (subcriticality),
- cooling of fuel elements (coolant inventory, heat transport and heat sink),
- confinement of radioactive material (in particular, integrity of the containment)

have been achieved, and thus the release of activity into the environment does not exceed the accident planning values. In case, a violation of one of the fundamental safety functions criteria is detected, then the respective procedures, oriented on the fundamental safety functions, are used to bring the plant parameters back into their normal range. If no violation of fundamental safety functions criteria is detected and the event may be assigned to a known type of accident, the further proceeding will be based on event-oriented procedures. If beyond-design basis plant conditions are detected, the shift personnel will also consult the decision trees for severe accidents and will employ the accident management measures. The transition from design basis accident procedures to accident management measures is described in the section "Fundamental-Safety-Functions-Oriented Procedures“ of the operating manual.

Irrespective of the procedure chosen to control a design basis accident, the fundamental safety functions criteria have to be reviewed cyclically, and the procedure has to be adapted if necessary.

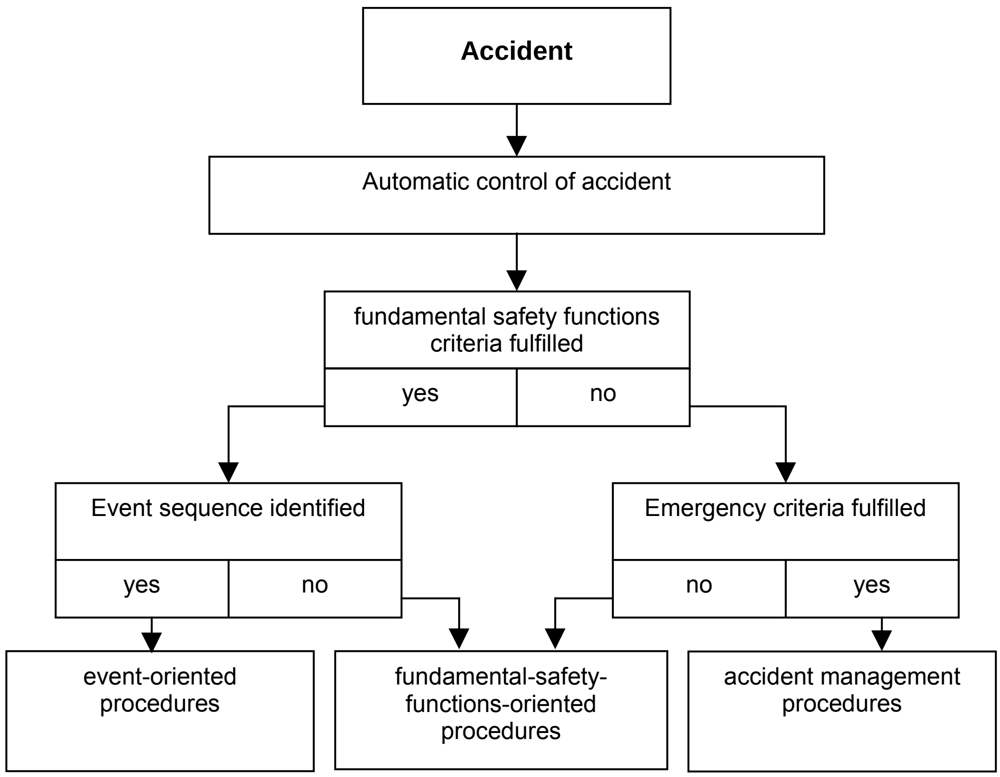


Figure 19-1 Accident Sequence Diagram

Fundamental-safety-functions-oriented procedures in case of design basis accidents

The fundamental-safety-functions-oriented procedures do not require the identification of the actual event but are rather guided by the observable plant conditions (symptoms). The operating manual lists the corresponding plant parameters for every fundamental safety function which have to be checked. Each description of the fundamental-safety-function-oriented procedures is structured as follows:

- definition,
- list of the important plant parameters,
- list of the important operating and limiting values,
- conditions under which the available measures are effective,
- description of the measures for ensuring that the protection criteria are met, and
- general remarks and pertinent diagrams.

If the fundamental safety function criteria cannot be met, the accident management measures, treated in the accident management manual (→ Chapter 18 (1)), have to be applied according to additionally specified criteria.

Event-oriented procedures in case of design basis accidents

Event-oriented procedures are applied if none of the fundamental safety functions is endangered and if the event can clearly be assigned to an accident type (e.g. loss-of-coolant accident, failure of heat removal without loss of coolant, external impact). By means of detailed step programmes, the plant is brought into a long-term safe condition. In parallel, it is checked regularly whether the fundamental safety functions criteria are met still. Detecting that one of the criteria failed, the event-oriented procedures will immediately be interrupted to return to the fundamental-safety-functions-oriented procedures in order to bring the respective plant parameters back into normal range.

Emergencies

The administrative measures that have been taken in all nuclear power plants to cope with emergencies include an emergency organisation with an emergency response team which will be supported by personnel from the operating staff. It should be possible for the emergency response team to be assembled and ready to work within one hour. Appropriate working rooms as well as working and communication equipment are kept in readiness at all times. Pre-arranged contracts assure the assistance of external institutions such as the plant manufacturer and the Kerntechnischer Hilfsdienst (Nuclear Emergency Service Company), a permanent organisation jointly installed by the licensees of German nuclear power plants to help in coping with emergencies and the removal of possible consequences. Corresponding alarm plans and organisational structures are specified in the operating manual, and individual technical measures and accident management measures in the accident management manual.

19 (v) Engineering and Technical Support

The qualification of the personnel for the engineering and technical support is based on the proven educational system for the technical professions in Germany and on the experience accumulated in more than four decades of industrial utilisation of nuclear energy. Depending on the individual activities, experience is concentrated in those institutions concerned with the design and construction, the safety assessment and licensing or the operation of nuclear power plants, and furthermore, in the nuclear training and research institutes.

In accordance with the Atomic Energy Act [1A-3], the licensee is required, among other things, to present proof that he has sufficient personnel with the required qualification at his disposal. This proof is furnished by special proofs of the technical qualification of the personnel, which also cover simulator training (→ Chapter 11 (2)).

In accordance with the organisational structure, as implemented at most of the German nuclear power plants, the production division which is directly responsible for plant operation is supported in its activities by the organisational units, e.g. for technology, maintenance and surveillance. These organisational units, whose integration into the organisational structure may differ from plant to plant, have well-defined tasks and keep the necessary technical expertise at their disposal for their fulfilment:

– Technology
  Maintenance and optimisation of the functionality and operational safety of the mechanical and electrical components and systems (specialised engineering knowledge of employed components and systems).
– Maintenance
  Planning, control, performance and surveillance of maintenance tasks and of technical modifications and backfitting.
– Surveillance
  Working out solutions for all technical problems that concern the nuclear installation or its operation, in physics, chemistry, radiation protection, environmental protection, fire protection and physical protection.

The nuclear safety commissioner takes part in all activities concerned with technical modifications as well as in the evaluation of operating experience and of the reportable events (→ Chapter 9).

For general service tasks, separate planning and engineering divisions or management positions are provided either at the nuclear installation or at the company's headquarters, depending on the individual licensee. With regard to questions beyond plant operation, the licensee can always receive support from the manufacturer of the nuclear installation or its components as well as support from scientific establishments, such as universities and other research institutions.

The extent of external service by contractors differs, depending on the company policy of the individual licensees. However, external personnel is used extensively during major plant revisions (refuelling outage), major modifications or maintenance and also to establish and maintain plant documentation. Outside contractors are usually engaged for the calculation of refuelling, revision of emergency diesel generators, pumps and valves, the non-destructive testing of materials, the regular inspection of steam generators, and for the compilation and update of the operating, testing and quality assurance manuals. In all cases, plant personnel supervises the activities of the external personnel and performs the activities regarding quality assurance. It carries responsibility for the technical supervision of the activities, especially with regard to the ensurance of plant safety. These tasks determine the minimum size of the plant personnel. To ensure the quality of work by outside contractors, all licensees basically use only those companies - aside from the manufacturers of the nuclear power plant - that have proven their trustworthiness and expertise over the years and have qualified personnel at their disposal. In order to avoid conflicting work schedules in the case of companies specialised in certain fields of maintenance (e.g. of primary coolant pumps or safety valves), the licensees co-ordinate their time schedules for the major maintenance activities and plant outages on a nation-wide scale.

19 (vi) Reporting of Events, Regulatory Reporting Procedure

An obligation to report accidents and other harmful occurrences to the competent supervisory authority had already been specified in the original version of the Atomic Energy Act in 1959 [1A-3]. In 1975, a central reporting system was established by the Federal States Committee for Nuclear Energy. Accordingly, the licensees of German nuclear power plants are obliged to report any reportable event to the supervisory authorities in accordance with nation-wide applicable reporting criteria. Then, in 1992, with the promulgation of the Ordinance on Reportable Events [1A-17], the obligation of the licensees of nuclear installations to report accidents, incidents or other events relevant to safety (reportable events) to the competent supervisory authority became legally formalised at the level of an ordinance. The nuclear installations concerned are nuclear power plants, research reactors with a thermal power larger than 50 kW and all facilities of the fuel cycle.

The regulatory reporting procedure is embedded in the regulatory supervision of nuclear installations. On this basis, the supervisory authority is in the position to detect possible deficiencies at an early stage. The event reports and the results of their evaluation are distributed in a nation-wide information system. This supports the taking of preventive measures against a recurrence of events from similar causes in other nuclear installations.

After an initial engineering evaluation, each reportable event is assigned to one of the individual reporting categories. These categories particularly take into account the aspect that the authority has to be able to take precautionary measures irrespective of the actual significance of the event.

Category S (immediate report - reporting deadline: without delay)
Category S events are those events where the supervisory authority must be quickly informed in order to allow the authority to be able to initiate immediate investigations or other measures. Any event that points to an acute safety deficiency would also be placed in this category.

Category E (quick report - reporting deadline: within 24 hours)
Although events in Category E do not call for an immediate action by the supervisory authority, safety reasons require that their cause is identified and that remedial action be taken within an appropriately short time period. These are, in general, events that may have a potential - but no direct - significance to safety.

Category N (normal report - reporting deadline: within 5 days)
Category N is for events with a low significance to safety. They are only slightly different from routine operational events while plant conditions and operation remain in full accord with the operating instructions. These events are, nevertheless, systematically evaluated with the purpose of detecting possible weak points at an early stage.

Category V (before initial core loading - reporting deadline: within 10 days)
This category V is used for events occurring during erection and commissioning of the nuclear power plant of which the supervisory authority should be informed with regard to the later safe operation of the plant.

Special reporting forms were developed for recording and categorising reportable events in accordance with approximately 80 reporting criteria. These reporting criteria are contained in the respective ordinance and are subdivided into radiation criteria which are the same for all nuclear installations and individual criteria applicable to nuclear power plants and to the installations of the nuclear fuel cycle.

Any event that is categorised as reportable in accordance with the corresponding reporting criteria is reported by the licensee to the competent supervisory authority. The licensee has the responsibility that the report is presented within the period stipulated and that it contains the correct and complete information on the reportable event. The supervisory authority, in turn, after its initial evaluation of the circumstances will inform the BMU which is responsible for federal supervision. At the same time, the Federal Office for Radiation Protection (BfS) and the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), an expert organisation working under contract of the BMU, are informed. In those cases where the information required in the reporting form is not completely available within the reporting deadline, the report will be marked as provisional. The supervisory authority receives a completed report (final report) as soon as the missing data is available.

The information required in the written report on the event is indicated by the outline of the corresponding reporting form. It includes the immediately available information on the radiation situation, a summary of the safety relevance of the event and additional details necessary for the evaluating organisations. The uniform arrangement of data in the reporting form simplifies both the comparison of different reports and the transfer of their contents to corresponding databases. The reporting form has four distinct parts:

– general information on the nuclear installation and on the event,
– information on the radiological impacts,
– a part with a detailed and properly arranged description, and
– identifying codification of the event and the affected components.

In addition to the regulatory reporting procedure in accordance with the Reporting Ordinance, the licensee also categorises the reportable events according to the seven levels of the INES scale (→ Chapter 19 (vii)). This scale is used to inform the general public of the significance of the particular event with special regard to the safety of the plant and to whether or not it had or could have had any radiological impact on the public or the environment.

19 (vii) Collecting, Analysing and Exchanging Operating Experience

From a very early stage in utilising nuclear energy in Germany, a system was established for the collection and sharing of operating experience from nuclear installations. This system has been improved over more than 25 years. The resulting feedback of operating experience has been a major contributing factor to the further development of safety in nuclear installations.

The operating experience is evaluated by the industry and the authorities at several levels, i.e. by the licensee of the nuclear installation concerned and by the operators of other installations, by the *Länder* authorities and their expert organisations at a national level, and at a federal level by BfS and GRS (by order of the BMU). At a federal level, an initial evaluation of the reportable events is carried out by BfS. These multiple-level and independent analyses ensure that each event is evaluated in detail and that the required remedies are taken.

Evaluation of operating experience by the utilities

The most important source for the feedback of experience are the deficiencies and abnormal occurrences in the nuclear power plants. Some of these will be classified as reportable events, however, the majority will stay below the reporting thresholds. Following the Safety Criteria [3-1], the licensee has to record and evaluate events below the reporting threshold and has to take the appropriate actions where necessary. This requirement is laid down in the individual operating manuals. All deficiencies and abnormal occurrences are recorded and documented, today mainly with the computer-based operational management system. In daily meetings, the deficiencies and abnormal occurrences are discussed and evaluated and the required measures are specified. The results of in-service inspections and maintenance as well as important measured values which can indicate deviations of process parameters are documented. This allows a life history to be created for every component. These data form the basis for a selected evaluation of individual components as well as for generic issues, for trend analyses or the determination of reliability parameter for plant-specific probabilistic safety assessments.

The operating experience is also systematically analysed by the licensee with regard to human errors and to possible improvements which may be derived from them (→ Chapter 12 (i)).

The utilisation of the feedback from plant-specific experience by other nuclear installations is essentially based on the reportable events. Parallel to submitting the report to the competent authority, the licensee also informs the Association of Large Power Plant Operators (VGB). VGB collects these reports and distributes them among its members independently of the reporting path via the authorities. The manufacturers participate in the information exchange via VGB and via the authorities.

The licensees are obliged to additionally evaluate the reportable events from other nuclear installations with respect to possible conclusions for their own installation.

In addition to this experience feedback through the system for handling reportable events, the licensees have installed a number of working groups which regularly meet for detailed discussion of operating experience. In addition to the experience from abnormal occurrences and deficiencies, modification and backfitting measures are also discussed. Furthermore, the utilities conduct joint investigation and research programmes on issues important to safety and on optimising the operation of nuclear power plants (→ Chapter 11 (1)).

In addition to the reporting system for events, there are further information systems. For instance, some licensees are connected to the respective manufacturer system on experience feedback. Also, several licensees of foreign nuclear power plants are members of the VGB and thereby participate in the exchange of experience.

The licensees also participate in the reporting system operated by WANO and perform a trend analysis with indicators of the WANO reporting system.

The licensees report to the supervisory authorities on the conclusions drawn from the evaluation of experience (relevance of events) and on the modification and backfitting measures performed in their monthly, maintenance outage and annual reports. Further, the licensees prepare annual reports to inform the Reactor Safety Commission. Moreover, the submission of additional reports of the licensees on ageing management at the individual plants is planned (→ Chapter 13).

Evaluation of operating experience by the authorities

The competent *Länder* authority and its expert organisation analyse a reportable event primarily with regard to the conclusions and the remedies to be taken for the affected installation. In a second step, however, the *Länder* authority and its expert organisation also investigate the significance of the event to other nuclear installations in their area of supervision.

Table 19-1 Number of Reportable Events in Nuclear Power Plants According to the Different Reporting Categories

Year	Number	Reporting category				INES-category		
		S	E	N	V	0	1	≥ 2
1994	161	1	1	159	0	158	3	0
1995	152	0	2	150	0	151	1	0
1996	137	0	2	135	0	131	6	0
1997	117	0	3	114	0	114	3	0
1998	136	0	4	132	0	132	3	1
1999	121	0	1	120	0	120	1	0
2000	94	0	2	92	0	91	3	0
2001	126	2	7	117	0	119	5	2
2002	167	0	10	157	0	154	13	0
2003	137	0	0	137	0	134	3	0

On behalf of the BMU, the BfS performs the central collection and documentation of information on all reportable events. The BfS performs an initial evaluation of the reported events and informs the *Länder* nuclear authorities, the expert organisations, the manufacturers and the licensees of nuclear power plants as well as the general public in quarterly reports which contain all reportable events in nuclear power plants and research reactors. Table 19-1 lists the reportable events that occurred over the last ten years also indicating both the German and the INES (see below) reporting categories.

Figures 19-2 and 19-3 show these events according to their kind of occurrence - spontaneously or detection during inspections and maintenance - and according to the operating condition at the time of detection of the event and the impact on operation. All events are included in these presentations, even those reported or re-classified at a later date. Figure 19-4 shows the development over the last ten years of the average number of reactor scrams, also indicating their essential causes.

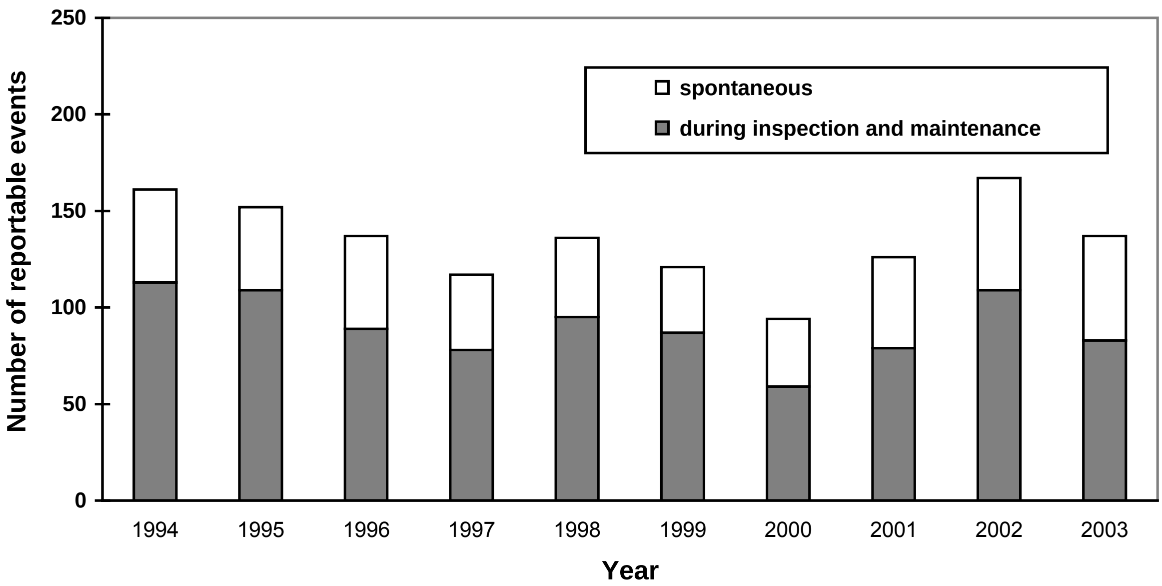


Figure 19-2 Number of Reportable Events from Nuclear Power Plants According to the Kind of Occurrence

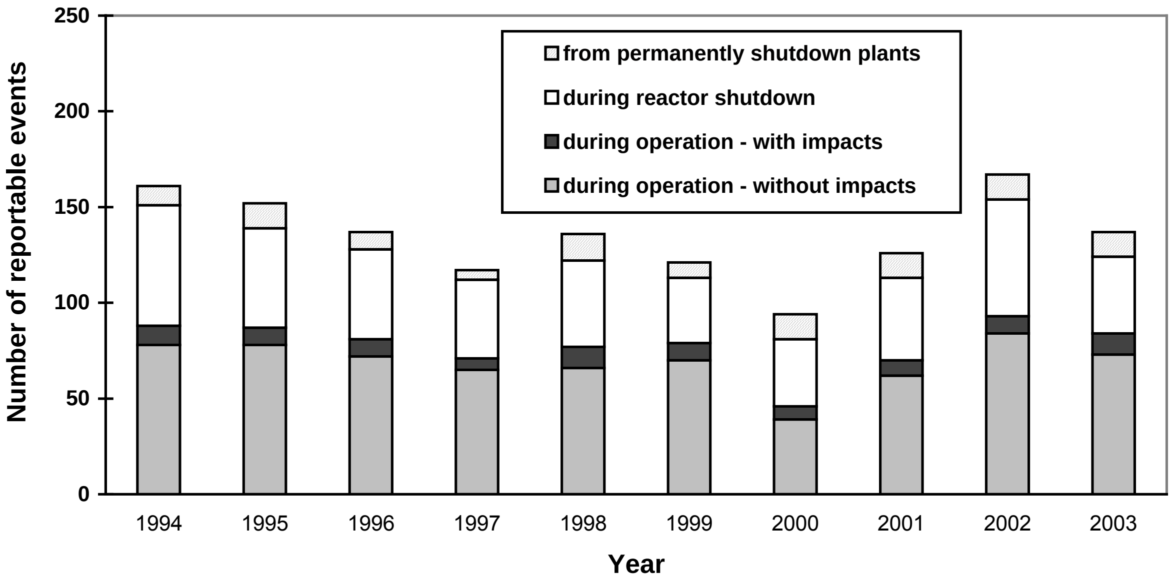


Figure 19-3 **Number of Reportable Events from Nuclear Power Plants According to Mode of and Consequence on Operation (power operation, start-up and shutdown operation)**

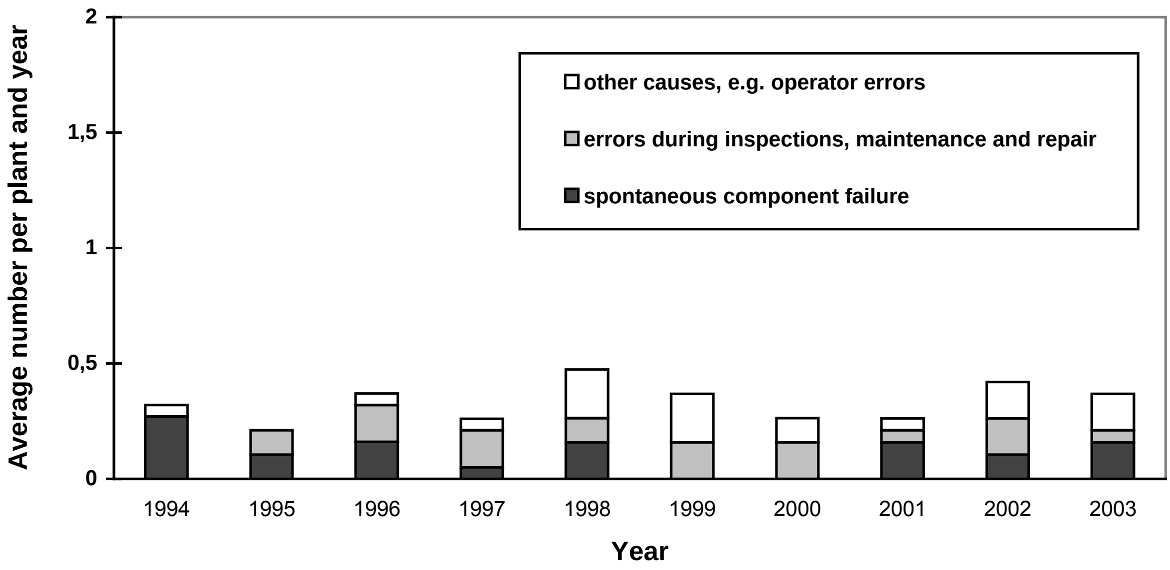


Figure 19-4 **Average Number of Unplanned Reactor Scrams per Plant and Year**

In addition to the German experience, another important source for operating experience is found at the international level. For this reason, internationally available operating experience is also utilised intensively in Germany. An important source for safety-related findings from international operating experience is the IRS of IAEA/NEA. The Federal Republic of Germany actively participates in this reporting system. The events reported within this system are systematically evaluated by GRS by order of the BMU. In its quarterly reports, GRS presents short descriptions for every IRS event and a comment regarding applicability and relevance to German nuclear power plants. These quarterly reports - together with the corresponding reports by IRS - are sent to the supervisory authorities and expert organisations as well as to the licensees and other competent institutions. In addition, GRS prepares annual reports containing detailed descriptions and evaluations of the most important events. These annual reports are distributed in the same way as the quarterly reports. The licensees evaluate these reports with regard to the applicability to their own plants.

GRS prepares information notices for all those events in German and foreign nuclear power plants where the in-depth analyses show a significance and applicability to the safety of other plants. These information notices are distributed by order of the BMU to the supervisory authorities and expert organisations as well as to the licensees and other competent institutions. These information notices cover a description of the circumstances of the event, the results of the root cause analysis, an evaluation regarding safety relevance, a description of the measures taken or planned, as well as recommendations regarding investigations and, possibly, remedial measures to be taken in other plants. In accordance with corresponding licensing provisions, the licensee submits a comment on each information notice to the competent supervisory authority with special emphasis on the implementation of the recommendations. These comments are evaluated by the competent expert organisation. GRS collects all comments on and evaluations of the information notices and prepares an annual assessment with particular regard to additional findings.

Moreover, GRS performs a generic assessment of German and international operating experience. Safety problems not to be assigned to a single event but to a group of events (event collective) and general safety issues arising from an event are subject to in-depth analysis. The results and conclusions from these generic assessments are presented in reports that are distributed in the same way as the information notices if they are also significant to other plants. The licensees again perform a plant-specific evaluation of these reports and possibly implement the issue.

The generic evaluations also include systematic precursor analyses which are performed by GRS for reportable events in German plants. The purpose is the identification of weak points by probabilistic methods and trend analyses of the safety status. Following international practice, GRS currently develops a method for the performance of trend analyses of parameters important to safety which can be derived from the reportable events.

Working groups similar to those of the licensees have also been installed by the authorities and expert organisations which meet regularly for the discussion of operating experience and of the conclusions drawn with respect to safety and to the general applicability of plant specific evaluations. Moreover, the reports of the licensees on plant operation and experience evaluation, and the information notices and evaluations of GRS on events in German and foreign countries are also discussed regularly by the Reactor Safety Commission.

19 (viii) Processing and Storage of Spent Fuel and Radioactive Waste

In accordance with Section 9a of the Atomic Energy Act [1A-3], anyone who produces residual radioactive material shall make provisions to ensure that they are utilised without detrimental effects or are disposed of as radioactive waste in an orderly manner.

Generation, processing, clearance and disposal of radioactive waste

Any activities concerning the management of radioactive waste are subject to regulatory supervision by the respective *Länder* authorities. The licensee submits a conceptual waste programme to the competent supervisory authority; it accounts for all waste accumulated in the restricted access area during operation of the nuclear power plant. By adequate operational management by the licensees and corresponding planning for major plant revisions (refuelling outages), the volume of radioactive waste is reduced substantially. Regarding treatment, conditioning and disposal of radioactive waste, the utilities are often supported by specialised outside contractors.

From the time of its generation, the accumulated radioactive material is sorted according to radioactivity and type. The intention is to recycle - with or without restrictions - as much of the material as possible after clearance measurement and decontamination, or, if the prescribed limits are not exceeded, to provide for their disposal as conventional waste.

The clearance levels for radioactive material with minor activity and the clearance procedure are specified in the amended Radiation Protection Ordinance [1A-8]. For about 300 radionuclides, the Radiation Protection Ordinance prescribes mass-specific clearance levels for solid and liquid material, for the clearance of buildings and land areas, as well as for the clearance for disposal at a domestic waste dump or an incineration plant on the basis of the 10 µSv-concept. Clearance is regulated by the supervisory authority. The measurements required for it are performed by the licensee and are subject to the supervision by the competent *Länder* authority which also performs control measurements.

Pre-treatment and treatment of radioactive waste minimises its volume and converts the primary waste to intermediate products that can be handled and properly conditioned for final disposal. During its generation, all radioactive waste is sorted and documented by type, content and radioactivity. The regulatory guideline on radioactive waste without heat generation [3-59] specifies the sorting criteria and the requirements regarding registration, determination of activity and documentation. By doing so, the waste producers will always be able to give information on the amount of activity and the storage place of the radioactive waste.

Packaging, pre-treatment and conditioning of the radioactive waste is carried out with qualified procedures and, as far as possible and practicable, on site. Treatment and conditioning is always performed with regard to the requirements of subsequent disposal. Pre-treatment and treatment equipment (e.g. to concentrate, sort, compact and package) is available in all nuclear power plants. Accordingly, non-combustible liquid waste is concentrated, and the non-combustible solids are compacted by high pressure. In many cases, conditioning in compliance with the requirements for repositories is performed by outside contractors that have mobile equipment available (e.g. in-drum drying facilities for liquid concentrates, remote underwater disassembling equipment for intermediate level wastes) and will transport this equipment to the nuclear power plant. The combustion of combustible waste and conditioning (cementing) of the resulting ashes is performed by outside contractors in off-site plants. The conditioned waste packages are returned to the nuclear power plants for storage at on-site facilities or transported to a central (external) interim storage facility.

The Federal Office for Radiation Protection (BfS) performs an annual survey on the accumulated radioactive waste in Germany, including the volume of radioactive waste produced at the nuclear power plants. The BfS generally differentiates between radioactive waste that produces heat and such whose heat generation is negligible.

Storage of spent fuel

In order to minimise the number of transports of spent fuel elements, the NPP operators have applied for the erection of local interim storage facilities for 12 sites (except Mülheim-Kärlich and Stade) in the years 1998 to 2000 (Table 19-2). These are planned as dry storage facilities for spent fuel elements in shipping and storage casks mainly of the Castor-type. The capacity of these storage facilities is designed to store all spent fuel elements accumulating until final cessation of nuclear power plant operation also after decommissioning of the respective plant until commissioning of a repository. The time of operation is limited to 40 years. The applications have been approved, and the Federal Government expects that these on-site interim storage facilities will be operable until 2006 at the latest. In the case of Emsland this has already happened. In order to avoid intermittent bottlenecks in storage capacities, the nuclear power plants Biblis, Brunsbüttel, Krümmel, Neckarwestheim and Philippsburg have applied for additional interim storage places with a capacity of 12 to 28 storage positions for casks. The licences were granted, some of these additional interim storage places are already in use. The licensing of all interim storage facilities falls within the competence of the Federal Office for Radiation Protection.

Table 19-2 Local Interim Storage Facilities for Spent Fuel Elements

Nuclear Power Plant (Site)	Type of Facility	Capacity HM [Mg]	Storage Positions for Casks	Date of License	Start of Construction
Biblis (KWB)	Interim storage	1400	135	22.09.2003	01.03.2004
	Additional storage places	300	28	20.12.2001	in operation
Brokdorf (KBR)	Interim storage	1000	100	28.11.2003	05.04.2004
Brunsbüttel (KKB)	Interim storage	450	80	28.11.2003	07.10.2003
	Additional storage places	140	18		
Grafenrheinfeld (KKG)	Interim storage	800	88	12.02.2003	22.09.2003
Grohnde (KWG)	Interim storage	1000	100	20.12.2002	10.11.2003
Gundremmingen (KRB B and C)	Interim storage	1850	192	19.12.2003	
Isar (KKI 1 and KKI 2)	Interim storage	1500	152	22.09.2003	14.06.2004
Krümmel (KKK)	Interim storage	775	80	19.12.2003	23.04.2004
	Additional storage places	120	12	20.06.2003	
Emsland (KKE)	Interim storage	1250	130	06.11.2002	in operation
Neckarwestheim (GKN 1 and GKN 2)	Interim storage	1600	151	22.09.2003	17.11.2003
	Additional storage places	250	24	10.04.2001	in operation
Philippsburg (KKP 1 and KKP 2)	Interim storage	1600	152	19.12.2003	17.05.2004
	Additional storage places	250	24	17.02.2003	in operation
Stade (KKS)	Interim storage	300	80	application withdrawn	
Unterweser (KKU)	Interim storage	800	80	22.09.2003	19.01.2004

Waste management

The legal basis of waste management is the Atomic Energy Act which was amended on 22 April 2002 in consideration of the agreement between the Federal Government and the power utilities of 11 June 2001. Accordingly, the waste management of nuclear power plants comprises

- interim storage of spent fuel elements at the local interim storage facilities at the plant sites, in exceptional cases at central (external) interim storage facilities, and in future direct disposal of the fuel elements,
- reprocessing of spent fuel elements until 30 June 2005 at the latest (transport date), utilisation of recovered nuclear fuel as well as the proper disposal of waste material,
- conditioning, interim storage and future disposal at a final repository of the radioactive waste from operation and decommissioning of the nuclear power plants.

The Morsleben repository (ERAM) for low-level and medium-level radioactive waste was in operation until September 1998. The Morsleben repository is not included in the future waste management concept. The repository will be backfilled and sealed. The plan approval procedure for the Schacht Konrad repository was finalised with plan approval decision on 22 May 2002. An action was filed against the plan approval decision. The Schacht Konrad repository will be kept open until the court issues a decision.

In accordance with the agreement initialled on 14 June 2000, the exploration of the Gorleben salt dome was interrupted on 1 October 2000 for the clarification of conceptual and safety-related issues for at least three, but no more than10 years. In December 1999, the BMU established the Committee on a Site Selection Procedure for Repository Sites (Auswahlverfahren Endlagerstandorte - AkEnd). In December 2002 the Committee submitted its recommendations for a new site selection procedure for repository sites. The commissioning of a repository is planned for the year 2030 approximately.

Until commissioning of a repository, the utilities can take precautionary measures for waste management by furnishing proof on sufficient interim storage capacities.

Chapter 19: Progress and changes since 2001

For all local interim storage facilities where applications were made, licences have been issued. In the meantime the erection started for several storage facilities.

Chapter 19: Future activities

The Nuclear Safety Officer and Reporting Ordinance [1A-17] and its reporting criteria shall be revised considering the experience from application of these criteria and current international developments.

Future Activities of the Federal Regulator

The Federal Regulator (BMU) sees a need for action both with regard to safety-related issues and regulatory issues in order to maintain and improve the safety level of the German nuclear power plants also during their remaining operating lives. In the following, the intended measures from the previous chapters are presented in summary form.

Ad Article 6
Continuation of the safety assessments as it is common practice within the framework of supervision and the comprehensive safety reviews to be performed on the statutory dates.

Ad Article 7
Within the framework of its revision of the rules and regulations, the BMU initiated a comparison of the national nuclear rules and regulations with IAEA standards, which is currently being performed, to determine potential improvements of the current German rules and regulations or the national safety practice and to take them into account for the revision of the German rules and regulations.

The Federal Government will continue the participation in the WENRA working group for the establishment of uniform and high reference levels for the safety of the nuclear power plants currently operated in the EU.

Ad Article 8
The results of the BMU studies on independence will be considered in the planning on a fundamental reform of the nuclear administration.

Ad Article 10
A process-oriented safety management system with indicators shall be implemented at all nuclear installations in accordance with the basic paper of the BMU.

As Federal Regulator, the BMU will therefore ensure that the regulations of the safety management systems of all German nuclear power plants will include provisions that oblige the utilities explicitly,

- to inform the competent supervisory authority without delay about findings that the demonstration of the control of design basis accidents could be put into question,
- to shut down the plant temporarily if the required control of design basis accidents cannot be demonstrated at short notice,
- to submit a work plan for analyses and backfitting measures.

Ad Article 12
The requirements on the safety specifications stated in the draft version shall serve as a basis for the revision of KTA 1201.

Ad Article 13
The BMU will initiate the submission of annual, plant-specific reports on ageing management by the licensees.

Ad Article 14
<u>High burn-up of fuel elements</u>
The licensees of the nuclear power plants intend to further increase the target burn-up of the fuel elements. The conservative accident analyses and analyses on damage extent with comprehensive consideration of high burn-up effects, which are required for safety assessment, are only available in part. In this respect, best-estimate analyses including uncertainty analyses will also be considered.

The Federal Regulator regards further experimental analyses on fuel behaviour both under operating and accident conditions as necessary and will monitor progress and result of the investigations started within the framework of OECD research programmes which are also supported by the German plant operators. Furthermore, the calculation methods for the assessment of the fuel and fuel rod behaviour will be examined with regard to the high burn-up effects expected.

The burn-up values reached in the nuclear reactor facilities were recorded. The licensees gave reports on operating experience and the available experimental database for the fuel rod behaviour in case of power ramps or during reactivity accidents. For reactivity accidents, the verifications are based on Japanese experiments at the Nuclear Safety Research Reactor (NSRR) and on the French CABRI-REP-Na test programme. The proof that the maximum permissible fuel rod loads are not exceeded is furnished with three-dimensional core models under realistic and also conservative assumptions. The German utilities participate in the OECD CABRI Water Loop Project in France aiming at the completion of the experimental database for higher burn-up and for representative cooling conditions of the fuel rods. The calculation methods to estimate the fuel rod behaviour in case of LOCAs are reviewed and further developed with regard to higher burn-up and cladding tube materials containing niobium.

<u>ATWS events</u>
The plans of the plant operators to increase the target burn-up of fuel elements and to increase the use of MOX fuel elements have caused the Federal Regulator to examine the safety margins with regard to the control of ATWS events. In Germany, the control of ATWS-events for PWRs is checked for each refuelling. The corresponding requirements are laid down in the RSK-Guidelines [4-1]. In addition, the RSK requires the control of such an event also without consideration of switch-off of the main coolant pumps. In additional analyses, the influence of the core design on the void reactivity curve and the sensitivity of the maximum pressure dependent of the effectiveness of different system functions will be determined.

<u>Boron dilution</u>
Thermal-hydraulic calculations showed that the necessary in-core boron concentration in case of small leak event might not be ensured continuously, thus jeopardizing subcriticality.

New test results from PKL and ROCOM and technical reports on the applicability of the test results to a nuclear reactor facility have been submitted. For the refuellings performed until now, qualified proof on the minimum boron concentration at the core entrance has been furnished. The activities on the validation of the analysis methods for the determination of the condensate quantity produced and accumulated, on the transport of the condensate to the core and on the mixing of the condensate with highly borated coolant, particularly in the lower plenum, are continued.

<u>Digital instrumentation and control (I&C)</u>
At present, digital I&C is already used at some German nuclear power plants for safety-relevant function. In the coming years, modification and backfitting measures of safety I&C at German nuclear power plants on the basis of computer-based systems are expected to be taken increasingly because analogue hard-wired systems are no longer produced and spare parts will not be available in future. Requirements for computer-bases systems with safety relevance currently exist in the nuclear rules and regulations as basic approach. The guidelines of the RSK only include general requirements for software-based safety I&C. However, for the practical examination and evaluation in the nuclear licensing procedure, they are insufficient. For the drafting of the necessary detailed requirements, the Federal Regulator will participate in the development of international standards to an increasing degree and will ensure the transferability to and compatibility with the safety requirements in

Germany. This applies especially to the use of prefabricated hard- and software in the safety system.

Hydrogen depletion in case of core melt accidents
At present, catalytic recombiners for hydrogen depletion after beyond-design basis accidents with core melt in the containment are being installed at all German PWRs, except for the Obrigheim and Biblis-A nuclear power plants. However, there are questions to be solved regarding the inadvertent ignition of hydrogen by the passive recombiners (→ Chapter 18 (i)).

Impairment of water suction from the containment sump
Findings obtained by tests performed in the USA caused the Federal Regulator to check the measures initiated in German nuclear power plants due to the event at the Swedish nuclear power plant Barsebäck once again. These measures are to ensure that in case of LOCAs, during which the core has to be cooled with water from the containment sump, the water suction will not be impaired severely by fibres of pipe insulation material or other material. After the re-examination it has to be ensured that the necessary conservatism is given. Experiments on individual aspects are currently being conducted and partly have already been completed. The RSK developed an evaluation basis considering national and international findings.

Ad Article 16
The revision of the Basic Recommendations for Emergency Preparedness in the Environment of Nuclear installations [3-15] will start in 2004.

Also in 2004, the potassium iodide tablets required as thyroid blocker will be produced and distributed to the *Länder* and stored at central storage facilities.

Ad Article 17
The revision of the standard on flood protection [KTA 2207] has not been finalised yet because the approach for the determination of the design basis flood still has to be subjected to an in-depth verification. Accordingly, the planned revision of plant design on the basis of the updated KTA safety standard is also still in process.

Further examinations to determine the safety margins in the plant design with regard to low water are intended.

In addition, generic studies on the impact of climate changes with extreme weather conditions on the safety of the nuclear power plants are provided in the long term.

Ad Article 19
The Nuclear Safety Officer and Reporting Ordinance [1A-17] and its reporting criteria shall be revised considering the experience from application of these criteria and current international developments.

Appendix 1 Nuclear Power Plants

Appendix 1-1 Nuclear Power Plants in Operation

	Nuclear power plants in operation Site	a) Licensee b) Manufacturer c) Major shareholder	Type Gross capacity MWe	Design gen./ Constr. line	a) Date of application b) First criticality
1	Obrigheim (KWO) Obrigheim Baden-Württemberg	a) Kernkraftwerk Obrigheim b) Siemens c) EnBW 95%	PWR 357	1.	a) 16.07.1964 b) 22.09.1968
2	Biblis A (KWB A) Biblis Hessen	a) RWE Power b) KWU c) RWE Power 100%	PWR 1225	2.	a) 11.06.1969 b) 16.07.1974
3	Biblis B (KWB B) Biblis Hessen	a) RWE Power b) KWU b) RWE Power 100%	PWR 1300	2.	a) 03.05.1971 b) 25.03.1976
4	Neckarwestheim 1 (GKN 1) Neckarwestheim Baden-Württemberg	a) Gemeinschaftskernkraftwerk Neckar b) KWU c) Neckarwerke 70%	PWR 840	2.	a) 02.04.1971 b) 26.05.1976
5	Brunsbüttel (KKB) Brunsbüttel Schleswig-Holstein	a) Kernkraftwerk Brunsbüttel b) AEG/KWU c) Vattenfall Europa 66,7%	BWR 806	69	a) 10.11.1969 b) 23.06.1976
6	Isar 1 (KKI 1) Essenbach Bayern	a) E.ON Kernkraft b) KWU c) E.ON Bayern 50% E.ON Kernkraft 50%	BWR 912	69	a) 25.06.1971 b) 20.11.1977
7	Unterweser (KKU) Esenshamm Niedersachsen	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 100%	PWR 1410	2.	a) 07.04.1971 b) 16.09.1978
8	Philippsburg 1 (KKP 1) Philippsburg Baden-Württemberg	a) EnBW b) KWU c) EnBW 100 %	BWR 926	69	a) 20.02.1970 b) 09.03.1979
9	Grafenrheinfeld (KKG) Grafenrheinfeld Bayern	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 100%	PWR 1345	3.	a) 07.06.1973 b) 09.12.1981
10	Krümmel (KKK) Krümmel Schleswig-Holstein	a) Kernkraftwerk Krümmel b) KWU c) Vattenfall Europa 50% E.ON Kernkraft 50%	BWR 1316	69	a) 18.02.1972 b) 14.09.1983
11	Gundremmingen B (KRB B) Gundremmingen Bayern	a) Kernkraftwerk Gundremmingen b) KWU c) RWE Power 75%	BWR 1344	72	a) 15.03.1974 b) 09.03.1984

Appendix 1-1 Nuclear Power Plants in Operation

	Nuclear power plants in operation Site	a) Licensee b) Manufacturer c) Major shareholder	Type Gross capacity MWe	Design gen./ Constr. line	a) Date of application b) First criticality
12	Grohnde (KWG) Grohnde Niedersachsen	a) Gemeinschaftskernkraftwerk Grohnde b) KWU c) E.ON Kernkraft 83,3%	PWR 1430	3.	a) 03.12.1973 b) 01.09.1984
13	Gundremmingen C (KRB C) Gundremmingen Bayern	a) Kernkraftwerk Gundremmingen b) KWU c) RWE Power 75%	BWR 1344	72	a) 15.03.1974 b) 26.10.1984
14	Philippsburg 2 (KKP 2) Philippsburg Baden-Württemberg	a) EnBW b) KWU c) EnBW 100 %	PWR 1458	3.	a) 24.06.1975 b) 13.12.1984
15	Brokdorf (KBR) Brokdorf Schleswig-Holstein	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 80%	PWR 1440	3.	a) 12.03.1974 b) 08.10.1986
16	Isar 2 (KKI 2) Essenbach Bayern	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 40%	PWR 1475	4. Konvoi	a) 13.02.1979 b) 15.01.1988
17	Emsland (KKE) Lingen Niedersachsen	a) Kernkraftwerke Lippe-Ems b) KWU c) RWE Power 87,5%	PWR 1400	4. Konvoi	a) 28.11.1980 b) 14.04.1988
18	Neckarwestheim 2 (GKN 2) Neckarwestheim Baden-Württemberg	a) Gemeinschaftskernkraftwerk Neckar b) KWU c) Neckarwerke 70%	PWR 1365	4. Konvoi	a) 27.11.1980 b) 29.12.1988

Appendix 1-2 Nuclear Power Plants Permanently Shut Down

	Nuclear power plants permanently shut down *) Site	a) Last licensee b) Manufacturer	Type Gross capacity MWe	a) First criticality b) Date of shutdown
1	Versuchsatomkraftwerk (VAK) Kahl Bayern	a) Versuchsatomkraftwerk Kahl b) AEG/General Electric	BWR 16	a) 13.11.1960 b) 25.11.1985
2	Mehrzweckforschungs-reaktor (MZFR) Karlsruhe Baden-Württemberg	a) Kernkraftwerk-betriebsgesellschaft b) Siemens/KWU	Pressurised heavy water reactor 57	a) 29.09.1965 b) 03.05.1984
3	Rheinsberg (KKR) Rheinsberg Brandenburg	a) Energiewerke Nord b) VEB Kernkraftwerksbau Berlin	PWR (WWER) 70	a) 11.03.1966 b) 01.06.1990
4	Gundremmingen A (KRB A) Gundremmingen Bayern	a) Kernkraftwerk RWE-Bayernwerk b) AEG/General Electric	BWR 250	a) 14.08.1966 b) 13.01.1977
5	Atomversuchskraftwerk (AVR) Jülich Nordrhein-Westfalen	a) Arbeitsgemeinschaft Versuchsreaktor b) BBC/Krupp Reaktorbau (BBK)	HTR 15	a) 26.08.1966 b) 31.12.1988
6	Stade (KKS) Stade Niedersachsen	a) E.ON Kernkraft b) KWU	PWR 672	a) 28.07.1967 b) 14.11.2003
7	Lingen (KWL) Lingen Niedersachsen	a) Kernkraftwerk Lingen b) AEG/KWU	BWR 268	a) 31.01.1968 b) 05.01.1977
8	Heißdampfreaktor (HDR) Großwelzheim Bayern	a) Forschungszentrum Karlsruhe b) AEG	Super heated steam-cooled reactor 25	a) 14.10.1969 b) 20.04.1971
9	Würgassen (KWW) Würgassen Nordrhein-Westfalen	a) PreussenElektra b) AEG/KWU	BWR 670	a) 22.10.1971 b) 26.08.1994
10	Niederaichbach (KKN) Niederaichbach Bayern	a) Forschungszentrum Karlsruhe Kernkraftwerkbetriebs-gesellschaft b) Siemens	Pressure tube reactor 106	a) 17.12.1972 b) 31.07.1974
11	Greifswald 1 (KGR 1) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) 03.12.1973 b) 18.12.1990

Appendix 1-2 Nuclear Power Plants Permanently Shut Down

	Nuclear power plants permanently shut down *) Site	a) Last licensee b) Manufacturer	Type Gross capacity MWe	a) First criticality b) Date of shutdown
12	Greifswald 2 (KGR 2) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) 03.12.1974 b) 14.02.1990
13	Greifswald 3 (KGR 3) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) 06.10.1977 b) 28.02.1990
14	Kompakte natriumgekühlte Reaktoranlage (KNK II) Karlsruhe Baden-Württemberg	a) Kernkraftwerkbetriebs-gesellschaft b) Interatom	FBR 21	a) 10.10.1977 b) 23.08.1991
15	Greifswald 4 (KGR 4) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) 22.07.1979 b) 02.06.1990
16	Thorium-Hochtemperatur-reaktor (THTR 300) Hamm-Uentrop Nordrhein-Westfalen	a) Hochtemperatur Kernkraftwerk b) BBC/HRB/NUKEM	HTR 308	a) 13.09.1983 b) 29.09.1988
17	Mülheim-Kärlich (KMK) Mülheim-Kärlich Rheinland-Pfalz	a) RWE Power b) BBR	PWR 1302	a) 01.03.1986 b) 09.09.1988
18	Greifswald 5 (KGR 5) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) 26.03.1989 b) 30.11.1989
	Projects stopped			
19	Greifswald 6 (KGR 6) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) b) Project stopped
20	Greifswald 7 (KGR 7) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) b) Project stopped
21	Greifswald 8 (KGR 8) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 440	a) b) Project stopped
22	SNR 300 Kalkar Nordrhein-Westfalen	a) Schnell-Brüter Kernkraftwerks-gesellschaft b) INTERATOM /BELGONUCLEAIRE /NERATOOM	FBR 327	a) b) Project stopped 20.03.1991

Appendix 1-2 Nuclear Power Plants Permanently Shut Down

	Nuclear power plants permanently shut down *) Site	a) Last licensee b) Manufacturer	Type Gross capacity MWe	a) First criticality b) Date of shutdown
23	Stendal A Stendal Sachsen-Anhalt	a) Altmark Industrie b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 1000	a) b) Project stopped
24	Stendal B Stendal Sachsen-Anhalt	a) Altmark Industrie b) VEB Kombinat Kraftwerksanlagenbau	PWR (WWER) 1000	a) b) Project stopped

*) decommissioned or shut down

Appendix 2 Research Reactors

Appendix 2-1 Research Reactors in Operation and Construction

	Research reactor Site	Licensee	Reactor type thermal output th. n-flow [cm^2s^{-1}]	First criticality
1	FRG-1 Geesthacht Schleswig-Holstein	GKSS-Forschungszentrum	Swimming pool/MTR 5 MW $< 1.4 \cdot 10^{14}$	23.10.1958
2	FRJ-2 (DIDO) Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Tank type D_2O reactor 23 MW $< 2 \cdot 10^{14}$	14.11.1962
3	SUR Berlin	Technische Universität Berlin Institut für Energietechnik	SUR-100 100 mW $< 5 \cdot 10^{6}$	26.07.1963 Last operation 2000
4	SUR Stuttgart Baden-Württemberg	Universität Stuttgart Institut für Kernenergetik und Energiesysteme	SUR-100 100 mW $< 5 \cdot 10^{6}$	24.08.1964
5	FRMZ Mainz Rheinland-Pfalz	Universität Mainz Institut für Kernchemie	Swimming pool/ TRIGA Mark II 0.1 MW $< 4 \cdot 10^{12}$	03.08.1965
6	SUR Aachen Nordrhein-Westfalen	RWTH Aachen Institut für elektrische Anlagen und Energiewirtschaft	SUR-100 100 mW $< 5 \cdot 10^{6}$	22.09.1965
7	SUR Ulm Baden-Württemberg	Fachhochschule Ulm Labor für Strahlenmess-technik und Reaktortechnik	SUR-100 100 mW $< 5 \cdot 10^{6}$	01.12.1965
8	SUR Kiel Schleswig-Holstein	Fachhochschule Kiel	SUR-100 100 mW $< 5 \cdot 10^{6}$	29.03.1966 Last operation 1997
9	SUR Hannover Niedersachsen	Universität Hannover Institut für Werkstoffkunde	SUR-100 100 mW $< 5 \cdot 10^{6}$	09.12.1971
10	SUR Furtwangen Baden-Württemberg	Fachhochschule Furtwangen	SUR-100 100 mW $< 5 \cdot 10^{6}$	28.06.1973
11	BER II Berlin	Hahn-Meitner-Institut Berlin	Swimming pool/MTR 10 MW $< 1.5 \cdot 10^{14}$	09.12.1973

Appendix 2-1 Research Reactors in Operation and Construction

	Research reactor Site	Licensee	Reactor type thermal output th. n-flow [cm^2s^{-1}]	First criticality
12	AKR Dresden Sachsen	Technische Universität Dresden Institut für Energietechnik	SUR type 2 W $< 3{\cdot}10^7$	28.07.1978 Currently not operated due to backfitting
13	ZLFR Zittau Sachsen	Hochschule Zittau/Görlitz Fachbereich Maschinenwesen	Tank type/WWR-M 10 W $< 2{\cdot}10^8$	25.05.1979
14	FRM-II Garching Bayern	Technische Universität München	Swimming pool/ compact core 20 MW $< 8{\cdot}10^{14}$	02.03.2004 Commissioning phase not yet completed

Appendix 2-2 Research Reactors in Decommissioning or Decommissioning Decided

	Research reactors in decommissioning or decommissioning decided Site	Licensee	Reactor type thermal output th. n-flow [cm^2s^{-1}]	a) First criticality b) Shutdown c) Status
1	FRM Garching Bayern	Technische Universität München	Swimming pool/MTR 4 MW $< 7{\cdot}10^{13}$	a) 31.10.1957 b) 28.07.2000 c) 14.12.1998 AS[1]
2	RFR Rossendorf Sachsen	Verein für Kernforschungstechnik und Analytik Rossendorf (VKTA)	Tank type/ WWR-S(M) 10 MW $< 1{,}2{\cdot}10^{14}$	a) 16.12.1957 b) 27.06.1991 c) 03.04.2001 3. TSG[2]
3	FR 2 Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	Tank type/ D_2O-Reaktor 44 MW $< 10^{14}$	a) 07.03.1961 b) 21.12.1981 c) 20.11.1996 SE[3]
4	FRJ-1 (MERLIN) Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Swimming pool/MTR 10 MW $< 10^{14}$	a) 23.02.1962 b) 22.03.1985 c) 31.07.2001 3. TSG
5	FRG-2 Geesthacht Schleswig-Holstein	GKSS-Forschungszentrum Geesthacht	Swimming pool/MTR 15 MW $< 1.5{\cdot}10^{14}$	a) 16.03.1963 b) 28.01.1993 c) 17.01.1995 Licence for decom. and partial dismantling
6	TRIGA HD I Heidelberg Baden-Württemberg	Deutsches Krebsforschungszentrum	Swimming pool/ TRIGA Mark I 0.25 MW $< 10^{13}$	a) 26.08.1966 b) 31.03.1977 c) 11.12.1980 SE
7	FMRB Braunschweig Niedersachsen	Physikalisch Technische Bundesanstalt Braunschweig	Swimming pool/MTR 1 MW $< 6{\cdot}10^{12}$	a) 03.10.1967 b) 19.12.1995 c) 02.03.2001 SG[4]
8	FRN Oberschleißheim Bayern	Forschungszentrum für Umwelt und Gesundheit (GSF)	Swimming pool/ TRIGA Mark III 1 MW $< 3{\cdot}10^{13}$	a) 23.08.1972 b) 16.12.1982 c) 24.05.1984 SE
9	FRH Hannover Niedersachsen	Medizinische Hochschule Hannover	Swimming pool/ TRIGA Mark I 0.25 MW $< 8.5{\cdot}10^{12}$	a) 31.01.1973 b) 01.01.1997 c) 22.02.2002 AS
10	TRIGA HD II Heidelberg Baden-Württemberg	Deutsches Krebsforschungszentrum	Swimming pool/ TRIGA Mark I 0.25 MW $< 10^{13}$	a) 28.02.1978 b) 30.11.1999 c) 25.02.2003 AS

Appendix 2-2 Research Reactors in Decommissioning or Decommissioning Decided

	Research reactors in decommissioning or decommissioning decided Site	Licensee	Reactor type thermal output th. n-flow [cm^2s^{-1}]	a) First criticality b) Shutdown c) Status
11	FRF 2 Frankfurt Hessen	Johann Wolfgang Goethe Universität Frankfurt	modifizierter TRIGA 1 MW $< 3 \cdot 10^{13}$	a) no criticality b) 1980 c) 25.10.1982 SG 01.09.2003 dismantling of remains applied for

1 AS Application for decommissioning
2 TSG Licence for partial decommissioning
3 SE Safe enclosure
4 SG Licence for decommissioning

Appendix 2-3 Research Reactors Completely Dismantled

	Decommissioned or dismantled research reactors Site	Licensee	Type thermal output th. n-flow [cm^2s^{-1}]	a) First criticality b) Shutdown
1	FRF 1 Frankfurt Hessen	Johann Wolfgang Goethe Universität Frankfurt	Homogeneous reactor 10 kW $< 10^{12}$	a) 10.01.1958 b) 19.03.1968 1970 partial dismantling, until 1977 conversion to FRF 2
2	BER I Berlin	Hahn-Meitner-Institut Berlin	Homogeneous reactor 50 kW $< 10^{12}$	a) 24.07.1958 b) 1972
3	SAR München Bayern	Technische Universität München	Argonaut 1 kW $< 2.4 \cdot 10^{11}$	a) 23.06.1959 b) 1968
4	SUA München Bayern	Technische Universität München	Subcritical assembly	a) 6/1959 b) 1968
5	AEG Prüfreaktor PR-10 Karlstein Bayern	Kraftwerk Union	Argonaut 180 W $2.5 \cdot 10^{10}$	a) 27.01.1961 b) 1976
6	SUR München Bayern	Technische Universität München	SUR-100 100 mW $< 5 \cdot 10^{6}$	a) 28.02.1962 b) 10.08.1981
7	RRR Rossendorf Sachsen	Verein für Kernforschungs-technik und Analytik Rossendorf (VKTA)	Argonau 1 kW $< 1.5 \cdot 10^{11}$	a) 16.12.1962 b) 7/1991
8	STARK Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	Argonaut 10 W $< 1.4 \cdot 10^{8}$	a) 11.01.1963 b) 3/1976
9	SUR Darmstadt Hessen	Technische Hochschule Darmstadt	SUR-100 100 mW $< 5 \cdot 10^{6}$	a) 23.09.1963 b) 22.02.1985
10	ANEX Geesthacht Schleswig-Holstein	GKSS-Forschungszentrum Geesthacht	Critical assembly 100 W $< 2 \cdot 10^{8}$	a) 5/1964 b) 05.02.1975
11	SUAK Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	Subcritical assembly	a) 20.11.1964 b) 07.12.1978

Appendix 2-3 Research Reactors Completely Dismantled

	Decommissioned or dismantled research reactors Site	Licensee	Type thermal output th. n-flow [cm^2s^{-1}]	a) First criticality b) Shutdown
12	SUR Hamburg	Fachhochschule Hamburg	SUR-100 100 mW $< 5{\cdot}10^6$	a) 15.01.1965 b) 1997
13	SUR Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	SUR-100 100 mW $< 5{\cdot}10^6$	a) 07.03.1966 b) 9/1996
14	SNEAK Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	Homogeneous reactor 1 kW $< 7{\cdot}10^6$	a) 15.12.1966 b) 11/1985
15	ADIBKA (L77A) Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Homogeneous reactor 100 W $< 2.8{\cdot}10^8$	a) 18.03.1967 b) 30.10.1972
16	AEG Nullenergie Reaktor Karlstein Bayern	Kraftwerk Union	Tank type/ Critical assembly 100 W $< 10^8$	a) 6/1967 b) 1973
17	SUR Bremen	Hochschule Bremen	SUR-100 100 mW $< 5{\cdot}10^6$	a) 10.10.1967 b) 17.06.1993
18	NS OTTO HAHN Geesthacht Schleswig-Holstein	GKSS-Forschungszentrum Geesthacht	DWR ship reactor 38 MW $< 2.8{\cdot}10^{13}$	a) 26.08.1968 b) 22.03.1979
19	RAKE Rossendorf Sachsen	Verein für Kernforschungs- technik und Analytik Rossendorf (VKTA)	Tank type/ Critical assembly 10 W $< 1{\cdot}10^8$	a) 03.10.1969 b) 26.11.1991
20	KEITER Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Critical assembly 1 W $< 2{\cdot}10^7$	a) 15.06.1971 b) 1982
21	KAHTER Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Critical assembly 100 W $< 2.2{\cdot}10^8$	a) 02.07.1973 b) 03.02.1984

Appendix 3 Design Basis Accidents and Beyond Design Basis Accidents, PWR and BWR

Level 3, design basis accidents — PWR

3-1 Transients

- Reactivity accident due to withdrawal of the most effective control rod or control rod group during start-up
- Loss of main heat sink caused by failure to open of the main steam bypass valve after turbine trip
- Loss of main feedwater supply
- Loss of auxiliary station supply (emergency power situation)
- Leakage in main steam piping up to 0.1F if manufactured in rupture preclusion quality, otherwise 2F
  (F: open cross section of the pipe)

3-2 Loss of coolant accidents

Leakage sizes to be considered for typical locations in the primary coolant pressure boundary:

- Leak cross section < 120 cm^2 for
  - overpressure protection devices stuck-open
  - rupture of connecting pipes
  - leakage at branch-off locations, penetrations or seals
  - leakage through open cracks
  - double-ended rupture of a steam generator tube
- Leak size 0.1F in the primary coolant line if manufactured in rupture preclusion quality, otherwise up to 2F

3-3 Radiologically representative accidents

- Loss of coolant with
  - leak size 2F for an instrumentation pipe in the annulus, assumed open for 30 minutes after rupture
  - leak size 2F for steam generator tube rupture and simultaneous leak in the main steam line behind the isolation valve, considering closing times of the isolation valve,
  - leak size 0.1F if manufactured in rupture preclusion quality, otherwise up to 2F
- Fuel element handling accidents
  - damage of all fuel rods at the outside of the fuel element
- Failure of auxiliary systems
  - pipe rupture in the off-gas treatment system
  - failure of the liquid waste evaporator in the coolant treatment system

3-4 Internal impacts

- Flooding due to leakage of pipes outside the primary coolant boundary, up to 0.1F if manufactured in rupture preclusion quality, otherwise up to 2F
- Other internal flooding (e.g. leakage of auxiliary service water pipes)
- Plant-internal fires
- Fragments with high kinetic energy resulting from component failure (e.g. turbine blade failure)

3-5 External impacts

- Site-specific events caused by nature
  (earthquake and weather condition, such as lightning, flooding, wind, ice and snow)

Level 4, beyond-design basis accidents — PWR

4-1 Specific, very rare events

- ATWS
- Site-specific, man-made external impacts (specific emergency situations)

4-2 Plant condition due to unavailability of activated safety equipment (emergencies)

- Loss of steam generator feed, with a trend to a total dry-out of the secondary side
- Loss of coolant from a small leak, with a trend to increase the primary coolant pressure beyond the feed pressure of the high pressure injection pumps
- Double-ended rupture of a steam generator tube and increasing main steam pressure, with a trend to open the main steam safety valves
- Loss of three-phase current supply - unless backed by batteries - for up to 2 hours
- Global long-term increase of containment pressure, with a trend to exceed the design pressure limit
- Increase of hydrogen concentration in the containment, with a trend to reach the ignition point

Level 3, design basis accidents BWR

3-1 Transients

- Reactivity accidents
  - limited failure of the most effective control rod
  - uncontrolled withdrawal of control rods during start-up
- Loss of main heat sink due to erroneous closing of the main steam containment penetration valves
- Loss of the main feedwater supply
- Loss of auxiliary station supply (emergency power situation)

3-2 Loss of coolant accidents

Leakage sizes to be considered for typical locations in the coolant pressure boundary:

- Leak cross section < 80 cm² for leaks through open cracks in the lower plenum of the reactor pressure vessel, in between the control rod drives
- Leak size < 0.1F in pipes if manufactured in rupture preclusion quality, otherwise up to 2F
  (F: open cross section of the pipe)

3-3 Radiologically representative accidents

- Loss of coolant with
  - leak size 2F for an instrumentation pipe with reactor coolant in the reactor building, assumed open for 30 minutes after rupture
  - leak size 0.1F for a residual heat removal train in the reactor building if manufactured in rupture preclusion quality, otherwise 1F, considering closing times of the isolation valve
  - leak size 0,1F if manufactured in rupture preclusion quality, otherwise up to 2F
  - Leak cross section 80 cm² for leaks through open cracks in the lower plenum of the reactor pressure vessel, in between the control rod drives
- Fuel element handling accidents
  - damage of all fuel rods at the outside of the fuel element
- Failure of auxiliary systems
  - pipe rupture in the off-gas treatment system
  - failure of the liquid waste evaporator in the coolant treatment system

3-4 Internal impacts

- Flooding due to leakage of pipes outside the reactor coolant boundary, up to 0.1F if manufactured in rupture preclusion quality, otherwise up to 2F
- Other internal flooding (e.g. leakage of auxiliary service water pipes)
- Plant-internal fires
- Fragments with high kinetic energy resulting from component failure (e.g. turbine blade failure)

3-5 External impacts

- Site-specific events caused by nature
  (earthquakes and weather condition, such as lightning, flooding, wind, ice and snow)

Level 4, beyond-design basis accidents BWR

4-1 Specific, very rare events

- ATWS
- site-specific, man-made external impacts (specific emergency situations)

4-2 Plant conditions due to unavailability of activated safety equipment (emergencies)

- Loss of coolant with subsequent overfeeding of a main steam pipe and the possibility of water hammer outside the penetration isolation
- Transients with a trend to decrease the coolant level within the reactor pressure vessel
  to the bottom of the core
- Loss of three-phase current supply - unless backed by batteries - for up to 2 hours
- Global long-term increase of containment pressure, with a trend to exceed the design pressure limit
- Increase of hydrogen concentration in the containment, with a trend to reach the ignition point

Appendix 4 Design Characteristics Important to Safety, PWR and BWR

1. Reactor Coolant Pressure Boundary PWR

Design Characteristics	1^{st} Design Generation	2^{nd} Design Generation	3^{rd} Design Generation	4^{th} Design Generation
Number of Loops	2 or 4	3 or 4	4	4
Suitability of the components for non-destructive testing	Yes, with minor restrictions		Yes	
Components				
- Seamless forged rings for vessels	Reactor pressure vessel, steam generators (primary side only)		Reactor pressure vessel, steam generators, pressuriser	
- Seamless pipes	Main coolant line with minor restrictions		Main coolant line	
Materials				
- Ageing-resistant ferritic fine-grained structural steels with stabilised austenitic cladding	All components and pipes with nominal diameter > 400 mm			Like 1^{st} to 3^{rd} generation, but with optimised qualities
- Ageing-resistant stabilised austenitic steels	All pipes with nominal diameter < 400 mm and component internals			
- Corrosion-resistant steam generator tube material (Incoloy 800)	Yes (exchange of steam generators in one plant)	Yes		
Application of the rupture preclusion concept	Post-commissioning qualification		Prior to commissioning	From the start of planning
Reduction of embrittlement from neutron radiation exposure	Use of dummy fuel elements and special fuel element management	Enlargement of reactor pressure vessel diameters to reduce neutron fluence		

1. Reactor Coolant Pressure Boundary BWR

Design Characteristics	Construction Line 69	Construction Line 72
Re-circulation pumps integrated in the reactor pressure vessel	8 to 10	8
Suitability of the components for non-destructive testing	Yes, with minor restrictions	Yes
Components		
- Seamless forged rings for reactor pressure vessels	No	Yes
- Seamless pipes	Yes, after replacement of pipes	Yes
Materials		
- Ageing-resistant ferritic fine-grained structural steels	Reactor pressure vessel, main-steam and feedwater pipes	
- Ageing-resistant stabilised austenitic steels	Pipes, partly backfitted by replacements, in addition reactor pressure vessel internals and cladding	
Application of the break preclusion concept	Post-qualification partly through pipe replacement	Prior to planning
Reduction of embrittlement from neutron radiation exposure	Special fuel element management	

2. Emergency Core Cooling PWR

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Number of emergency core cooling trains / capacity	4 trains of at least 50 % each			
Pump head of high-pressure pumps	Approximately 110 bar			
Secondary circuit shutdown in case of small leaks	Manually or fully automatic	Automatic partial shutdown or fully automatic	fully automatic	
Number of borated water flooding tanks	3 or 5	4, in some cases twin tanks		
Pump head of low-pressure injection pumps	1 plant 8 bar 1 plant 18 bar	Approximately 10 bar		
Accumulators (injection pressure)	1 per loop (26 bar); 1 plant without accumulators	1 or 2 per loop (25 bar)	2 per loop (25 bar)	
Sump pipe before outer penetration isolation valve	Single pipe (1 plant without sump suction pipe)	Guard pipe construction, some with leakage monitoring	Guard pipe construction with leakage monitoring	
Place of installation of the active emergency core cooling systems	Separate building, reactor building or annulus	Annulus		

2. Emergency Core Cooling BWR

Design Characteristics	Construction Line 69	Construction Line 72
Number of trains of the high-pressure safety injection system (capacity)	1 train (steam turbine, up to 50 bar main steam pressure, approx. 300 kg/s)	3 trains (electric pumps, 3 x 70 kg/s)
Diversified high-pressure safety injection system	1 train (electric pump approx. 40 kg/s)	No
Pressure relief	7 to 11 safety and pressure relief valves, additionally 3 to 6 motorised pressure relief valves	11 safety and pressure relief valves, additionally 3 motorised pressure relief valves
Intermediate-pressure injection system	No	1 train (electric pump, 40 bar)
Number of low-pressure emergency core cooling trains / capacity	4 trains of 50% each	3 trains of 100% each
Low-pressure safety system with diversified injection	1 x 100% core flooding system	No
Backfeed from containment sump	Yes, via active systems	Yes, via passive systems with 4 overflow pipes
Place of installation of the emergency core cooling systems	In separate rooms of the reactor building	In separate rooms of the reactor building, intermediate-pressure system in a bunkered building

3. Containment Vessel PWR

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Type	Spherical steel vessel with surrounding concrete enclosure, annular gap and constant internal subatmospheric pressure			
Design pressure (overpressure)	1 plant 2.99 bar, 1 plant 3.78 bar	4.71 bar	5.3 bar	5.3 bar
Design temperature	1 plant 125°C 1 plant 135°C	135°C	145°C	145°C
Material of steel vessel	BH36KA; HSB50S	FB70WS; FG47WS; BHW33	FG51WS; 15 MnNi 63; Aldur 50/65D	15 MnNi 63
Wall thickness of steel vessel in the spherical region remote from discontinuities	Up to 25 mm	Up to 29 mm	Up to 38 mm	38 mm
Airlocks				
- Equipment airlock	Single or double seals without evacuation	Double seals with evacuation		
- Personnel airlock	Single or double seals without evacuation	Double seals with evacuation		
- Emergency airlock	One with single seal	One with double seals and evacuation	Two with double seals and evacuation	
Penetrations				
- Main steam line	One isolation valve outside of containment			
- Feedwater line	One isolation valve each inside and outside of containment			
- Emergency core cooling and auxiliary systems	With a few exceptions, one isolation valve each inside and outside of containment			One isolation valve each inside and outside of containment
- Ventilation systems	One isolation valve each inside and outside of containment			

3. Containment Vessel BWR

Design Characteristics	Construction Line 69	Construction Line 72
Type	Spherical steel vessel with pressure suppression pool located in the thorus	Cylindrical pre-stressed concrete shell with annular pressure suppression pool
Design pressure (overpressure)	Up to 3.5 bar	3.3 bar
Design temperature	Approximately 150°C	
Material of steel vessel	WB25; Aldur50D, BHW25	TTSTE29
Wall thickness of steel vessel outside the concrete support	Depending on geometry and design: 18 mm to 50 mm, 18 mm to 65 mm, 20 mm to 70 mm, 25 mm to 70 mm	8 mm steel liner
Number of pipes in the pressure suppression pool	Depending on the plant: 58, 62, 76 or 90	63
Immersion depth of pipes in the pressure suppression pool	2.0 or 2.8 m	4.0 m
Inertisation of the air in the pressure suppression pool	Yes	Yes
Inertisation of the drywell	Yes	No
Airlocks	In all cases double seals with evacuation	
- Equipment airlock	None	
- Personnel airlock	Leading to control rod drive chamber, for personnel and for equipment transports	
- Emergency airlock	One, from control rod drive chamber	One from control rod drive chamber and one above pressure suppression pool
Penetrations		
- Main steam line/ Feedwater line	One isolation valve each inside and outside of containment	
- Emergency core cooling and auxiliary systems	Emergency core cooling system in the area of the pressure suppression pool and several small pipes with two isolation valves outside of containment, otherwise one isolation valve each inside and outside of containment	
- Ventilation system	Two isolation valves outside of containment	

4. Limitations and Safety Actuation Systems PWR

4.1 Limitations

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Reactor power limitation	1 plant yes, 1 plant no	Yes		
Control rod movement limitation	Yes (monitoring of shut-down reactivity)			
Limitations of coolant pressure, coolant mass and temperature gradient	Coolant pressure	Partially	Yes	

4.2 Safety Actuation Systems

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Actuation criteria derived from accident analysis	Largely, yes	Yes		
Different physical actuation criteria for reactor protection system	Yes, or higher-grade redundancy	Yes, or diverse actuation channels		
Failure combinations	Random failure, systematic failure, consequential failures, non-availability due to maintenance			
Testing of reactor protection system is possible during power operation	Yes, largely by automatic self-monitoring (of functional readiness)			
Actuation of protection systems	Apart from a few exceptions, all actions are performed automatically, and manual actions are not required within the first 30 min after the onset of an accident.			

4. Limitations and Safety Actuation Systems BWR

4.1 Limitations

Design Characteristics	Construction Line 69	Construction Line 72
Fixed reactor power limitation	Yes, speed reduction of forced-circulation pumps	
Variable reactor power limitation	Yes, control rod withdrawal interlock start-up interlock of forced-circulation pumps	
Local power limitation	Yes, control rod withdrawal interlock	Yes, control rod withdrawal interlock and speed reduction of forced-circulation pumps

4.2 Safety Actuation Systems

Design Characteristics	Construction Line 69	Construction Line 72
Actuation criteria derived from accident analysis	Largely, yes	Yes
Different physical actuation criteria for reactor protection system	Yes, or higher level of redundancy	Yes, or diversified actuation channels
Failure combinations	Random failure, systematic failure, consequential failures, non-availability due to maintenance	
Testing of reactor protection system is possible during power operation	yes, largely by automatic self-monitoring (of functional readiness)	
Actuation of protection systems	Apart from a few exceptions, all actions are performed automatically, and manual actions are not required within the first 30 min after the onset of an accident.	

5. Electric Power Supply PWR

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

5. Electric Power Supply BWR

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

6. Protection against External Impacts PWR

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Earthquake	Design of components important to safety in accordance with site-specific load assumptions			
Aircraft crash and pressure waves from explosions	Not considered in the design, later risk assessment, separate emergency systems	Different designs, separate emergency systems	Specific design in accordance with the nuclear safety regulations (see Chapter 17 (i)), emergency systems integrated in the safety system	

6. Protection against External Impacts BWR

Design Characteristics	Construction Line 69	Construction Line 72
Earthquake	Design of components important to safety in accordance with site-specific load assumptions	
Aircraft crash and pressure waves from explosions	Different designs, up to status of construction line 72, emergency systems separate, or integrated in the safety system	Specific design in accordance with the nuclear safety regulations (see Chapter 17 (i)), emergency systems integrated in the safety system

Appendix 5 Reference List of Nuclear Safety Regulations

(A selection concerning nuclear power plants, structure and order
as in the "Handbuch Reaktorsicherheit und Strahlenschutz" www.bfs.de)

Content

1 Rechtsvorschriften
1A Nationales Atom- und Strahlenschutzrecht

1A-1 Gesetz zur **Ergänzung des Grundgesetzes** vom 23. Dezember 1959, betreffend §§ 74a Nr. 11, 87c (BGBl.I, S. 813)

1A-3 Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (**Atomgesetz** - AtG) vom 23. Dezember 1959, Neufassung vom 15. Juli 1985 (BGBl.I, Nr. 41), zuletzt geändert durch die 8. ZuständigkeitsanpassungsVO vom 25. November 2003 (BGBl.I 2003, Nr.56)

1A-4 **Fortgeltendes Recht der Deutschen Demokratischen Republik** aufgrund von Artikel 9 Abs. 2 in Verbindung mit Anlage II Kapitel XII Abschnitt III Nr. 2 und 3 des Einigungsvertrages vom 31. August 1990 in Verbindung mit Artikel 1 des Gesetzes zum Einigungsvertrag vom 23. September 1990 (BGBl.II, S. 885, 1226), soweit dabei radioaktive Stoffe, insbesondere Radonfolgeprodukte, anwesend sind:

- Verordnung über die Gewährleistung von Atomsicherheit und Strahlenschutz vom 11. Oktober 1984 und Durchführungsbestimmung zur Verordnung über die Gewährleistung von Atomsicherheit und Strahlenschutz vom 11. Oktober 1984 (GBl.(DDR) I 1984, Nr. 30, berichtigt GBl.(DDR) I 1987, Nr. 18)
- Anordnung zur Gewährleistung des Strahlenschutzes bei Halden und industriellen Absetzanlagen und bei Verwendung darin abgelagerter Materialien vom 17. November 1990 (GBl.(DDR) I 1990, Nr. 34)

1A-5 Gesetz zum vorsorgenden Schutz der Bevölkerung gegen Strahlenbelastung (**Strahlenschutzvorsorgegesetz** - StrVG) vom 19. Dezember 1986 (BGBl.I, S. 2610), zuletzt geändert durch die 8. ZuständigkeitsanpassungsVO vom 25. November 2003 (BGBl.I 2003, Nr.56)

1A-8 Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (**Strahlenschutzverordnung** - StrlSchV) vom 20. Juli 2001 (BGBl.I 2001, Nr. 38), zuletzt geändert durch VO vom 18. Juni 2002 (BGBl.I 2002, Nr. 36)

1A-10 Verordnung über das Verfahren bei der Genehmigung von Anlagen nach § 7 des Atomgesetzes (**Atomrechtliche Verfahrensverordnung** - AtVfV) vom 18. Februar 1977, Neufassung vom 3. Februar 1995 (BGBl.I 1995, Nr. 8), letzte (ausführbare) Änderung durch Gesetz vom 27. Juli 2001 (BGBl.I 2001, Nr. 40)

1A-11 Verordnung über die Deckungsvorsorge nach dem Atomgesetz (**Atomrechtliche Deckungsvorsorge-Verordnung** - AtDeckV) vom 25. Januar 1977 (BGBl.I 1977, S. 220), zuletzt geändert durch durch VO vom 18. Juni 2002 (BGBl.I 2002, Nr. 36)

1A-13 Verordnung über Vorausleistungen für die Einrichtung von Anlagen des Bundes zur Sicherstellung und zur Endlagerung radioaktiver Abfälle (**Endlagervorausleistungs-verordnung** - EndlagerVlV) vom 28. April 1982 (BGBl.I, S. 562), zuletzt geändert durch VO vom 18. Juni 2002 (BGBl.I 2002, Nr. 36)

1A-17 Verordnung über den kerntechnischen Sicherheitsbeauftragten und über die Meldungen von Störfällen und sonstigen Ereignissen (Atomrechtliche **Sicherheitsbeauftragten- und Meldeverordnung** - AtSMV) vom 14. Oktober 1992 (BGBl.I 1992, Nr. 48), zuletzt geändert durch VO vom 18. Juni 2002 (BGBl.I 2002, Nr. 36)

1A-18 Verordnung über die Verbringung radioaktiver Abfälle in das oder aus dem Bundesgebiet (**Atomrechtliche Abfallverbringungsverordnung** - AtAV) vom 27. Juli 1998 (BGBl.I 1998, Nr. 47), zuletzt geändert durch VO vom 20 Juli 2001 (BGBl.I 2001, Nr.38)

1A-19 Verordnung für die Überprüfung der Zuverlässigkeit zum Schutz gegen Entwendung oder erhebliche Freisetzung radioaktiver Stoffe nach dem Atomgesetz (**Atomrecht-liche Zuverlässigkeits-überprüfungs-Verordnung** - AtZüV) vom 1. Juli 1999 (BGBl.I 1999, Nr. 35), zuletzt geändert durch G vom 11. Oktober 2003 (BGBl.I 2003, Nr. 73)

1A-20 Verordnung zur Abgabe von kaliumiodidhaltigen Arzneimitteln zur Iodblockade der Schilddrüse bei radiologischen Ereignissen (**Kaliumiodidverordnung** - KIV) vom 5. Juni 2003 (BGBl. I 2003, Nr. 25)

1B Rechtsvorschriften, die im Bereich der Sicherheit kerntechnischer Anlagen anzuwenden sind

1B-1 **Verwaltungsverfahrensgesetz** vom 25. Mai 1976 (BGBl.I 1976, S. 1253), Neufassung vom 23. Januar 2003 (BGBl.I 2003, Nr. 4)

1B-2 **Umweltinformationsgesetz** vom 8. Juli 1994 (BGBl.I 1994, Nr. 42), Neufassung vom 23. August 2001 (BGBl.I 2001, Nr. 45)
Umweltinformationskostenverordnung vom 7. Dezember 1994 (BGBl.I 1994, Nr. 88), Neufassung vom 23. August 2001 (BGBl.I 2001, Nr. 45)

1B-3 **Umweltverträglichkeitsprüfungsgesetz** vom 12. Februar 1990 (BGBl.I, S. 205), Neufassung vom 5. September 2001 (BGBl.I 2001, Nr. 48)

1B-4 **Umweltauditgesetz** vom 7. Dezember 1995 (BGBl.I 1995, 1591), Neufassung vom 4. September 2002 (BGBl.I 2002, S. 3490)

1B-10 **Umwelthaftungsgesetz** vom 10. Dezember 1990 (BGBl.I 1990, S. 2634)

1B-11 **Strafgesetzbuch** vom 15. Mai 1871 (RGBl. S. 127) in der Fassung der Bekanntmachung vom 10. März 1987 (BGBl.I 1987, S. 945+1160

1B-14 **Bau- und Raumordnungsgesetz** 1998 vom 18. August 1997 (BGBl.I 1997, Nr. 59)

1B-16 Gesetz zum Schutz vor schädlichen Umwelteinwirkungen durch Luftverunreinigungen, Geräusche, Erschütterungen und ähnliche Vorgänge (**Bundes-Immissionsschutzgesetz**) in der Fassung der Bekanntmachung vom 14. Mai 1990 (BGBl.I 1990, S. 880), mit diversen Verordnungen

1B-24 **Kreislaufwirtschafts- und Abfallgesetz** vom 27. September 1994 (BGBl.I 1994, Nr. 66)

1B-27 Gesetz zur Ordnung des Wasserhaushalts (**Wasserhaushaltsgesetz**) vom 27. Juli 1957, Neufassung vom 12. November 1996 (BGBl.I 1996, Nr. 58), Neufassung vom 19. August 2002 (BGBl.I 2002, Nr. 59)

1B-29 Gesetz über Naturschutz und Landschaftspflege (**Bundesnaturschutzgesetz**) vom 12. März 1987 (BGBl.I 1987, S. 889), Neufassung vom 21. September 1998 (BGBl.I 1998, Nr. 66), Neufassung vom 25. März 2002 (BGBl.I 2002, Nr. 22)

1B-31 Verordnung zum Schutz vor gefährlichen Stoffen (**Gefahrstoffverordnung**), Neufassung vom 15. November 1999 (BGBl.I 1999, Nr. 52)

1B-32 Verordnung über Trinkwasser und über Wasser für Lebensmittelbetriebe (**Trinkwasserverordnung**) vom 5. Dezember 1990 (BGBl.I 1990, S. 2612, BGBl.I 1991, S. 227)

1B-33 Gesetz über technische Arbeitsmittel (**Gerätesicherheitsgesetz**) vom 24. Juni 1968, Neufassung vom 23. Oktober 1992, (BGBl.I 1992, Nr. 49)

1B-34 **Betriebssicherheitsverordnung** vom 27. September 2002 (BGBl.I 2002, S. 3777)

1B-37 **Unfallverhütungsvorschrift Kernkraftwerke** (VBG 30) und Durchführungsanweisung zur Unfallverhütungsvorschrift vom 1. Januar 1987

1B-39 Gesetz über **Betriebsärzte, Sicherheitsingenieure** und andere Fachkräfte für Arbeitssicherheit vom 12. Dezember 1973 (BGBl.I 1973, S. 1885

1B-41 Gesetz über den Verkehr mit Lebensmitteln, Tabakerzeugnissen, kosmetischen Mitteln und sonstigen Bedarfsgegenständen (**Lebensmittel- und Bedarfsgegenständegesetz**) vom 15. August 1974 (BGBl.I 1975, S. 2652), Neufassung vom 9. September 1997 (BGBl.I 1997, Nr. 63), mit diversen Verordnungen

1E Multilaterale Vereinbarungen über nukleare Sicherheit und Strahlenschutz mit nationalen Ausführungsvorschriften

Nukleare Sicherheit und Strahlenschutz

1E-1 Übereinkommen über die Umweltverträglichkeitsprüfung im grenzüberschreitenden Rahmen - **Espoo-Konvention** (Convention on the Environmental Impact Assessment in a Transboundary Context) vom 25. Februar 1991 und Änderungen vom Februar 2001, in Kraft seit 10. September 1997
Gesetz dazu vom 7. Juni 2001 (BGBl.II 2001, Nr. 22)
in Kraft für Deutschland seit 8. August 2002

1E-2 Konvention über den Zugang zu Informationen, die Öffentlichkeitsbeteiligung an Entscheidungsverfahren und den Zugang zu Gerichten in Umweltangelegenheiten - **Aarhus-Konvention** (Convention on Access to Information, Public Participation in Decision-Making and Access to Justice in Environmental Matters) vom 25. Juni 1998, in Kraft seit 30. Oktober 2001
von Deutschland gezeichnet am 21. Dezember 1998

1E-3 Übereinkommen **Nr. 115** der Internationalen Arbeitsorganisation vom 22. Juni 1960 über den Schutz der Arbeitnehmer vor ionisierenden Strahlen (Convention Concerning the Protection of Workers against Ionising Radiations) vom 22. Juni 1960, in Kraft seit 17. Juni 1962
Gesetz hierzu vom 23. Juli 1973 (BGBl.II 1973, Nr. 37),
in Kraft für Deutschland seit 26. September 1974 (BGBl.II 1973, Nr. 63)

1E-4 Ratsbeschluß der Organisation für Wirtschaftliche Zusammenarbeit und Entwicklung (OECD) vom 18. Dezember 1962 über die Annahme von Grundnormen für den Strahlenschutz (**OECD-Grundnormen**) (Radiation Protection Norms)
Gesetz hierzu vom 29. Juli 1964 (BGBl.II 1964, S. 857),
in Kraft für Deutschland seit 3. Mai 1965
Neufassung vom 25. April 1968 (BGBl.II 1970, Nr. 20)

1E-5 Übereinkommen vom 26. Oktober 1979 über den **physischen Schutz von Kernmaterial** (Convention on the Physical Protection of Nuclear Material (INFCIRC/274 Rev.1), entry into force 8 February 1987),
Gesetz hierzu vom 24. April 1990 (BGBl.II 1990, S. 326), zuletzt geändert durch das Strafrechtsänderungsgesetz vom 27. Juni 1994 (BGBl.I 1994, Nr. 40),
in Kraft für Deutschland seit 6. Oktober 1991 (BGBl.II 1995, Nr. 11)

1E-6 Übereinkommen über die **frühzeitige Benachrichtigung** bei nuklearen Unfällen vom 26. September 1986 und Übereinkommen über **Hilfeleistung bei nuklearen Unfällen** oder radiologischen Notfällen vom 26. September 1986, (Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (INFCIRC/336), Convention on Early Notification of a Nuclear Accident (INFCIRC/335), entry into force 27 October 1986, both),
Gesetz zu den beiden IAEA-Übereinkommen vom 16. Mai 1989 (BGBl.II 1989, Nr. 18),
in Kraft für Deutschland seit 15. Oktober 1989 (BGBl.II 1993, Nr. 34)

1E-7 **Übereinkommen über nukleare Sicherheit** (Convention on Nuclear Safety, INFCIRC/449), vom 17. Juni 1994, in Kraft seit 24. Oktober 1996)
Gesetz hierzu vom 7. Januar 1997 (BGBl.II 1997, Nr. 2)
in Kraft für Deutschland seit 20. April 1997 (BGBl.II 1997, Nr. 14)

1E-8 Gemeinsames Übereinkommen über die Sicherheit der Behandlung abgebrannter Brennelemente und über die Sicherheit der Behandlung radioaktiver Abfälle - **Übereinkommen über nukleare Entsorgung** (Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, INFCIRC/546) vom 5. September 1997, in Kraft seit 18. Juni 2001,
Gesetz hierzu vom 13. August 1998 (BGBl.II 1998, Nr. 31)
in Kraft für Deutschland seit 18. Juni 2001 (BGBl.II 2001, Nr. 36)

1E-9 Vertrag über die Nichtverbreitung von Kernwaffen - **Atomwaffensperrvertrag** (Treaty on the Non-Proliferation of Nuclear Weapons, INFCIRC/140)
vom 1. Juli 1968, in Kraft seit 5. März 1970 Gesetz dazu vom 4. Juni 1974 (BGBl.II 1974, S. 785)
in Kraft für Deutschland seit 2. Mai 1975 (BGBl.II 1976, S. 552),
Verlängerung des Vertrages auf unbegrenzte Zeit am 11. Mai 1995 (BGBl.II 1995, S. 984)

1E-10 Übereinkommen vom 5. April 1973 zwischen dem Königreich Belgien, dem Königreich Dänemark, der Bundesrepublik Deutschland, Irland, der Italienischen Republik, dem Großherzogtum Luxemburg, dem Königreich der Niederlande, der Europäischen Atomgemeinschaft und der Internationalen Atomenergie-Organisation in Ausführung von Artikel III Absätze 1 und 4 des Vertrages vom 1. Juli 1968 über die Nichtverbreitung von Kernwaffen (**Verifikationsabkommen**), (INFCIRC/193), entry into

force for all Parties 21 February 1977
Gesetz hierzu vom 4. Juni 1974 (BGBl.II 1974, S. 794),
Ausführungsgesetz hierzu vom 7. Januar 1980 (BGBl.I 1980, S. 17), zuletzt geändert durch Gesetz vom 27. Dezember 1993 (BGBl.I 1993, S. 2378)
Zusatzprotokoll vom 22. September 1998,
Gesetz zum Zusatzprotokoll vom 22. September 1998 vom 29. Januar 2000 (BGBl.I 2000, Nr. 4)
Ausführungsgesetz zum Verifikationsabkommen und zum Zusatzprotokoll vom 29. Januar 2000 (BGBl.I 2000, Nr. 5)

Haftung

1E-11 Übereinkommen über die Haftung gegenüber Dritten auf dem Gebiet der Kernenergie - Pariser Übereinkommen (Convention on Third Party Liability in the Field of Nuclear Energy - **Paris Convention**) vom 29. Juli 1960, ergänzt durch das Protokoll vom 28. Januar 1964 und das Protokoll vom 16. November 1982, in Kraft seit 1. April 1968
Gesetz hierzu vom 8. Juli 1975 (BGBl.II 1975, S. 957)
in Kraft für Deutschland seit 30. September 1975 (BGBl.II 1976, S. 308),
Gesetz hierzu vom 21. Mai 1985 (BGBl.II 1985, S. 690)
in Kraft für Deutschland seit 7. Oktober 1988 (BGBl.II 1989, S. 144)

1E-12 Zusatzübereinkommen zum Pariser Übereinkommen vom 29. Juli 1960 - Brüsseler Zusatzübereinkommen (Convention Supplementary to the Paris Convention of 29 July 1960 on Third Party Liability in the Field of Nuclear Energy - **Brussels Convention**) vom 31. Januar 1963, ergänzt durch das Protokoll vom 28. Januar 1964 und das Protokoll vom 16. November 1982, in Kraft
Gesetz hierzu vom 8. Juli 1975 (BGBl.II 1975, S. 957)
in Kraft für Deutschland seit 1. Januar 1976 (BGBl.II 1976, S. 308)
Gesetz hierzu vom 21. Mai 1985 (BGBl.II 1985, S. 690)
in Kraft für Deutschland seit 1. August 1991 (BGBl.II 1995, S. 657)

1E-13 Vienna Convention on Civil Liability for Nuclear Damage - **Vienna Convention** (Wiener Übereinkommen) of 21 May 1963, (INFCIRC/500), entry into force 12 November 1977, amended by Protocol of 29 September 1997

1E-14 Joint Protocol Relating to the Application of the **Vienna Convention and the Paris Convention** (**Joint Protocol** - Gemeinsames Protokoll) of 21 September 1988 (INFCIRC/402),
entry into force 27 April 1992
Gesetz hierzu vom 5. Mai 2001 (BGBl.II 2001, Nr.7)
in Kraft für Deutschland seit 13. September 2001 (BGBl.II 2001, Nr. 24)

1E-15 Convention on **Supplementary Compensation** for Nuclear Damage of 29 September 1997 (INFCIRC/567), not yet in force

1E-16 Übereinkommen über die zivilrechtliche **Haftung bei der Beförderung von Kernmaterial auf See** (Convention Relating to Civil Liability in the Field of Maritime Carriage of Nuclear Materials)
vom 17. Dezember 1971, in Kraft seit 15. Juli 1975
Gesetz hierzu vom 8. Juli 1975 (BGBl.II 1975, S. 957), geändert durch Gesetz vom 9. Juni 1980 (BGBl.II 1980, S. 721)
in Kraft für Deutschland seit 30. Dezember 1975 (BGBl.II 1976, S. 307)

1E-17 Abkommen zwischen der Bundesrepublik **Deutschland und der Schweizerischen Eidgenossenschaft** über die Haftung gegenüber Dritten auf dem Gebiet der Kernenergie vom 22. Oktober 1986,
Gesetz dazu vom 28. Juni 1988 (BGBl.II 1988, S. 598),
in Kraft für Deutschland seit 21. September 1988 (BGBl.II 1988, S. 955)

1F Recht der Europäischen Union

Verträge, Allgemeines

1F-1 Vertrag vom 25. März 1957 zur Gründung der Europäischen Atomgemeinschaft (**EURATOM**) in der Fassung des Vertrages über die **Europäische Union** vom 7. Februar 1992, geändert durch den Beitrittsvertrag vom 24. Juni 1994 in der Fassung des Beschlusses vom 1. Januar 1995 (BGBl.II 1957, S. 753, 1014, 1678; BGBl.II 1992, S. 1251, 1286; BGBl.II 1993, S. 1947; BGBl.II 1994, S. 2022; ABl.EG 1995, Nr. L1), der Vertrag ist in seiner ursprünglichen Fassung am 1. Januar 1958 in Kraft getreten (BGBl. 1958 II S. 1), die Neufassung trat am 1. November 1993 in Kraft (BGBl. 1993 II S. 1947), Berichtigung der Übersetzung des EURATOM-Vertrages vom 13. Oktober 1999 (BGBl.II 1999, Nr. 31)

1F-2 Empfehlung 91/444/EURATOM der Kommission vom 26. Juli 1991 zur **Anwendung von Artikel 33** des EURATOM-Vertrages (ABl.EG 1991, Nr. L238)

1F-3 Empfehlung 2000/473/EURATOM der Kommission vom 8. Juni 2000 zur **Anwendung des Artikels 36** des EURATOM.Vertrages (ABl.EG 2000, Nr. L191)

1F-4 Empfehlung 1999/829/EURATOM der Kommission vom 6. Dezember 1999 zur **Anwendung des Artikels 37** des EURATOM-Vertrages (ABl.EG 1999, Nr. L324)

1F-5 Verordnung (EURATOM) 2587/1999 des Rates vom 2. Dezember 1999 zur Bestimmung der **Investitionsvorhaben**, die der Kommission **gemäß Artikel 41** des Vertrages zur Gründung der Europäischen Atomgemeinschaft **anzuzeigen** sind (ABl.EG 1999, Nr. L315), Durchführungsbestimmungen dazu Verordnung (EG) 1209/2000 (??), zuletzt geändert durch Verordnung (EURATOM) 1352/2003 der Kommission (ABl.EU 2003, Nr. L192)

1F-6 Bekanntmachung über die Meldung an die Behörden der Mitgliedsstaaten auf dem Gebiet der **Sicherungsmaßnahmen gemäß Artikel 79** Abs. 2 des EURATOM-Vertrages vom 19. August 1999 (BGBl.II 1999, S. 811)

1F-7 **Verifikationsabkommen** siehe [1E-10]

1F-8 Verordnung (EURATOM) 3227/76 der Kommission vom 19. Oktober 1976 zur Anwendung der Bestimmungen der **EURATOM-Sicherungsmaßnahmen** (ABl.EG 1976, Nr. L363), geändert durch Verordnung EURATOM 2130/93 der Kommission vom 27. Juli 1993 (ABl.EG 1993, Nr. L191)

1F-9 Abkommen über die Zusammenarbeit zwischen der **EURATOM und** der internationalen Arbeitsorganisation (**ILO**) vom 26. Januar 1961 (ABl.EG 1961, Nr. L18)

1F-10 Abkommen über die Zusammenarbeit zwischen der **EURATOM und** der Internationalen Atomenergie-Organisation (**IAEO**) vom 1. Dezember 1975 (ABl.EG 1975, Nr. L329)

1F-11 Agreement for Co-operation in the Peaceful Uses of Nuclear Energy between EURATOM and the United States of America, signed on March 29, 1996 (ABl.EG 1996, Nr. L120) in Kraft seit 12. April 1996

1F-12 Richtlinie 85/337/EWG des Rates vom 27. Juni 1985 über die **Umweltverträglichkeitsprüfung** bei bestimmten öffentlichen und privaten Projekten (Abl.EG 1985, Nr. L175), geändert durch die Richtlinie 97/11/EG des Rates vom 3. März 1997 (ABl.EG 1997, Nr. L73)

1F-14 Richtlinie 90/313/EWG des Rates vom 7. Juni 1990 über den **freien Zugang zu Informationen über die Umwelt** (ABl.EG 1990, Nr. L158)

1F-15 Richtlinie 98/34/EG des Europäischen Parlaments und des Rates vom 22. Juni 1998 über ein **Informationsverfahren** auf dem Gebiet der Normen und **technischen Vorschriften** (ABl.EG 1998, Nr. L204)

1F-16 Richtlinie 98/37/EG des Europäischen Parlaments und des Rates vom 22. Juni 1998 zur **Angleichung der Rechts- und Verwaltungsvorschriften** der Mitgliedstaaten für Maschinen (ABl.EG 1998, Nr. L207)

Strahlenschutz

1F-18 Richtlinien des Rates, mit denen die Grundnormen für den Gesundheitsschutz der Bevölkerung und der Arbeitskräfte gegen die Gefahren ionisierender Strahlungen festgelegt wurden (**EURATOM-Grundnormen**)

- Richtlinie vom 2. Februar 1959 (ABl.EG 1959, Nr. 11),
- Richtlinie vom 5. März 1962 (ABl.EG 1962, S. 1633/62),
- Richtlinie 66/45/EURATOM (ABl.EG 1966, Nr. 216),
- Richtlinie 76/579/EURATOM vom 1.6.1976 (ABl.EG 1976, Nr. L187),
- Richtlinie 79/343/EURATOM vom 27.3.1977 (ABl.EG 1979, Nr. L83),
- Richtlinie 80/836/EURATOM vom 15.7.1980 (ABl.EG 1980, Nr. L246),
- Richtlinie 84/467/EURATOM vom 3.9.1984 (ABl.EG 1984, Nr. L265),

- Neufassung mit Berücksichtigung der ICRP 60 in Richtlinie 96/29/EURATOM vom 13. Mai 1996 (ABl.EG 1996, Nr. L159)

1F-20 Richtlinie 90/641/EURATOM des Rates vom 4. Dezember 1990 über den **Schutz externer Arbeitskräfte**, die einer Gefährdung durch ionisierende Strahlung bei Einsatz im **Kontrollbereich** ausgesetzt sind (ABl.EG 1990, Nr. L349)

1F-21 Richtlinie 94/33/EG des Rates vom 22. Juni 1994 über **Jugendarbeitsschutz** (ABl.EG 1994, Nr. L216)

1F-22 Richtlinie 2003/122/EURATOM des Rates vom 22. Dezember 2003 zur **Kontrolle hochradioaktiver Strahlenquellen** und **herrenloser Strahlenquellen** (ABl.EG 2003, Nr. L346)

Radiologische Notfälle

1F-29 Richtlinie 89/618/EURATOM des Rates vom 27. November 1989 über die **Unterrichtung der Bevölkerung** über die bei einer radiologischen Notstandssituation geltenden Verhaltensmaßregeln und zu ergreifenden Gesundheitsschutz-maßnahmen (ABl.EG 1989, Nr. L357)
- Mitteilung der Kommission betreffend die Durchführung der Richtlinie 89/618/EURATOM (ABl.EG 1991, Nr. C103)

1F-30 Verordnungen zur Festlegung von **Höchstwerten an Radioaktivität** in Nahrungsmitteln und Futtermitteln im Fall eines nuklearen Unfalls oder einer anderen radiologischen Notstandssituation:
- Ratsverordnung (EURATOM) 3954/87 vom 22.12.1987 (ABl.EG 1987, Nr. L371) geändert durch Ratsverordnung (EURATOM) 2218/89 vom 18.7.1989 (ABl.EG 1989, Nr. L211),
- Kommissionsverordnung (EURATOM) 944/89 vom 12.4.89 (ABl.EG 1989, Nr. L101),
- Kommissionsverordnung (EURATOM) 770/90 vom 29.3.1990 (ABl.EG 1990, Nr. L83)

1F-31 Ratsverordnung (EWG) 2219/89 vom 18.7.1989 über **besondere Bedingungen für die Ausfuhr** von Nahrungsmitteln und Futtermitteln im Falle eines **nuklearen Unfall**s oder einer anderen radiologischen Notstandssituation (ABl.EG 1989, Nr. L211)

1F-32 Ratsverordnung (EWG) 3955/87 vom 22. Dezember 1987 über die **Einfuhrbedingungen** für landwirtschaftliche Erzeugnisse mit Ursprung in Drittländern **nach dem Unfall** im Kernkraftwerk **Tschernobyl** (ABl.EG 1987, Nr. L371), mit diversen Verordnungen und Durchführungsbestimmungen

2 Allgemeine Verwaltungsvorschriften

2-1 Allgemeine Verwaltungsvorschrift zu § 45 Strahlenschutzverordnung: **Ermittlung der Strahlenexposition** durch die Ableitung radioaktiver Stoffe aus kerntechnischen Anlagen oder Einrichtungen vom 21. Februar 1990 (BAnz. 1990, Nr. 64a), in Überarbeitung

2-2 Allgemeine Verwaltungsvorschrift zu § 62 Abs. 2 Strahlenschutzverordnung (**AVV Strahlenpaß**) vom 3. Mai 1990 (BAnz. 1990, Nr. 94a), in Überarbeitung

2-3 Allgemeine Verwaltungsvorschrift zur Ausführung des Gesetzes über die **Umweltverträglichkeitsprüfung** (UVPVwV) vom 18. September 1995 (GMBl. 1995, Nr. 32)

2-4 Allgemeine Verwaltungsvorschrift zum Integrierten Meß- und Informationssytem nach dem **Strahlenschutzvorsorgegesetz** (AVV-IMIS) vom 27. September 1995 (BAnz. 1995, Nr. 200a)

3 Bekanntmachungen des Bundesumweltministeriums und des vormals zuständigen Bundesinnenministeriums

3-1 Sicherheitskriterien für Kernkraftwerke vom 21.10.1977 (BAnz. 1977, Nr. 206)

3-2 Richtlinie für den Fachkundenachweis von Kernkraftwerkspersonal vom 14.4.1993 (GMBl. 1993, Nr. 20)

3-3 Richtlinie für den Fachkundenachweis von Forschungsreaktorpersonal vom 16.2.1994 (GMBl. 1994, Nr. 11)

3-4 Richtlinien über die Anforderungen an Sicherheitsspezifikationen für Kernkraftwerke vom 27.4.1976 (GMBl. 1976, S. 199)

3-5 Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor vom 26.7.1976 (GMBl. 1976, S. 418)

3-6 Richtlinie für den Schutz von Kernkraftwerken gegen Druckwellen aus chemischen Reaktionen durch Auslegung der Kernkraftwerke hinsichtlich ihrer Festigkeit und induzierten Schwingungen sowie durch Sicherheitsabstände vom 13.9.1976 (BAnz. 1976, Nr. 179)

3-7-1 Zusammenstellung der in atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen (ZPI) vom 20.10.1982 (BAnz. 1983, Nr. 6a)
3-7-2 Zusammenstellung der zur bauaufsichtlichen Prüfung kerntechnischer Anlagen erforderlichen Unterlagen vom 6.11.1981 (GMBl. 1981, S. 518)

3-8 Grundsätze für die Vergabe von Unteraufträgen durch Sachverständige vom 29.10.1981 (GMBl. 1981, S. 517)

3-9-1 Grundsätze zur Dokumentation technischer Unterlagen durch Antragsteller /Genehmigungsinhaber bei Errichtung, Betrieb und Stillegung von Kernkraftwerken vom 19.2.1988 (BAnz. 1988, Nr. 56)
3-9-2 Anforderungen an die Dokumentation bei Kernkraftwerken vom 5.8.1982 (GMBl. 1982, S. 546)

3-12 Bewertungsdaten für Kernkraftwerksstandorte vom 11. Juni 1975 (Umwelt 1975, Nr. 43)

3-13 Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk vom 20.4.1983 (GMBl. 1983, S. 220), in Überarbeitung

3-15
1. Rahmenempfehlungen für den Katastrophenschutz in der Umgebung kerntechnischer Anlagen vom 9.8.1999 (GMBl. 1999, Nr. 28/29)
2. Radiologische Grundlagen für Entscheidungen über Maßnahmen zum Schutz der Bevölkerung bei unfallbedingten Freisetzungen von Radionukliden vom 9.8.1999 (GMBl. 1999, Nr. 28/29)

3-23 Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen (REI) vom 30.6.1993 (GMBl. 1993, Nr. 29) , in Überarbeitung
3-23-2 ergänzt um die Anhänge B und C vom 20.12.1995 (GMBl. 1996, Nr. 9/10), in Überarbeitung

3-24 Richtlinie über Dichtheitsprüfungen an umschlossenen radioaktiven Stoffen vom 20.8.1996 (GMBl. 1996, Nr. 35), in Überarbeitung

3-25 Grundsätze zur Entsorgungsvorsorge für Kernkraftwerke vom 19.3.1980 (BAnz. 1980, Nr. 58)

3-27 Richtlinie über die Gewährleistung der notwendigen Kenntnisse der beim Betrieb von Kernkraftwerken sonst tätigen Personen vom 30.11.2000 (GMBl. 2001, S. 153)

3-31 Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 27.12.1976 (GMBl. 1977, S. 48)

3-32 Änderung der Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 18.10.1977 (GMBl. 1977, S. 664)

3-33 Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV (Störfall-Leitlinien) vom 18.10.1983 (BAnz. 1983, Nr. 245a) Störfallberechnungsgrundlagen für die Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gemäß § 28 Abs. 3 StrlSchV vom 18.10.1983 (BAnz. 1983, Nr. 245a), Neufassung des Kapitels 4 "Berechnung der Strahlenexposition" vom 29. Juni 1994 (BAnz. 1994, Nr. 222a), in Überarbeitung
3-34 Rahmenrichtlinie über die Gestaltung von Sachverständigengutachten in atomrechtlichen Verwaltungsverfahren vom 15.12.1983 (GMBl. 1984, S. 21)

3-37-1 Empfehlung über den Regelungsinhalt von Bescheiden bezüglich der Ableitung radioaktiver Stoffe aus Kernkraftwerken mit Leichtwasserreaktor vom 8.8.1984 (GMBl. 1984, S. 327)

3-38 Richtlinie für Programme zur Erhaltung der Fachkunde des verantwortlichen Schichtpersonals in Kernkraftwerken vom 1.9.1993 (GMBl. 1993, Nr. 36)

3-39 Richtlinie für den Inhalt der Fachkundeprüfung des verantwortlichen Schichtpersonals in Kernkraftwerken vom 23.4.1996 (GMBl. 1996, Nr. 26)

3-40 Richtlinie über die Fachkunde im Strahlenschutz vom 17.9.1982 (GMBl. 1982, S. 592), in Überarbeitung

3-41 Richtlinie für das Verfahren zur Vorbereitung und Durchführung von Instandhaltungs- und Änderungsarbeiten in Kernkraftwerken vom 1.6.1978 (GMBl. 1978, S. 342), in Überarbeitung

3-42 Richtlinie für die physikalische Strahlenschutzkontrolle zur Ermittlung der Körperdosen (§§ 62, 63, 63a StrlSchV; §§35, 35a RöV) vom 20.12.1993 (GMBl. 1994, Nr. 7), in Überarbeitung

3-42-1 Richtlinie für die Ermittlung der Körperdosen bei innerer Strahlenexposition gemäß den §§ 63 und 63a der Strahlenschutzverordnung (Berechnungsgrundlage) vom 13. März 1997 (BAnz. 1997, Nr. 122a), in Überarbeitung

Richtlinie für den Strahlenschutz des Personals bei der Durchführung von Instandhaltungsarbeiten in Kernkraftwerken mit Leichtwasserreaktor;

3-43 Teil I: Die während der Planung der Anlage zu treffende Vorsorge vom 10.7.1978 (GMBl. 1978, S. 418), in Überarbeitung

3-43-1 Teil II: Die Strahlenschutzmaßnahmen während der Inbetriebsetzung und des Betriebs der Anlage vom 4.8.1981 (GMBl. 1981, S. 363), in Überarbeitung

3-44 Kontrolle der Eigenüberwachung radioaktiver Emissionen aus Kernkraftwerken vom 5.2.1996 (GMBl. 1996, Nr. 9/10)

3-49 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke; Einzelfehlerkonzept - Grundsätze für die Anwendung des Einzelfehlerkriteriums vom 2.3.1984 (GMBl. 1984, S. 208)

3-50 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 17.5.1979 (GMBl. 1979, S. 161)
zu Sicherheitskriterium 2.6: Einwirkungen von außen
zu Sicherheitskriterium 8.5: Wärmeabfuhr aus dem Sicherheitseinschluß

3-51 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 28.11.1979 (GMBl. 1980, S. 90)
zu Sicherheitskriterium 2.2: Prüfbarkeit
zu Sicherheitskriterium 2.3: Strahlenbelastung in der Umgebung
zu Sicherheitskriterium 2.6: Einwirkungen von außen
zu Sicherheitskriterium 2.7: Brand- und Explosionsschutz
ergänzende Interpretation zu Sicherheitskriterium 4.3:
Nachwärmeabfuhr nach Kühlmittelverlusten

3-52-2 Meldung meldepflichtiger Ereignisse in Anlagen zur Spaltung von Kernbrennstoffen (RS I 5 - 14009/13, Mai 1993)
- Erläuterungen zu den Meldekriterien für meldepflichtige Ereignisse in Anlagen zur Spaltung von Kernbrennstoffen (2/91 ersetzt durch die überarbeitete Fassung 12/97)
- Zusammenstellung der in den Meldekriterien verwendeten Begriffen (Anlagen zur Spaltung von Kernbrennstoffen) (2/91)
- Meldeformular (Anlagen zur Spaltung von Kernbrennstoffen) (12/92)

3-52-3 Erläuterungen zu den Meldekriterien für meldepflichtige Ereignisse in Anlagen, die nicht der Spaltung von Kernbrennstoffen dienen (1/97)
- Meldeformular (Anlagen die nicht der Spaltung von Kernbrennstoffen dienen) (12/92)

3-52-4 Meldung eines Befundes bzgl. Kontamination oder Dosisleistung bei der Beförderung von entleerten Brennelement-Behältern, Behältern mit bestrahlten Brennelementen und Behältern mit verglasten hochradioaktiven Spaltproduktlösungen (8/00)
- Meldeformular (Behälter) (7/00)

3-52-5 Erläuterungen zu den Meldekriterien für meldepflichtige Ereignisse in Anlagen zur Spaltung von Kernbrennstoffen - für die Anwendung in Forschungsreaktoren (11/92)

3-53 Richtlinie für den Inhalt der Fachkundeprüfung des verantwortlichen Schicht-personals in Forschungsreaktoren vom 14. November 1997 (GMBl. 1997, Nr. 42)

3-54 Rahmenempfehlung für die Fernüberwachung von Kernkraftwerken vom 6.10.1980 (GMBl. 1980, S 577)

3-54-1 Empfehlung zur Berechnung der Gebühr nach § 5 AtKostV für die Fernüberwachung von Kernkraftwerken (KFÜ) vom 21.1.1983 (GMBl. 1983, S. 146)

3-57 Anforderungen an den Objektsicherungsdienst und an Objektsicherungsbeauftragte in kerntechnischen Anlagen der Sicherungskategorie I vom 8.4.1986 (GMBl. 1986, S. 242)

3-57-3 Richtlinie für den Schutz von Kernkraftwerken mit Leichtwasserreaktoren gegen Störmaßnahmen oder sonstige Einwirkungen Dritter vom 6.12.1995 (GMBl. 1996, Nr. 2, ohne Wortlaut)

3-59 Richtlinie zur Kontrolle radioaktiver Abfälle mit vernachlässigbarer Wärmeentwicklung, die nicht an eine Landessammelstelle abgeliefert werden vom 16.1.1989 (BAnz. 1989, Nr. 63a), letzte Ergänzung vom 14.1.1994 (BAnz. 1994, Nr. 19), in Überarbeitung

3-61 Richtlinie für die Fachkunde von Strahlenschutzbeauftragten in Kernkraftwerken und sonstigen Anlagen zur Spaltung von Kernbrennstoffen vom 10.12.1990 (GMBl. 1991, S. 56)

3-62 Richtlinie über Maßnahmen für den Schutz von Anlagen des Kernbrennstoffkreislaufs und sonstigen kerntechnischen Einrichtungen gegen Störmaßnahmen oder sonstige Einwirkungen zugangsberechtigter Einzelpersonen vom 28.1.1991 (GMBl. 1991, S. 228)

3-65 Anforderungen an Lehrgänge zur Vermittlung kerntechnischer Grundlagenkenntnisse für verantwortliches Schichtpersonal in Kernkraftwerken - Anerkennungskriterien vom 10. Oktober 1994

3-66 Meldung an die Behörden der Mitgliedstaaten auf dem Gebiet der Sicherungsmaßnahmen gemäß Artikel 79 Abs. 2 des EURATOM-Vertrages vom 12.8.1991 (BAnz. 1991, Nr. 158)

3-67 Richtlinie über Anforderungen an Personendosismeßstellen nach Strahlenschutz- und Röntgenverordnung vom 10.12.2001 (GMBl. 2002, Nr. 6)

Richtlinie für die Überwachung der Radioaktivität in der Umwelt nach dem Strahlenschutzvorsorgegesetz
3-69 Teil I: Meßprogramm für den Normalbetrieb (Routinemeßprogramm) vom 28. Juli 1994 (GMBl. 1994, Nr. 32), in Überarbeitung
3-69-2 Teil II: Meßprogramm für den Intensivbetrieb (Intensivmeßprogramm) vom 19. Januar 1995 (GMBl. 1995, Nr. 14), in Überarbeitung

3-71 Richtlinie für die Fachkunde von verantwortlichen Personen in Anlagen zur Herstellung von Brennelementen für Kernkraftwerke vom 30. November 1995 (GMBl. 1996, Nr. 2)

3-72 Richtlinie über Anforderungen an Inkorporationsmeßstellen vom 30. September 1996 (GMBl. 1996, Nr. 46), in Überarbeitung

3-73 Leitfaden zur Stillegung von Anlagen nach § 7 des Atomgesetzes vom 14. Juni 1996 (BAnz. 1996, Nr. 211a), in Überarbeitung

Leitfäden zur Durchführung von Periodischen Sicherheitsüberprüfungen (PSÜ) für Kernkraftwerke in der Bundesrepublik Deutschland, in Überarbeitung
3-74-1 - Grundlagen zur Periodischen Sicherheitsprüfung für Kernkraftwerke
- Leitfaden Sicherheitsstatusanalyse
- Leitfaden Probabilistische Sicherheitsanalyse
Bekanntmachung vom 18. August 1997 (BAnz. 1997, Nr. 232a)
3-74-2 - Leitfaden Deterministische Sicherungsanalyse
Bekanntmachung vom 25. Juni 1998 (BAnz. 1998, Nr. 153)

3-79 Schadensvorsorge außerhalb der Auslegungsstörfälle, RdSchr. des BMU vom 15. Juli 2003

4 Empfehlungen der RSK

4-1 RSK-Leitlinien für Druckwasserreaktoren, 3. Ausgabe vom 14. Oktober 1981 (BAnz. 1982, Nr. 69a) mit den Änderungen in Abschn. 21.1 (BAnz. 1984, Nr. 104), in Abschn. 21.2 (BAnz. 1983, Nr. 106) und in Abschn. 7 (BAnz. 1996, Nr. 158a) mit Berichtigung (BAnz. 1996, Nr. 214) und den Anhängen vom 25. April 1979 zu Kapitel 4.2 der 2. Ausgabe der RSK-LL vom 24. Januar 1979 (BAnz. 1979, Nr. 167a)
Anhang 1: Auflistung der Systeme und Komponenten, auf die die Rahmenspezifikation Basissicherheit von druckführenden Komponenten anzuwenden ist
Anhang 2: Rahmenspezifikation Basissicherheit; Basissicherheit von druckführenden Komponenten: Behälter, Apparate, Rohrleitungen, Pumpen und Armaturen (ausgenommen: Einbauteile, Bauteile zur Kraftübertragung und druckführende Wandungen < DN 50)

5 Regeln des Kerntechnischen Ausschusses (KTA) www.kta-gs.de

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
	1000 KTA-interne Verfahrensregeln					
	1100 Begriffe und Definitionen		-		-	-
	(KTA-Begriffesammlung KTA-GS-12) *(KTA-Collection of Definitions)*	1/04		6/91 1/96		
	1200 Allgemeines, Administration, Organisation *General, Administration, Organization*					
1201 *	Anforderungen an das Betriebshandbuch *Requirements for the Operating Manual*	6/98	172 a 15.09.98	2/78 3/81 12/85	-	+
1202	Anforderungen an das Prüfhandbuch *Requirements for the Testing Manual*	6/84	191 a 09.10.84	-	15.06.99	+
	1300 Radiologischer Arbeitsschutz *Radiological (aspects of) industrial safety*					
1301.1	Berücksichtigung des Strahlenschutzes der Arbeitskräfte bei Auslegung und Betrieb von Kernkraftwerken; Teil 1: Auslegung *Radiation Protection Considerations for Plant Personnel in the Design and Operation of Nuclear Power Plants;* *Part 1: Design*	11/84	40 a 27.02.85	-	15.06.99	+
1301.2	Berücksichtigung des Strahlenschutzes der Arbeitskräfte bei Auslegung und Betrieb von Kernkraftwerken; Teil 2: Betrieb *Radiation Protection Considerations for Plant Personnel in the Design and Operation of Nuclear Power Plants;* *Part 2: Operation*	6/89	158 a 24.08.89 Berichtigung 118 29.06.91	6/82	15.06.99	+
	1400 Qualitätssicherung *Quality Assurance*					
1401	Allgemeine Forderungen an die Qualitäts-sicherung *General Requirements Regarding Quality Assurance*	6/96	216 a 19.11.96	2/80 12/87	19.06.01	+
1404	Dokumentation beim Bau und Betrieb von Kernkraftwerken *Documentation During the Construction and Operation of Nuclear Power Plants*	6/01	235 a 15.12.01	6/89	-	+
1408.1	Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitäts-führende Komponenten in Kernkraftwerken; Teil 1: Eignungsprüfung *Quality Assurance for Weld Filler Materials and Weld Additives for Pressure and Activity Retaining System in Nuclear Power Plants;* *Part 1: Suitability Testing*	6/85	203 a 29.10.85	-	19.06.01	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
1408.2	Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitätsführende Komponenten in Kernkraftwerken; Teil 2: Herstellung *Quality Assurance for Weld Filler Materials and Weld Additives for Pressure and Activity Retaining System in Nuclear Power Plants; Part 2: Manufacturing*	6/85	203 a 29.10.85 Berichtigung 229 10.12.86	-	19.06.01	+
1408.3	Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitätsführende Komponenten in Kernkraftwerken; Teil 3: Verarbeitung *Quality Assurance for Weld Filler Materials and Weld Additives for Pressure and Activity Retaining System in Nuclear Power Plants; Part 3: Processing*	6/85	203 a 29.10.85	-	19.06.01	+
	1500 Strahlenschutz und Überwachung *Radiological Protection and Monitoring*					
1501 *	Ortsfestes System zur Überwachung von Ortsdosisleistungen innerhalb von Kernkraftwerken *Stationary System for Monitoring Area Dose Rates within Nuclear Power Plants*	6/91	7 a 11.01.92	10/77	11.06.96	-
1502.1 *	Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 1: Kernkraftwerke mit Leichtwasserreaktor *Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 1: Nuclear Power Plants with Light Water Reactors*	6/86	162 a 03.09.86 Berichtigung 195 15.10.88	-	11.06.96	+
(1502.2)	Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 2: Kernkraftwerke mit Hochtemperaturreaktor *Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 2: Nuclear Power Plants with High Temperature Reactors*	6/89	229 a 07.12.89	-	-	+
1503.1	Überwachung der Ableitung gasförmiger und an Schwebstoffen gebundener radioaktiver Stoffe; Teil 1: Überwachung der Ableitung radioaktiver Stoffe mit der Kaminfortluft bei bestimmungsgemäßem Betrieb *Monitoring and Assessing of the Discharge of Gaseous and Dispersed Particle Bound Radioactive Substances; Part 1: Monitoring and Assessing of the Stack Discharge of Radioactive Substances during Specified Normal Operation*	6/02	172 a 13.09.02	2/79 6/93	-	-

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
1503.2	Überwachung der Ableitung gasförmiger und aerosolgebundener radioaktiver Stoffe; Teil 1: Überwachung der Ableitung radioaktiver Stoffe mit der Kaminfortluft bei Störfällen *Monitoring and Assessing of the Discharge of Gaseous and Aerosolbound Radioactive Substances; Part 2: Monitoring and Assessing of the Stack Discharge of Radioactive Substances during Anticipated Operational Occurrences and Accident Conditions*	6/99	243 b 23.12.99	-	-	-
1503.3	Überwachung der Ableitung gasförmiger und aerosolgebundener radioaktiver Stoffe; Teil 1: Überwachung der nicht mit der Kaminluft abgeleiteten radioaktiven Stoffe *Monitoring and Assessing of the Discharge of Gaseous and Aerosolbound Radioactive Substances; Part 3: Monitoring and Assessing of Radioactive Substances not Discharged via the Stack*	6/99	243 b 23.12.99	-	-	-
1504	Überwachung der Ableitung radioaktiver Stoffe mit Wasser *Monitoring and Assessing of the Discharge of Radioactive Substances in Liquid Effluents*	6/94	238 a 20.12.94 Berichtigung 216 a 19.11.96	6/78	15.06.99	+
1505	Nachweis der Eignung von Strahlungsmesseinrichtungen *Verification of Suitability of Radiation Measuring Equipment*	11/03	26 a 07.02.04	-	-	-
1506 *	Messung der Ortsdosisleistung in Sperrbereichen von Kernkraftwerken *Measuring Local Dose Rates in Exclusion Areas of Nuclear Power Plants*	6/86	162 a 03.09.86 Berichtigung 229 10.12.86	-	11.06.96 1)	+
1507	Überwachung der Ableitungen gasförmiger, aerosolgebundener und flüssiger radioaktiver Stoffe bei Forschungsreaktoren *Monitoring the Discharge of Gaseous, Aerosol-bound and Liquid Radioactive Materials from Research Reactors*	6/98	172 a 15.09.98	3/84	-	-
1508 *	Instrumentierung zur Ermittlung der Ausbreitung radioaktiver Stoffe in der Atmosphäre *Instrumentation to Determine Atmospheric Diffusion of Radioactive Substances*	9/88	37 a 22.02.89	-	20.06.00	+
	2100 Gesamtanlage *Plant*					
2101.1	Brandschutz in Kernkraftwerken; Teil 1: Grundsätze des Brandschutzes *Fire Protection in Nuclear Power Plants; Part 1: Basic Principles*	12/00	106 a 09.06.01	12/85	-	+
2101.2	Brandschutz in Kernkraftwerken; Teil 2: Brandschutz an baulichen Anlagen *Fire Protection in Nuclear Power Plants; Part 2: Structural Components*	12/00	106 a 09.06.01	-	-	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
2101.3	Brandschutz in Kernkraftwerken; Teil 3: Brandschutz an maschinen- und elektrotechnischen Anlagen *Fire Protection in Nuclear Power Plants;* *Part 3: Mechanical and Electrical Components*	12/00	106 a 09.06.01	-	-	+
2103	Explosionsschutz in Kernkraftwerken mit Leichtwasserreaktoren (Allgemeine und fallbezogene Anforderungen) *Explosion Protection in Nuclear Power Plants with Light Water Reactors (General and Case-Related Requirements)*	6/00	231 a 08.12.00	6/89	-	+
	2200 Einwirkungen von außen *External Events*					
2201.1	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 1: Grundsätze *Design of Nuclear Power Plants against Seismic Events;* *Part 1: Principles*	6/90	20 a 30.01.91	6/75	20.06.00	+
2201.2	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 2: Baugrund *Design of Nuclear Power Plants against Seismic Events;* *Part 2: Subsurface Materials (Soil and Rock)*	6/90	20 a 30.01.91	11/82	20.06.00	+
2201.4	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 4: Anforderungen an Verfahren zum Nachweis der Erdbebensicherheit für maschinen- und elektrotechnische Anlagenteile *Design of Nuclear Power Plants against Seismic Events;* *Part 4: Requirements for Procedures for Verifying the Safety of Mechanical and Electrical Components against Earthquakes*	6/90	20 a 30.01.91 Berichtigung 115 25.06.96	-	20.06.00	+
2201.5	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 5: Seismische Instrumentierung *Design of Nuclear Power Plants against Seismic Events;* *Part 5: Seismic Instrumentation*	6/96	216 a 19.11.96	6/77 6/90	19.06.01	+
2201.6	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 6: Maßnahmen nach Erdbeben *Design of Nuclear Power Plants against Seismic Events;* *Part 6: Post-Seismic Measures*	6/92	36 a 23.02.93	-	18.06.02	+
2206	Auslegung von Kernkraftwerken gegen Blitzeinwirkungen *Design of Nuclear Power Plants against Lightning Effects*	6/00	159 a 24.08.00	6/92	-	-
2207 *	Schutz von Kernkraftwerken gegen Hochwasser *Flood Protection for Nuclear Power Plants*	6/92	36 a 23.02.93	6/82	-	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
	<u>2500 Bautechnik</u> *<u>Civil Engineering</u>*					
2501	Bauwerksabdichtungen von Kernkraftwerken *Waterproofing of Structures of Nuclear Power Plants*	6/02	172 a 13.09.02	9/88	-	-
2502	Mechanische Auslegung von Brenn-elementlagerbecken in Kernkraftwerken mit Leichtwasserreaktoren *Mechanical Design of Fuel Storage Pools in Nuclear Power Plants with Light Water Reactors*	6/90	20 a 30.01.91	-	20.06.00	+
	<u>3000 Systeme allgemein</u> *<u>Genaral Systems</u>*					
	<u>3100 Reaktorkern und Reaktorregelung</u> *<u>Reactor Core and Reactor Control</u>*					
3101.1 *	Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 1: Grundsätze der thermohydraulischen Auslegung *Design of Reactor Cores of Pressurized Water and Boiling Water Reactors;* *Part 1: Principles of Thermohydraulic Design*	2/80	92 20.05.80	-	20.06.00 2)	+
3101.2 *	Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 2: Neutronenphysikalische Anforderungen an Auslegung und Betrieb des Reaktorkerns und der angrenzenden Systeme *Design of Reactor Cores of Pressurized Water and Boiling Water Reactors;* *Part 2: Neutron-Physical Requirements for Design and Operation of the Reactor Core and Adjacent Systems*	12/87	44 a 04.03.88	-	10.06.97	+
(3102.1)	Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 1: Berechnung der Helium-Stoffwerte *Reactor Core Design for High Temperature Gas-Cooled Reactors;* *Part 1: Calculation of the Material Properties of Helium*	6/78	189 a 06.10.78	-	15.06.93	+
(3102.2)	Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 2: Wärmeübergang im Kugelhaufen *Reactor Core Design for High Temperature Gas-Cooled Reactors;* *Part 2: Heat Transfer in Spherical Fuel Elements*	6/83	194 14.10.83	-	15.06.93	+
(3102.3)	Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 3; Reibungsdruckverlust in Kugelhaufen *Reactor Core Design for High Temperature Gas-Cooled Reactors;* *Part 3: Loss of Pressure through Friction in Pebble Bed Cores*	3/81	136 a 28.07.81	-	15.06.93	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
(3102.4)	Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 4: Thermohydraulisches Berechnungsmodell für stationäre und quasistationäre Zustände im Kugelhaufen *Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 4: Thermohydraulic Analytical Model for Stationary and Quasi-Stationary Conditions in Pebble Bed Cores*	11/84	40 a 27.02.85 Berichtigung 124 07.07.89	-	15.06.93	+
(3102.5)	Auslegung der Reaktorkerne von gasgekühlten Hochtemperaturreaktoren; Teil 5: Systematische und statistische Fehler bei der thermohydraulischen Kernauslegung des Kugelhaufenreaktors *Reactor Core Design for High Temperature Gas-Cooled Reactors; Part 5: Systematic and Statistical Errors in the Thermohydraulic Core Design of the Pebble-Bed Reactor*	6/86	162 a 03.09.86	-	15.06.93	+
3103	Abschaltsysteme von Leichtwasserreaktoren *Shutdown Systems of Light Water Reactors*	3/84	145 a 04.08.84	-	15.06.99	+
3104	Ermittlung der Abschaltreaktivität *Determination of the Shutdown Reactivity*	10/79	19 a 29.01.80	-	15.06.99	+
	3200 Primär- und Sekundärkreis Primary and Secondary Circuits					
3201.1 *	Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 1: Werkstoffe und Erzeugnisformen *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 1: Materials and Product Forms*	6/98	170 a 11.09.98	2/79 11/82 6/90	-	+
3201.2 *	Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 2: Auslegung, Konstruktion und Berechnung *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 2: Design and Analysis*	6/96	216 a 19.11.96	10/80 3/84	-	+
3201.3 *	Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 3: Herstellung *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 3: Manufacture*	6/98	219 a 20.11.98	10/79 12/87	-	+
3201.4 *	Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 4: Wiederkehrende Prüfungen und Betriebsüberwachung *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 4: Inservice Inspections and Operational Monitoring*	6/99	200 a 22.10.99	6/82 6/90	-	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
3203	Überwachung des Bestrahlungsverhaltens von Werkstoffen der Reaktordruckbehälter von Leichtwasserreaktoren *Surveillance of the Irradiation Behaviour of Reactor Pressure Vessel Materials of LWR Facilities*	6/01	235 a 12.12.01	-	-	+
3204 *	Reaktordruckbehälter-Einbauten *Reactor Pressure Vessel Internals*	6/98	236 a 15.12.98 Berichtigung 129 13.07.00 136 22.07.00	3/84	-	-
3205.1	Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen; Teil 1: Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen für Primärkreis-komponenten in Leichtwasserreaktoren *Component Support Structures with Non-integral Connections;* *Part 1: Component Support Structures with Non-integral Connections for Components of the Reactor Coolant Pressure Boundary*	6/02	189 a 10.10.02	6/82 6/91	-	-
3205.2	Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen; Teil 2: Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen für druck- und aktivitätsführende Komponenten in Systemen außerhalb des Primärkreises *Component Support Structures with Non-integral Connections;* *Part 2: Component Support Structures with Non-Integral Connections for Pressure and Activity-Retaining Components in Systems Outside the Primary Circuit*	6/90	41 a 28.02.91	-	20.06.02	+
3205.3	Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen; Teil 3: Serienmäßige Standardhalterungen *Component Support Structures with Non-integral Connections;* *Part 3: Series-Production Standard Supports*	6/89	229 a 07.12.89 Berichtigung 111 17.06.94	-	15.06.99	+
3211.1	Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 1: Werkstoffe *Pressure and Activity Retaining Components of Systems outside the Reactor Coolant Pressure Boundary;* *Part 1: Materials*	6/00	194 a 14.10.00 Berichtigung 132 19.07.01	6/91	-	+
3211.2 *	Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 2: Auslegung, Konstruktion und Be-rechnung *Pressure and Activity Retaining Components of Systems outside the Reactor Coolant Pressure Boundary;* *Part 2: Design and Analysis*	6/92	165 a 03.09.93 Berichtigung 111 17.06.94	-	-	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
3211.3	Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 3: Herstellung *Pressure and Activity Retaining Components of Systems outside the Reactor Coolant Pressure Boundary; Part 3: Manufacture*	11/03	26 a	07.02.04	6/90	-	-
3211.4	Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 4: Wiederkehrende Prüfungen und Betriebsüberwachung *Pressure and Activity Retaining Components of Systems outside the Reactor Coolant Pressure Boundary; Part 4: Inservice Inspections and Operational Monitoring*	6/96	216 a	19.11.96	-	19.06.01	+
	<u>3300 Wärmeabfuhr</u> *<u>Heat Removal</u>*						
3301	Nachwärmeabfuhrsysteme von Leichtwasserreaktoren *Residual Heat Removal Systems of Light Water Reactors*	11/84	40 a	27.02.85	-	15.06.99 3)	+
3303	Wärmeabfuhrsysteme für Brennelementlagerbecken von Kernkraftwerken mit Leichtwasserreaktoren *Heat Removal Systems for Fuel Assembly Storage Pools in Nuclear Power Plants with Light Water Reactors*	6/90	41 a	28.02.91	-	20.06.00 2)	+
	<u>3400 Sicherheitseinschluß</u> *<u>Containment</u>*						
3401.1 *	Reaktorsicherheitsbehälter aus Stahl; Teil 1: Werkstoffe und Erzeugnisformen *Steel Containment Vessels; Part 1: Materials and Product Forms*	9/88	37 a	22.02.89	6/80 11/82	16.06.98	-
3401.2	Reaktorsicherheitsbehälter aus Stahl; Teil 2: Auslegung, Konstruktion und Berechnung *Steel Containment Vessels; Part 2: Analysis and Design*	6/85	203 a	29.10.85	6/80	20.06.00	+
3401.3 *	Reaktorsicherheitsbehälter aus Stahl; Teil 3: Herstellung *Steel Containment Vessels; Part 3: Manufacture*	11/86	44 a	05.03.87	10/79	10.06.97	+
3401.4	Reaktorsicherheitsbehälter aus Stahl; Teil 4: Wiederkehrende Prüfungen *Steel Containment Vessels; Part 4: Inservice Inspections*	6/91	7 a	11.01.92	3/81	19.06.01	-
3402	Schleusen am Reaktorsicherheitsbehälter von Kernkraftwerken - Personenschleusen *Air Locks Through the Containment Vessel of Nuclear Power Plants - Personnel Locks*	11/76	38	24.02.77	-	15.06.99	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
3403	Kabeldurchführungen im Reaktorsicherheits-behälter von Kernkraftwerken *Cable Penetrations through the Reactor Containment Vessel*	10/80	44 a 05.03.81	11/76	19.06.01	+
3404	Abschließung der den Reaktorsicherheits-behälter durchdringenden Rohrleitungen von Betriebssystemen im Falle einer Freisetzung von radioaktiven Stoffen in den Reaktorsicherheitsbehälter *Isolation of Operating System Pipes Penetrating the Containment Vessel in the Case of a Release of Radioactive Substances into the Containment Vessel*	9/88	37 a 22.02.89 Berichtigung 119 30.06.90	-	16.06.98	+
3405	Integrale Leckratenprüfung des Sicherheits-behälters mit der Absolutdruckmethode *Integral Leakage Rate Testing of the Containment Vessel with the Absolute Pressure Method*	2/79	133 a 20.07.79	-	15.06.99	+
3407	Rohrdurchführungen durch den Reaktor-sicherheitsbehälter *Pipe Penetrations through the Reactor Containment Vessel*	6/91	113 a 23.06.92	-	19.06.01	+
3409	Schleusen am Reaktorsicherheitsbehälter von Kernkraftwerken - Materialschleusen *Air-Locks for the Reactor Containment Vessel for Nuclear Power Plants - Material Locks*	6/79	137 26.07.79	-	15.06.99	+
3413	Ermittlung der Belastungen für die Auslegung des Volldrucksicherheitsbehälters gegen Störfälle innerhalb der Anlage *Determination of Loads for the Design of a Full Pressure Containment Vessel against Plant-Internal Incidents*	6/89	229 a 07.12.89	-	15.06.99	+
	3500 Instrumentierung und Reaktorschutz *Instrumentations and Reactor Protection*					
3501	Reaktorschutzsystem und Überwachungs-einrichtungen des Sicherheitssystems *Reactor Protection System and Monitoring Equipment of the Safety System*	6/85	203 a 29.10.85	3/77	20.06.00	+
3502	Störfallinstrumentierung *Incident Instrumentation*	6/99	243 b 23.12.99	11/82 11/84	-	-
3503 *	Typprüfung von elektrischen Baugruppen des Reaktorschutzsystems *Type Testing of Electrical Modules for the Reactor Protection System*	11/86	93 a 20.05.87	6/82	10.06.97	-
3504 *	Elektrische Antriebe des Sicherheits-systems in Kernkraftwerken *Electrical Drives of the Safety System in Nuclear Power Plants*	9/88	37 a 22.02.89	-	16.06.98	-

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
3505 *	Typprüfung von Meßwertgebern und Meßumformern des Reaktorschutzsystems *Type Testing of Measuring Transmitters and Transducers of the Reactor Protection System*	11/84	40 a	27.02.85	-	10.06.97	+
3506	Systemprüfung der leittechnischen Einrichtungen des Sicherheitssystems in Kernkraftwerken *Tests and Inspections of the Instrumentation and Control Equipment of the Safety System of Nuclear Power Plants*	11/84	40 a	27.02.85	-	18.06.02	+
3507	Werksprüfungen, Prüfungen nach Instandsetzung und Nachweis der Betriebsbewährung für leittechnische Einrichtungen des Sicherheitssystems *Factory Tests, Post-Repair Tests and Demonstration of Successful Service for the Instrumentation and Control Equipment of the Safety System*	6/02	27 a	08.02.03	11/86	-	-
	<u>3600 Aktivitätskontrolle und -führung</u> <u>Activity Control and Activity Management</u>						
3601 *	Lüftungstechnische Anlagen in Kernkraftwerken *Ventilation and Air Filtration Systems in Nuclear Power Plants*	6/90	41 a	28.02.91	-	13.06.95 4)	-
3602	Lagerung und Handhabung von Brennelementen und zugehörigen Einrichtungen in Kernkraftwerken mit Leichtwasserreaktoren *Storage and Handling of Nuclear Fuel Assemblies and Pertinent Equipment in Nuclear Power Plants with Light Water Reactors*	11/03	26 a	07.02.04	6/82 6/84 6/90		-
3603	Anlagen zur Behandlung von radioaktiv kontaminiertem Wasser in Kernkraftwerken *Facilities for Treating Radioactively Contaminated Water in Nuclear Power Plants*	6/91	7 a	11.01.92	2/80	19.06.01 4)	+
3604 *	Lagerung, Handhabung und innerbetrieblicher Transport radioaktiver Stoffe (mit Ausnahme von Brennelementen) in Kernkraftwerken *Storaging, Handling and On-Site Transportation of Radioactive Substances (other than Fuel Elements) in Nuclear Power Plants*	6/83	194	14.10.83	-	14.06.94	+
3605	Behandlung radioaktiv kontaminierter Gase in Kernkraftwerken mit Leichtwasserreaktoren *Treatment of Radioactively Contaminated Gases in Nuclear Power Plants with Light Water Reactors*	6/89	229 a	07.12.89	-	15.06.99	+

Regel-Nr. KTA	Titel	Letzte Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
	3700 Energie- und Medienversorgung *Energy and Media Supply*					
3701	Übergeordnete Anforderungen an die elektrische Energieversorgung in Kernkraftwerken *General Requirements for the Electrical Power Supply in Nuclear Power Plants*	6/99	243 b 23.12.99	3701.1 (6/78) 3701.2 (6/82) 6/97	-	+
3702	Notstromerzeugungsanlagen mit Dieselaggregaten in Kernkraftwerken *Emergency Power Generating Facilities with Diesel-Generator Units in Nuclear Power Plants*	6/00	159 a 24.08.00	3702.1 (6/88) 3702.2 (6/91)	-	-
3703	Notstromanlagen mit Batterien und Gleichrichtergeräten in Kernkraftwerken *Emergency Power Generating Facilities with Batteries and Rectifier Units in Nuclear Power Plants*	6/99	243 b 23.12.99	6/86	-	+
3704	Notstromanlagen mit Gleichstrom-Wechselstrom-Umformern in Kernkraftwerken *Emergency Power Facilities with Rotary Converters and Static Inverters in Nuclear Power Plants*	6/99	243 b 23.12.99	6/84	-	+
3705	Schaltanlagen, Transformatoren und Verteilungsnetze zur elektrischen Energieversorgung des Sicherheitssystems in Kernkraftwerken *Switchgear Facilities, Transformers and Distribution Networks for the Electrical Power Supply of the Safety System in Nuclear Power Plants*	6/99	243 b 23.12.99	9/88	-	+
3706	Sicherstellung des Erhalts der Kühlmittelverlust-Störfallfestigkeit von Komponenten der Elektro- und Leittechnik in Betrieb befindlicher Kernkraftwerke *Measures to Preserve Resistance of Electrical and I & C Components against Loss of Coolant Accident Conditions of Operating Nuclear Power Plants*	6/00	159 a 24.08.00	-	-	-
	3900 Systeme, sonstige *Other Systems*					
3901 *	Kommunikationsmittel für Kernkraftwerke *Communication Devices for Nuclear Power Plants*	3/81	136 a 28.07.81 Berichtigung 155 22.08.81	3/77	11.06.96	+
3902	Auslegung von Hebezeugen in Kernkraftwerken *Lifting Equipment in Nuclear Power Plants*	6/99	144 a 05.08.99	11/75 6/78 11/83 6/92	-	+
3903	Prüfung und Betrieb von Hebezeugen in Kernkraftwerken *Inspection, Testing and Operation of Lifting Equipment in Nuclear Power Plants*	6/99	144 a 05.08.99	11/82 6/93	-	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
3904 *	Warte, Notsteuerstelle und örtliche Leitstände in Kernkraftwerken *Control Room, Emergency Control Room and Local Control Stations in Nuclear Power Plants*	9/88	37 a	22.02.89	-	16.06.98	+
3905	Lastanschlagpunkte an Lasten in Kernkraftwerken *Load Attaching Points on Loads in Nuclear Power Plants*	6/99	200 a	22.10.99	6/94	-	+

* Standard in revision

( ) Safety standard related to high temperature reactors no longer included in the reaffirmation process according to sec. 5.2 of the procedural statutes.

1) This standard will be withdrawn after revision of KTA 1501.

2) The KTA decided during its 54th meeting to start a revision process for these standards as soon as there are draft standards issued for KTA BR 01 and KTA-BR 02.

3) The KTA issued on its 43th meeting "Instructions for the user of KTA 3301 (11/84)".

4) In this safety standard, the HTR (High-temperatrure-reactor)-related requirements were deleted.