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Environment, Nature Conservation
and Nuclear Safety

Convention on Nuclear Safety

Report by the Government of the
Federal Republic of Germany
for the Fourth Review Meeting in April 2008

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Abbreviations

AtG	*Atomgesetz* Atomic Energy Act
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
BMBF	*Bundesministerium für Bildung und Forschung* Federal Ministry for Education and Research
BMWi	*Bundesministerium für Wirtschaft und Technologie* Federal Ministry of Economics and Technology
BWR	Boiling Water Reactor
GRS	*Gesellschaft für Anlagen- und Reaktorsicherheit*
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
KTA	*Kerntechnischer Ausschuß* Nuclear Safety Standards Commission
LAA	*Länderausschuß für Atomkernenergie* *Länder* Committee for Nuclear Energy
OECD/NEA	Organisation for Economic Co-operation and Development/ Nuclear Energy Agency
PSR	PeriodicSafety Review
PWR	Pressurised Water Reactor
RSK	*Reaktor-Sicherheitskommission* Reactor Safety Commission
SSK	*Strahlenschutzkommission* Commission on Radiological Protection
SR	Safety Review
StrlSchV	*Strahlenschutzverordnung* Radiation Protection Ordinance
VGB	VGB Power Tech e. V., formerly “Technische Vereinigung der Großkraftwerksbetreiber“
WANO	World Association of Nuclear Operators

Introduction

The fourth German report under the Convention on Nuclear Safety was prepared jointly by the organisations in Germany which are concerned with the safety of nuclear power plants. These are the nuclear authorities of the Federation and the *Länder*, supported by their expert organisations, as well as the four power utilities which operate nuclear power plants in Germany.

The report by Germany as Contracting Party was approved by the Cabinet of Ministers of the Federal Government at its meeting on 24 October 2007.

General conditions for the use of nuclear energy in the Federal Republic of Germany

The "Act on the structured phase-out of the utilisation of nuclear energy for the commercial generation of electricity" of 22 April 2002 - which is based on the Agreement between the Federal Government and the power utilities of 14 June 2000 (signed on 11 June 2001) - created new basic conditions in Germany for the use of nuclear power. One of the purposes of the amended Atomic Energy Act (AtG) is now the structured phase-out of nuclear power instead of its promotion. The agreed starting point for a step-wise phase-out of the operation of the nuclear power plants is an average total operating lifetime of 32 years. The restriction to 32 years has no technical justification but is based on a political decision in conjunction with balancing of benefits and risks of nuclear power by the legislator.

The Agreement confirms that during the residual operating lifetimes, a high level of safety of the nuclear power plants according to international standards will continue to be ensured.

After the federal elections in 2005 in the course of forming the Federal Government, the parties forming the coalition stated in their coalition contract that there was a difference in opinion on the use of nuclear power. The above mentioned Agreement between the Federal Government and the power utilities of 11 June 2001 and the procedures contained therein as well as the associated rules provided in the amendment of the Atomic Energy Act therefore remain unchanged. The safe operation of the nuclear power plants has the highest priority for the ruling parties. It was decided to continue and expand research into the safe operation of the existing nuclear power plants. Furthermore, the coalition contract stipulates that federal and *Länder* governments work together on a basis of trust regarding nuclear regulatory supervision.

This trustful co-operation - added by the involvement of the plant operators - is mirrored in the process of the drafting of the 4th report under the Convention on Nuclear Safety.

Irrespective of the structured phase-out of the use of nuclear energy for commercial electricity generation, Germany is committed to its international obligations, especially to the fulfilment of its obligations under the Convention on Nuclear Safety.

Procedure upon the preparation of the report

Like the previous reports, this report has been conceived as a complete and closed representation and does therefore not merely confine itself to the changes since the Third Review Meeting. The individual chapters have been largely rewritten. Following the Guidelines regarding National Reports for the preparation of the report, additional information is provided at the end of each individual chapter about what developments there have been since the last report, what measures have been implemented and what measures are to be carried out in future. The structure and content of the 4th report follows the Articles of the

Convention and takes the guidelines for report preparation into account. The numbering of the chapters is in line with the numbering of the Articles of the Convention. Articles that do not contain any obligations of the Contracting Parties are not dealt with any further. Each obligation is commented on separately. As suggested in the Guidelines regarding National Reports, the information provided in the report is kept generic; plant-specific information is provided wherever this illustrates the fact that the obligations under the Convention have been fulfilled in specific cases. In the chapter on Article 6 and in Appendix 2, information on research reactors has been included - as already in the third report - even though research reactors are not nuclear installations as defined by the Convention.

To demonstrate that the obligations have been fulfilled, the relevant acts of law, ordinances and regulations pertaining to each Article of the Convention are indicated. There then follow explanations of how the essential safety requirements are fulfilled and what corresponding measures have been taken by the operators of the nuclear installations. Focal issues of this 4th national report are again the licensing procedure and regulatory supervision as well as the measures taken to ensure nuclear safety.

The Appendix contains a list of the nuclear power plants and research reactors that are presently in operation or decommissioned, a compilation of the design basis accidents and beyond-design-basis events to be taken into account in the safety review, an overview of safety-relevant features of the operating nuclear power plants (nuclear installations as defined by the Convention), itemised by type and construction line of the nuclear power plants, and a comprehensive list of the legal provisions, administrative regulations, rules and guidelines in the nuclear area that are relevant to the safety of the nuclear power plants according to the Convention and which are referred to in the report.

Those involved in the preparation of the German report have based their work on:

- the results of the Third and previous Review Meetings,
- the focal points of the questions that were posed to Germany as Contracting Party on the occasion of the Third Review Meeting,
- the results of the consultations within Country Group 5 of the Third Review Meeting, and
- the announcements made by Germany as Contracting Party at the previous Review Meetings.

The above items were used to derive major points for in-depth analysis, verification and discussion among those involved during report preparation.

The plant operators are responsible for statements of the plant operators. In all other respects, the responsibility rests with the Federal Republic of Germany, represented by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety.

Progress and Changes Since 2004

During the review period, deliberations continued on a possible reform of the nuclear administration structure in Germany. There are no plans to change the existing basic structure of nuclear administration - i.e. the distribution of functions among federal and *Länder* governments - in Germany.

The staffing situation at the regulatory body still needs to be improved considerably in view of the high and still increasing requirements to be met. This applies in particular to the federal nuclear authority and the subordinate Federal Office for Radiation Protection.

In the reporting period, external experts have drafted and publicly commented on "Safety Requirements for Nuclear Power Plants" under the leadership of the BMU. In addition, the General Committee decided in November 2006 to set up a federal/*Länder* Working Group "for preliminary work on a legal regulation or general administrative procedure". This Working Group is to consider in its work the current German nuclear regulations, the draft "Safety Requirements for Nuclear Power Plants", the WENRA Reference Levels and the current body of IAEA rules and is tasked with portraying the safety philosophy of the German nuclear power plants.

In the reporting period, business and management principles have undergone further development. The concepts worked out jointly by the plant operators regarding safety culture and its development and trend monitoring as well as concerning the optimisation of safety management have continued to be implemented in the individual plants. Apart from the organisational structure, process organisation has been documented in the form of process descriptions. General indicators have been introduced and developed plant-specifically. In many nuclear power plants, organisation was certified according to DIN EN ISO 9001:2000.

After intense discussions with the Federal Ministry for the Environment, the plant operators have put their concept to implement measures resulting from safety-relevant findings in concrete terms, especially for the case that such findings give rise to doubts about the ability to control a design basis accident.

Future Activities

The efficiency of the regulatory body in Germany is to be further developed and optimised on the basis of the existing competences at federal and *Länder* level. The competent federal and *Länder* authorities will collaborate closely to this end.

The IRRS Mission planned for 2008 shall in particular serve to review the staffing situation and competence of the nuclear authorities as well as the effective fulfilment of the tasks attributed to these authorities by law.

The management system of the nuclear authorities is developed further with consideration of the relevant IAEA Safety Standards and supplemented by processes of federation/*Länder* co-operation.

It is intended to bring the BMU project "Safety Requirements for Nuclear Power Plants" as well as the deliberations of the federal and *Länder* governments regarding a future nuclear ordinance or administrative procedure to an end.

Further co-ordination with the *Länder* of the action plan to implement the WENRA Reference Levels in Germany and the performance of the corresponding measures concerning the nuclear non-mandatory guidance instruments and their implementation in the nuclear power plants shall take place until 2010.

The plant operators intend to process and assess their experience with the enhanced safety culture and safety management procedure and will make appropriate improvements. The plant operators assured the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety to perform a status review on the implementation of the self-learning safety management system and to develop and implement necessary amendments and further improvements within one year.

Any experiences made and their relevance to safe plant operation will be classified by the authorities and discussed with the respective plant operators, especially with regard to any conclusions for supervisory instruments and practice. Further nuclear power plants are to be certified according to DIN EN ISO 9001:2000.

6 Existing Nuclear Installations

ARTICLE 6 EXISTING NUCLEAR INSTALLATIONS

Each Contracting Party shall take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention enters into force for that Contracting Party is reviewed as soon as possible. When necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible. The timing of the shut-down may take into account the whole energy context and possible alternatives as well as the social, environmental and economic impact.

Nuclear installations as defined by the Convention

In Germany, there are 18 nuclear installations as defined by the Convention. Currently, 17 nuclear power plant units are in operation at 12 different sites producing a total of 21,457 MWe. Appendix 1-1 presents an overview of the nuclear power plants in operation. Figure 6-1 shows the geographical location of the individual sites.

The Obrigheim nuclear power plant (KWO - 357 MWe PWR) is also still a nuclear installation as defined by the convention. It was shut down on 11 May 2005 after the legally allowed amount of electricity to be produced had been reached. On 21 December 2004, the plant operator filed an application for decommissioning. The fuel elements have been removed from the core but are still in the plant. The authority has not yet given its approval to a decommissioning programm.

For the Mülheim-Kärlich nuclear power plant (KMK - 1,302 MWe PWR), 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 licence was granted on 16 July 2004, another one on 23 February 2006. The fuel elements have completely been removed from the plant. Thus, the Mülheim-Kärlich nuclear power plant is no longer a nuclear installation as defined by the Convention.

The Stade nuclear power plant (KKS - 672 MWe PWR) was shut down for decommissioning on 14 November 2003. The licence for decommissioning and dismantling of the plant was granted on 7 September 2005, a second licence was granted on 15 February 2006. The last fuel elements were removed on 27 April 2005. Thus, the Stade nuclear power plant is no longer a nuclear installation as defined by the Convention.

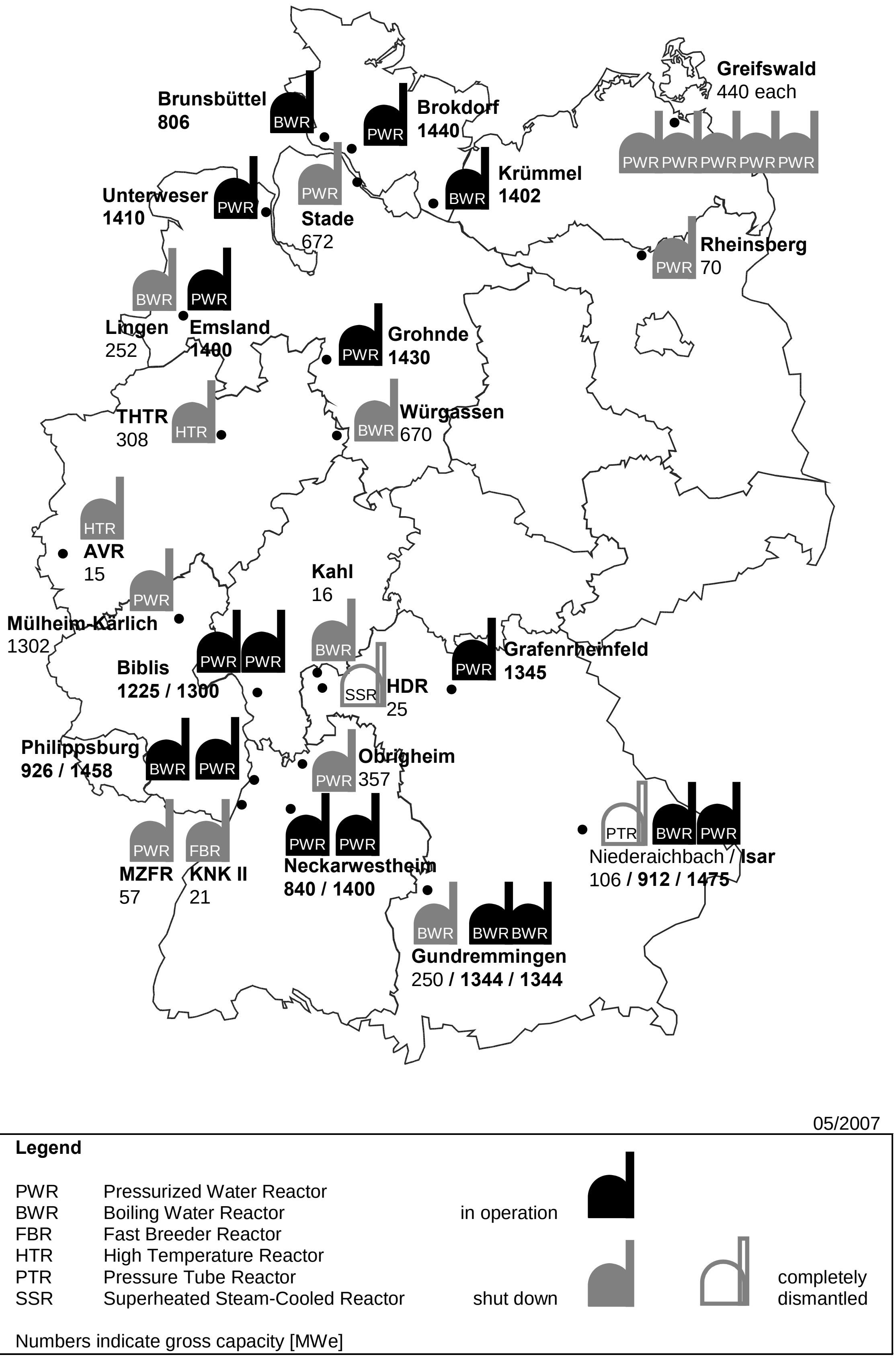


Figure 6-1 Nuclear Power Plants in Germany

According to the time of their construction, the nuclear power plants with pressurised water reactors can be classified according to four construction lines, whereas those with boiling water reactors belong to two different construction lines. The construction lines of the plants are noted in Appendix 1-1 and will be used throughout the report in the results presented. The two plants of the 1st construction line of pressurised water reactors have meanwhile been shut down. 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.

Operation of the nuclear installations as defined by the Convention

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 2006 (2005), the electricity generated by German nuclear power plants amounted to 167.4 (163.0) TWh.

Table 6-1 Average Availability of German Nuclear Power Plants

Year	Time availability %	Energy availability %	Energy utilisation %
2006	91.1	90.8	89.1
2005	88.8	88.0	86.3
2004	89.8	89.2	87.4
2003	87.7	87.0	84.3
2002	85.7	86.0	83.8

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* 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. Most of the nuclear power plants have already been issued licence permits to increase the initial enrichment of U-235 to values of up to 4.4 weight % and fissile plutonium in MOX fuel elements. It will then be possible to achieve a burn-up of more than 60 GWd per ton of heavy metal. In pressurised water reactors, this may require the use of boric acid enriched in B-10.

From 2004 to 2006, a total of 22 modification licences were granted for both units of the Biblis nuclear power plant which concerned safety-related improvements.

For four other nuclear plants, licences were granted for the increase of the initial enrichment of the fuel up to 4.4 weight % (Neckarwestheim 1, Philippsburg 1 and Philippsburg 2) and 4.77 weight % (Krümmel). Licences for the use of advanced mixed oxide fuel elements were granted for the two units in Gundremmingen.

For the Brokdorf nuclear power plant (KBR - 1,440 MWe PWR), a licence for increase of the thermal reactor power from 3,765 to 3,900 MWth was granted in May 2006. The electric power of the power plant will be 1,500 MWe.

For the nuclear power plants Isar 1, Philippsburg 1 and Brokdorf, the safety reviews required by law have been submitted.

Research for the safety of nuclear installations as defined by the Convention

For the Federal Government, the safe operation of the nuclear power plants has top priority. In this context, research for the safe operation of nuclear power plants is continued and extended.

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. The Federal Ministry of Economics and Technology currently provides approximately € 17 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.

Since 2004, institutional funding of nuclear safety and repository research by the Federal Ministry for Education and Research (BMBF) has been relatively constant with an amount of about € 31 millions per year. From the point of view of the Federal Government, the capability shall be maintained by research projects to judge the safety of nuclear power plants also in neighbouring countries. Further, international developments are watched to determine to which extent the objectives of further increased reactor safety, increased economical operation, proliferation resistance and reduction of radioactive wastes are actually achieved.

The plant operator also gives top priority to research and development in the field of nuclear safety. This circumstance is reflected, among others, in the budget of the ad-hoc committee on systems engineering (“Sonderausschuss Anlagentechnik”) used by the plant operators to fund joint research and development projects under the umbrella of the VGB. In the period from 2003 to the middle of 2006, more than € 20 millions were invested into a total of 243 projects within the framework of this programme existing since 1989. Focal points are, among others,

- materials science,
- systems and component engineering,
- accident analysis,
- non-destructive tests,
- PSA,
- fuel behaviour,
- radiation protection,
- issue of hydrogen, and

– seismic qualification.

Since 2001, research projects dealing with research on new and innovative reactor concepts have no longer been financed by public funds.

Research reactors

Here, account shall also be given on research reactors although they 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.

In Germany, four research reactors with a capacity of more than 50 kW thermal power and eight small training reactors are in operation (→ Appendix 2-1). Ten research reactors have been decommissioned and are being dismantled (→ Appendix 2-2), another 24 have already been dismantled completely (→ Appendix 2-3). Figure 6-2 shows the sites of research reactors with a capacity of more than 50 kW thermal power. The research reactors in operation have a capacity of up to 20 MW, and the FRM II reactor commissioned in 2004 reaches a maximum thermal neutron flux density of $8x10^{14}$ per cm² and s with a capacity of 20 MW.

In Germany, research reactors are licensed pursuant to the regulations for power reactors with application limitations according to the physical characteristics. The safety concept for the new FRM II, for example, meets the highest requirements which are reached by passive and active safety systems. Protection against earthquake and aircraft crash also was provided in the same way as for nuclear power plants. Further, research reactors with a capacity of more than 50 kW thermal power are, as the power reactors, also subject to, e.g. the obligations to report in case of reportable events (→ Article 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.

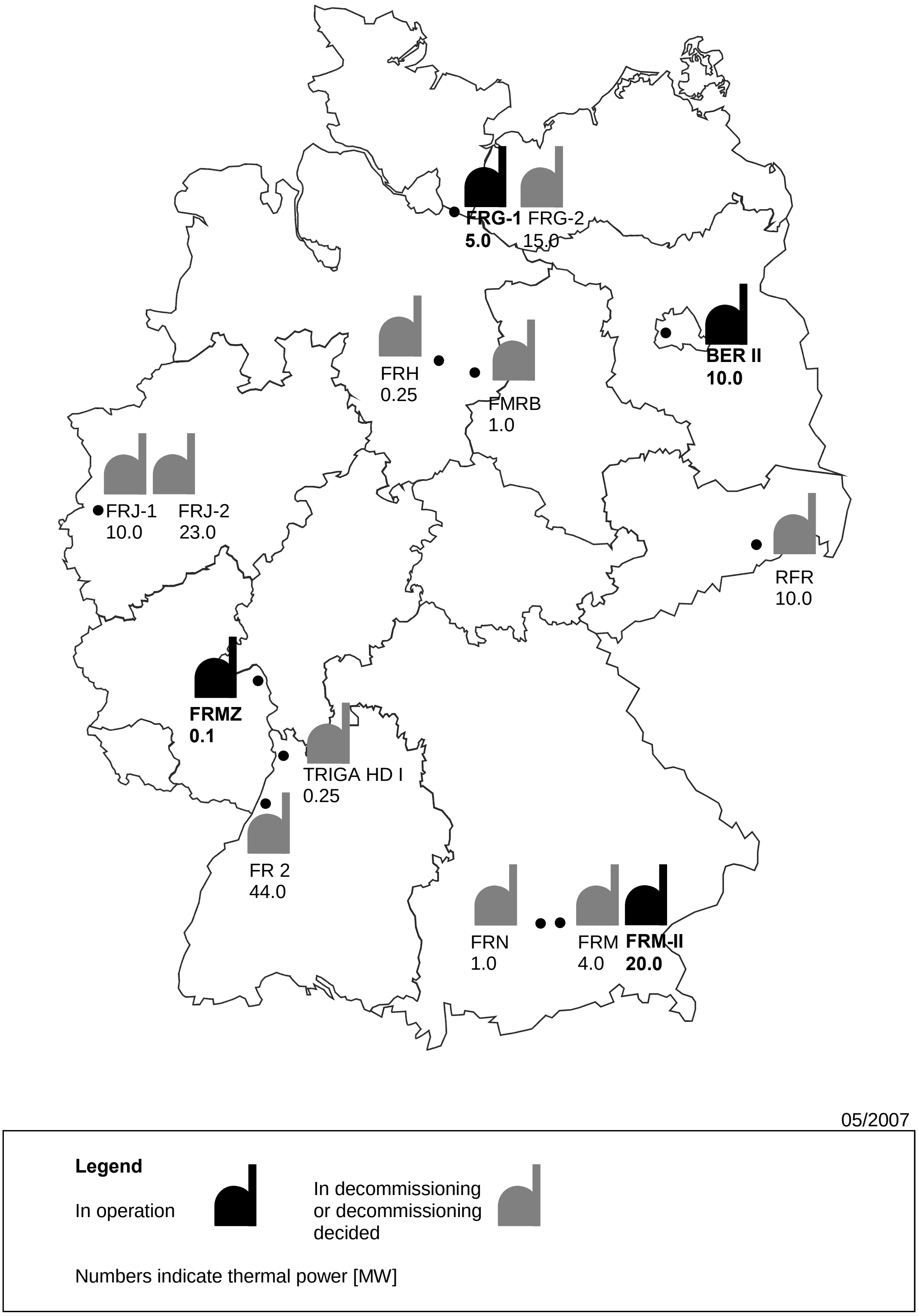


Figure 6-2 Research Reactors > 50 kW in Germany

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. However, some of these installations are subject to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management about which last report was given by Germany within the framework of the Review Meeting in May 2006.

Altogether, 19 nuclear power plants have been decommissioned (→ Appendix 1-2). From these, 15 nuclear power plants are currently being dismantled, two nuclear power plants are in safe enclosure and two plants have already been completely dismantled. The Obrigheim nuclear power plant has been shut down for decommissioning, but it is currently still a nuclear installation as defined by the Convention. Six other 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. Several fuel fabrication plants were decommissioned and have completely been dismantled. The facility for molybdenum production at the Rossendorf research site is in the decommissioning phase. 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. Central interim storage facilities and local interim storage facilities serve the purpose of interim storage of spent fuel elements (→ Article 19 (viii)). The licensing procedure for the pilot spent fuel conditioning plant in Gorleben (PKA) was completed in December 2000 with the granting of the third partial construction 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.

The Morsleben repository (ERAM) for low-level and medium-level radioactive waste with short half-lives was in operation until September 1998. The plan approval procedure for decommissioning has been initiated. The plan approval procedure for decommissioning the Schacht Konrad repository was ended with the plan approval decision which became final in April 2007. The exploration works in the Gorleben mine were interrupted in 2000; the duration of this moratorium shall be at least 3 and at most 10 years.

Safety review for nuclear installations as defined by the Convention

Even if the Atomic Energy Act limits the remaining electricity output still to be generated via the operating life of the plants, all currently operated nuclear power plants, as listed in Appendix 1-1, have an operating licence unlimited in time. These licences for nuclear power plants were granted after the applicant had proven to the nuclear licensing authority that the required protection against damage 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 (→ Article 7 (2ii)). These licensing prerequisite applies both to the licences granted after that and the licences still to be granted for major modifications of the plant itself or of its mode of operation. This way, each major modification, performed within the framework of the object of the modification procedure, results in a safety review and, where required, in an adaptation to necessary precautions against damage according to the state of the art in science and technology.

Further, the authority may revoke the licence if this licensing prerequisite is no longer fulfilled and cannot be fulfilled within a reasonable time.

Over the past years, numerous improvements have been realised at all nuclear power plants in the course of their operating lives, in particular also by measures in the area of beyond-design basis accidents, as presented in Table 6-2. Thus, safety precautions and risk prevention for the nuclear power plants were further developed in accordance with the progress in the state of the art in science and technology. Table 6-2 shows which improvements, for modern nuclear power plants already considered in the design, were implemented by backfitting measures at older plants.

The safety of the plant is continuously reviewed within the framework of regulatory supervision. In case of new safety-relevant findings, the necessity of improvements is determined. This also contributes to further developing plant safety.

Until 2002, periodic safety reviews (PSRs) were performed voluntarily or due to requirements specified in the licensing decisions to supplement the continuous regulatory supervision process. Since 2002, they have been made mandatory by the Atomic Energy Act (→ Article 14). Within the reporting period, safety reviews were submitted for the nuclear power plants Isar 1, Philippsburg 1 and Brokdorf.

In summary, the German Federal Government ascertains that - with the safety assessments for modification licences, within the framework of regulatory supervision and the safety reviews performed so far - reviews in terms of Article 6 of the Convention were performed and will also be performed in future. Necessary improvement measures were and will be performed, in particular on the basis of results of safety reviews.

Table 6-2 Major Backfitting Measures in Nuclear Power Plants According to Construction Line

	Objective of improvement Improvement measures	PWR construction line 1	2	3	4	BWR constr.line 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	●
	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.	**Improvement of safety during 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 ● already covered by the design

Article 6: Progress and Changes Since 2004

The nuclear power plants Mülheim-Kärlich (KMK) and Stade (KKS) do no longer present nuclear installations as defined by the Convention. The Obrigheim nuclear power plant (KWO - 357 MWe PWR) was shut down for decommissioning on 11 May 2005 after the legally allowed amount of electricity to be produced had been reached.

In the period from 2004 to 2006, licences were granted for six nuclear power plants to use advanced fuel elements or such with increased enrichment. For the Brokdorf nuclear power plant, a licence was granted to increase the thermal power. A total of 22 modification licences were granted for both units of the Biblis nuclear power plant which concerned safety-related improvements.

For the nuclear power plants Isar 1, Philippsburg 1 and Brokdorf, the safety reviews required by law have been submitted.

Improvements were realised that mainly concerned the increase of effectiveness and reliability of specified normal operation and of safety systems and equipment, the control of specific emergency situations, the mitigation of fire consequences, the improvement of barriers and accident management. The reviews and measures were performed in accordance of the respective state of the art in science and technology.

Article 6: Future Activities

The safety assessments are continued as it is common practice within the framework of licensing and supervision, including the mandatory safety reviews.

In order to be able to make use of findings and technological developments due to the progress in the state of the art for the best possible prevention against damage and risks also in future, research for the safe operation of nuclear power plants shall be continued and extended.

7 Legislative and Regulatory Framework

ARTICLE 7 LEGISLATIVE AND REGULATORY FRAMEWORK

1. Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.

2. The legislative and regulatory framework shall provide for:

   i) the establishment of applicable national safety requirements and regulations;

   ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a licence:

   iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licences;

   iv) the enforcement of applicable regulations and of the terms of licences, including suspension, modification or revocation.

7 (1) Legislative and Regulatory Framework

Framework requirements due to the federal structure of the Federal Republic of Germany

The Republic of Germany is a federal state. Responsibilities for legislation and execution are assigned to the organs of the Federation and the *Länder* according to their scope of functions. Specifications are given by provisions of the Basic Law [1A-1] of the Federal Republic of Germany.

The Federal Government has the legislative competence for the peaceful use of nuclear energy. So far, the Federal Government had concurrent legislative power in areas concerning the Convention. The Federal Government made extensive use of this concurrent legislation and thus excluded the *Länder* from legislation. After entry into force of the federalism reform on 1 September 2006, the Federal Government now has the exclusive legislative power for the above-mentioned matter according to Article 73 (1) 14 of the Basic Law. Therefore, there must be no need for a federal rule - as it was required so far - that the Federal Government has the right to legislate on these matters. According to Articles 87c and 85 of the Basic Law, the Atomic Energy Act is executed as before - with some exceptions - by the *Länder* on behalf of the Federal Government. In this respect, the *Länder* authorities are under the supervision of the Federation with regard to the lawfulness and expediency of their actions.

> ***Article 85***
> ***[Implementation by the Länder for the Federation (federal executive administration)]***
>
> ***(1)*** *Where the Länder implement federal legislation for the Federation, the establishment of the authorities shall remain their concern except in so far as federal legislation with the consent of the Bundesrat (Federal Council) provides otherwise.*

> ***(2)*** *The Federal Government may, with the consent of the Bundesrat, issue general administrative rules. It may provide for the uniform training of civil servants and other public employees. The heads of intermediate authorities shall be appointed with its approval.*
>
> ***(3)*** *The Land authorities shall comply with directives from the supreme federal authorities concerned. Such directives shall be addressed to the supreme Land authorities unless the Federal Government deems the matter urgent. Compliance with directives shall be ensured by the supreme Land authorities.*
>
> ***(4)*** *Federal supervision shall relate to the legality and expediency of implementation. For this purpose the Federal Government may call for reports and documents and the submission of files and send commissioners to any authority.*

The competent supervisory and licensing authorities report to the Federation on law enforcement. The Federation has the right to request additional information and reports, the right to full access to files and may, in the individual case, issue binding directives to the *Land* authority. The Federation may assume the competence for the subject matter, i.e. the decision in the cause, by exercising its right to issue directives. However, the competence to execute the duties, i.e. the execution of the decision towards the applicant or licensee, remains with the competent *Land* authority.

Within the framework of nuclear procedures, other legal regulations, such as immission control act, water law and construction law, also have to be considered. Legal regulations on assessing the environmental impact usually are part of the nuclear licensing procedure.

In Germany, decisions of the public administration, so-called administrative acts, can be appealed before the administrative courts by the party concerned, e.g. by applicants and licensees and also by concerned third parties of the public (guarantee of recourse to the courts according to Article 19 (4) of the Basic Law). An action is brought against that authority which issued the notice/administrative act, i.e. the competent Land authority. This also applies to the case that the *Land* took a decision due to a directives issued by the Federation. The parties concerned may also take legal actions in case of failure of the authorities to act. So, e.g. the plant operators may claim for granting of licences applied for or the residents for cessation of the operation of a nuclear installation.

In Germany, the legislation and its execution must also take into account any binding requirement from regulations of the European Communities. With respect to radiation protection, these are, among others, the EURATOM Basic Safety Standards 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.

Further development of the nuclear law and legal provision, general administrative provisions and guidelines issued on its basis lies within the responsibility of the Federation.

7 (2i) Nuclear Safety Regulations

National nuclear safety regulations

<u>Acts, ordinances and administrative provisions</u>

Figure 7-1 presents the hierarchy of the national rules and regulations, the authority or institution issuing the regulation and their degree of bindingness.

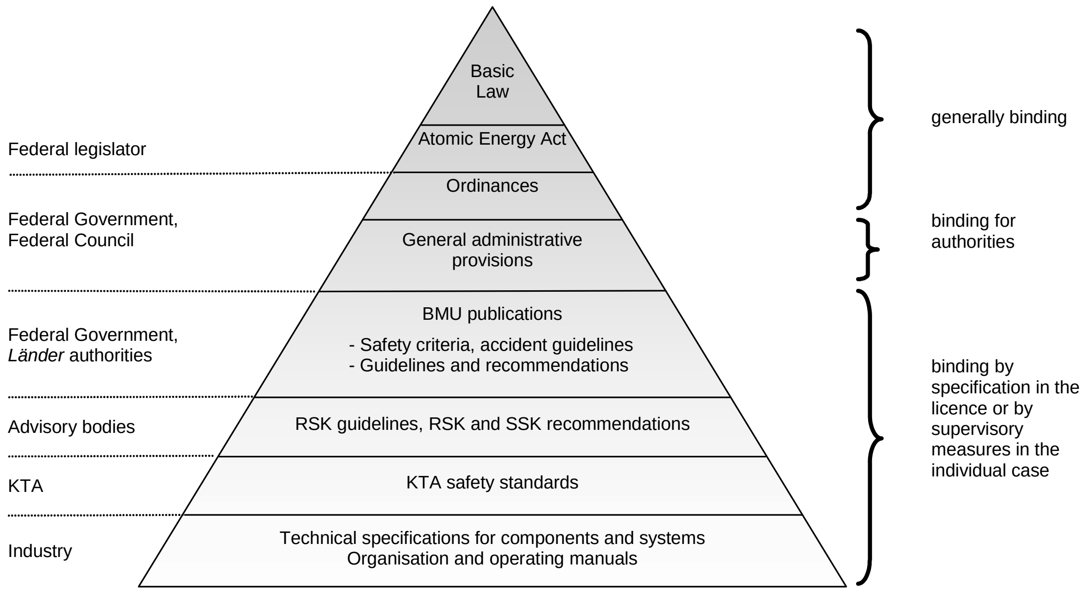


Figure 7-1 Regulatory Pyramid

Basic Law

The Basic Law [1A-1] includes provisions on the competencies of the Federation and the *Länder* regarding the use of nuclear energy (→ Article 8). It established fundamental principles that are also applicable to the nuclear law. With the basic rights, in particular the right to life and physical integrity, it determines the standard to be applied to the protective and preventive measures at nuclear power plants which is further specified in the above hierarchy levels of the pyramid. The principle of proportionality and guaranty of property, laid down in the Basic Law, must also be considered.

Atomic Energy Act

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 amended several times. The purpose of the Atomic Energy Act after the amendment of 2002 is to end the use of nuclear energy for the commercial production of electricity 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.

The Atomic Energy Act includes the general national regulations for protective and preventive measures, radiation protection, disposal of radioactive waste and irradiated fuel elements in Germany and is the basis for the associated ordinances.

Further to purpose and general provisions, the Atomic Energy Act also comprises surveillance regulations, general regulations on competencies of the administrative authorities, liability provisions and provisions on the payment of fines.

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. Prerequisites and procedures for licensing and performance of supervision are specified, including the regulations for consulting experts (Section 20 of the Atomic Energy Act) and charging of costs (Section 21 of the Atomic Energy Act).

However, most of the regulations laid down there are not exhaustive and are further specified both regarding the procedures and the substantive legal requirements by ordinances and non-legally binding regulatory guidance instruments.

According to Section 7 of the Atomic Energy 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 licensing prerequisites stated in Section 7 para 2 of the Atomic Energy Act are fulfilled, i.e. if

- the necessary precautions against damage have been taken in the light of the state of the art in science and technology,
- trustworthiness and technical qualification of the responsible personnel is given,
- it is assured that the persons who are otherwise engaged in the operation of the installation have the necessary knowledge concerning the safe operation of the installation, the possible hazards and the protective measures to be taken,
- the necessary protection has been provided against disruptive action or other interference by third parties,
- the necessary financial security has been provided to comply with the legal liability to pay compensation for damage, and
- the choice of the site of the installation does not conflict with overriding public interests, in particular in view of its environmental impacts.

These requirements for the licensing of nuclear power plants are also assessment criteria for supervision during operation. Today, they are only relevant as licensing prerequisite for modifications or the decommissioning of existing plants, since Section 7 para 1 sentence 2 of the Atomic Energy Act stipulates that no further licences will be issued for the construction and operation of nuclear power plants and reprocessing facilities.

The undefined legal terms used by the legislator, such as the “the necessary precautions in the light of the state of the art in science and technology“, were chosen to facilitate a dynamic further development of the precautions according to the latest state of the art. Thus, legislation largely left it to the executive - be it by way of ordinances according to the relevant authorisations, be it in case of individual decisions also under consideration of the non-legally binding regulatory guidance instruments - to decide on the kind and, in particular, the extent of risks to be accepted or not to be accepted. The Atomic Energy Act does not include specific regulations about the procedure for the assessment of such risks.

In addition to the Atomic Energy Act, the Radiation Precautionary Act [1A-5] of 1986, which came about in the wake of the reactor accident at Chernobyl, specifies the tasks of environmental monitoring also in the case of events with significant radiological effects.

Another legal basis to be mentioned is the “Act on the Establishment of a Federal Office for Radiation Protection“ [1A-22] by which certain tasks regarding the safety of nuclear power plants are delegated to this office to support the nuclear federal authorities.

Ordinances

For more details regarding the legal regulations, the Atomic Energy Act includes authorisations for issuing ordinances. These ordinances require approval by the *Bundesrat* (Federal Council). The *Bundesrat* is a constitutional body of the Federation in which the governments of the *Länder* are represented.

Table 7-1 presents the current ordinances on protective and preventive measures.

In Germany, regulations on the technical plant safety or safe operation, publications of operating experience for the purpose of safety improvements, requirements on training and technical qualification or on various protective measures have not been specified in ordinances so far - although possible according to the Atomic Energy Act - and are subject of the non-legally binding regulatory guidance instruments.

General administrative provisions

Ordinances may include additional authorisations for issuing general administrative provisions. Such regulate the actions of the authorities, but they only have a direct binding effect for the administration. They have an indirect effect since they are considered in the administrative decisions.

In the nuclear sector, there are general administrative provisions relevant to

- the calculation of radiation exposure during specified normal operation of nuclear power plants [2-1],
- the radiation passport [2-2],
- the environmental impact assessment [2-3]
- the environmental monitoring [2-4], and
- the monitoring of foodstuffs and feedingstuffs [2-5], [2-6].

Table 7-1 Ordinances on Protective and Preventive Measures at Nuclear Power Plants

	Brief description on the legislative content	[Ref.]
StrlSchV	**Radiation Protection Ordinance** Principles and limits of radiation protection, requirements on organisation of radiation protection, personal monitoring, environmental monitoring, accident management, design against incidents and accident planning values	1A-08
AtVfV	**Nuclear Licensing Procedure Ordinance** Application documents (one safety analysis report), involvement of the public, safety specifications (operational limits and conditions for safe operation), procedures and criteria for major modifications (public participation)	1A-10
AtSMV	**Nuclear Safety Officer and Reporting Ordinance** Position, duties, responsibilities of the nuclear safety officer, reporting of special events in nuclear installations	1A-17
AtZüV	**Nuclear Reliability Assessment Ordinance** Checking of personal reliability for protecting against the diversion or major release of radioactive material	1A-19
AtDeckV	**Nuclear Financial Security Ordinance** Financial security pursuant to the Atomic Energy Act	1A-11
AtKostV	**Cost Ordinance under the Atomic Energy Act** Fees and costs in nuclear procedures	1A-21
KIV	**Ordinance Concerning Potassium Iodide Tablets** Provision and distribution of medicine containing potassium iodide as thyroid blocker in case of radiological events	1A-20
AtAV	**Nuclear Waste Transfer Ordinance** Transfer of radioactive wastes into or out of the territory of the Federal Republic of Germany	1A-18
Endlager VlV	**Repository Prepayment Ordinance** Advance payments for the construction of radioactive waste disposal facilities of the Federation for the long-term engineered storage and disposal of radioactive waste	1A-13

<u>Regulatory guidelines published by BMU</u>

After having consulted the *Länder*, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) prepares regulatory guidelines. These are, among others, safety criteria, accident and other guidelines and recommendations. In general, these are regulations passed by consensus with the competent licensing and supervisory authorities of the *Länder* on the uniform application of the Atomic Energy Act. The recommendations of the BMU, however, describe its view 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. The regulatory guidelines are not binding for the *Länder* authorities in contrast to the general administrative provisions. Their relevance is also given by the right of the BMU to issue binding individual directives for particular cases to the *Länder* authorities. On two important regulatory guidelines - the above mentioned safety criteria and the accident guidelines - Section 49 of the Radiation Protection Ordinance explicitly stipulates that the licensing authorities may apply these two guidelines if they have to judge whether the design of a nuclear power plant against accidents complies with the criteria of Section 7 of the Atomic Energy Act.

Currently, about 60 BMU regulatory guidelines exist in the field of nuclear technology (→ Appendix 5 under “3 Regulatory Guidelines Published by BMU and the Formerly Competent Ministry of the Interior “).

These are regulations pertaining to

- general safety requirements for nuclear power plants (”safety criteria”),
- details on the design basis accidents to be considered in the design of pressurised water reactors (since 1982 for the last three nuclear power plants built of construction line 4),
- dispersion calculations,
- accident management measures to be planned by the plant operators 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,
- reporting criteria for reportable events at nuclear power plants and research reactors,
- monitoring of emissions and radioactivity in the environment,
- periodic safety reviews for nuclear power plants,
- technical documents to be prepared regarding construction, operation and decommissioning of nuclear power plants,
- documents to be supplied with the application for a licence, and
- qualification of the personnel in nuclear installations.

<u>Other rules and regulations on the safety of nuclear power plants</u>

Recommendations of the RSK and SSK, RSK guidelines

The BMU requests the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK) for advice on important issues related to licensing and supervisory procedures, development of rules and regulations or safety research. In addition, the commissions may also give advice on their own initiative. Depending on the issues to be discussed, *Länder* authorities, plant operators or the industry also participate in the discussions. The results of these discussions are statements or recommendations for the BMU. After own verification, the BMU implements the results in the respectively appropriate manner.

A special role play the so-called RSK guidelines [4-1]. In the last version of these guidelines of 1996, the RSK compiled the fundamental safety requirements for nuclear power plants with pressurised water reactors. The RSK uses these guidelines as a basis of its advisory work and recommendations. The RSK deviates from them if the state of the art in science and technology has meanwhile changed in specific areas.

The nuclear licensing authorities of the *Länder* have taken the RSK guidelines as an assessment basis of the non-legally binding regulatory guidance instruments for plants whose licences on the site and safety concept were to be granted after entry into force of the RSK guideline and made them binding for the plant operator by the licence permit. For plants that were granted a licence before, the RSK guidelines were referred to for assessing the adequacy of the further development of plant safety.

KTA safety standards

The Nuclear Safety Standards Commission (KTA) was established at the BMU. 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. of the unions, the industrial safety and the liability insurers.

The office of the KTA is affiliated to the Federal Office for Radiation Protection (BfS).

In accordance with its statutes, the KTA formulates detailed safety standards 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.

The KTA safety standards pertain to

– organisational issues,
– 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 and quality management occupy 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.

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.

On the basis of the regular reviews and, where required, 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, the KTA has issued a total of 90 safety standards and 2 draft standards. 12 draft standards are in preparation and 40 safety standards are in the process of being revised.

Within the framework of the KTA work, the following draft safety standards have been in preparation in the last years:

– [KTA 1203] "Requirements for the Emergency Manual",
– [KTA 2301] "Age Management in Nuclear Power Plants",

- [KTA 3101.3] "Design of Reactor Cores of Pressurized Water and Boiling Water Reactors; Part 3: Mechanical and Thermal Design", and
- [KTA 3107] "Nuclear Criticality Safety Requirements during Refuelling"

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.

Revision of the nuclear rules and regulations

National nuclear rules and regulations

The Atomic Energy Act, associated ordinances and administrative provisions as well as the non-legally binding regulatory guidance instruments largely date back to the seventies and eighties of the 20th century. Since then, the practices of licensing and regulatory enforcement, the rules and regulations of the KTA safety standards and also at the international organisations and in other countries have continuously been further developed. This applies to the German national regulatory rules and regulations only in parts.

Therefore, the BMU initiated a project at the end of 2003 on the development of an integrative and consistent set of rules and regulations for assessing the safety of German nuclear power plants. This framework shall include the fundamental safety requirements of the existing safety criteria [3-1] and of the existing RSK guidelines [4-1], shall describe the state of the art in science and technology and, among others, consider the international regulations as well as practical experience from the application of the existing German nuclear rules and regulations. A first draft was discussed (workshops and Internet) with the stakeholders. A new draft for “Safety Requirements for Nuclear Power Plants” (Revision B) was submitted in September 2006.

At the beginning of 2006, the *Länder* Committee for Nuclear Energy (LAA) (→ Article 8 (1)) decided that an update of the nuclear rules and regulations is necessary and is to be performed, in the interest of the safe operation of the nuclear power plants, by agreement between the BMU and the nuclear licensing and supervisory authorities of the *Länder*.

At the LAA meeting in November 2006, it was also decided that a working group of the Federal Government and the *Länder* under the leadership of the BMU shall develop a document as preparatory work for an ordinance or general administrative provision which includes the fundamental safety requirements for nuclear power plants under consideration of

- the German nuclear rules and regulations,
- the draft of the Safety Requirements for Nuclear Power Plants,
- the WENRA reference levels, and
- the IAEA safety standards,

and which represents the safety philosophy.

Further, an impact assessment shall be performed by means of a target-performance comparison of the fundamental safety requirements with the features of the German nuclear power plants and of an assessment of the deviations.

Development of international rules and regulations

Persons responsible for safety from Germany participate in the international development of rules and regulations. On the one hand, the aim is to make a contribution that international regulations promote the best possible precaution against damage and lead to the further development of national rules and regulations accordingly. On the other hand, the international developments shall be incorporated in the German nuclear rules and regulations and make a contribution to harmonisation.

In this respect, several projects were performed within the reporting period by authorities and their experts and, in some cases, also by plant operators:

- *Evaluation of rules and regulations during the development of the WENRA reference levels (WENRA RL)*
  During the development of the WENRA RL, all relevant safety standards of the International Atomic Energy Agency (IAEA) were evaluated. As a result of this work, a direct comparison for each of the 18 WENRA issues with current German regulations and IAEA rules and regulations was performed. The results are used for the development of the German WENRA action plan and the update of the German rules and regulations.
- *Comparison of the German rules and regulations with current IAEA safety standards*
  The current IAEA requirements and guides were assessed according to safety-relevant priorities and those selected were systematically compared and commented and, where appropriate, recommendations were given for the German rules and regulations. Altogether, it showed that in most of the cases many different German rules and regulations have to be referred to for a comparison with an IAEA safety standard.
- Participation in the development of IAEA rules and regulations and in safety standards commissions and committees such as CSS and NUSSC
  The BMU and experts from many areas actively participate in the development of the IAEA safety standards. The BMU is represented in the CSS and in the NUSSC. It has been practice for many years, to formally involve the public before IAEA rules and regulations are passed: the drafts are published in the Federal Bulletin with the request for comments.

  In 2006, the BMU started to prepare annual summary reports on the work of the IAEA on rules and regulations. These reports are submitted to the nuclear authorities and their experts and are accessible to the public. By means of the reports it shall be determined which projects on rules and regulations are relevant for Germany, in which projects German institutions want to participate actively and how progress in the work on rules and regulations shall be implemented in the German practice.
- *Action plan for the implementation of the WENRA reference levels into the German legal framework*
  In November 2005, WENRA published the reference levels for harmonisation and further development of safety of nuclear power plants in Europe. The WENRA report and the assessment of the fulfilment of the reference levels by the German rules and regulations and the safety practice at German nuclear power plants was published on the Internet.

  In November 2006, the BMU submitted, like all other WENRA countries, a first draft for a national action plan for the implementation of the WENRA requirements. This plan was published on the Internet. The plan will be further developed in agreement with the *Länder* authorities and after having heard the plant operators.

7 (2ii) System of Licensing

General provisions

The licensing of nuclear installations is regulated in the Atomic Energy Act [1A-3]. According to Section 7 of this Act, certain facilities (in particular, nuclear power plants) require a licence for the construction, operation, essential modifications of the plant or its operation and also for decommissioning. When issuing a licence, obligations may generally be imposed for achieving the purpose of protection. 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].

According to the applicable law (amendment of the Atomic Energy Act of 2002), licences for the construction of nuclear power plants for the commercial production of electricity are no longer issued. The licences for operation of the existing nuclear power plants are not limited in time and thus do not require a renewal. The authorisation for power operation of the existing nuclear power plants expires once the electricity volume for the respective plant as stipulated in the Atomic Energy Act or the electricity volume derived from transfers has been produced. Therefore, licensing procedures are only performed for the modification of existing nuclear installations and for decommissioning.

Thus, the following presentation concentrates on licensing procedures for major modifications of the existing nuclear power plants or their operation. Decommissioning is object of reporting within the framework of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.

The planned modifications of a nuclear power plant or its operation are to be assessed systematically with regard to their impacts on the necessary protective and preventive measures and are to be treated in the procedure accordingly. Modifications that may have greater than obviously insignificant impacts on the safety level of the nuclear installations are subject to licensing pursuant to Section 7 (1) of the Atomic Energy Act. In addition, there are modifications obviously having only insignificant impacts on the safety level and therefore are not subject to licensing but require accompanying inspections by the safety authorities within the framework of the supervisory procedure. For modifications requiring a licence, the fulfilment of the licensing prerequisites is to be verified according to Section 7 of the Atomic Energy Act.

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-12] and with the consideration of other licensing requirements (e.g. regarding the possible release or discharge of non-radioactive pollutants into air or water (→ Article 17 (ii)).

The Paris Convention on Third Party Liability in the Field of Nuclear Energy [1E-11] and the Joint Protocol [1E-14] 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

<u>Licence application</u>

The written licence application is submitted to the competent licensing authority of that *Land* in which nuclear installation is sited. The licence application is accompanied by documents that are stated in the Nuclear Licensing Procedure Ordinance [1A-10] and specified in guidelines.

In case of applications for modification licences, the examination of the licensing prerequisites does not only refer to the object of modification where major modifications are concerned but also to those plant components and procedural steps of the licensed plant on which the modification will have an impact. The documents have to cover these plant components and procedural steps.

In order to verify that the licensing prerequisites are fulfilled, appropriate - e.g. detailed plans, drawings and descriptions documents - are to be submitted on the issues concerned by the modification. Further details that have been object of the examinations of the licensing prerequisites in previous licensing procedures are referred to 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 specifications,
– the financial security,
– the type of residual radioactive material and its disposal, and
– the intended environmental protection measures.

<u>Examination of the application</u>

On the basis of the submitted documents, the licensing authority examines whether or not the licensing 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, e.g. - depending on the object of licensing - authorities responsible under the building code, the water code, for regional planning and for 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. They have no autonomous decision-making powers. The licensing authority assesses and decides on the basis of its own judgment. The authority is not bound by the findings of their authorised experts.

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 its advisory commissions, the RSK and the SSK, and in many cases the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) for advice and technical support. The BMU states, where required, its position on the project to the competent *Land* authority.

Participation of the general public

Public participation shall offer the citizens the opportunity to bring in their safety interests into the procedure themselves. 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.

The application documents to be submitted depend on whether the procedure is to be performed with our without the participation of the public.

In case of public participation, an additional safety report is to be submitted for informing the public. This report mainly serves to describe effects related to the modification, including the possibly changed effects of design basis accidents and the associated precautionary measures such that the citizens that might be affected can judge whether their rights are being violated. This report is not a “safety analysis report” for the determination and update of the licensing status.

Participation of the public was obligatory for construction licences. In case of major modifications, the authority may waive a public participation according to the Nuclear Licensing Procedure Ordinance if the modification does not give rise to concern that there may be adverse effects on the public. However, the public has to be involved if this is required pursuant to the Act on the Assessment of Environmental Impacts.

Environmental impact assessment

The Act on the Assessment of Environmental Impacts [1F-12] in conjunction with the Nuclear Licensing Procedure Ordinance [1A-10] 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 and, if available, the statement by the BMU and the authorities involved as well as 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. Supervision is performed by the *Länder* authorities. The *Länder* act on behalf of the Federal Government also with regard to the supervisory procedure, (→ Article 7 (1)), 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. The decisions on supervisory measures to be performed are taken by the supervisory authority.

As in licensing, the supreme objective of 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 licence notices,
- the fulfilment of the requirements of the Atomic Energy Act, the nuclear ordinances and the other nuclear safety standards and guidelines, and
- the fulfilment of any supervisory order.

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

- the compliance with the operating procedures,
- the performance of in-service inspections of components and systems important to safety,
- the evaluation of reportable events,
- the implementation of modifications of the nuclear installation or its operation,
- the radiation protection monitoring of the nuclear power plant personnel,
- the radiation protection monitoring in the vicinity of the nuclear installation, including the operation of the independent authority-owned remote monitoring system for nuclear reactors,
- the compliance with the authorised limits for radioactive discharge,
- the measures taken against 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, and
- the quality assurance measures.

In accordance with the Atomic Energy Act, the authorised experts called in by the supervisory authority have access to the nuclear installation at any time and are authorised to perform necessary examinations and to demand pertinent information. The supervisory authority is not bound by the result of its examinations.

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 to physical protection must be reported to the authorities [1A-17]. The regulations and procedures regarding reportable events and their evaluation are described in Article 19 (vi) - (vii). In addition, the plant operators regularly report on specific issues.

In addition to the continuous regulatory supervision, Section 19a of the Atomic Energy Act stipulates the performance of safety reviews and presentation of the results on fixed dates (→ Article 14 (i)).

On-site supervisory activities of the supervisory authority during normal operation are performed, on average, once per week and plant. The experts consulted even show greater presence. The involvement of the different management levels of the plant operators is always ensured. During plant revisions with refuelling outages and after reportable events, on-site supervision also takes place every working day.

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, or
- 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 related 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, or
- 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 related 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/her position of responsibility.

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 (→ Article 7 (2iii)) carried out in Germany in the course of design, construction, commissioning, operation and decommissioning of nuclear installations, 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.

The instruments presented have proven their effectiveness since, in the normal case, they ensure that the authorities have appropriate sanction possibilities and authorisations for the enforcement of regulations and provisions if required.

Article 7: Progress and Changes Since 2004

Within the reporting period, drafts on “Safety Requirements for Nuclear Power Plants” have been prepared by external experts under the leadership of the BMU and publicly commented on. In addition, the LAA General Committee decided in November 2006 to establish a joint federal and *Länder* working group for preparatory work on an ordinance or general administrative provision. In conducting its work, the working group shall consider the current German nuclear rules and regulations, the draft of the “Safety Requirements for Nuclear Power Plants”, the WENRA reference levels and the current IAEA rules and regulations and represent the safety philosophy of the German nuclear power plants.

Within the reporting period, a draft on the action plan for the implementation of the WENRA reference levels into the German legal framework and into the safety practice at the German nuclear power plants was submitted to the BMU. It was decided to further develop the action plan in agreement with the *Länder*.

Within the reporting period, the BMU participated intensively in the further development of the IAEA safety standards. For the first time, an annual report on the further development of the IAEA standards was submitted to the General Committee, that will be prepared regularly in future, to also increase the involvement of the nuclear authorities of the *Länder*. Within the reporting period, experts from authorities, technical expert organisations and plant operators have actively participated in the development of IAEA safety standards.

Article 7: Future Activities

The participation procedure on the BMU project “Safety Requirements for Nuclear Power Plants” will be finalised.

The consultation process of the federal and *Länder* working group regarding a future nuclear ordinance or administrative provision and initiation of related follow-up activities will be finalised.

Further concertation of the action plan for implementation of the WENRA reference levels with the *Länder* will take place. The related measures in the area of the nuclear rules and regulations will be performed and implemented at the nuclear power plants by 2010.

A continuous systematic exchange on further development of the nuclear rules and regulations in Germany with further development of the IAEA safety standards will take place.

8 Regulatory Body

ARTICLE 8 REGULATORY BODY

1. Each Contracting Party shall establish or designate a regulatory body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence and financial and human resources to fulfil its assigned responsibilities.
2. Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.

8 (1) Authorities, Committees and Organisations

Composition of the regulatory body

Germany is a federal republic. Unless otherwise specified, the execution of federal laws lies in principle within the sole responsibility of the federal states, the *Länder*. The "Regulatory body" is therefore composed of federal government and *Länder* government authorities (→ Figure 8-1).

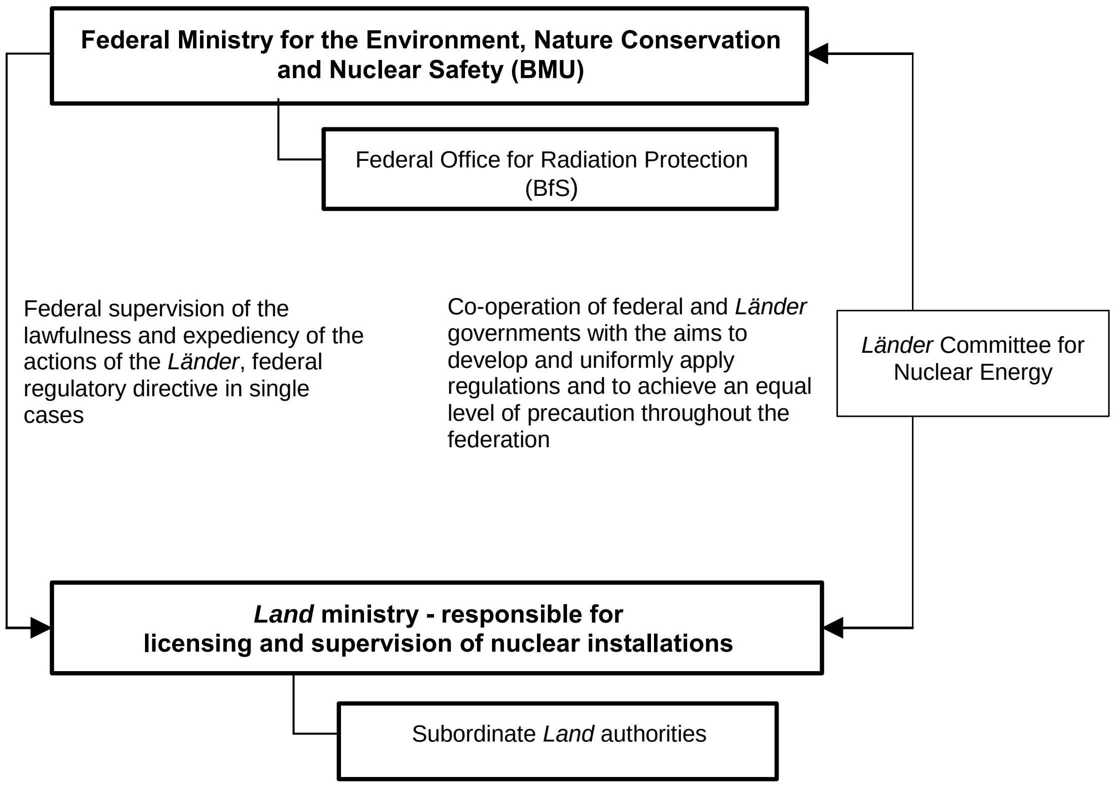


Figure 8-1 Organisation of the Regulatory Body

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

Safety (BMU) [1A-23]. Previously, the Federal Ministry of the Interior had been competent for environmental protection as well as for atomic law. The responsibility for the organisation, staffing and financing of the Federal Government's nuclear regulatory authority thus lies with the BMU. The BMU has the organisational powers and applies for the requisite human and financial resources from the annual federal budget.

Hence the BMU is the supreme regulatory authority in charge of nuclear safety and security in Germany. Regarding the obligations under the Convention, the BMU carries overall state responsibility both towards the interior of Germany and towards the international community that those in charge at applicants and plant operators, federal and *Länder* authorities, and at the authorised experts' organisations ensure at any time and with a lasting effect the effective protection of man and the environment against the hazards involved in nuclear energy and the harmful effects of ionising radiation.

The fundamental regulations for the further official competences are contained in the Atomic Energy Act (AtG) [1A-3] in Sections 22 - 24. According to Section 24, the respective *Länder* governments determine the supreme *Länder* authorities in charge of the licensing and supervision of nuclear power plants. Hence the responsibility for the organisation, staffing and financing of these executive authorities lies solely with the respective *Länder* governments. In individual cases, subordinate authorities may also be tasked with supervisory functions.

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>Environment Ministry
in agreement with
Economics Ministry and Interior Ministry</td><td>Environment Ministry</td></tr>
<tr><td>Bavaria</td><td>Isar 1
Isar 2
Grafenrheinfeld
Gundremmingen B
Gundremmingen C</td><td>State Ministry of the Environment, Public Health and Consumer Protection
In agreement with
State Ministry of the Economy, Infrastructure, Transport and Technology</td><td>State Ministry of the Environment, Public Health and Consumer Protection</td></tr>
<tr><td>Hesse</td><td>Biblis A
Biblis B</td><td colspan="2">Ministry of the Environment, Rural Areas and Consumer Protection</td></tr>
<tr><td>Lower Saxony</td><td>Unterweser
Grohnde
Emsland</td><td colspan="2">Environment Ministry</td></tr>
<tr><td>Schleswig-Holstein</td><td>Brunsbüttel
Krümmel
Brokdorf</td><td colspan="2">Ministry for Social Affairs, Health, the Family, Youth and Senior Citizens</td></tr>
</table>

Assignment of functions and competences of the regulatory body to the federal and *Länder* government authorities

The regulatory body tasked with the implementation of the framework for legislation and execution defined in Article 7 (1) thus consists essentially of the BMU and the competent supreme *Länder* authorities. According to Article 7 (2), this "regulatory body" has to fulfil four basic functions:

- the development of safety procedures and regulations,
- licensing procedures,
- regulatory examination and assessment, and
- execution and inspection.

From the Articles of the Convention listed below ensue the following further functions to be fulfilled by the "regulatory body":

- regulatory safety research (Art. 14, 18, 19),
- system for the application of operating experience (Art. 19),
- radiation protection (Art. 15),
- emergency preparedness (Art. 16) and
- international co-operation (Preamble vii and viii, Art. 1).

In Germany, these functions are distributed among federal and *Länder* government authorities. Nuclear regulatory authorities exist in all off the *Länder*. Table 8-1 lists the licensing and supervisory authorities of those *Länder* in which nuclear installations according to the Convention are located.

In principle, federal as well as *Länder* government authorities are involved in all functions, albeit with different competences, responsibilities and duties to co-operate. This distribution is shown in Table 8-2. Further details are provided in the respective relevant chapters of this report.

Within the reporting period, deliberations continued on a possible reform of the nuclear administration structure in Germany. The current Federal Government has no plans to change the existing basic structure of nuclear administration - i.e. the distribution of functions among federal and *Länder* governments - in Germany. The effectiveness of the regulatory body in Germany, however, is to be developed further and optimised. The IRRS mission planned for 2008 shall contribute to such optimisation.

<u>Subordinate federal government authority - Federal Office for Radiation Protection</u>

The subordinate authority to the BMU in the area of radiation protection and nuclear safety is the Federal Office for Radiation Protection (BfS), which was established by the corresponding Act of Parliament of 9 October 1989 [1A-22]. The four technical departments of the Federal Office for Radiation Protection deal with the tasks provided by the Act in the areas of environmental and industrial radiation protection, radiation biology, radiation medicine, nuclear fuel supply and waste management, and nuclear installation safety. The issues concerning the Convention on Nuclear Safety are mainly dealt with by the "Nuclear Safety" department. It supports the BMU technically and scientifically, especially in the execution of federal supervision, the preparation of legal and administrative procedures, and in intergovernmental co-operation.

Table 8-2 Assignment of the Regulatory Functions to the Nuclear Authorities of the Federal and *Länder* Governments

Regulatory function	Functions and responsibilities of the regulatory body Federal Government authorities	*Länder* government authorities
Main functions		
Establishment of national safety requirements and regulations [Art. 7 (2i)]	Further development of the legal requirements (decision by Parliament in the case of formal Acts, by Federal Government with approval of the Bundesrat in the case of ordinances) and the non-legally binding regulatory guidance instruments	Participation on the basis of consolidated findings and needs in connection with execution; supplementary administrative procedures of the respective *Länder*
Licensing system with regard to nuclear installations [Art. 7 (2ii)]	Supervision on lawfulness and expediency* Checking of consolidated findings with regard to their relevance to standard national requirements	Checking of applications and notifications according to Section 7 AtG, granting of licences and approvals
System of regulatory inspection and assessment of nuclear installations [Art. 7 (2iii)]	Supervision on lawfulness and expediency* Checking of consolidated findings with regard to their relevance to standard national requirements	Controls and inspections in the nuclear facilities, checking and assessment with regard to the relevance to the safety of the installation as well as to protection and prevention measures
Enforcement of applicable regulations and of the terms of licenses [Art. 7 (2iv)]	Supervision on lawfulness and expediency* Checking of consolidated findings with regard to their relevance to standard national requirements	Implementation of necessary measures to avert hazards and concerning necessary safety improvements and improvements of protection and prevention measures
Secondary functions		
Regulatory safety research	Investigation of safety issues for standard requirements	Plant-specific studies
Monitoring of events, operating experience and implementation	Examination and assessment of events in Germany and abroad with regard to generic relevance to the safety of the installations as well as to protection and prevention measures, national organisation of experience feedback	Examination and assessment of events with regard to relevance to the safety of the installations as well as to protection and prevention measures
Radiation protection, environmental monitoring	Monitoring of the radiation exposure of the population and the federal territory	Plant-specific monitoring of emissions and immissions (radiation exposure of workers and in the environment)
Emergency preparedness	Preparation and planning of general requirements; cross-national emergency preparedness, international reporting systems	Participation in the preparation and planning of general requirements, plant-specific emergency protection
International co-operation	Participation in international activities to determine the state of the art in science and technology and regarding the nuclear regulations, and provision for national purposes; Fulfilment of international obligations; assertion of German safety interests	Consideration of the internationally documented state of the art in science and technology Participation in the co-operation with neighbouring countries in the case of installations close to the border, especially on the basis of bilateral agreements

Grey	Leading function, execution within area of competence
Light grey	Function with separate competences but common objectives
White	"Federalism function" federal supervision or participation (e.g. in the *Länder* Committee for Nuclear Energy (LAA), by provision of information)

* This also means that the Federal Government may execute its power to decide the respective matter in hand itself and initiate on its own authority the corresponding detailed examinations.

Subordinate authorities in the *Länder*

As nuclear licensing and supervision is a function assigned to the supreme *Länder* authorities (ministries), only a few tasks are fulfilled by subordinate *Länder* authorities. In Baden-Württemberg, for example, measurements for the environmental monitoring of the nuclear power plants are performed by the Regional Office for Environmental Protection, which is subordinate to the Environment Ministry. This Regional Office also operates the computer and monitoring networks of the NPP remote monitoring (KFÜ) system.

Co-operation between the authorities of the regulatory body - *Länder* Committee for Nuclear Energy

The *Länder* Committee for Nuclear Energy (LAA) is a permanent Federation-*Länder* Committee composed of representatives from the *Länder* nuclear licensing and supervisory authorities and the BMU. It serves for the preparatory co-ordination of Federal and *Länder* authorities in connection with the execution of the Atomic Energy Act as well as for the preparation of amendments and the further development of legal and administrative provisions as well as of the non-legally binding guidance instruments. In the interest of an execution of nuclear law that is as uniform throughout Germany as possible, the competent nuclear licensing and supervisory authorities of the *Länder* and the BMU draft any regulations on the uniform handling of nuclear law in consensus. These regulations are then promulgated by the BMU. The BMU chairs the LAA and also manages its affairs. The Committee's decisions are usually by mutual consent.

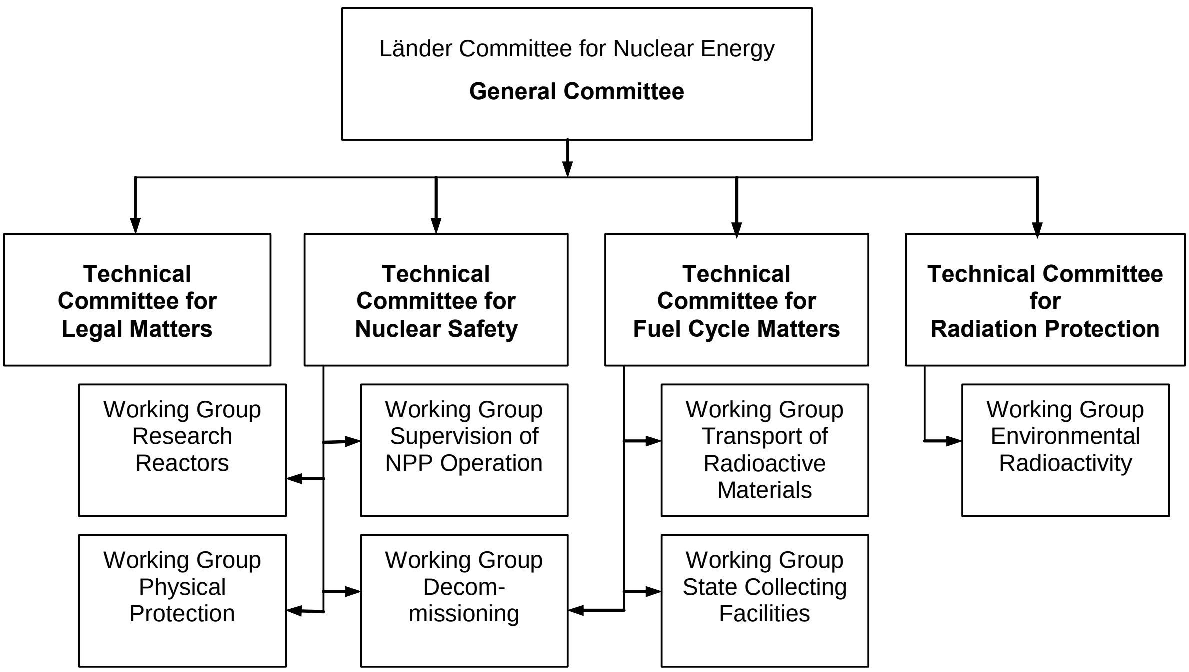


Figure 8-2 ***Länder*** **Committee for Nuclear Energy**

For preparing decisions to be taken by the General Committee, the *Länder* Committee for Nuclear Energy (→ Figure 8-2) avails itself of several Technical Committees on the issues of "Legal Matters", "Nuclear Safety", "Radiation Protection" and "Fuel Cycle Matters" as well as of the Working Groups assigned to these Technical Committees for special permanent tasks. If need be, the Technical Committees may set up *ad hoc* Working Groups for special and above all urgent individual issues. The Technical Committees and the permanent Working Groups convene at least twice a year and more frequently if necessary. The General Committee convenes at least once a year.

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).

Organisation and staffing of the nuclear authorities of Federation and *Länder*

<u>Nuclear authority of the Federation</u>

The nuclear authority of the Federation is a Technical Department (Directorate General) of the BMU. It comprises three Directorates. The entities of Directorate General RS dealing with the fulfilment of the obligations under the Convention on Nuclear Safety are Directorate RS I and some Divisions of Directorate RS II.

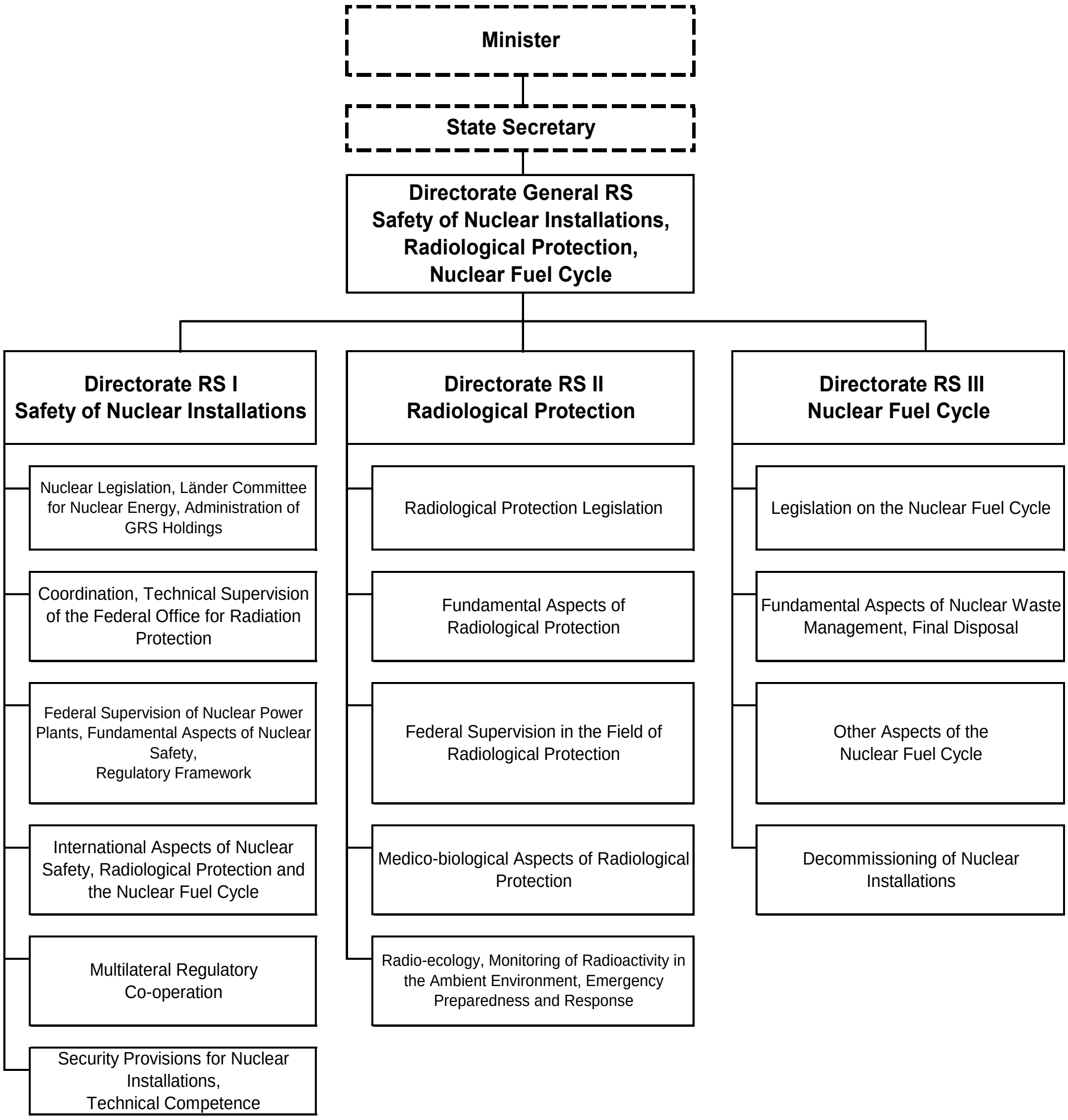


Figure 8-3 **Organisation of the Directorate-General Nuclear Safety and Radiation Protection**

Ministry staff are usually civil servants appointed for life or public sector workers.

The legal civil servants or public sector workers are required to have qualified at university and to have passed the corresponding examinations. The scientific-technical civil servants (senior service) of the Directorate-General RS are required to have completed a corresponding course at a university or a university of applied sciences (higher service). Other than that, there are no relevant regulations concerning training and qualification.

Department RS I is the one mainly responsible for the fulfilment of the BMU's obligations under the Convention. As at mid-2007, staffing of Directorate RS I with *legal experts* under permanent contract (including staff of other non-technical disciplines such as business administration or economics) and with scientific and technical civil servants or public-sector employees of the higher or senior service - i.e. *technical staff* - was as follows (Figure 8-4):

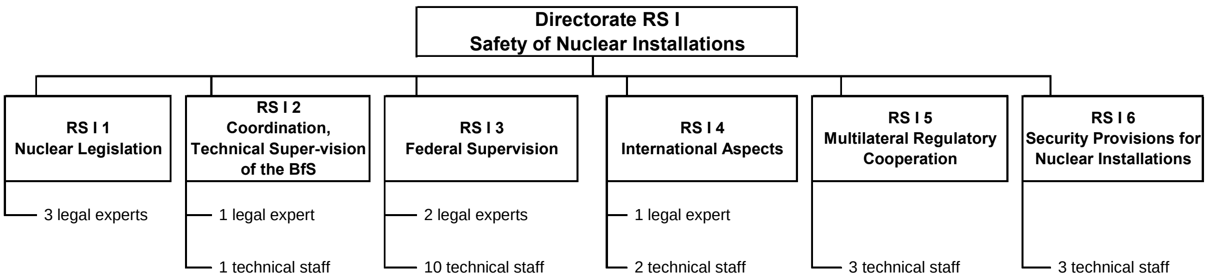


Figure 8-4 Organisation and Staffing of Directorate RS I

In Directorate RS II there are a total of 10 scientific-technical staff concerned with topics of radiation protection/emergency preparedness as affecting the scope of this Convention.

To bring about a change in the downward trend concerning the provision of competent personnel, corresponding demands have been filed in connection with the budget negotiations for the year 2008. Regarding the staffing of the federal nuclear authority it has to be taken into account that the latter avails itself of the scientific and technical support of BfS, of GRS in its function as authorised expert organisation of the Federal Government as well as of other authorised experts. Budget resources to the amount of about € 22 million are provided each year for such contract placing (see section on financial resources of the "Regulatory Body").

Nuclear authorities of the *Länder*

The nuclear authorities of the *Länder* for the supervision of nuclear energy are the supreme *Länder* authorities (ministries) determined by the *Länder* governments. The assignment of the competence to the ministries is by ordinance or by other organisational decree of the *Länder* governments. Table 8-1 shows the ministries competent for nuclear installations according to the Convention. Within these ministries, the functions of the nuclear authority are usually fulfilled by ministerial directorates. The structure of such directorates depends on the kind and scope of the nuclear activities and installations in the *Land* concerned. The directorates are in turn subdivided into divisions for the execution of the licensing and supervisory procedures for the nuclear installations and are supported by additional divisions dealing with radiation protection and environmental radioactivity, waste management, fundamental issues, and legal affaires. In some *Länder*, nuclear fuel cycle facilities not pertaining to the scope of the Convention have to be supervised in addition to nuclear power plants. The directorate for the supervision of nuclear energy is usually supported by a further organisational unit of the ministry; often this is a directorate for central tasks (e.g. human resources and budget-related affaires, infrastructure tasks and general services). The illustration in Figure 8-5 shows a basic organisation chart of a directorate for the supervision of nuclear power at *Länder* level.

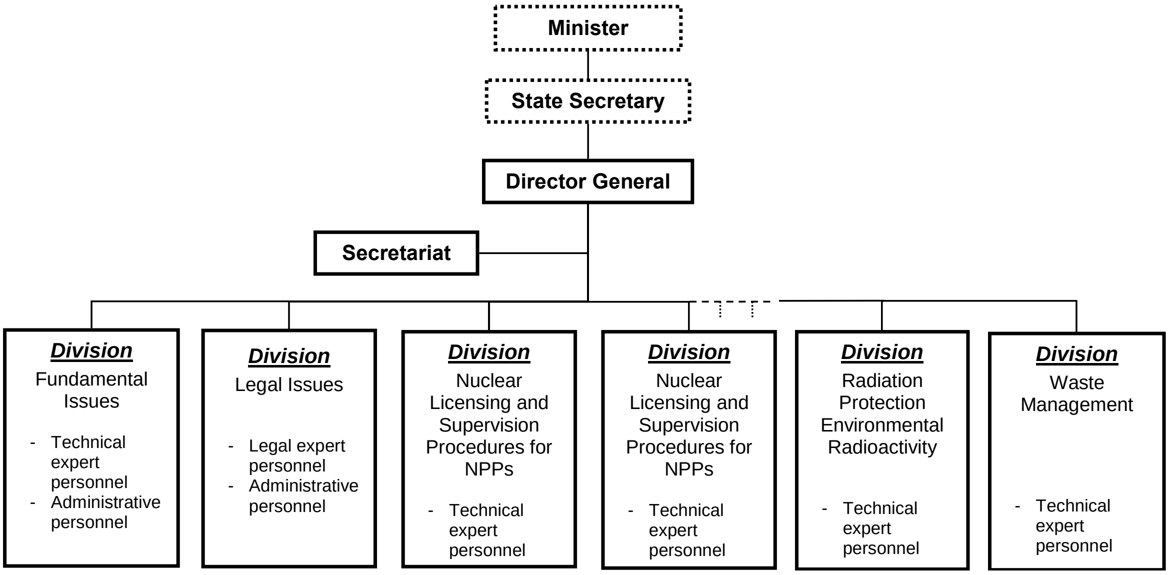


Figure 8-5 Basic Organisation of a *Länder* Ministry Directorate for the Supervision of Nuclear Energy

The directorates for the supervision of nuclear energy mainly employ technical specialist staff, especially engineers and scientists. They also have legal experts and administrative staff. All these directorates mainly carry out reviews and assessments as well as tasks related to the execution of the nuclear licensing and supervisory procedure as described in more detail in the following chapters. There is no strict allocation of staff to the tasks "Review and Assessment" and "Licensing" nor to "Inspection". These staff are furthermore tasked with the management and deployment of the authorised experts consulted as well as with the review and assessment of expert opinions.

The majority of the work has to do with the work on licences and the execution of concrete nuclear power plant supervision. The collaboration in the co-ordination of a uniform framework for licensing and supervision in the Federal/*Länder* committees as well as in the drafting of the safety requirements and regulations mentioned in Article 7 (2i) require a working effort that is not to be ignored. Its share depends on the extent of the nuclear energy programme of the *Land* and of the size of the respective organisation. Usually, it takes up between 10 - 15 % of the total effort.

Regarding the staffing of the nuclear authorities of Federation and *Länder* it has to be taken into account that according to Section 20 of the Atomic Energy Act authorised experts may be consulted in the nuclear administration procedure. The *Länder* nuclear licensing and supervisory authorities make regular and extensive use of this option due to the large extent of the inspections and the associated wide scope of different scientific and technical disciplines required as well as the special technical equipment and computer systems needed. To carry out the nuclear licensing and supervisory procedures, about 30 - 40 man-years are required for one single nuclear power plant each year. This includes the work of the authority staff and of the authorised experts consulted. For the most part, the scientific-technical competence needed is contributed by authorised experts.

Competence of the "regulatory body" staff

In its former reports under the Convention on Nuclear Safety, the Federal Government has affirmed that efficient and competent regulatory supervision is still necessary during the nuclear phase-out: "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."

A large number of experienced personnel of the nuclear licensing and supervisory authorities has already reached retirement age and left in the last few years or will do so in the years to come. This generation change represents a great challenge for the nuclear authorities, which have to compensate the loss of informed and experienced personnel by suitable measures in order to maintain the competence of the regulatory body in the field of nuclear safety and radiation protection. The situation is further aggravated by the fact that government saving measures often mean that positions that become vacant in particular at the federal nuclear authorities (BMU, BfS) are either not re-filled at all or only partly, usually with university graduates without any special nuclear knowledge.

A first step was to be a systematic "competence loss analysis" of the nuclear authorities and the subordinate authorities. The aim of the study was to find out which expert personnel were going to retire and what technical competences needed for the authorities' work would thereby be lost. However, this competence loss analysis could not be brought to an end for reasons, for example, of data protection.

<u>Competence and personnel development at the federal nuclear authority</u>

A personnel development concept to ensure personnel levels could not be established due to the budgetary boundary conditions. In the area of federal supervision, however, competence could be strengthened in particular by the recruitment of junior personnel. The loss of experience was largely compensated by the documentation of knowledge and by interviewing those who were about to retire, and the commitment of the junior personnel was successfully steered towards the acquisition of the knowledge thus preserved. As a first step to improve the situation, the BMU has applied for seven new positions within the nuclear safety administration for the 2008 federal budget.

The search for suitable personnel is carried out mainly by placing job advertisements in daily newspapers and technical journals and by internal advertising of vacant positions. An employment condition for technical personnel is a university degree in the relevant discipline. The knowledge needed for the special tasks of federal supervision (expert nuclear knowledge, administrational knowledge, etc.) is imparted in special courses during an introductory phase as well as by on-the-job training. The technical qualification and further education of newly employed personnel is mainly done by means of training courses and mentoring carried out by the expert organisation GRS as well as by participation in external events, such as specialist seminars and simulator training.

Further qualification as well as advanced training and professional development are addressed in the regular appraisal interviews held with all members of the personnel.

<u>Competence and personnel development at the *Länder* nuclear authority</u>

The *Länder* nuclear authorities, too are faced with special challenges regarding competence maintenance due to general budget saving measures. However, the situation compared with federal supervision is a different one since according to the Nuclear Costs Ordinance, the cost of the work of the authority has to be born by the plant operators. Nevertheless, special efforts are required to maintain the necessary staff levels and ensure the timely introduction of succeeding personnel to their particular fields of work. Reviews have led to a strengthening of personnel organisation and an improvement in the ratio of filled positions at some authorities.

New employees are to take part in the process of knowledge transfer of the authorities on the basis of a policy of overlapping re-occupation of positions. Their introduction to their respective fields of work is based on individual on-the-job training plans. Each individual on-the-job training plan comprises different training and further qualification measures, the introduction to special fields of work, and guidance for independent acting. Depending on the intended area of work and already available knowledge, the junior personnel is trained in all relevant technical and legal areas.

In addition, the personnel of the *Land* authority who already look back on many years in employment there and who have gained a large amount of experience are officially obliged to keep their technical qualification constantly up to date and to take part in the corresponding measures for their further qualification.

An employment condition for technical personnel is a university degree in the relevant discipline. The knowledge needed for the special tasks of regulatory supervision (expert nuclear knowledge, administrational knowledge, etc.) is imparted in special courses during an introductory phase as well as by on-the-job training with guidance by a mentor. Continuous checks of the working performance and results are made by the superior. Further qualification is addressed in the regular appraisal interviews. The search for suitable personnel is carried out mainly by placing job advertisements in daily newspapers and technical journals and by internal advertising of vacant positions. In the past, university graduates have employed as well as persons who have gained professional experience at commercial inspectorates, authorised experts, in industry and in science.

The fact that authorised experts are consulted for various different licensing and supervisory procedure demands that the regulatory officials have a broad, generalist knowledge. For example, they have to verify whether the authorised experts' comments cover all relevant areas and have to come to a decision of the basis of different comments. Some *Land* authorities have appointed so-called technical co-ordinators for individual technical fields in which these excel by having special knowledge.

Information and knowledge management system

To preserve part of the knowledge gained from past experience and to make it accessible to future personnel, an information management system is set up at the BMU in close collaboration with GRS. For this purpose, compilations of documents and technical information relevant to nuclear authorities and expert organisations are classified, structured and provided electronically. The personnel are to be given direct access to the information relevant to their work on their desktop computers, doing away with traditional files and time-consuming searches.

As the international exchange of information and knowledge is becoming increasingly important for the execution of the Atomic Energy Act and for regulatory co-operation, there is close networking with international information services and databases.

Financial resources of the "regulatory body"

The financial means available to the authorities for their own personnel and for the consultation of experts are fixed by the Federal Parliament (Bundestag) and the *Länder* parliaments in their respective budgets. The applicants and licensees are invoiced by the *Länder* for the project-specific costs of licensing and supervision. There is no refinancing of the activities of the federal nuclear authority.

In principle, the granting of licenses for nuclear power plants and the supervision activities of the *Länder* countries are with costs. The costs are paid by the licensee to the treasury of the respective *Land*. For a construction and operating license of a nuclear power plant, altogether 2 tenths of a percent of the construction costs have to be paid. A modification subject to licensing costs between € 500 and € 500,000. The costs of supervision are invoiced by the actual effort involved in the individual activities and lie between € 25 and € 250,000. The remuneration for the authorised experts consulted is also refunded by the applicant or licensee as expenses.

The BMU can dispose of an approximate annual € 22 million from the federal budget for studies related to nuclear safety. These funds are used for the financing of the work of the advisory commissions, for the direct support of the federal nuclear authority, for scientific and technical studies as well as for the participation of external experts in international co-operation. Corresponding programmes are perpetuated by the BMU every year and concern particularly the evaluation and assessment of operating experience, studies into special safety-related issues, the further development of technical requirements for nuclear installations as well as work on technical and other individual questions in connection with the licensing and supervision of nuclear installations. These programmes are administered by the BfS and in part also technically controlled. In addition, an approximate annual total of € 9 million is spent on studies related to radiation protection.

Management systems of the "regulatory body"

The management system of the Directorate-General RS is based on organisational decrees, responsibility schedules, rules of internal procedure, and procedural instructions as they generally apply to supreme federal authorities.

In addition, a special process-based quality management system has been introduced for the BMU Directorate-General RS.

Over a period of about two years, all relevant processes were initially identified by means of an as-is analysis, and the processes sequences were described. In a next step, the processes were summarised in process groups and optimised in collaboration with all those involved in the individual processes. Since August 2005, the descriptions of the process sequences have been available to all personnel of Directorate-General RS in the form of an electronic manual (for the process model see Fig. 8-6) and are continuously revised.

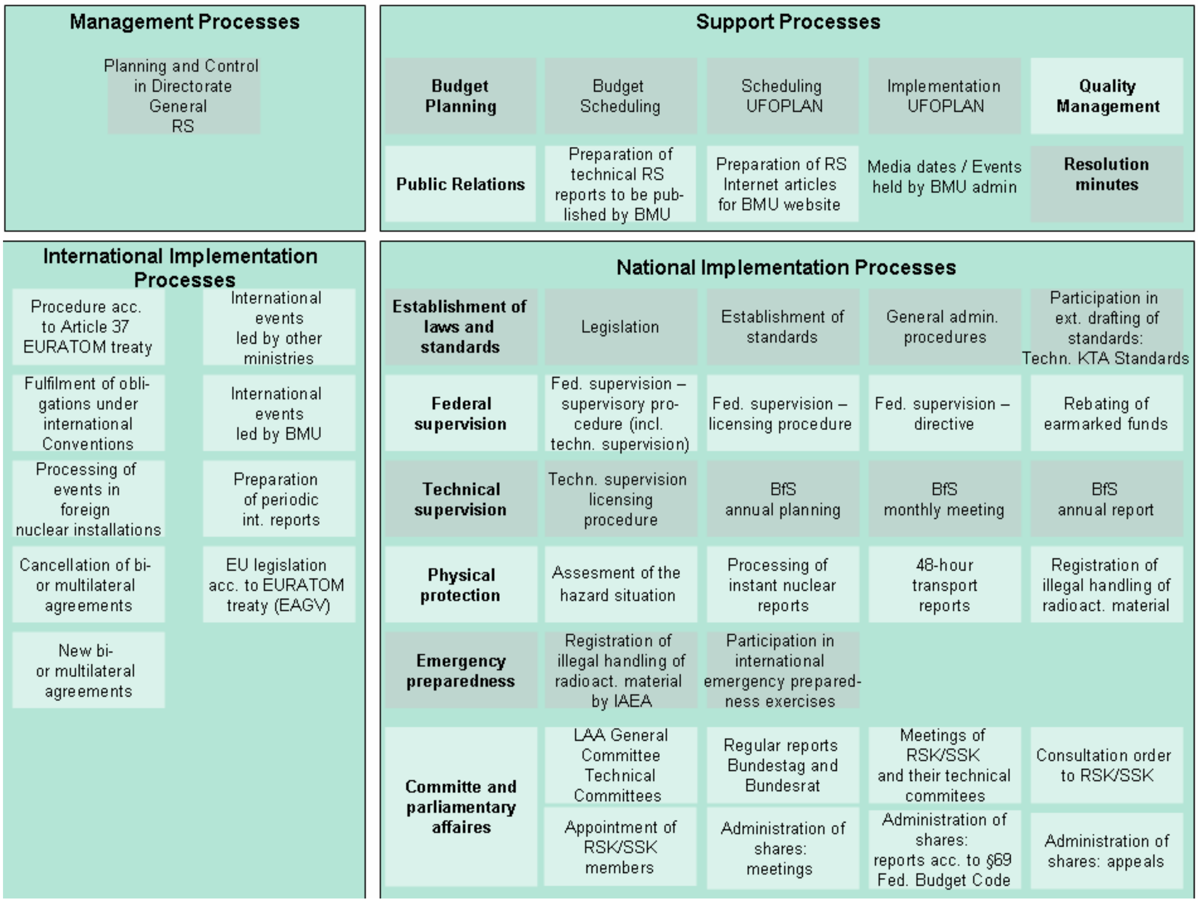


Figure 8-6 **Process Model of the Directorate-General RS (Nuclear Safety)**

The quality management system in the chosen form is both working principle and instrument of effective administrational control, supporting the senior personnel in carrying out their managerial functions. It should contribute to raising the quality and efficiency of the work and offers the individual help in better coping with the increasing workload. In addition, it is ensured by documentation of the processes and work instructions that experience is passed on specifically and is not lost as a result of the retirement of personnel.

The central process of the management system is the management process "Planning and Control in Directorate-General RS". The aim of this process description is to lay down how the planning and steering within the Directorate-General RS is to take place. With this process description it is to be ensured in particular that

– by elaborating the strategic aims of the Directorate-General RS, the definition of the key political strategies of the BMU is prepared,
– the measures to reach the strategic aims of the Directorate-General RS are implemented systematically and consistently with participation of the personnel,
– the responsibilities within the Directorate-General RS as well as the target values of each individual strategic target are defined,
– the Strategic Plan the Directorate-General RS is defined and continuously checked for the degree of its fulfilment, its topicality, content and time frame and that it is revised and supplemented if necessary, and
– the control of the fulfilment of the strategic aims the Directorate-General RS and the control of the status of the individual measures takes place on a continuous basis.

Development of management systems at *Länder* authorities

Irrespective of the fact that work sequences and processes of nuclear regulatory authorities are already largely regulated by the established organisational procedures for *Land* ministries, further developed approaches to these management systems that are specific to nuclear regulation are employed where applicable.

The authorised expert organisations consulted by the *Länder* licensing and supervisory authorities are certified according to the international quality assurance standards ISO 9001/2000. Some of the *Länder* authorities have their own quality management systems; others are in the process of building up such systems. Here, the activities are focused on the description and analysis of process sequences in connection with the nuclear licensing and supervisory procedure.

Support by the Federal Office for Radiation Protection, advisory commissions and authorised experts

Federal Office for Radiation Protection (BfS)

The support of the BMU by the BFS is mainly provided by the department "Nuclear Safety" (SK). At present, there are 22 scientific and technical personnel employed in its five sections.

The kind and extent of the support is co-ordinated on an annual basis between the BMU and BfS as part of their annual planning. With the annual planning, a catalogue of products including the envisaged labour input is defined, and the tasks to be done for reaching the products are assigned to the respective working units. As for the topics that concern the Convention on Nuclear Safety, these are mostly dealt with by the department "Nuclear Safety" (SK):

- documentation of the licensing status and the residual electricity volumes of nuclear power plants,
- documentation and initial assessment of reportable events,
- methods and status of the safety reviews,
- selected safety issues,
- international co-operation,
- national and international regulations,
- keeping of a register of occupational radiation exposure, and
- Support and administration of regulatory study projects.

Within the reporting period, the entire BfS as well as its department "Nuclear Safety" (SK) has lost personnel and technical competence to some extent due to the restrictive requirements of the federal budget legislation. Several technical issues can no longer extensively be dealt with.

Expert opinions on how the BfS may be modernised are available and being discussed.

Reactor Safety Commission, Commission on Radiological Protection

The Federal Environment Ministry receives regular 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 14 and 17 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 issues of fundamental importance, but they also initiate developments directed at furthering safety technology. The results of the deliberations of the two commissions are formulated as general recommendations and as statements on individual cases, which are then published (www.rskonline.de, www.ssk.de).

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.

Regarding the regulatory supervision of the peaceful use of nuclear energy, Germany - on the basis of the market-orientated structure of the Federal Republic - has like in other areas of technical supervision given preference to the co-operative relief of the state by private-sector forces of society over the build-up of large state authorities which would have to be staffed with sufficient personnel to deal with all the tasks involved exhaustively themselves. The special technical knowledge and independence are the decisive criteria for the involvement of authorised experts. Today, this is mainly ensured by the Technical Inspection Agencies (TÜV), which act on behalf of the authority as so-called "main consultants" of the *Länder* authorities.

Over the past decades, the Technical Inspection Agencies have built up large and powerful nuclear divisions or independent subsidiaries with considerable expert resources of about 1,000 specialists of the most varied disciplines. This is added by their experience from their work in the conventional, non-nuclear field. With only a few exceptions, they all dispose of the requisite knowledge in all relevant technical fields and ensure its sustained provision by taking suitable steps towards the acquisition and maintenance of competence as well as by a diversified exchange of experience in association with all other Technical Inspection Agencies.

The *Länder* authorities are not bound by the authorised experts' evaluation results in making their decisions. They have the necessary competences to fulfil their functions, which also involve the management of the authorised experts consulted.

In performing their licensing and supervisory activities, the *Länder* ministries may engage expert organisations or individual experts. These are engaged with regard to almost all technical issues related to the assessment of the safety of the installations and their operation. They are particularly involved in all licensing procedures as well as in the supervisory procedures, like e.g. in the evaluation of operating experience, the assessment of reportable events, in in-service inspection, and in applications for smaller modifications.

Section 20 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.

Further details regarding these requirements are specified in the regulatory guidelines [3-8] and [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, preferably with methods and computer codes different from those used by the licence applicant. The persons involved in preparing the expert opinions are not bound by any technical directives and are reported to the respective authority by name. In making their decisions, the authorities are not bound by the authorised experts' evaluation results. For the federal supervisory activities, the BMU will equally consult national and international experts if necessary. The work of the experts is financed by the plant operators via reimbursement, in some cases directly to ease administration.

Gesellschaft für Anlagen- und Reaktorsicherheit

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) is a central expert organisation. GRS performs scientific research in the field of nuclear safety technology, predominantly sponsored form federal funds, and is the main expert organisation advising the Federal Environment Ministry on technical issues. A limited amount of work is also performed by order of the licensing and supervisory authorities of the *Länder*. GRS employs more than 300 experts in the different fields related to nuclear safety.

IRRS self assessment and mission for the "regulatory body" in Germany

During the review period, the IAEA conducted two information workshops in Germany - in April 2006 and March 2007 - on the kind, procedure, contents and extent of an IRRS mission.

It has now been decided that the IRRS for the BMU and the nuclear authority of the *Land* of Baden-Württemberg is to be carried out in the first half of 2008.

The BMU is currently performing the self-assessment according to the unitised IRRS catalogue of questions.

As for the nuclear authority of the *Land* of Baden-Württemberg, a range of national reviews have already been carried out.

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 critically questioned the organisational separation of functions within German nuclear authorities as required by the Convention. In the centre was the question how far compliance with Article 8 (2) of the Convention is affected by the fact that the functions of nuclear regulatory supervision and energy industry promotion in parts 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 installations.

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 entails the 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 [1A-1], 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 [1A-1]) 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.

On the level of the *Länder*, the principle of separation of Article 8 (2) of the Convention is adhered to on the basis of the organisational provisions realised in the *Länder*. The effective separation of the competent authorities for the area of nuclear licensing and supervision from other authorities which - as part of the overall energy policy or energy industry support - also deal with matters of nuclear energy is ensured by the fact that different ministries or different and independent organisational units within one and the same ministry are in charge of and responsible for the different functions.

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 (→ Articles 7 (2ii) and (2iii) and Article 8 (1)).

The authority of the Federation to give directives concerning issues related to the licensing and supervision of nuclear installations - which is derived from Articles 85 (3) and 87c 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.

No other Federal Government agencies promote the utilisation of nuclear power, either. The policy of the Federal Government aims instead 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.

In relation to the above-mentioned state agencies, the licensees of nuclear power plants - in their function as users and perhaps promoters of nuclear power - represent commercial enterprises under civil law. They are either power utilities themselves or are composed of shareholders from the ranks of the German power utilities. These power utilities are also commercial enterprises under civil law, usually joint-stock companies (→ Article 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, the Federal Government also made it clear from the very beginning that there will be no cut-back in safety.

The governmental organisation in Germany fulfils the requirements of Article 8 (2) of the Convention.

Article 8: Progress and Changes since 2004

During the review period, deliberations continued on a possible reform of the nuclear administration structure in Germany. There are no plans to change the existing basic structure of nuclear administration - i.e. the distribution of functions among federal and *Länder* governments - in Germany.

The staffing situation at the regulatory body still needs to be improved. This applies in particular to the Federal Office for Radiation Protection.

Some nuclear authorities have managed to maintain the number of personnel and their competence. However, the loss of competence of expert personnel with long-standing experience has progressed due to retirements. The federal authority in particular has not been able to compensate for this loss as job cuts have increased at the same time.

The regulatory body continues to be reliant - and in some areas increasingly so - on the support by technical expert organisations such as GRS or the Technical Inspection Agencies. The expert organisations have established programmes that ensure the requisite numbers of personnel and their competence in the long run.

Article 8: Future Activities

The effectiveness of the regulatory body in Germany is to be further developed and optimised on the basis of the existing competences at federal and *Länder* level. The staffing situation shall be improved. The competent federal and *Länder* authorities collaborate to this end.

For the year 2008 an IRRS mission is planned that will involve inspections of the Federal authority as well as of the competent ministry in Baden-Württemberg.

The management systems of the nuclear authorities will be developed further, taking relevant IAEA Standards into account, and will be added by collaboration processes between federal and *Länder* bodies.

9 Responsibility of the Licence Holder

ARTICLE 9 RESPONSIBILITY OF THE LICENCE HOLDER

Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant licence and shall take the appropriate steps to ensure that each such licence holder meets its responsibility.

Regulatory requirements

The principle of responsibility is based on the regulations of the Atomic Energy Act [1A-3] on licensing and supervision. According to Section 7 of the Atomic Energy Act, the licence for construction and operation is only granted if the applicant proves that the necessary technical and organisational precautions for a safe operation have been taken. During operation, the plant operator has to fulfil his responsibility continuously. This is verified and ensured by the licensing and supervisory authority which has the means of Sections 17 and 19 of the Atomic Energy Act at its disposal (→ Articles 7 (2iv) and (2iii)).

Further, Section 7 of the Atomic Energy Act stipulates that the licence for construction and operation of a plant may only be granted if, among others, if there are no doubts as to the trustworthiness of the applicant and the responsible persons and these persons have the necessary technical qualification.

The nuclear regulations include general requirements for the fulfilment of the responsibility for safety regarding personnel and organisation. In terms of the Radiation Protection Ordinance [1A-8], the holder of the licence is the “radiation protection supervisor“ (Section 31 of the Radiation Protection Ordinance). In the case of corporate enterprises, the tasks of the radiation protection supervisor are fulfilled by a person authorised to represent the operating organisation. According to Section 33 of the Radiation Protection Ordinance, the radiation protection supervisor is obliged to take protective measures to protect man and the environment from harmful effects of ionising radiation, taking due account of the state of the art in science and technology. This responsibility includes, among other things, the provision of appropriate installations and equipment as well as the adequate regulation of the operating procedures with sufficient and qualified personnel.

In addition, officers are to be appointed for special tasks: the radiation protection officer according to the Radiation Protection Ordinance and the nuclear safety officer according to the Nuclear Safety Commissioner and Reporting Ordinance [1A-17], whose rights and obligations are specified in these ordinances in a legally binding form.

The responsibilities of the responsible personnel and otherwise engaged personnel are specified in the regulatory guidelines on technical qualification [3-2] and [3-27].

According to regulatory guideline [3-2], the <u>plant manager</u> 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 department or section.

The <u>heads of department or section</u> are responsible for their technical areas and are authorised to give orders to their subordinate personnel.

The <u>responsible shift personnel</u> - 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 (direct operating process). This also includes the necessary measures in case of alarms and emergencies.

In addition to the above-mentioned persons, regulatory guideline [3-2] specifies the tasks of the training manager, the head of the quality assurance division and the physical protection officer.

When using external personnel, the applicant has to make sure that the necessary knowledge is ensured according to guideline [3-27] and, where required, by persons in support of them. This also applies to the case that knowledge is communicated by the contractor. This is to be demonstrated to the supervisory authority upon request.

Implementation and measures by the utilities

The licensees of German nuclear power plants fulfil their responsibility for the safety of the nuclear power plants by many measures. Some of them are presented in the following.

Company principles

All German companies operating nuclear power plants committed themselves in fundamental documents, such as management principles or missions statements, to give priority to safety over all other business objectives. These documents include, among other things, binding objectives for the entire company:

- The safety of the nuclear power plants has top priority. It is based on proven technology, adequate organisational specifications and qualified personnel.
- Safety-relevant processes are critically analysed, monitored and further developed.
- All actions/activities/measures are performed with the necessary safety awareness.
- The technical safety level reached and the condition of the plant in compliance with the requirements of the licence are maintained and further developed by means of adequate monitoring and maintenance concepts and by plant modifications.
- Electricity is produced in an environmentally friendly manner.
- Fast and comprehensive exchange of experiences on safety-relevant events or findings is of great importance for the German nuclear power plants.

A special German feature is the co-operation of the plant operation in the VGB PowerTech e. V. (VGB) under whose umbrella research and development work is jointly promoted. The development of concepts, activities on updating the state of the art in science and technology and the exchange of experience among the plant operators is also organised via the VGB. Examples of the joint concept development are the following VGB documents: "Leitfaden zur Sicherheitskultur in deutschen Kernkraftwerken" (guideline on safety culture at German nuclear power plants), the framework paper "Sicherheitskultur in deutschen Kernkraftwerken - Konzept zur Bewertung und Trendverfolgung" (safety culture at German nuclear power plants - concept for assessment and trend analyses) and the "Konzept zur Optimierung des Sicherheitsmanagementsystems" (concept for the optimisation of the safety management system) as well as the jointly commissioned development of a system for integral event analysis under consideration of human errors and possibilities for organisational optimisation.

National and international reviews

The operators of the German nuclear power plants perform national peer reviews in the style of the WANO peer reviews. The aim of this initiative is to obtain representative information on the quality of the administrative/operative plant management, analogous to the WANO peer reviews, and to perform optimisations, if required. Nine representative processes (among others maintenance, evaluation of operating experiences, technical qualification, engineering/contracting) were selected for these reviews which are periodically performed by experts of other German plants for about three review days each. In general, a national peer review is performed at every German nuclear power plant once a year.

In addition, safety-relevant processes are reviewed within the framework of WANO peer reviews (plant operator initiative) by international experts (→ Article 14 (ii)).

Altogether, a large number of recommendations was developed in the reviews which led to optimisations at the plants.

However, the benefit for the German nuclear power plants is not only generated by the recommendations of the teams but also by the gain in experience of the peers from German nuclear power plants who frequently participate in the WANO peer reviews.

Parties responsible for the safety at the nuclear power plants

The plant manager is responsible for the safe operation of a German nuclear power plant. In particular, he is responsible for the compliance with the provisions and requirements under the Atomic Energy Act and licence permits as well as for the co-operation of all divisions. The plant manager is authorised to give orders to the heads of department or section.

In addition, different organisational parties - required by law and independent of the company hierarchy - supervise the safe operation of the German nuclear power plant within the company.

For the assurance of radiation protection during work, radiation protection officers are appointed in writing at all German nuclear power plants for directing or controlling such work in accordance with the Radiation Protection Ordinance [1A-8]. They are appointed by the radiation protection supervisor. Radiation protection supervisor is - depending on the organisational integration of the nuclear power plant concerned in the companies or corporations - the company or corporation itself. A board member or the director of the company fulfils the tasks of the radiation protection supervisor. He specifies appropriate regulations on the operating procedures and provides sufficient and qualified personnel and ensures, together with the radiation protection officer, that the protective provisions and control provisions of the Radiation Protection Ordinance are complied with. If the radiation protection officer cannot agree with the radiation protection supervisor with regard to the removal of deficiencies proposed by the former, then the radiation protection supervisor shall inform the radiation protection officer in writing of his rejection of the proposal, stating his reasons, and shall send a copy to, among others, the competent authority.

At each nuclear power plant, a nuclear safety officer is appointed in accordance with the Nuclear Safety Commissioner and Reporting Ordinance [1A-17]. The nuclear safety officer has to ensure, among other things, the evaluation of internal and external reportable events and other malfunctions at the plant. Information on reportable events in other plants are evaluated for their significance for the own plant. The nuclear safety officer co-operates in the preparation of corrective actions and improvements and checks that any reportable event has been correctly and completely reported. For fulfilment of his tasks, support by different organisational units (e.g. engineering) is available. The nuclear safety officer has the right to report directly to the management if agreement with the plant manager could not be reached and if he considers it to be necessary due to the particular importance of the matter in question.

Proceeding of the plant operators in case of doubts about the management and control of design basis accidents

The German nuclear power plant operators developed a common understanding on the handling of information and findings which raise doubts about the management and control of design basis accidents. They define “information” as data, facts or assessments by third parties that could be relevant for PWR and BWR plants of German licensees. Information will result in a finding if own knowledge is obtained by own assessment after evaluation of the information.

The following approach is practised. If the own engineering judgement of information under consideration of operating experience and known plant behaviour results in the finding that the management and control of a design basis accident is put into question, the licensee will, in the short term,

- initiate a work programme for further clarification of the issue and, if required, for the development of corrective measures, and
- inform the supervisory authority accordingly.

At the same time, the licensee decides whether the plant is to be shut down.

Regulatory review

For the German nuclear power plants, the organisation charts, the responsible persons and their area of responsibility are documented in the plant personnel organisation. The plant personnel organisation is part of the safety specification (→ Article 19 (ii)) and a licensing document. During the licensing procedure, the licensing authority checks whether the responsibilities are specified in an appropriate manner. The plant operator informs the authority about changes in the organisation chart or of responsible persons. A major modification of the plant personnel organisation requires the approval or licence by the competent authority.

In addition to the required technical qualification (→ Article 11 (2)), the supervisory and licensing authorities also evaluate the trustworthiness of the responsible persons of the plant operation and all persons working in safety-relevant areas. For assessment of the trustworthiness, an enquiry is made about findings of the police authorities. The persons may only start to work if the supervisory authority has no doubts as to their trustworthiness [1A-19].

Moreover, the nuclear authority also checks the trustworthiness of the applicant or licensee (of a corporation) or the persons representing him (e.g. the board members or directors).

The supervisory authority holds meetings with the board members or directors of the licensee to check how the persons responsible of the plant operators fulfil their responsibility for nuclear safety. Here, general questions relating to safety and the relationship between supervisory authority and plant operator can be brought up for discussion during which the supervisory authority ensures that the primary responsibility of the plant operator for safe operation is not impaired.

The supervisory authority regards all its activities performed within the framework of regulatory supervision as independent review to determine to which extent the license fulfils his responsibility for the nuclear safety of the plant.

The operators of the German nuclear power plants described how they intend to react to new findings and information which raise or may raise doubts about the management and control of design basis accidents. From the point of the view of the BMU, the criteria presented will need further specification.

Article 9: Progress and Changes Since 2004

After the event at Unit I of the Swedish Forsmark nuclear power plant in July 2006, the BMU and representatives of the German nuclear plant operators discussed the introduction of a reporting system of the plant operators for international events at nuclear installations with safety relevance for existing German installations.

A concept was presented which provides that

- the operators with power plant activities at the international group level accelerate the internal flow of information on events with safety relevance. Information potentially relevant for German plants shall be provided directly to other German plant operators.
- by means of the international reporting system NEWS of the IAEA all nuclear power plants are informed by own initiative about safety-relevant events within 48 hours without waiting for an assessment by the regulatory authority. The information shall generally be accompanied with a qualified report to serve as a basis for an in-depth plant specific review for applicability to other plants or, if necessary, for initiation and performance of first measures.

Assessment and forwarding functions of this new international reporting system of the plant operators shall be performed centrally at the nuclear reactor manufacturer AREVA.

According to a preliminary assessment by the BMU, these new reporting structures of the plant operators present a qualitative enhancement which may substantially improve the reaction of the plant operators to safety-relevant events. A final assessment of the system by the BMU is still pending.

Within the reporting period, company and management principles have been further developed. The concepts on safety culture, its assessment and trend analyses as well as on the optimisation of the safety management system - jointly developed by the plant operators under the umbrella of the VGB - have further be implemented at the different nuclear power plants.

The plant operators informed about conceptual deliberations on how they intend to react in case of doubts about the management and control of design basis accidents.

Article 9: Future Activities

The plant operators announced that they will evaluate and assess their experiences with the further developed approach on safety culture and safety management and perform appropriate improvements.

The BMU and the plant operators agreed that the plant operators will perform an appraisal of the self-learning safety management system and will develop and realise necessary additions and further improvement possibilities within one year.

10 Priority to Safety

ARTICLE 10 PRIORITY TO SAFETY

Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear installations shall establish policies that give due priority to nuclear safety.

Regulatory requirements

The protection of the public and property from the hazards of nuclear energy and ionising radiation and thus the assurance of nuclear safety is the central purpose of the Atomic Energy Act [1A-3] which is stated in its Section 1 subpara 2. This general purpose is served by the provisions of the Atomic Energy Act on the licensing of nuclear power plants and on the supervision during plant operation (→ Article 7). In particular, Section 17 of the Atomic Energy Act stipulates that the authority may impose obligations subsequently if it is required to achieve this protection purpose. The authority may also revoke the licences if such revocation is necessary to avoid substantial hazards to the public.

According to Section 33 of the Radiation Protection Ordinance [1A-8] one obligation of the radiation protection supervisor and the radiation protection officer is to ensure protection of the public and the environment in case of radiation exposure and protection against safety-relevant events.

From this interrelation of the Atomic Energy Act and the Radiation Protection Ordinance it follows that the plant operator, who has the primary responsibility for the safety of the nuclear installation, can get a licence for construction and operation of a nuclear power plant and can maintain it on a sustained basis only if he gives the required priority to safety.

With the agreement between the Federal Government and the power utilities of 11 June 2001, these declared that high safety level of the German nuclear plants, compared internationally, is also maintained during the remaining operating lives agreed upon. In particular, it was laid down that

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

In 2004, the BMU published the “Fundamentals of Safety Management Systems“ [3-81] with requirements for a process-based management system. In this publication, the BMU states that safety, as integral part of the safety policy of the plant operator, shall have priority over all other business objectives.

Implementation and measures of the plant operators (according to their own statement)

<u>Safety policy</u>

All German companies operating nuclear power plants committed themselves in the management principles or missions statements to give priority to safety over all other business objectives (→ Article 9). For the implementation of these principles, measures have been taken in the last years to further enhance the safety-directed behaviour of the personnel, keyword “safety culture", and to further develop the safety management system.

Safety culture

In 1999, the German operators published the VGB guideline "Sicherheitskultur in deutschen Kernkraftwerken" on the safety culture at German nuclear power plants. In this publication, the operators documented how they interpret and practice safety culture and according to which criteria safety culture can be assessed. Thus, there is an integral representation of the understanding of the German nuclear power plants regarding the topic of safety culture. Moreover, it is shown how the German nuclear power plant operators let themselves be guided by a common basis regarding the attitude to safety in spite of different company structures.

In 2002, the operators of the German nuclear power plants published the framework paper "Sicherheitskultur in deutschen Kernkraftwerken - Konzept zur Bewertung und Trendverfolgung" on safety culture at German nuclear power plants and the concept for assessment and trend analyses. According to this concept, a self-assessment of the safety culture was performed at all nuclear power plants which highlights the various elements of safety culture (e.g. motivation, information and communication) from different perspectives (e.g. managing personnel, staff) and assesses them. Prerequisite for the success of this self-assessment is a trustful, sanction-free and responsible use of the tool since the results concern sensitive areas of communication and interaction of the organisational units, teams and staff.

Safety management

In 2003, the German nuclear power plant operators published the "VGB-Konzept zur Optimierung des Sicherheitsmanagementsystems" on the optimisation of the safety management system. As orientation guide, this concept describes necessary and expedient elements and instruments of safety management and their interactions. It is based on the IAEA documents INSAG 13, NS-G-2.4 and ISO 9001:2000. However, by no means it requires an uniform approach to organisational structures and procedures in all elements and instruments. In fact, different forms are permissible and even desired for fostering of established company structures if the general objective of a comprehensive, effective and self-contained and thus adaptive safety management is achieved. On the basis of this VGB concept, safety management systems at the German power plants were further developed as process-based and thus adaptive systems. The process orientation is expressed in the stipulation of requirements and specifications, in the application of appropriate instruments for verifying the effectiveness of safety-relevant processes and by derivations of optimisation measures from these reviews. The process-based safety management at the nuclear power plants uses diverse instruments. Such instruments are, e.g., the safety culture assessment system of the VGB (VGB-SBS), audits, management review and the systematic introduction and use of safety indicators on the basis of IAEA TECDOC-1141 as an additional internal management instrument. They contribute to an objective and also quantifying assessment of the “lived” safety at the respective site. For many German nuclear power plants, realisation of a process-based organisation directed to continuous improvement was confirmed by certification according to DIN EN ISO 9001:2000.

Review by the authority

Within the framework of licensing of a nuclear power plant and within the framework of supervision of plant operation, the authority checks which provisions are implemented by the applicant to fulfil his responsibility for the safe operation of the plant (→ Article 9) and to give priority to safety.

In meetings with the managing personnel of the licensee, the supervisory authority satisfies itself that priority is given to the safe operation of the plant at the strategy level. In this respect, the statements and the behaviour of the managing personnel of the plant operator (top management) are of particular importance.

The supervisory authorities obtain information about the safety-directed behaviour of the operating personnel of the plant operator e.g. by extensive controls during on-site inspections and the evaluation of reportable events and other occurrences (→ Article 19 (vii)). The evaluation of this information supplements the assessment of the strategic area and its practical implementation.

Within the framework of accompanying reviews of the supervisory authorities on introduction and application of the safety management systems of the plant operators it is checked, among others, whether and how priority to safety is anchored in the basic principles of the safety management system. In addition to the basic principles, the focus of supervision is on those processes in which priority to safety becomes particularly evident (e.g. company objectives, management review).

The safety culture assessment system (VGB-SBS) is a tool for self-assessment applied by the operators of the nuclear power plants. The supervisory authorities of the *Länder* informed themselves about the methods and proceedings of the plant operators. The supervisory authorities are informed about the performance and main results of the VGB-SBS.

Implementation by the authority

Priority to safety is also applied as basic principle for the work of the nuclear authorities of the Federal Government and the *Länder*. This principle is implemented in the task descriptions of the supervisory and licensing authorities and is concretised in supervisory practice.

Supervision by the *Länder* is structured systematically according to the different fields of supervision (e.g. maintenance, in-service inspections, radiation protection). The regular evaluation of the findings from supervisory procedures allows that the *Länder* can organise their supervision by, e.g., additional inspections in case of indications, such that safety-relevant issues are given due attention.

Article 10: Progress and Changes Since 2004

In the last years, the management systems of the nuclear power plants have been further developed. In addition to the structural organisation, the procedural organisation was documented by process descriptions. General indicators were introduced and further develop plant-specifically. At many nuclear power plants, the organisation was certified according to DIN EN ISO 9001:2000.

Article 10: Future Activities

The certification according to DIN EN ISO 9001:2000 for further nuclear power plants is being continued.

The introduction of the process-based safety management system will be finalised at all nuclear power plants.

11 Financial Means and Human Resources

ARTICLE 11 FINANCIAL AND HUMAN RESOURCES

1. Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation throughout its life.

2. Each Contracting Party shall take the appropriate steps to ensure that sufficient numbers of qualified staff with appropriate education, training and retraining are available for all safety-related activities in or for each nuclear installation, throughout its life.

11 (1) Financial Means

Regulatory requirements

According to Section 7 para 2 of the Atomic Energy Act [1A-3] a licence for the construction, operation or essential modification of a nuclear power plant may only be granted if, among others, the necessary precautions have been taken in the light of the state of the art in science and technology to prevent damage resulting from the construction and operation of the installation. This precaution against damage is also the criterion for supervision during operation (→ Article 7 (2iii)). Prerequisite for ensuring the necessary precautions in accordance with the state of the art in science and technology is, among others, the provision of the necessary resources.

The requirement to provide the necessary financial means is also implicitly derived from the obligations of the radiation protection supervisor to establish and maintain a plant condition and operation that is in compliance with the licences and the safety requirements. Legally, the radiation protection supervisor is the license holder of a nuclear installation.

In Section 33, the Radiation Protection Ordinance [1A-8] regulates the obligations of the radiation protection supervisor. Accordingly, the radiation protection supervisor shall make provisions to ensure, in the light of the state of the art in science and technology, by appropriate protection measure, in particular by

– the provision of suitable rooms, equipment and devices,
– adequate regulation of the operating procedures, and
– provision of sufficient and qualified personnel,

that the provisions for the protection of the public, the environment and workers during the performance of their occupation from radiation exposure are complied with, that sufficient protection against safety-relevant events (design basis accidents, radiation emergencies and beyond-design basis accidents) is ensured and that the necessary measures against inadvertent criticality of fissile nuclear material are taken.

In order to be prepared for the follow-up costs connected with the operation of a nuclear power plant, the plant operators are obliged pursuant to the commercial law to build up financial reserves during the plant's operating life for the decommissioning and dismantling of the installations, and the treatment and disposal of radioactive material including spent fuel elements. For providing financial security for facilities for the final storage of radioactive wastes, the Federal Office for Radiation Protection collects advance payments for expenses for the exploration and construction of repositories according to the Waste Disposal Advance Payments Ordinance [1A-13].

Implementation and measures of the plant operators (according to their own statement)

The nuclear power plants in operation in Germany are run by private corporate enterprises. These are large power utilities and financially stable. Within the framework of management principles and mission statements, the utilities committed themselves to maintain a high safety level, to perform backfitting measures and to provide sufficient financial resources.

Consequentially, the German operators make extensive investments to maintain and further enhance the safety level of their nuclear power plants. For example, about € 970 millions were invested at the Biblis site from 1999 to 2005 for the optimisation of safety (45 %), routine tests/inspections (24 %) and modernisations (28 %). Other nuclear power plants made investments in a similar range.

Further modernisation measures are planned to be performed at all nuclear power plants. Thus, the plant operators fulfil their obligation from the agreement between the Federal Government and the power utilities of 11 June 2001 to maintain a high level of safety independent of the remaining operating lives of the plants.

To cover the follow-up costs resulting from plant operation, the operators built up adequate financial reserves for the treatment and disposal of radioactive material including decommissioning and dismantling of the nuclear installations which are adjusted on an annual basis.

Review by the authority

Within the framework of licensing of a nuclear power plant, the licensing authority examines whether safe operation is to be expected due to appropriate financial means of the applicant.

Essential changes in the legal form of the company that may have an influence on the financial means of the licensee are subject to licensing.

The operation of a nuclear power plant is subject to the continuous supervision by the authority. Should the regulatory supervision reveal that investments important with regard to safety are not made, the authority may order measures to be taken.

The valuation of reserves for waste treatment and disposal in the field of nuclear energy including decommissioning is regularly reviewed by independent accountants and the financial authorities.

11 (2) Human resources

Regulatory requirements

The obligations of the radiation protection supervisor according to Section 33 of the Radiation Protection Ordinance mentioned in Article 11 (1) also include the requirement for the provision of sufficient and qualified personnel.

The required qualification of the personnel responsible for the construction and operation is a licensing prerequisite according to Section 7 of the Atomic Energy Act and thus also to be fulfilled as prerequisite for operation in the long run. 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. 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].

Detailed requirements for the technical qualification of the responsible personnel are specified in guideline [3-2] and for the specific knowledge of the personnel otherwise engaged in guideline [3-27]. As responsible personnel, guideline [3-2] describes the following functions:

- plant manager,
- head of department or section,
- responsible shift personnel,
- training manager,
- head of quality assurance,
- radiation protection officer,
- nuclear safety officer, and
- physical protection officer.

The above mentioned guidelines [3-2] and [3-27] are supplemented by the guidelines on the certification of the qualification of responsible shift personnel, on the maintenance of qualification of responsible shift personnel, and on the specific qualification of personnel responsible for radiation protection [3-38], [3-39], [3-40], [3-61], [3-65]. 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. Guidelines [3-2] and [3-27] also include requirements for the maintenance of qualification of responsible personnel and measures for maintenance of knowledge of otherwise engaged personnel.

When using external personnel, the applicant has to make sure that the necessary knowledge is ensured according to guideline [3-27] and, where required, by persons in support of them. This also applies to the case that knowledge is communicated by the contractor. This is to be demonstrated to the supervisory authority upon request.

Implementation and measures of the plant operators (according to their own statement)

<u>Personnel development</u>

German nuclear power plants currently in operation are staffed with personnel that has a long experience in the operation of nuclear power plants. In addition to own plant personnel, use is also made of external personnel. On average, about 350 own plant employees and about 150 employees of contractors are employed all year round per unit. During plant outage for refuelling and annual inspection, the number of external personnel is increased to another 1000 employees approximately.

The demographic change and the generational change at the nuclear power plants have a decisive influence on the personnel structure of the operating organisations. Figure 11-1 shows the age distribution of personnel with technical know-how at a site with double-unit plant. The situation at other sites is comparable. The future personnel development at the plants will also be influenced by early retirement provisions of the last years. Therefore, the German nuclear power plants compare the long-term personnel planning with the demographic personnel development for early identification of future shortage of personnel and compensation by early recruitment of new personnel.

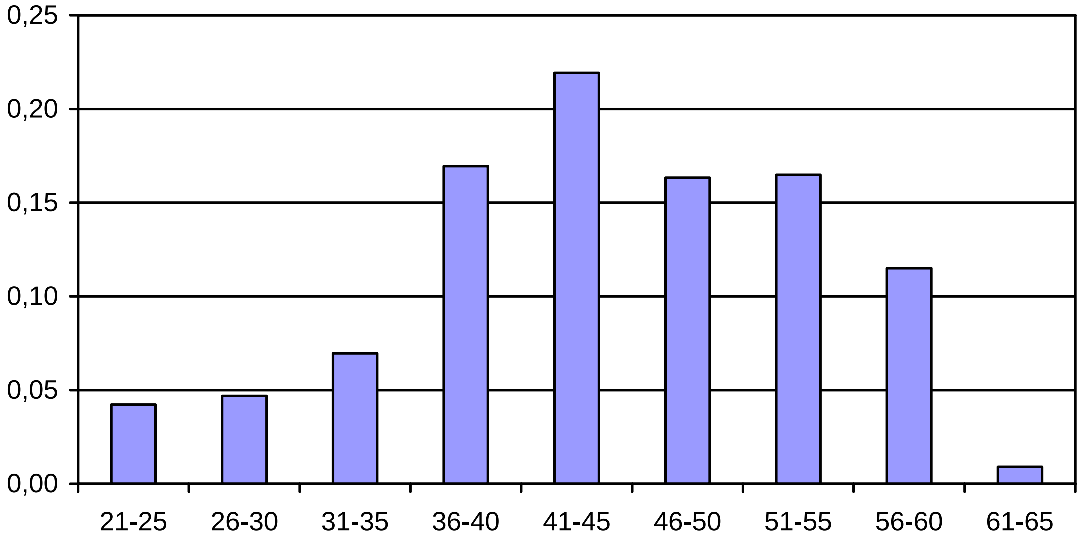


Figure 11-1 Age Distribution of Personnel with Technical Know-how at the Biblis Site

An equally important issue of personnel development at the nuclear power plants is the succession management. This also includes, among others, that adequate successors are identified and promoted by assessing their potential. This way, leadership positions that become vacant can be filled with staff best suited for the position in due time. The preparation for taking over of such positions is performed by appropriate training.

For technical positions, the transfer of technical competences from experienced senior staff that will leave the company in the foreseeable future to junior staff is ensured by a period of "overlap" according to the requirements which may well last several years.

Further, continuous promotion of junior staff takes place by intensive co-operation of the plant operators with the universities and the nuclear research institutions which comprises the promotion of professorships in the field of nuclear engineering, funding of doctoral studies as well as professional practical training and courses for students.

Personnel qualification

The German public vocational training system provides excellent conditions to ensure that the operating organisations 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. To supplement the public vocational training system, in 1970 the utilities founded a power plant training centre to correspond to the requirements regarding the specific skills of the nuclear power plant personnel.

The concrete requirements on the qualification of nuclear power plant personnel belonging to responsible personnel or otherwise engaged personnel according to guidelines [3-2] and [3-27] are specified in the training manuals of the nuclear power plants. They also document the measures for acquisition, maintenance and verification of the technical qualification. So, e.g., the responsible shift personnel must have passed the examination of technical qualification by the time they first act in the respective function. The measures regarding control of success and documentation of the training performed are also part of the training manual.

Plant-specific full-scope simulators are available for all nuclear power plants. Simulator training is an essential part of the programmes for the maintenance of technical qualification. The training is regularly adapted to new findings and technical facts. The training deals, among others, also with methods for coping with stress situations and communication. Particular attention is being paid to the feedback of operating experience.

The technical personnel - during initial training and repeatedly during advanced training - is regularly made aware of the importance of safety-oriented actions (→ Article 10). By means of practical examples (findings from the evaluation of operating experience), the particular importance of safety-oriented actions is concretised.

<u>Qualification of external personnel</u>

The requirements for otherwise engaged personnel from above-mentioned guidelines also apply to external personnel. In accordance with the respective duties, occupational qualification, practical experience and certification of knowledge are already required within the commissioning procedure. In addition, special instructions are given at the nuclear power plants. Here, plant-specific knowledge is imparted, at least in the fields of radiation protection, fire protection, industrial safety, and plant organisational structure and procedures. For persons in special positions (e.g. radiation protection planner, person responsible for the performance of the work), additional training is required.

Review by the authority

Within the framework of licensing of a nuclear power plant, it has to be verified to the licensing authority by the plant operator that a sufficient number of qualified personnel is provided for plant operation. The verifications of the plant operator are reviewed by the authority within the framework of the licensing procedure.

The supervisory authority gathers information about the long-term human resources planning of the plant operator. Essential changes in the number of personnel of the plant operator which may negatively influence the safe operation are subject to licensing and review by the competent authority.

Prior to the deployment of personnel stated in guideline [3-2] relating to the proof of the technical qualification of nuclear power plant personnel (management personnel), the supervisory authority requires the submission of documents which verify the necessary technical qualification and practical experience. It reviews these documents for compliance with the requirements of the guideline.

Members of the responsible shift personnel (shift supervisors, deputy shift supervisors and reactor operators) must additionally have passed the examination of technical qualification by the time they first act with responsibility in the respective function which consists of a written and an oral part [3-39]. After the written examination, the plant operator submits, among others, the result of the written examination and a compilation of the other proofs of technical qualification. Members of the examination board for the oral examination with a vote are, in addition to representatives of the plant operator, representatives of the supervisory authority and their experts consulted. The supervisory authority makes a written decision about the admission to the intended function as soon as the candidate has passed the examination and has met all other prerequisites.

For the otherwise engaged personnel it has to be verified that they have the necessary knowledge concerning the safe operation of the installation, of the possible hazards and of the protective measures to be taken as far as this knowledge is required for the proper performance of the tasks and for the protection of the person itself. This is verified by random inspections within the framework of regulatory supervision.

The plant operator submits the verifications on advanced training of his personnel and his three years programme on the maintenance of technical qualification to the supervisory authority. The supervisory authority reviews the appropriateness of the measures on the basis of the requirements of the guidelines on technical qualification [3-2] and [3-27].

In some cases, events during the reporting period also indicate deficiencies in the technical qualification or a behaviour of the plant personnel that is not always safety oriented.

Article 11: Progress and Changes Since 2004

The general duties of the plant operators and the competent authorities according to the regulatory requirements in terms of a continuously improving safety culture and in terms of the requirements of the Convention are a guide for action and measures. Furthermore, no specific measures have been required in the past three years.

Article 11: Future Activities

The general duties of the plant operators and the competent authorities according to the regulatory requirements in terms of a continuously improving safety culture and in terms of the requirements of the Convention are a guide for action and measures. Furthermore, no specific measures are provided for the next three years.

12 Human Factors

ARTICLE 12 HUMAN FACTORS

Each Contracting Party shall take the appropriate steps to ensure that the capabilities and limitations of human performance are taken into account throughout the life of a nuclear installation.

Regulatory requirements

The safety criteria for nuclear power plants [3-1] stipulate as a matter of principle that high requirements have to be imposed on the design and quality of the nuclear installations. Further, safety-enhancing operating principles are to be realised. These general requirements include, among others,

- ease of maintenance of the systems and plant components under special consideration of possible radiation exposure of the personnel,
- ergonomic measures at the workplaces, and
- reliable monitoring of the operating conditions.

Criterion 2.5 [3-1] requires that workplaces and work procedures are to be designed under ergonomic aspects in such a way that they create the prerequisites for the personnel's optimal behaviour in terms of safety.

KTA safety standard [KTA 3501] requires that necessary manual initiations of protective actions for controlling incidents are not required before a time span of 30 minutes. This requirement has considerable influence on the automation of protective actions and the design of the control room.

The general requirements on man, technology and organisation are specified in guidelines and KTA safety standards.

The general procedure of maintenance measures is described in the guideline for the procedure for preparation and performance of maintenance and modifications at nuclear power plants [3-41].

KTA safety standard [KTA 1201] includes the requirements on the operating manual, KTA safety standard [KTA 1202] analogously the requirements for the testing manual.

KTA safety standard [KTA 3904] makes requirements on control room, emergency control room and local control stations in nuclear power plants.

Implementation and measures of the plant operators (according to their own statement)

The 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. For the design of the control room, great importance was attached to ergonomic aspects.

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 routine actions and the control room staff is able to concentrate on the monitoring of safety.

The concept of the reactor protection system design includes the automatic control of 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, the aim of this concept is to ensure sufficient time to diagnose the situation and take appropriate actions.

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. 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.

Irrespective of the numerous technical provisions for preventing human errors, tools were implemented to further optimise the man-machine interface. In addition to the human factors programme (optimisation of the man-machine interface), already established in the nineties, the analyses of events (also referred to as HF events) was further optimised by introducing optimised methods (safety through organisational learning). In this respect, the operators of the German nuclear power plants developed a guideline on an integrated event analyses within the framework of the VGB Power Tech e.V. (VGB). Apart from the reportable events, reports about other abnormal occurrences and voluntary reports made by staff members are also recorded and investigated. For the analysis and determination of the contributing factors that have led to an event, generally accepted ergonomic methods are applied. The investigations comprise the areas man, technology and the organisation.

With the introduction of the VGB guideline on safety culture in 1999, the operators of the German nuclear power plants also showed how they interpret and practise safety culture and by means of which criteria effectiveness can be assessed. Accordingly, a high safety culture results in an awareness of the persons involved in plant processes that the safety of the nuclear power plant has the highest priority, that allows an appropriate handling of errors and that there is a working atmosphere based on trust. Essential principles of this guideline have been integrated in the management principles, company concepts, etc. and thus binding for the company's staff.

Regulatory review

The fulfilment of the requirements on the man-machine interface is checked by the licensing authority in the licensing procedures for construction and operation of the plant according to the requirements of the rules and regulations. To this end, verifications of the plant operator are subjected to extensive reviews on behalf of the authority. Modifications of safety-relevant plant components and written operating procedures are subject to regulatory review within the framework of the modification procedure. Scope and depth of the review depend on the safety relevance of the modification. For the handling of reportable events and other occurrences, the authority also considers the contributing factors from the areas man and organisation.

The concept of the plant operators on an integrated event analysis was reviewed for its suitability by the *Länder* authorities within the framework of their supervisory activities, partly with the support of the authorised experts. The plant operators give report to the supervisory authority on the application, results and effectiveness of their integrated event analysis. Moreover, the supervisory authority reviews the methods of the plant operator for the analysis of events and experience feedback in technical meetings. Main objective of the review is to ensure that the plant operator analyses the events in an integrated manner under consideration of all contributing factors from the areas man, technology and organisation and derives remedial measures with regard to the contributing factors.

In individual cases, the supervisory authority requires the performance of additional independent event analyses for in-depth assessment of contributing factors from the fields of man and organisation.

Article 12: Progress and Changes Since 2004

The plant operators developed methods for the analysis of events, here also referred to as HF events, on the basis of the guideline on an integrated event analysis of the VGB Power Tech e.V. (VGB). Apart from the reportable events, reports about other abnormal occurrences and voluntary reports made by staff members shall also be recorded and investigated. The analyses cover the areas man, technology and organisation.

The qualification of methods and experiences with their application are examined by the authorities. In individual cases, an in-depth analysis of contributing factors from the areas man and organisation were required.

Article 12: Future Activities

The general duties of the plant operators and the competent authorities according to the regulatory requirements in terms of a continuously improving safety culture and in terms of the requirements of the Convention are a guide for action and measures. Furthermore, no specific measures are provided for the next three years.

13 Quality Assurance

ARTICLE 13 QUALITY ASSURANCE

Each Contracting Party shall take the appropriate steps to ensure that quality assurance programmes are established and implemented with a view to providing confidence that specified requirements for all activities important to nuclear safety are satisfied throughout the life of a nuclear installation.

Regulatory requirements

All operators of German nuclear power plants are obliged to perform comprehensive quality management that is based on the provisions for quality assurance specified in the Safety Criteria [3-1] and in the KTA nuclear safety standards.

The general requirements regarding quality assurance are contained in KTA safety standard [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 plant operator 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 his own quality assurance system.

Implementation/measures

All German operators of nuclear power plants have implemented comprehensive quality assurance programmes on the basis of the provisions for quality assurance specified in the Safety Criteria and in the KTA nuclear safety standards. Main objective of these programmes is to assure the quality required for the safety of the plants in a comprehensive manner. By the high quality of plant operation systems, a sound and environmentally compatible operation is established and accidents are prevented.

The concrete implementation of the requirements from KTA safety standard [KTA 1401] and the Safety Criteria [3-1] is described in plant-specific documents (e.g. QM framework descriptions). These 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 verified. Procedures are described for the initiation of corrective measures in case of non-compliance with the quality requirements. Furthermore, the structure of the organisation implemented for quality assurance is described and reference is made to work procedures to perform quality assurance.

Quality assurance is independently performed by the plant operator within the framework of his responsibility for the safety of his plant.

With the introduction of ISO 9001:2000 and the associated discussion about management systems, e.g. also safety management, the plant operators improved quality assurance to become a process-oriented and thus adaptive quality management. Some nuclear power plants have their quality management system already certified according to ISO 9001:2000 by independent auditors.

To ensure that contractors for supplies and services, including their subcontractors, plan and implement quality assurance in accordance with the requirements of the quality assurance system of the nuclear power plant, they are monitored generally by certification according to KTA safety standard [KTA 1401] and specifically by contractor assessment for each individual order. The information about the contractor are stored in a central database and are available to each nuclear power plant. Any detected gaps and deficiencies are communicated immediately. Remedial actions are initiated.

Introduction and review of the quality management system is performed at each nuclear power plant by a staff unit independent of the monitored organisational units. This staff unit, with the quality management officer, is authorised to have access to all relevant information, propose solutions to pending problems, and monitor compliance with the quality assurance measures.

In exercising their responsibility for safety operation, the plant operators regularly review the effectiveness of their QA systems by own internal audits.

Review by the authority

The supervisory authority gathers information about the following topics:

- Results of the internal audits,
- implementation of the measures derived,
- further development of quality management towards an integrated management system,
- certification of the management systems,
- evaluation of indicators, and
- results of the management review.

On the basis of the information, the supervisory authority satisfies itself with regard to the effective implementation of the quality assurance system. Moreover, the supervisory authority controls the results of the audits performed by the plant operator and the implementation of measures derived from it within the framework of on-site inspections. Assessments and regulatory requirements refer to the effectiveness of quality assurance. The overall organisational responsibility for an effective quality assurance system remains with the licensee.

Article 13: Progress and Changes Since 2004

The general duties of the plant operators and the competent authorities according to the regulatory requirements in terms of a continuously improving safety culture and in terms of the requirements of the Convention are a guide for action and measures. Furthermore, no specific measures have been required in the past three years.

Article 13: Future Activities

For the further development of quality assurance during operation of a nuclear power plant, KTA safety standard [KTA 1401] is currently being revised.

14 Assessment and Verification of Safety

ARTICLE 14 ASSESSMENT AND VERIFICATION OF SAFETY

Each Contracting Party shall take the appropriate steps to ensure that:

i) comprehensive and systematic safety assessments are carried out before the construction and commissioning of a nuclear installation and throughout its life. Such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information, and reviewed under the authority of the regulatory body;

ii) verification by analysis, surveillance, testing and inspection is carried out to ensure that the physical state and the operation of a nuclear installation continue to be in accordance with its design, applicable national safety requirements, and operational limits and conditions.

14 (i) Assessment of Safety

Requirements for safety assessments in licensing procedures

For the application for the construction, operation and essential modifications of a nuclear power plant, it has to be demonstrated to the competent authority in detail that the licence prerequisites stated in Section 7 (2) of the Atomic Energy Act [1A-3] have been met (→ Article 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,
– 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,
– 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.

In accordance with guideline [3-5], the safety analysis report has to describe the effects of the project and the precautionary measures provided to be taken into consideration for the decision on the application. In this respect, third parties shall have the possibility to assess whether their rights could be violated by the nuclear installation and its operation. 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 facilities.

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. The main items are

- site,
- the nuclear power plant itself and protection against internal and external impacts,
- organisational structure and responsibilities,
- radioactive material and the corresponding physical protection measures taken,
- operation of the nuclear power plant, and
- 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.

The safety analysis report mainly serves the purpose of general assessment of the project under consideration of possible objections raised by members of the public.

For demonstrating the fulfilment of the licensing prerequisites and applicable safety requirements, supplementary documents and verifications are required pursuant to the Nuclear Licensing Procedure Ordinance [1A-10]. All documents are subject to regulatory review.

The safety specifications stated in the Nuclear Licensing Procedure Ordinance [1A-10] and specified in safety standard [3-4] are to be submitted with the application. They have to comprise

- 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, and
- treatment of reportable events.

These safety specifications (→ Article 19 (ii)) have to comprise all organisational regulations as well as relevant data, limits and measures which are essential for a safe condition and operation of a nuclear power plant. In particular, those procedures are to be described that are provided to cope with abnormal operation and design basis accidents. Any changes with respect to the safety specifications require the approval of the licensing and supervisory authorities.

All documents prepared for verification purposes according to Section 7 of the Atomic Energy Act are to be compiled systematically under consideration of the results of the expert assessments and regulatory assessments. For this purpose, the licensee has to prepare and keep up to date a “safety documentation” according to the guidelines [3-9.1] and [3-9.2] which includes all technical documents for verifications in nuclear licensing and supervisory procedures as defined by Section 7 para 2 subparas 3 and 5 and Section 19 paras 2 and 3 of the Atomic Energy Act.

These are, for example,

- documents pertaining to the specifications to be complied with in the realisation and testing of the nuclear installation and its parts and systems,
- documents pertaining to safety-related purposes and the mode of functioning of the various parts and systems of the nuclear installation,
- specifications regarding design, materials, construction and testing as well as specifications concerning maintenance and repairs,
- documents pertaining to the results of safety-related measurements and tests including radiographs and material samples for mechanical tests,
- documents pertaining to the fulfilment of safety-related specifications, e.g. calculatory demonstrations and design plans or drawings for the nuclear installation and its parts and systems,
- significant safety-related operating records,
- documents pertaining to the radiological protection of personnel and the environment, and
- other documents proving the fulfilment of safety-related specifications, conditions (Section 17 para 1 of the Atomic Energy Act) and directives (Section 19 para 3 of the Atomic Energy Act).

From the point of view of the authorities, the purpose and function of documentation is to show the existence or the fulfilment of the legal preconditions, e.g. the licensing conditions as laid down in Section 7 para 2 of the Atomic Energy Act, in a way that can be traced back and proved and to put the operating organisation or licensee of a nuclear power plant in a position to fulfil its obligation to inform the authorities in charge of government supervision (Section 19 para 2 of the Atomic Energy Act).

Thus, for nuclear power plants, the conclusion is drawn that safety assessments are to be updated in the light of operating experience and significant new safety information. If required, report is to be made on the results of these assessments and measures that may have to be taken in accordance with the Nuclear Safety Commissioner and Reporting Ordinance [1A-17], the operating manual and corresponding regulations from licensing and supervision. Moreover, comprehensive safety reviews for nuclear power plants are required according to Section 19a of the Atomic Energy Act. For their performance, guidelines are available [3-74.1 - 3]. Guidelines and dates prescribed by law for submission of such safety reviews are based on reviews after every ten years of operation.

Safety assessments in the supervisory procedure

Safety assessments are not only performed within the framework of licensing but also during the entire lifetime of a nuclear power plant. The operator of the nuclear power plant is obliged to do so, e.g. by the Atomic Energy Act or by licensing provisions. The safety assessments are to be submitted to the nuclear supervisory authority and reviewed with consultation of authorised experts. In addition to deterministic methods, probabilistic methods are also used for safety assessments.

A safety assessment comprising the entire nuclear power plant is performed within the framework of the so-called safety review. The safety review is dealt with in detail further below.

Safety assessments concentrating on a specific section of the nuclear power plant are, e.g., the safety demonstrations on the new reactor core after refuelling. In these safety demonstrations, the calculation of essential physical parameters and the fulfilment of the safety-related boundary conditions are demonstrated to the supervisory authority.

Safety assessments are also submitted to the supervisory authority in the course of modification processes. The performance of major modifications requires a licence pursuant to Section 7 of the Atomic Energy Act. The procedure is basically performed according to the same regulations described above for the granting of a construction licence. This also applies to the documents to be submitted and the safety assessment based on them. However, the extent of the documents and assessment is, by nature, limited to the object of modification (including its impacts) (→ Article 7 (2ii)). For modifications of the nuclear power plant or its operation not subject to licensing pursuant to Section 7 of the Atomic Energy Act due the negligibility of safety impacts other regulations are implemented. These specify the types of modification requiring prior approval by the supervisory authority and modifications that only have to be reported to the supervisory authority. For the approval of a planned modification by the supervisory authority, safety demonstrations may be required to a major extent that shall serve to verify that the performance of the modification and the modified plant condition do not impair the safety of the plant.

After safety-relevant occurrences at a nuclear power plant, the supervisory authority may require the performance of safety assessments, in particular if measures against recurrence or for improvement are to be taken. Safety assessments may also be required in case of relevant events at other plants with regard to their applicability and, where necessary, on improvements. New findings from plant operation and from science and technology may require an update of safety demonstrations already submitted.

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 regulatory 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 the end of the eighties, the operators of the German nuclear power plants had committed themselves voluntarily to the performance of PSRs at 10 year intervals. For seven nuclear power plants, such a PSR was already a mandatory requirement that had been specified in the corresponding licensing decision.

The amended version of the Atomic Energy Act of April 2002 stipulates the performance of safety reviews (SRs) every ten years. Due to the limitations imposed on the operating lives of the nuclear power plants, the safety reviews are no longer referred to as “periodic”. The dates for submission of the next SRs were included in the Atomic Energy Act (→ Table 14-1). 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 for submission of the SR mentioned in the Atomic Energy Act.

Table 14-1 Safety Reviews of the Nuclear Power Plants
(According to Appendix 4, Atomic Energy Act: Safety review pursuant to Section 19a para 1)

	NPP		Type	Date***)
1	Obrigheim*)	KWO	PWR	31.12.1998
2	Stade **)	KKS	PWR	31.12.2000
3	Biblis A	KWB A	PWR	31.12.2011
4	Biblis B	KWB B	PWR	31.12.2010
5	Neckarwestheim 1	GKN 1	PWR	31.12.2007
6	Brunsbüttel	KKB	BWR	30.06.2011
7	Isar 1	KKI 1	BWR	31.12.2004
8	Unterweser	KKU	PWR	31.12.2011
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	31.12.2010
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

*) End of power operation on 11 May 2005
**) End of power operation on 14 November 2003
***) Date for plants in operation according to Section 19a (1) of the Atomic Energy Act, i.e.: the date corresponds to the date mentioned in Appendix 4, Atomic Energy Act as far as it is after the 27 April 2002, for the deviant cases 10 years after the date mentioned in Appendix 4, Atomic Energy Act

The performance of the SR of nuclear power plants is based on the respective current national guidelines [3-74] for the deterministic and probabilistic safety analysis. 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 (→ Article 18 (i)) to cope with beyond-design basis conditions.

Deterministic safety status analyses and probabilistic safety analyses were performed for all 17 operating nuclear power plants and the Stade and Obrigheim nuclear power plants that meanwhile have been shut down.

As a result it can be stated that on the basis of the analyses performed (safety status analysis, probabilistic safety analysis, deterministic security analysis) it was demonstrated that the German nuclear power plants fully meet the protection goals - in the IAEA standards referred to as fundamental safety functions - necessary for achieving the safety requirements.

Probabilistic safety analyses

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

Since the seventies, the development of probabilistic methods and their exemplary application has mainly been performed by GRS on behalf of the Federal Government.

Extensive probabilistic analyses of Level 1 were finalised 2005 within the framework of a GRS research project for a BWR plant of construction line 69 for power, low power and shutdown operation, including analyses on the event of fire. The analyses were continued for power operation up to Level 2 with the objective to test the PSA methods of Level 2 for power operation. These have meanwhile been included, just as the PSA methods of Level 1 for low power and shutdown operation and updated and improved methods for fire PSA (Level 1 PSA), in a technical document on PSA methods pertaining to the regulatory PSA guidelines.

The methods and data applied for the probabilistic safety analysis are described and published in supplementary documents ("PSA Methods" and "PSA Data") [4-7] 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 was revised and republished in November 2005 [3-74.3].

In the years 1990 to 2000, the operators of the German nuclear power plants performed probabilistic safety analyses for all German nuclear power plants as part of the periodic safety review. Probabilistic safety analyses of Level 1 according to [3-74.1] now exist for all German nuclear power plants. They have led to technical and procedural improvements at the plants.

Of particular significance for the results of PSAs are common cause failures (CCF). Thus, an important aspect of the improvements from the methodical point of view is the further development of modelling and quantification of CCF. In this respect, reference is also to be made to the GRS-developed coupling model. It calculates the CCF probability with a coupling parameter which yields a phenomenon-specific estimate.

A measure that may be taken to counteract CCF is the additional introduction of diversities. Corresponding measures are also described in the following exemplary list of plant improvements derived from the full power PSA for the Philippsburg 1 nuclear power plant:

- Replacement of older level transmitters and selection of the combinations for minimisation of CCF by diversity and monitoring of the level transmitters, and
- actuation of the low pressure injection system by the core temperature as criterion diverse to the filling level.

Results from low power and shutdown PSAs mainly led to modifications in the area of administrative specifications.

The removal of deficiencies and improvement of the balance of the precautionary measures led to an increase in safety which is also reflected in the PSA results.

Involvement of authorised experts and subordinate authorities

The licensing and supervisory authorities normally consult external experts in accordance with Section 20 of the Atomic Energy Act for the assessment of specific technical aspects (→ Article 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.

The licensing and supervisory authorities themselves and subordinate authorities commissioned by them will also carry out own measurements and inspections.

14 (ii) Verification of Safety

Regulatory requirement

During plant operation, the provisions of the Atomic Energy Act and the statutory ordinances in pursuance thereof have to be complied with. The orders and directions issued hereunder and thereunder by the supervisory authorities and the terms and conditions of the notice granting the licence or general approval, as well as subsequently imposed obligations have to be adhered to.

With the licence, the licensee is obliged by law to verify regularly by means of in-service inspections that the quality characteristics and system functions essential for the safety of the plant have not deteriorated below acceptable levels. The corresponding provisions are included in the licences, the safety specifications and in the safety documentation. Detailed requirements for monitoring, recurrent tests and inspections are to be laid down in the operating manual according to KTA safety standard [KTA 1201] and in the testing manual according to KTA safety standard [KTA 1202].

Routine verification of safety by the plant operator

Within his responsibility, the nuclear power plant operator has to ensure that the safety of the plant is in compliance with the required level over its entire operating life. Whenever new safety-relevant findings are available, they check the necessity and adequacy of possible improvements.

To this end, the safety systems are subjected to in-service inspections by the plant operator 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 plant operator plans and performs regular and preventive maintenance of the plant systems during operation and evaluates the operational experience (→ Article 19 (vii)).

The in-service inspections of systems important to safety are performed in accordance with the requirements specified in the testing manual (→ Article 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 14-2 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 14-2 Annually Performed In-service Inspections, Typical for a PWR (Construction Line 3) with one Major Refuelling Outage per Year

Items	During operation	During outage	Total
Visual and functional tests	2 716	652	3 368
Radiation Protection	335	9	344
Lifting equipment	65	10	75
Non-destructive tests	1	6	7
Civil engineering	10	3	13
Plant security	85	-	85
Total	3 212	680	3 892

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

In addition, the plant operator performs the tests and inspections required by law in accordance with the conventional rules and regulations (e.g. according to the Operational Safety Ordinance).

To ensure that deficiencies and abnormal occurrences requiring remedial measures safely pass through the process of technical clarification up to the performance of the necessary measures after their detection, a corresponding operational management system, generally computer-based, is implemented. In this way it is ensured that deviations from specified plant conditions lead to the necessary repairs. Further details on this issue are included in Article 19.

Ageing management

The measures for maintaining quality over a long period of time (ageing management) are an integral part of the quality requirements specified in the German nuclear rules and regulations, particularly in the KTA safety standards. These deal with ageing phenomena under the term operational influences.

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,
– the monitoring of systems and operating conditions with respect to detecting any deterioration important to safety,
– the regular replacement of system component parts known to be susceptible to failure by preventive maintenance (→ Article 19 (iii)),
– an upgrading or replacement of technical systems in case weaknesses important to safety are found (→ Article 18 (ii)),
– the optimisation of technical systems and of operating conditions,
– continuous evaluation of operating experience, including the implementation of findings from experience feedback (→ Article 19 (vii)),
– acquisition and maintenance of qualification at a sufficiently high level (→ Article 11 (2)).

This practice is supplemented by appropriate research and development.

By evaluation of the results of the in-service inspections with special attention to systematic deficiencies, failure causes due to ageing shall be detected at an early stage. There are specific regulatory requirements regarding ageing of certain plant components (e.g. fatigue analyses for components of the pressure boundary according to KTA safety standard [KTA 3201.2], or type tests of instrumentation and control equipment according to KTA safety standard [KTA 3503] or type tests of electrical drives according to KTA safety standard [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 countermeasures 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. 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.

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.

In July 2004, 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. In 2005, the BMU asked the *Länder* to request the plant operators to implement the RSK recommendation. Plant-specific basic reports on ageing management are provided which are to be updated at regular intervals. The BMU and the *Länder* observe the implementation of the ageing management systems in the Technical Committee for Nuclear Safety of the *Länder* Committee for Nuclear Energy and its working group on supervision and reactor operation.

In November 2005, the Nuclear Safety Standards Committee (KTA) passed the decision to initiate project work on a new KTA safety standard [KTA 2301]. The work was started in May 2006.

Verification of safety by regulatory inspections

The nuclear licensing and supervisory authority monitors and, if necessary, enforces the fulfilment of the obligations of the plant operator relating to the licence.

In addition to the inspections performed by the plant operator, safety verifications are performed within the framework of regulatory supervision. The supervisory authorities verify by means of different methods whether the plant operators meet their obligations. The choice of the methods depends, among other things, on the plant state, such as construction, operation, outage or implementation of modification.

Accompanying inspections during construction, commissioning and modification

During the construction and commissioning phase, the authorised experts called in by the supervisory authority will perform accompanying inspections in order to supervise the compliance with the licence provisions and those of the supervisory procedure. 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 compliance with the boundary conditions for the accident analysis are checked (→ Article 19 (i)).

Inspections during operation

For tests and controls at the nuclear power plant, the supervisory authority of the respective *Land* performs on-site inspections at regular intervals by consulting of authorised experts. Such inspections may be directed to the clarification of specific issues or be performed with the objective of a general plant walkdown. Here, e.g., compliance with provisions related to radiation protection, the handling of fuel elements or the correct performance of tag-out and work order approval procedures is controlled. Moreover, the in-service inspections on safety-relevant components are accompanied by the experts of the supervisory authorities at specified intervals. The on-site inspections also cover the documentation to be prepared by the plant operator (e.g. shift logs, records on personal dosimetry). In addition to such inspections not bound to a specific occasion, there are also on-site inspections due to reportable events or other indications during which the supervisory authority and their experts build their own picture.

The plant operators are obliged, e.g. pursuant to licensing provisions, to submit reports on different subject areas. These include, e.g., subject matters related to operation, safety and radiation protection, including environmental monitoring, and on the radioactive material inventory and use. Such reports are evaluated by the supervisory authority, subordinate authorities or by experts consulted. Findings are handled by further investigations.

The current operating conditions of the nuclear power plants are directly monitored by the supervisory authority of the respective *Land* or a subordinate authority by means of the remote monitoring system for nuclear reactors (KFÜ) (→ Article 15). With this transmission system, the authority staff can monitor essential operating parameters and emission dates of the plant online. The values transmitted are updated at short intervals and stored so that they will also be available, if required, for future investigations. If specified limits are exceeded, the authority is alerted automatically.

International safety evaluations

Upon invitation, the IAEA has so far conducted five OSART missions at German nuclear power plants. The dates of the missions are mainly concentrated on the first period at the end of the eighties and at the beginning of the nineties: Biblis A (PWR) 1986, Krümmel (BWR) 1987, Philippsburg 2 (PWR) 1987 and Grafenrheinfeld (PWR) 1991 (with follow-up mission in 1993).

The last OSART mission in Germany was performed in 2004 at Philippsburg 2. The follow-up mission in November 2006 showed that all suggestions and recommendations resulting from the mission had already been dealt with or implemented to a large extent.

So, improvement measures were performed with regard to

- competencies and responsibilities relating to emergency preparedness,
- communication, team organisation and processes,
- structuring and accessibility of plant documentation under consideration of the hierarchy level and safety relevance of the individual documents, and
- proceeding in case of plant modifications limited in time.

For the Neckarwestheim 1 nuclear power plant (PWR), a further OSART mission is scheduled for 8 to 24 October 2007.

In Germany, WANO peer reviews were conducted successively for all plants in operation (with the exception of Philippsburg where the OSART missions mentioned took place). From 1997 to 2006, the plants Grohnde (1997), Grafenrheinfeld (1999), Gundremmingen (2000), Neckarwestheim (2001), Brunsbüttel (2001), Isar (2003), Emsland (2004), Brokdorf (2005), Biblis (2005), Unterweser (2005) and Krümmel (2006) were subjected to an audit.

For a second cycle for the performance of WANO peer reviews, the following proposal on scheduling was made: Grohnde (2007), Gundremmingen (2007), Grafenrheinfeld (2007), Brunsbüttel (2008), Isar (2009), Philippsburg (2009), Emsland (2010), Brokdorf (2010), Neckarwestheim (2011), Biblis (2011), Unterweser (2011) and Krümmel (2011).

Backfitting measures and improvements performed and current activities

The constantly increasing knowledge and additional requirements imposed by the authorities for the operating times of the nuclear power plants have led to safety-related backfits and improvements of the plants in different subject areas. Here, findings from the safety reviews including PSR/PSA were also referred to. Some important measures that give a good picture of the backfitting measures at German plants since 2003 are described in the following.

<u>Impairment of water suction from the containment sump</u>

In order to prevent impairment of water suction from the containment sump, measures were performed to optimise sump suction under the condition that the sump water contains insulation material. After experimental verification that the penetration of insulation material through sump strainers with smaller mesh size is significantly less, new strainers were installed at German PWR plants. An additional measure was the use of optimised insulations material, in particular avoidance of the use of both particulate and fibrous insulation material.

With regard to loss-of-coolant accidents with release of insulation material, the issue of attachment of microparticles to the fibrous insulation material at sump strainers and the resulting increase of flow resistance and pressure losses is currently of importance. Analyses on this issue are performed in response to the RSK statement of July 2004. To ensure core cooling in case of increased pressure losses due to depositions from insulation material and other substances on the strainers, strategies and procedures are being developed. To this end, it is planned to switch off the emergency core cooling and residual heat removal pumps for a short period and to perform a backflushing of the strainers to remove the depositions. The penetration of insulation material and other substances through the strainers into the emergency core cooling and residual heat removal lines or into the reactor core is limited by strainers with small mesh sizes.

Prevention of radiolysis gas accumulation

In December 2001, a radiolysis gas explosion occurred at the Brunsbüttel nuclear power plant (BWR) in a line of the RPV head spray system with rupture of parts of the line. The applicability of this occurrence to other plants was checked for all German nuclear power plants. In response to the results of these checks, plant modifications were performed at all German boiling water reactors. The measures served for the detection of radiolysis gas accumulations in areas close to the reactor and their prevention. The preventive measures were mainly consisted in simplifications of the piping system (removal of pipes without through-flow in which radiolysis gas may accumulate and modification of the pipe run) and in ensuring minimum flow rates through valves by drilling of vent holes. The temperature monitoring system was optimised by installation of additional sensors at high points. For PWR plants, no measures were required.

Accident behaviour of fuel elements with cladding tubes made of zirconium-niobium

The German plant operators participate in the OECD CABRI water loop project in France to complete the experimental database for higher burn-ups and for representative cooling conditions of the fuel elements. However, due to delays, first results of the experiments are not expected before 2009. The data expected shall also serve the validation of the computer codes.

Fuel element behaviour at higher burn-up

The continuous improvement of manufacturing and materials allows a continuous increase of fuel element burn-up. In this respect, it is always to be checked to which extent the protection goals and the acceptance criteria are fulfilled. The corresponding verifications on fulfilment of the accident criteria are to be performed. For the validation of the models used, the manufacturers performed experimental studies at the research centres Karlsruhe and Studsvik (Halden Reactor Project). If accepted ranges are to be exceeded, corresponding investigations are required and will be performed.

Boron dilution

Small leak events may lead to a decrease of boron concentration in the core as a result of heat removal in reflux condenser mode. For the refuellings performed until now, qualified proof on sufficient boron concentrations at the core entrance has been furnished under conservative assumptions. In case of increased requirements due to modifications of the core design, corresponding verifications have to be performed.

With regard to the analysis methods, further activities are planned to clarify details, particularly in connection with mixing processes (mixing of deborated and highly borated fluid in the RPV downcomer and in the lower plenum).

Software-based instrumentation and control (I&C)

At present, software-based I&C is used at German nuclear power plants for functions that are not assigned to the highest safety relevance (i.e. without direct significance for reliable design basis accident control but for accident prevention). These functions are provided, e.g., in the reactor control and limitation system as well as for I&C in auxiliary emergency systems.

Corresponding backfitting measures using digital I&C in the area of reactor and turbine control were performed, e.g., at Biblis A (2003), Isar 2 (2003), Isar 1 (2004), Grohnde (2004), Philippsburg 2 (2005), Krümmel (2005) and Emsland (2005).

A reactor protection system with completely computer-based I&C was installed for the first time at the FRM II research reactor commissioned in 2004. Such a backfit is planned for the power reactors Neckarwestheim 1 and Biblis B. For Neckarwestheim 1, an application has already been filed.

Article 14: Progress and Changes Since 2004

Results from low power and shutdown PSAs mainly led to modifications in the area of administrative specifications.

The PSA guideline [3-74.3] and the associated documents PSA Methods and PSA Data [4-7] were revised and supplemented. Further safety reviews were performed at nuclear power plants.

In July 2004, the RSK issued 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.

Since 2004, the nuclear power plants Emsland (2004), Brokdorf (2005), Biblis (2005), Unterweser (2005) and Krümmel (2006) have been subjected to audits within the framework of WANO peer reviews.

The constantly increasing knowledge and additional requirements imposed by the authorities for the operating times of the nuclear power plants have led to safety-related backfits and improvements of the plants in different subject areas.

Article 14: Future Activities

For further WANO peer reviews, the following proposal on scheduling was made: Grohnde (2007), Gundremmingen (2007), Grafenrheinfeld (2007), Brunsbüttel (2008), Isar (2009), Philippsburg (2009), Emsland (2010), Brokdorf (2010), Neckarwestheim (2011), Biblis (2011), Unterweser (2011) and Krümmel (2011).

Within the framework of safety reviews, PSAs of Level 2 will also be performed in future.

In future, the plant operators will submit annual plant-specific reports on ageing management.

15 Radiation Protection

ARTICLE 15 RADIATION PROTECTION

Each Contracting Party shall take the appropriate steps to ensure that in all operational states the radiation exposure to the workers and the public caused by a nuclear installation shall be kept as low as reasonably achievable and that no individual shall be exposed to radiation doses which exceed prescribed national dose limits.

Basic regulatory requirements

The Radiation Protection Ordinance [1A-8] is the legal basis for the handling of radioactive material. Over the years, it has repeatedly been amended and adapted to the respective EURATOM Basic Safety Standards [1F-18] which prescribe the framework for radiation protection in the European Union. The ordinance 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 defined according to Section 7 of the Atomic Energy Act [1A-3]. The Radiation Protection Ordinance specifies requirements for organisational/administrative and physical/technical protection measures and for medical surveillance. Moreover, it also specifies licensing obligations for the handling of radioactive material, for their import, export and their transport.

Relevant for performing practices in terms of the Radiation Protection Ordinance are, in addition to the principles of justification and limitation of radiation exposure included therein, the following radiation protection principles specified in Section 6 of the Radiation Protection Ordinance:

- Anyone who plans or performs a practice shall avoid any unnecessary radiation exposure or contamination of man and environment.
- Anyone who plans or performs a practice shall minimize any unnecessary radiation exposure or contamination of man and environment, even if below the respective limit, by taking into consideration the state of the art and by taking into account all circumstances of individual cases.

Together with the principle of proportionality - a constitutional principle to be accounted for in all cases - these principles lead to an obligation to optimise radiation protection in terms of the ALARA principle.

The main dose limits for the annual effective dose, organ doses and the lifetime dose specified in the Radiation Protection Ordinance are addressed in the following and listed in Table 15-1.

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

Sec-tion	Scope of Applicability	Time period	Limit [mSv]
	Design and operation of nuclear installations		
46	**Environment of nuclear installations**		
	Effective dose: external radiation exposure from the installation and contributions from its discharges	Calendar year	1,0
	Organ dose: eye lens	Calendar year	15
	Organ dose: skin	Calendar year	50
47	**Limits for the discharges with exhaust air or waste water during normal operation**		
	Effective dose	Calendar year	0,3
	Organ dose: bone surface, skin	Calendar year	1,8
	Organ dose: gonads, uterus, red bone marrow	Calendar year	0,3
	Organ dose: great gut, lung, stomach, bladder, breast, liver, gullet, thyroid, other organs or tissues unless specified above	Calendar year	0,9
49	**Design basis accident limits**		
	Effective dose	Event	50
	Organ dose: thyroid and eye lens	Event	150
	Organ dose: skin, hands, forearms, feet, ankles	Event	500
	Organ dose: gonads, uterus, red bone marrow	Event	50
	Organ dose: bone surface	Event	300
	Organ dose: great gut, lung, stomach, bladder, breast, liver, gullet, other organs or tissues unless specified above	Event	150
	Dose limits for occupationally exposed persons		
55	Occupationally exposed persons of **Category A**		
	Effective dose	Calendar year	20
	Organ dose: eye lens	Calendar year	150
	Organ dose: skin, hands, forearms, feet, ankles	Calendar year	500
	Organ dose: gonads, uterus, red bone marrow	Calendar year	50
	Organ dose: thyroid, bone surface	Calendar year	300
	Organ dose: great gut, lung, stomach, bladder, breast, liver, gullet, other organs or tissues unless specified above	Calendar year	150
	Occupationally exposed persons of **Category B**		
	Effective dose	Calendar year	6
	Organ dose: eye lens	Calendar year	45
	Organ dose: skin, hands, forearms, feet, ankles	Calendar year	150
	Effective dose for **persons under age 18**	Calendar year	1
	Trainees and students age 16 - 18 with agreement by the authority	Calendar year	6
	Organ dose: uterus of **women of childbearing age**	Month	2
	Foetus	Time of pregnancy	1
56	Effective dose	Entire life	400
58	**Radiation exposure** permitted in exceptional circumstances (only volunteers of Category A, after approval by the authority)		
	Effective dose	Professional life	100
	Organ dose: eye lens	Professional life	300
	Organ dose: skin, hands, forearms, feet, ankles	Professional life	1000
59	Regarding measures for **removal of pending danger to persons** it is to be achieved that an effective dose of more than 100 mSv only occurs once per calendar year and an effective dose of more than 250 mSv only once in a lifetime (only volunteers over age 18).		

Occupationally exposed persons

Regulatory requirements

The radiation exposure of the personnel is limited by the Radiation Protection Ordinance (→ Table 15-1). 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. A foetus shall 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 shall not exceed the life time dose of 400 mSv to ensure that radiation exposure of the personnel during the professional life is limited to an acceptable degree.

Exposures to radiation exceeding these limits per calendar year may be allowed up to 100 mSv in order to perform necessary work under exceptional circumstances. Regarding measures to avert danger to persons it shall be achieved that an effective dose effective dose of more than 100 mSv only occurs once per calendar year and an effective dose of more than 250 mSv only once in a lifetime.

The body doses are determined for persons spending any time in the radiologically controlled area. This is usually done by measuring the personal dose by means of electronic dosimeters of the plant operator and by official passive dosimeters. In addition, the dose due to inhalation is usually determined by monitoring of the airborne activity concentration. Further details are specified in the guidelines for the determination of body doses from external and internal radiation exposure [3-42] and [3-42.1]. Beside the operational dosimetry of the plant operator, an independent official dosimetry of the personnel is performed. This official dosimetry is based on passive dosimeters issued and evaluated by measuring institutions, which are designated by the competent authority. The usually monthly measured values are transmitted by the measuring institutions to the radiation protection supervisor or radiation protection officer of the nuclear installation and to the central Radiation Protection Register.

For occupationally exposed persons, distinction is made between Categories A and B. For persons of Category A, the effective dose limit is 20 mSv per year. Persons of this category are examined by authorised physicians once a year. For persons of Category B, the effective annual dose shall not exceed 6 mSv. Their medical examination is performed as stipulated by the authority. Moreover, a radiation passport is to be maintained for persons working in foreign radiologically controlled areas. The corresponding regulations [2-2] were revised in 2004 and now ensure that exposures from activities outside of nuclear power plants (e.g. during radiography in the conventional industry sector) are also taken into consideration.

The protection of the workers was already considered during the design of the nuclear power plants by implementing the provision of the Radiation Protection Ordinance and subordinate legislation, such as guideline [3-43] and KTA safety standard [KTA 1301.1]. The design-related aspects are also taken into consideration with regard to plant modifications and upgrades. For further development of provisions for modification and design, it is being discussed whether guideline [3-43] should be revised. At an early stage, organisational and technical measures for reduction of the radiation exposure of persons working at the plant were required as specified in KTA safety standard [KTA 1301.2] and in the guideline on radiation protection measures during operation of a nuclear installation [3-43.1].

In the last years, the rules and regulations were updated, placing special emphasis with regard to occupationally exposed personnel on the guidelines for the determination of body doses from external and internal radiation exposure [3-42] and [3-42.1] and the guideline on radiation protection measures during operation of a nuclear installation [3-43.1].

During the revision of the guidelines for the determination of body doses, different threshold values (e.g. for investigations, for the required incorporation monitoring) were adapted under consideration of the limit values of the Radiation Protection Ordinance modified in 2001 and the state of the art in science and technology. Moreover, in the guideline on internal radiation exposure, the calculation methods to be applied were modified and the related nuclide-specific dose coefficients for the different age groups and foetus were recalculated.

With revision of the guideline on radiation protection measures during operation of a nuclear installation [3-43.1], the dose reference levels above which a very detailed radiation protection planning must be made were reduced, i.e. from 50 mSv to 20 mSv for the collective dose and from 10 mSv to 6 mSv for the individual effective dose. Moreover, radiation protection is considered in the planning of all activities at a very early stage. The requirements of this guideline together with the increased radiation protection awareness among the personnel and involvement of the supervisory authorities in the planning of the radiation protection measures and their implementation form a good basis for the implementation of the ALARA concept with the aim of exposure reduction and optimisation at the plants.

<u>Experiences during operation of the nuclear power plants</u>

Collective and individual doses have clearly decreased until about 2000. Then, the construction lines showed different behaviour to some extent. Figure 15-1 shows the average collective doses per year and plant. The exposures at PWR plants of construction line 4 (Konvoi plants) constantly remain at the same low level. This is due to consequent abandoning the use of any materials containing cobalt in almost all components of the primary system. The PWR plants of construction lines 1 and 3 show a decreasing trend of the collective dose which, for construction line 3, is to be attributed to the improvements in radiation protection and small scope of back fitting activities compared to previous years and, for construction line 1, to the pending decommissioning in 2005. For plants of construction line 2, the change between years without any revision activities during outage and years with implementation of dose-intensive back fitting activities led to clear differences from year to year. However, the long-term reduction of the collective doses also becomes observable.

With regard to BWR plants, there is a stabilisation of the collective doses for construction lines 69 at a low level for BWRs while at both plants of construction line 72 slightly increased outage doses led to a minor increase of the collective doses (→ Figure 15-2).

For 2005, Figure 15-2 shows the collective doses of the plants in operation according to the different construction lines for both PWRs and BWRs. Further, it shows the different durations of planned outages and the distribution of the collective dose according to the different modes of operation. Here, it becomes apparent that for all plants the highest annual collective dose occurs during plant planned outage.

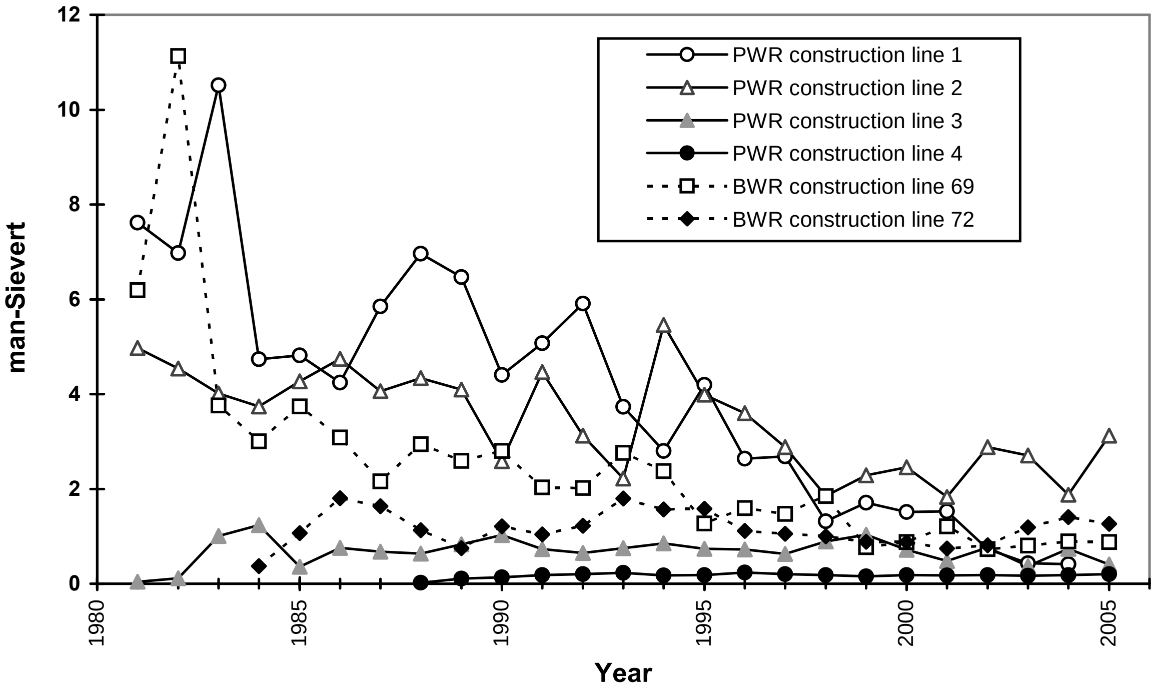
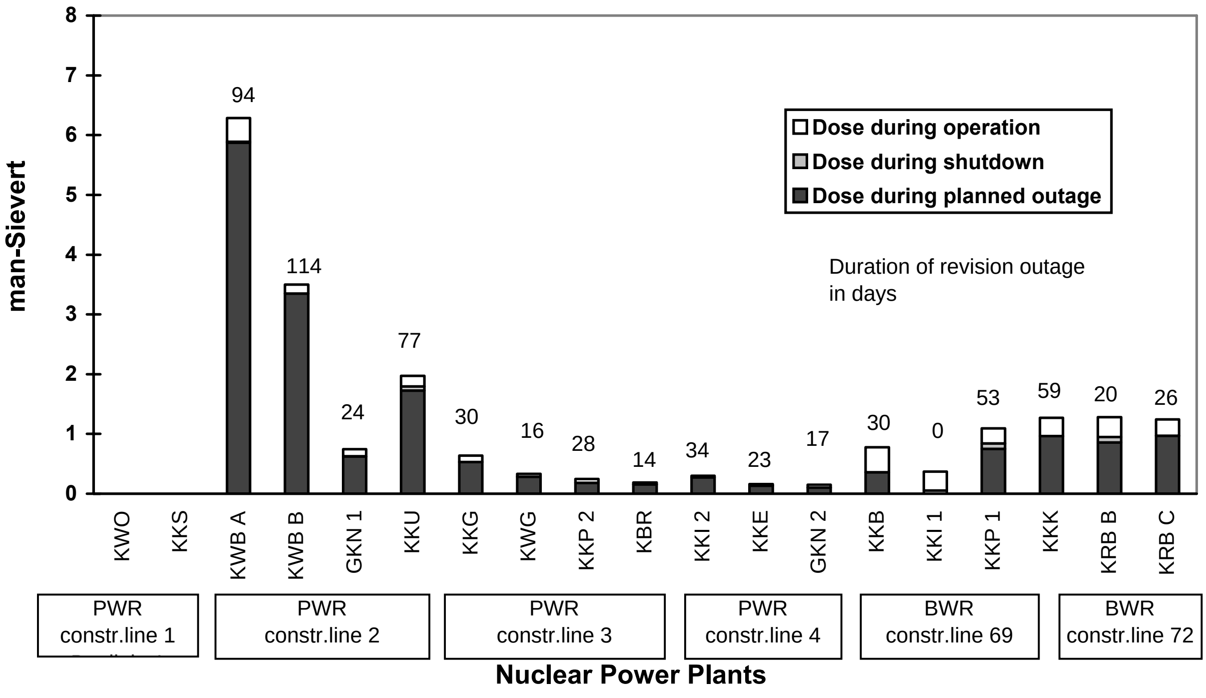


Figure 15-1 **Average Annual Collective Dose per Year and Plant**



Dose during operation: Collective dose during normal power operation
Dose during unplanned outage: Collective dose during plant shutdowns other than planned outages
Dose during planned outage: Collective dose during plant outage (scheduled plant revision and / or refuelling)

Figure 15-2 **Annual Collective Dose in Nuclear Power Plants in 2005 According to Mode of Operation, KWO and KKS Decommissioned**

Plant-specific measures were performed to reduce exposures, such as exchange of the neutron flux measuring lances at BWRs for reducing the dose rate in control rod drive rooms in some cases to 50 % of the initial value and work-specific chemical or mechanical decontamination of vessels and systems. In particular for some BWRs, measures for limiting the steam moisture (and thus reducing entry of activity into the turbine building systems) led to collective dose savings.

In “high dose ranges”, manipulators have increasingly been used: among others, vessel inspections by means of submarine systems for inspection, use of remotely controlled seal grinders at the reactor pressure vessel and use of improved manipulators for eddy current tests at steam generators. Dose-intensive radiographic testing is increasingly replaced by ultra sonic testing with manipulator technology. By means of zinc dosing at PWRs, it was possible to reduce the dose rate in rooms by up to 50 %. For the performance of outages, the newly introduced methods for the reduction of activity releases from opened systems into the room air (tarpaulin for reactor cavity of PWRs, chemical dosing for iodine retention) proved to be effective.

In the ten-year period of 1997 - 2006, 19 nuclear power plants were in operation which corresponds to 185 reactor operating years. From the plants in operation or permanently shutdown, a total of 1,316 events was reported from 1997 to 2006 which were reportable according to the criteria of the Nuclear Safety Officer and Reporting Ordinance [1A-17] (→ Article 19 (vi)). 45 of these events led to radiological impacts, but did not cause any excess of permissible personal doses.

Emission monitoring and emissions

Emission monitoring

According to Section 47 of the Radiation Protection Ordinance [1A-8], any uncontrolled release of radioactive material into the environment must be avoided. The basis for monitoring and balancing of the emissions is established in Section 48 of the Radiation Protection Ordinance. The programmes for emission monitoring during specified normal operation and in case of design basis accidents are in accordance with the Guideline on Emission and Immission Monitoring [3-23] and KTA safety standards [KTA 1503.1], [KTA 1503.2], [KTA 1503.3] and [KTA 1504]. The operators of nuclear installations perform these monitoring measures and submit the results to the supervisory authorities.

The sampling and measurement methods are oriented toward the two tasks of monitoring by continuous monitoring on the one hand, and sampling for balancing the discharge of radioactive material via the paths exhaust air and waste water according to type and amount on the other hand.

Continuous measurement is performed to monitor the discharge of the nuclides or nuclide groups with exhaust air for radioactive noble gases, radioactive aerosols and for iodine-131 and with waste water for gamma-emitting nuclides. For the determination of releases that may occur as a result of design basis accidents, instruments with extended measurement ranges are applied. In addition to the measuring instruments of the plant operators, there are also instruments of the supervisory authorities whose data are transmitted online via the KFÜ data network.

The balancing of the discharge with exhaust air comprises the following nuclides and nuclide groups: radioactive noble gases, radioactive aerosols, radioactive gaseous iodine, tritium, radioactive strontium, alpha emitters and carbon-14. For the waste water path, gamma-emitting nuclides, radioactive strontium, alpha emitters, tritium, iron-55 and nickel-63 are balanced. Reports on the balanced discharges are generally submitted to the supervisory authority every quarter as well as yearly [KTA 1503.1], [KTA 1504].

The external radiation from the plant is monitored by dose rate measurements at the fence.

According to the guideline on the control of the radiation measurement programme performed under the responsibility of the plant operator [3-44], the Federal Office for Radiation Protection performs a programme to control the operator’s measurement programme. For that, for controlling the monitoring of emissions with exhaust air, control measurements are performed on aerosol filter samples, iodine filter samples, tritium samples and carbon-14 samples and comparative measurements at the plant for determining the emission of radioactive noble gases. For controlling the monitoring of emissions with water, samples are analysed for gamma-emitting nuclides, tritium, strontium and alpha emitters. The results of the control measurements are submitted to the supervisory authorities. According to the above-mentioned guideline [3-44], the plant operators are also obliged to participate in round-robin tests. By means of round-robin tests, comprehensive quality control can be ensured.

In order to be able to evaluate the consequences of the discharge of radioactive material, the plant operator records the site-specific meteorological and hydrological parameters important to the dispersion and deposition of radioactive material. The requirements for meteorological Instrumentation are included in KTA safety standard [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.

Emissions

The discharge of radioactive material is permitted with the operating licenses. The licensing authorities stipulate maximum permissible activity amounts and concentrations for discharges that are calculated such that, under consideration of the site-specific dispersion conditions and exposure pathways, the potential radiation exposure for members of the public resulting from discharges to the extent of the permissible activity amounts and concentrations does not exceed the limits of Section 47 of the Radiation Protection Ordinance (→ Table 15-1). Together with the contribution by external radiation, the limits of Section 46 of the Radiation Protection Ordinance shall not be exceeded.

Section 6 of the Radiation Protection Ordinance stipulates that discharges of radioactive material shall be kept as low as possible taking due account of the state of the art and paying attention to the merits of each individual case, even where the values are below the limits of the operating license. Thus, high demands are placed on the quality of the fuel elements, the composition of the materials and the purity of the water used in the primary system for activity limitation and for preventing the contamination of components and systems. In addition, the plants are equipped with devices for the retention of radioactive material.

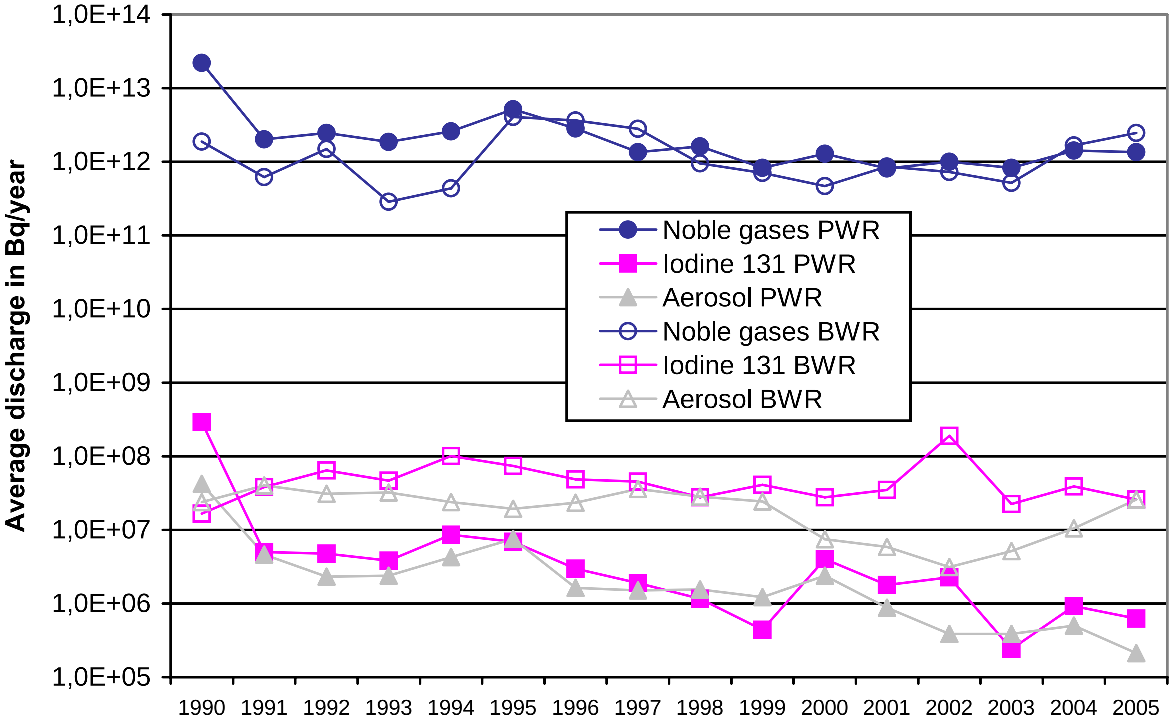


Figure 15-3 **Average Annual Discharge with Exhaust Air from PWRs and BWRs in Operation**

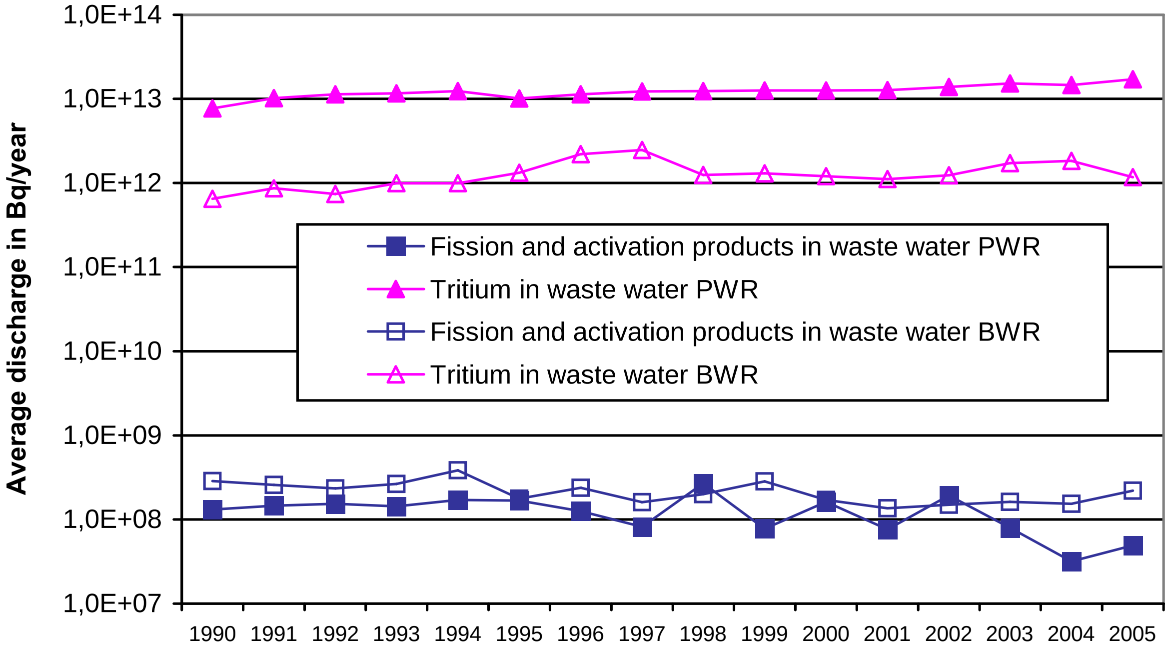


Figure 15-4 **Average Annual Discharge with Waste Water from PWRs and BWRs in Operation**

Except for tritium, the annual discharges reach just a few percentage points of the permitted values. Increased discharges and uncontrolled releases (i.e. releases through pathways not provided for controlled release) occur only very rarely. Within the reporting period (2004 - 2006), only three of these cases were reported to the authorities according to the radiological reporting criteria [1A-17]. In one case in 2004, there was an excess of the permitted values related to discharges with waste water. It was a slight excess of the moving half year permitted value for the discharge of tritium at the Biblis A nuclear power plant without having reached the annual permitted value. In the other two cases, radioactive material was released to a very small amount, far below the permitted values for discharges, with water into the receiving water through a pathway not provided for it.

The data on discharges of radioactive material with exhaust air and water 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. Figures 15-3 and 15-4 show the average annual discharges from German nuclear power plants.

Radiation exposure of the public

<u>Radiation exposure of the public during specified normal operation</u>

The dose limits and requirements applying to the radiation exposure of the public during operating conditions of nuclear installations are laid down in Sections 46 and 47 of the Radiation Protection Ordinance (→ Table 15-1).

A limit of 1 mSv per calendar year is specified for the effective dose by external radiation and the radiation exposure from discharges. In addition, there are limits for specified organs and tissues. For determining the exposure from external radiation, permanent stay is to be assumed in the plant design unless there are justifications for times of stay deviating from it.

The contributions of discharges to exposure are limited by Section 47 of the Radiation Protection Ordinance. For planning, construction and operation of nuclear installations, a maximum effective dose of 0.3 mSv per calendar year is applicable to radiation exposures of members of the public resulting from discharges of radioactive material with exhaust air or with waste water each. Further limits apply 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 shall be calculated for a reference person and all exposure pathways at the most unfavourable receiving points such that the radiation exposure to be expected will not be underestimated.

The results show (→ Figure 15-5 to 15-7) that the discharges with exhaust air only lead to doses in the range of a few µSv per year due to the measures of the plants in operation, filtering and only small fuel element defects. The relevant limits of 0.3 mSv for the effective dose and 0.9 mSv for the thyroid dose are only reached to a fractional amount for the highest exposure groups. For waste water, the resulting exposures are even lower with values, in general, of less than 1 µSv. Except for the first years, the time histories of the emissions with exhaust air and the results of the calculation of the doses of the public (→ Figure 15-7) do not show a direct correlation as due to the very low emissions, the dose is dominated by the discharged carbon-14 for which detection methods and balancing have been changed and improved over the years.

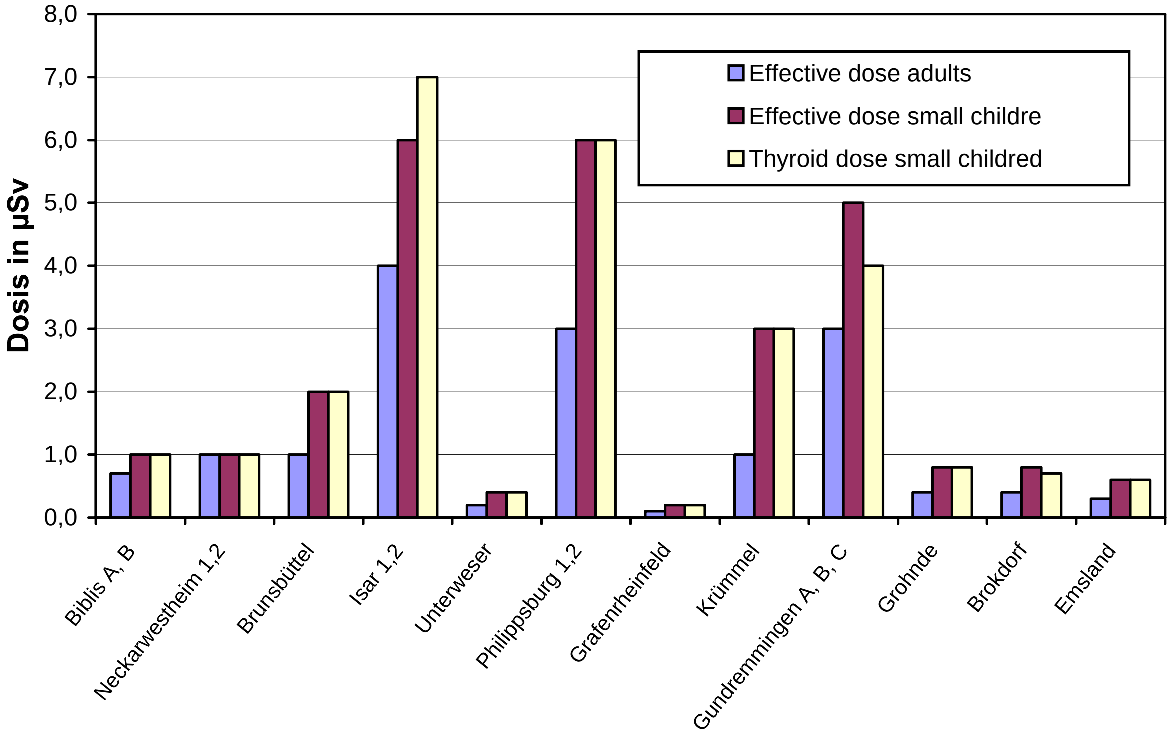


Figure 15-5 **Dose from Discharges with Exhaust Air from Plant in Operation in 2005**

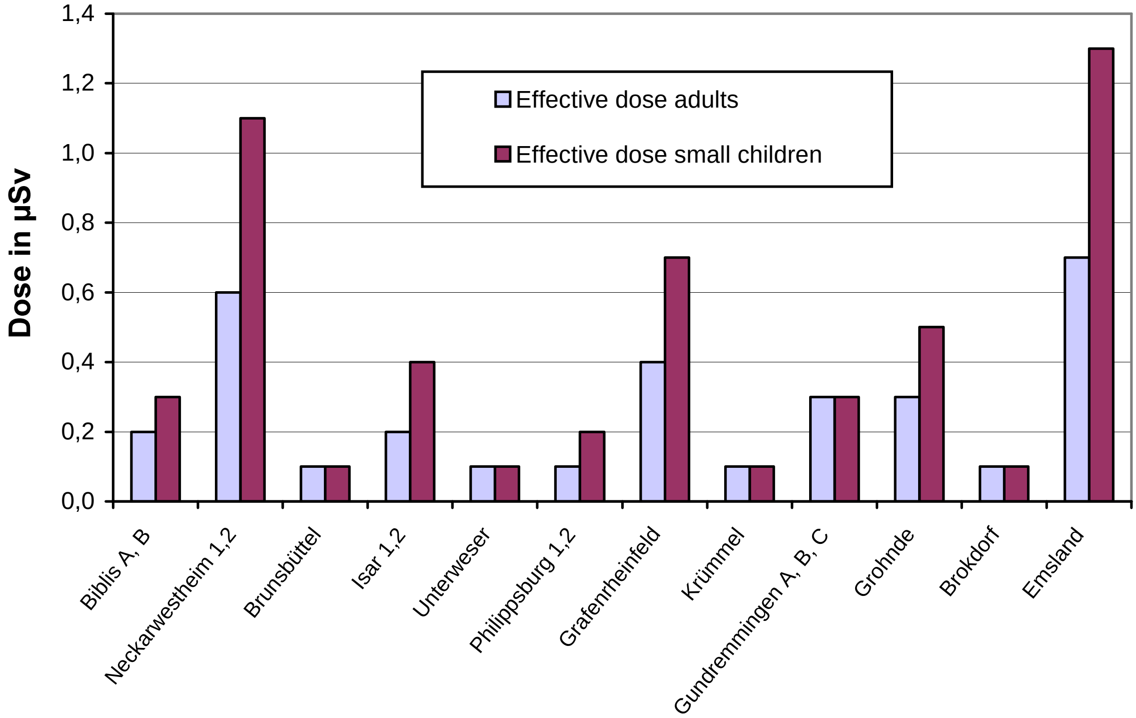


Figure 15-6 **Dose from Discharges with Waste Water from Plant in Operation in 2005**

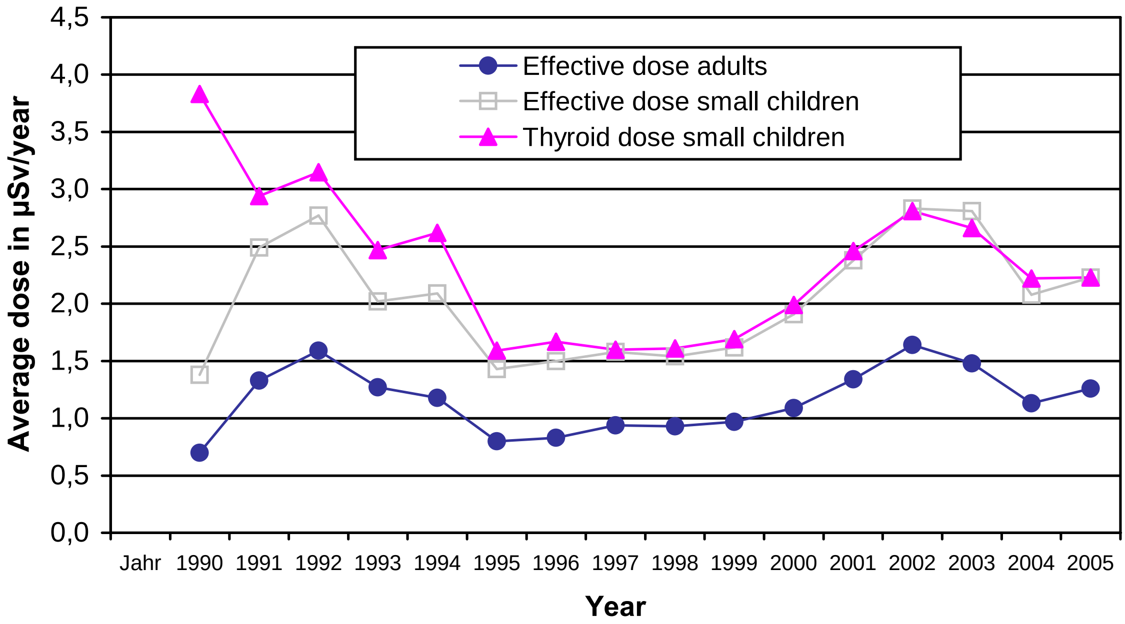


Figure 15-7 **Average Dose from Discharges with Exhaust Air from all Plants in Operation**

Radiation exposure of the public in case of design basis accidents

Central issues evaluated during the licensing procedure of a nuclear power plant are the planned structural and technical measures for the control of design basis accidents (→ Article 18 (i)). In accordance with Section 49 of the Radiation Protection Ordinance it has to be shown that, under consideration of the requirements of Section 6 of the Radiation Protection Ordinance (prevention of unnecessary radiation exposure, ALARA principle), 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 and 70-year dose commitment). Further planning values apply to specified organs and tissues. Regulatory guideline "Incident calculation bases" [3-33] specifies the analytical models and assumptions to be applied for these verifications.

Radiation exposure of the public in case of beyond-design basis accidents

Due to the design of the plants, these accidents are very improbable. Specification of dose limits and reference values as set targets for the protection of the public is not practicable. Instead, among others confirmed by the results of risk studies and probabilistic safety analyses, organisational and technical measures were taken within the framework of plant-internal accident management for the protection of the public in order to prevent accidents or at least to mitigate their consequences on-site and off-site the plant (→ Article 18). This is to prevent radiological situations which require drastic actions, such as evacuations or long-term resettlements. Notwithstanding this on-site emergency response, additional measures can be taken, if required, for the protection of the public within the framework of off-site emergency planning (→ Article 16) if there are significant releases or the risk of such releases.

Immission monitoring and immissions

Immission monitoring

According to Section 48 of the Radiation Protection Ordinance [1A-8], the plant operators perform a programme on immission monitoring in the vicinity of the plant as ordered by the authorities. In addition, measurements are performed by independent measuring institutions on behalf of the authority.

Immission monitoring supplements emission monitoring. It allows additional controls of the discharges and controls to verify compliance with the dose limits in the vicinity of the plant. The Guideline on Emission and Immission Monitoring [3-23] specifies programmes for immission monitoring prior to commissioning, during specified normal operation, during incidents or accidents and in the phase of decommissioning and safe enclosure. These programmes are to be implemented by the plant operator and the independent measuring institution. Site-specific circumstances and conditions are considered additionally.

Measurements prior to commissioning comprise the still uninfluenced environmental radioactivity and radiation exposure. Monitoring measures during operation serve, among other things, to monitor long-term changes that may occur due to the discharge of radioactive material. Incident and accident measurement programmes, set up in advance, are the basis for taking samples and for the measurements and evaluations in the event of a design basis accident or beyond-design basis accident. Immission monitoring considers exposure pathways that may lead to radiation exposure of the public. The sampling and measurement methods ensure that relevant dose contributions by external exposure, inhalation and ingestion can be identified during specified normal operation and can be determined in the case of design basis or beyond-design basis accidents.

The results of immission monitoring are submitted to the authority. Data on immissions are centrally recorded, evaluated and published at the BfS.

Immissions

Immissions resulting from discharges with exhaust air are not detected in the environment even by using the most sensitive analysis methods. The analysis of the ground level air, the precipitation, the soil, the vegetation and the foodstuffs of vegetable and animal origin shows that the content of long-lived radioactive substances, such as caesium-137 and strontium-90 does not differ from the values measured at other locations in Germany. Short-lived nuclides that might originate from discharges from plants with exhaust air also are not detected.

In surface waters, immissions of the water pathway are detected in individual cases. Occasionally, tritium is measured in samples directly taken at discharge structures. The values are mostly below 100 Bq/l. The nuclide contents of other fission and activation products are generally below the detection limit required for these analyses. The content of long-lived radioactive substances, such as caesium-137 and strontium-90 does not differ from the values measured at other locations in Germany also in this case. Also in sediment samples, the average radionuclide contents are below the required detection limits. In only a few samples directly taken at discharge structures, cobalt-60 can be detected in a small concentration. In 2004, values of 0.5 to 3 Bq/kg dry matter were detected at three sites. No radioactive material was found in fishes, aquatic plants and ground and drinking water that could be attributed to the operation of nuclear power plants. The increase of contents of fission and activation products caused by discharges of radioactive material with water is thus negligibly small.

Remote monitoring of nuclear power plants (KFÜ)

In addition to the radiation measurement programme performed under the responsibility of the licensee, the supervisory authorities of the *Länder* have their own systems for continuous acquisition of measurement data regarding emission and immission behaviour of the plant (*Kernkraftwerk-Fernüberwachungssystem* (KFÜ)). Together with the fast transfer of operational data, this continuous monitoring is an effective instrument of regulatory supervision according to Section 19 of the Atomic Energy Act.

The basic requirements for the remote monitoring system are laid down in the basic recommendations for the remote monitoring system for nuclear power plants [3-54]. The actual details are specified under the responsibility of the respective supervising *Land*.

Main function of the KFÜ is the continuous emission monitoring which is partly designed redundant to the radiation measurement programme performed under the responsibility of the plant operator and the immission monitoring in the vicinity of the plants. Further, meteorological data are continuously transmitted to the supervisory authority. Different operating parameters give indications to the operating status of the plants.

The use of the data acquired within the KFÜ mainly cover the regulatory supervision of the operational processes and automatically initiated alerting of the supervisory authority in the case of excess of permitted values. Further processing of these data in connection with meteorological factors in suitable computer codes allows assessing and predicting of the radiological exposure in the vicinity of the plants, in particular after release of radioactive material in case of incidents or accidents. Thus, the results also serve for the purposes of emergency response.

Monitoring of environmental radioactivity/Integrated measurement and information system

In addition to the site-specific monitoring of the vicinities of the nuclear power plants according to the Guideline on Emission and Immission Monitoring [3-23], the general radioactivity in the environment is recorded by extensive measurements in the entire territory 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]. This system is operated by the BfS. Monitoring comprises all relevant environmental areas from the atmosphere and the surface waters up to sampling of foodstuffs and drinking water. Core piece is the network comprising more than 2,000 measurement stations for measuring the local gamma dose rate. All data measured 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. IMIS is permanently in operation.

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. In the event of increased values in the territory of the Federal Republic of Germany, 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 extent and procedures for the corresponding measurements are specified in the general administrative provisions [2-4] for routine and intense operation. The results from these measurements are also used within the framework of international information exchange (→ Article 16 (2)). At present, the measured values of airborne activity and the local gamma dose rate in Germany are displayed in maps placed in the Internet (www.bfs.de) and updated on a weekly basis.

Article 15: Progress and Changes Since 2004

Within the reporting period, the guidelines for the determination of body doses [3-42] and [3-42.1], on the planning of radiation protection measures during activities at nuclear installations [3-43.1] and the general administrative provisions relating to the radiation passport
[2-2] were revised and further developed in order to improve protection of the personnel.

For improved monitoring of the discharge of radioactive material into the environment, the Guideline on Emission and Immission Monitoring [3-23] and the basic recommendations for the remote monitoring system [3-54] were updated. Regarding the monitoring of environmental radioactivity, the revised version of the general administrative provisions related to the Integrated Measurement and Information System for the Monitoring of Environmental Radiation (IMIS) [2-4] was passed in 2006. It replaces the old version of 1995 and considers technical and organisational changes after extensive innovations in the IT area as well as progress in the field of instrumentation and measurement techniques and adaptations due to changes in the environmental media.

Moreover, modifications were made in individual cases due to the revision of the guideline on technical qualification in radiation protection [3-40] which are applicable in the non-nuclear sector but are of relevance for radiography institutions or also for external storage facilities for radioactive material (except nuclear fuel).

The incident calculation bases were revised in 2003. The revised version is currently available as a recommendation of the Commission on Radiological Protection [3-33].

Article 15: Future Activities

Current and future activities concern the introduction of electronic personal dosimeters in the German official dosimetry and the update and further elaboration of general administrative provisions for calculating the radiation exposure of the public resulting from the discharge of radioactive material and for incidents during the decommissioning of nuclear facilities.

16 Emergency Preparedness

ARTICLE 16 EMERGENCY PREPAREDNESS

1. Each Contracting Party shall take the appropriate steps to ensure that there are on-site and off-site emergency plans that are routinely tested for nuclear installations and cover the activities to be carried out in the event of an emergency.
For any new nuclear installation, such plans shall be prepared and tested before it commences operation above a low power level agreed by the regulatory body.

2. Each Contracting Party shall take the appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.

3. Contracting Parties which do not have a nuclear installation on their territory, insofar as they are likely to be affected in the event of a radiological emergency at a nuclear installation in the vicinity, shall take the appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.

Structure and objectives of emergency preparedness

Nuclear emergency preparedness comprises on-site and off-site planning and preparedness for emergencies (→ Figure 16-1).

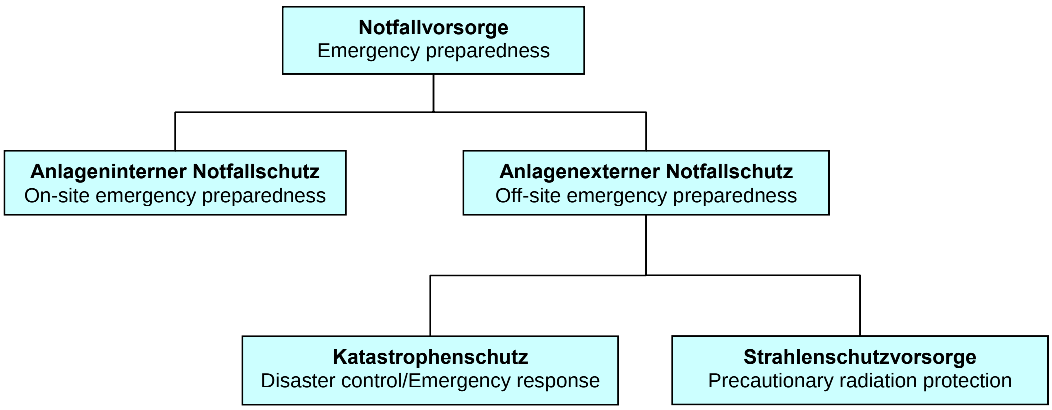


Figure 16-1 Structure of Emergency Preparedness

On-site emergency preparedness is realised by technical and organisational measures taken at nuclear power plants to control an event or to mitigate its consequences.

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

In Germany, the International Nuclear Event Scale (INES) is used for the classification of events at nuclear plants with or without radiological significance. The classification of the different event groups of the INES according to the categories of off-site emergency preparedness, i.e. disaster control and precautionary radiation protection, is included in Table 16-1.

Table 16-1 Grouping of Events for Off-site Emergency Preparedness

	Event	Classification according to INES scale	Classification disaster control vs. precautionary radiation protection
domestic	Incident	3	Precautionary radiation protection
	Accident	4 to 7	Disaster control (local site area) Precautionary radiation protection
abroad	Incident (neighbouring foreign country)	3	Precautionary radiation protection
	Accident (neighbouring foreign country)	4 to 7	Disaster control (local site area) Precautionary radiation protection
	Accident (far away foreign country)	4 to 7	Precautionary radiation protection

16 (1) Emergency Preparedness, Emergency Plans

Tasks and competencies

On-site emergency preparedness is a duty of the operator of a nuclear installation. Off-site emergency preparedness falls within the competence of the authorities of the *Länder* and the Federal Government (→ Figure 16-2).

Operator of the nuclear installation

According to the protection provisions of the Atomic Energy Act [1A-3] und Section 51 of the Radiation Protection Ordinance [1A-8], the operator is responsible - within the framework of on-site emergency preparedness - to keep the risk of potential hazards for man and the environment as low as possible in case of incidents and accidents. The measures of the operator are divided into preventive and mitigative measures. Main objective of the preventive measures is to reach and maintain a plant condition which cannot lead to dangerous consequences. The mitigative measures serve for limiting consequences.

In case of an emergency, the operator immediately informs the competent authorities. The operator is obliged to make information necessary for averting danger available to the authorities in time and appropriate to the situation, to support the authorities in assessing the situation and to advise and support them in taking decisions on protective actions for the public.

The emergency plans of the plant operators' ensure that these measures can be taken without any undue delay.

Authorities of the *Länder*

Pursuant to Article 70 of the Basic Law [1A-1], averting of danger by *disaster control* is a task of the *Länder* which, to this end, passed the disaster control laws. The implementation falls under the responsibility of the authorities of the interior of the *Länder* and, depending on the respective *Land*, is delegated to the regional or also to the local level. The nuclear supervisory authorities and the radiation protection authorities of the *Länder* provide their support (→ Figure 16-2).

Authorities of the Federal Government and the *Länder*

The BMU co-ordinates the measures of the *Länder* if more than one *Land* is affected by an accident. In case of need, the BMU makes its resources, including those of the BfS or its advisory committees RSK and SSK, available for providing support and advice to the *Länder*.

The nationwide co-ordination of disaster control planning by recommendations agreed upon under the management of the BMU is jointly performed by the Federal Government and the *Länder*. This includes the Basic Recommendations for Emergency Preparedness in the Environment of Nuclear Facilities [3-15.1.], the Radiological Bases for Decisions on Measures for the Protection of the Population against Accidental Releases of Radionuclides [3-15.2.] and the Recommendations for the Planning of Emergency Control Measures by the Licensees of Nuclear Power Plants [3-31], [3-32].

Within the framework of *precautionary radiation protection*, the Federal Government is authorised to specify limits and measures. However, as far as events with exclusively regional impact are concerned, the *Land* authority competent for precautionary radiation protection may determine measures to be taken for preventive health protection. By means of the Integrated Measurement and Information System for the Monitoring of Environmental Radiation (IMIS) [2-4], the Federal Government monitors and assesses the radiological situation in Germany both during routine operation and under incident and accident conditions, but in this case, measurements and samples will be taken more frequently (→ Article 15).

The BMU is responsible for the fulfilment of the international information and reporting obligations, e.g. for the implementation of the Convention on Early Notification of a Nuclear Accident [1E-6], the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency [1E-6] and the information exchange for radiological emergencies according to bi-lateral agreements.

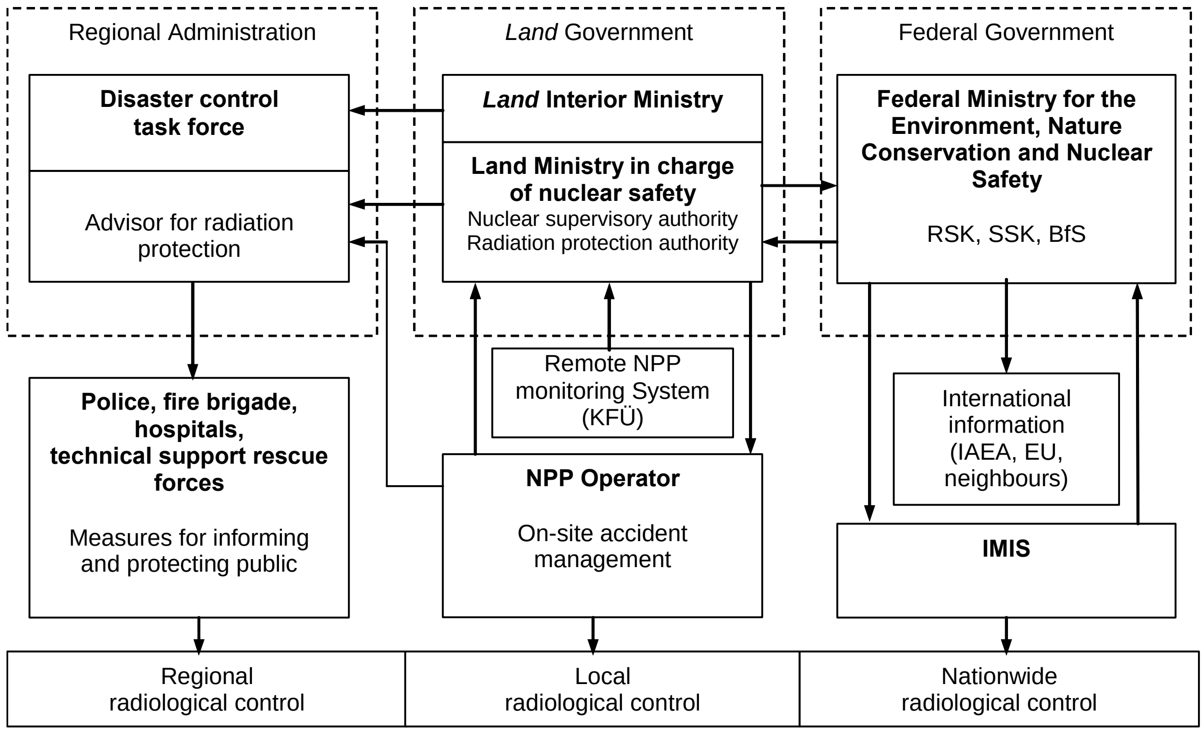


Figure 16-2 Emergency Preparedness Organisation

Rules and regulatory requirements

Based on the regulations of the Atomic Energy Act [1A-3], the Precautionary Radiation Protection Act [1A-5], the Radiation Protection Ordinance [1A-8] and the disaster control laws of the *Länder*, planning of emergency preparedness is described by the subordinate regulations and by recommendations.

The measures to cope with emergencies (→ Article 19 (iv)) implemented by the plant operator and laid down in the alarm regulation contained in the operating manual and the accident management manual are based on recommendations of the RSK and a joint recommendation of RSK and SSK [4-2] which includes the alarm criteria which, when reached, require alerting of the disaster control authorities. In addition, there are the reporting criteria of the Nuclear Safety Officer and Reporting Ordinance [1A-17] for the events to be reported to the supervisory authority (→ Article 19 (vi)).

In off-site emergency preparedness, the required planning scope of disaster control is established by the Basic Recommendations for Emergency Preparedness in the Environment of Nuclear Facilities [3-15.1.]. Principles and explanations are described in the associated Radiological Bases for Decisions on Measures for the Protection of the Population against Accidental Releases of Radionuclides [3-15.2.]. As a recommendation jointly prepared by the Federal Government and the *Länder*, the “Basic Recommendations” [3-15.1.] form the basis for planning of disaster control in the vicinity of the plant. They determine, among others, the planning areas, measures and further provision of the authorities and the documents required. In recent years, they were revised and have been updated now. After agreement by the responsible committees, which is expected for 2007, they will be newly published.

Besides the adaptations due to updates of the nuclear rules and regulations, in this revision, importance was attached to the consideration of “events with rapidly developing accident sequences“ and to the development of an improved communication and information concept.

The Precautionary Radiation Protection Act [1A-5] stipulates the competencies of the authorities of the Federal Government and the *Länder* in precautionary radiation protection. It specifies the responsibilities in the event of a not insignificant release, i. e. a significant release, of radioactive material 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 (Integrated Measurement and Information System for the Monitoring of Environmental Radiation, 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, feedingstuffs, drugs or other substances, and
- authorisations concerning cross-border traffic.

While after the reactor accident at Chernobyl, the European Union specified limits of radioactivity in foodstuffs and feedingstuffs immediately to be applied by the EU Commission in a radiological emergency [1F-30], [1F-31], general administrative provisions [2-5] and [2-6] were passed at the national level for verifying the compliance with the limits.

A guideline important for determining the situation is the Guideline on Emission and Immission Monitoring [3-23] which specifies, in addition to the necessary measurements during normal operation, kind and scope of the measuring tasks in case of incidents and accidents (→ Article 15).

Emergency plans and alerts

The alarm regulation of the plant operator includes the regulations on alerting in emergencies. It is part of the operating manual and belongs to the safety specifications. For coping with emergencies, the plant operator establishes a crisis management team. The individual organisational regulations are described in a separate document, the accident management manual (→ Article 19 (iv)). In their entirety, the regulations mentioned represent the emergency plan of the plant operator, which includes, among others,

- measures to make emergency organisation operable,
- criteria for alerting the responsible authorities,
- technical measures for prevention and mitigation of damages,
- measurement programmes for determining the radiological situation, and
- measures for efficient communication with the responsible authority and for informing the public.

Assistance is provided by the crisis management team of the plant manufacturer and by the Kerntechnischer Hilfsdienst GmbH (a permanent organisation jointly installed by the operators of German nuclear power plants). The crisis management team of the manufacturer advises the plant operator in technical questions of situation assessment and restoration of safe plant condition, while the Kerntechnische Hilfsdienst with its manipulators and measurement equipment may be employed at the site inside and outside the plant. In addition, contractual agreements exist between the plant operators on mutual support.

The responsible disaster control authorities prepare special disaster control plans for the vicinity of the plants. Primary objective of the planning of disaster control is, in case of accidental release, to prevent or mitigate direct consequences from the accident on the public. The content of the planning in the 25 km zone is based on the Basic Recommendations [3-15.1.]. The disaster control plans focus on the co-action of the planning of the disaster control authorities and of measures of the plant operator and on the implementation of the measures for protection of the public.

In the previous, but now revised version of the Basic Recommendations for Emergency Preparedness in the Environment of Nuclear Facilities [3-15.1.], the planning zones in which the distribution of iodine tablets to children and youths under the age of 18 and to pregnant women is to be prepared covered an area of up to 25 km. In the now revised version these planning zones are amended by a long distance zone up to 100 km. Moreover, part of the planning are the measurements required for determining the situation.

Decisions going beyond this planning in terms of space and time are taken within the framework of precautionary radiation protection by the emergency organisation of the BMU In this respect, the measure strategies and reference values as defined in the Catalogue of Measures [4-3] serve as decision basis. In the Catalogue of Measures, the recommendations of the Radiological Bases [3-15.2.] and the maximum permitted levels of the EU n regarding the radioactive contamination of foodstuffs and of feedingstuffs [1F-30] [1F-31] are considered. If necessary, disaster control measures are also implemented by the disaster control authorities outside the planning area thus complementing the measures of precautionary radiation protection.

An important aspect of planning is the information transfer between the authorities and, in particular, the alerting of the authorities by the plant operator. In this respect, RSK and SSK recommended criteria for alerting the disaster control authority by the operator of a nuclear installation [4-2], [4-2.1]. According to these, the plant operator defines in the alarm regulation plant-specific emission and immission criteria and technical criteria for early warning or an emergency alert which, when reached, require alerting the disaster control authorities with specification of the respective alert level. Here, the technical criteria, e.g. very high temperature or low water level in the RPV, are of particular importance, since they give an early indication to the violation of safety objectives and allow rapid alerting. In addition, alerting the disaster control authorities is also possible by the responsible supervisory authority.

A special disaster control planning is performed in agreement with the neighbouring countries concerned for such foreign nuclear power plants which may require disaster control measures in German territory due to their location near the border.

Situation assessment

The determination of the situation is performed at a radiological situation centre with the available information about plant state, meteorological situation and emission and immission situation. First, it is based on prognoses and later increasingly on measurement in the surrounding area.

In the pre-release phase, the radiological situation to be expected in the vicinity of the plant is estimated on the basis of forecast data of the source term and the meteorological situation. Use is made of the decision support system RODOS of the BfS in combination, where appropriate, with the remote monitoring system for nuclear power plants (KFÜ) of the *Land* (→ Article 15). As an alternative, specific systems are applied by the individual *Länder*. RODOS is able to calculate local and regional consequences of releases as well as the effect of protective actions, thus making available situation information and impact assessment to the authorities as decision support. Data on the source term are provided by the operator based on his expectation on the situation. Meteorological data required for the systems result from data measured at the site with KFÜ and the numerical weather forecast of the German Meteorological Service, the Deutscher Wetterdienst (DWD).

In the release phase, the plant operator determines the source term, also additional data of the KFÜ may be available. In this phase, there are also data for assessment of the radiological situation available. These data will be obtained from the local dose rate probes of the KFÜ and from the integrated measuring and information system IMIS, both permanently installed in the vicinity of the plant; in addition, as the case may be, first data of measuring teams will be available. Here, again, the decision support systems described are applied. As soon as data are available according to the measurement programmes provided (→ Figure 16-3), the situation predicted is checked and adapted to the situation determined by measurements.

In the post-release phase, the measurement and sampling services of the plant operator and of the authorities (by independent measurement organisations) provide data for the determination of the radiological situation, in accordance with the requirements of the Guideline on Emission and Immission Monitoring [3-23], supplemented by simple follow-up measurements of radiation detection teams. The soil contamination in the more distant surroundings of the plant and the identification of areas with increased dose rate (hot spots) is shown by means of aircraft hosted gamma spectrometry. All involved teams performing measurements are led by the radiological situation centre.

The development of the wide-range radiological situation in Germany is determined and presented by means of the IMIS which provides information used as support in taking decisions on measures of precautionary radiation protection.

The necessity to inform a large number of authorities and organisations about the current situation in case of a radiological event at short notice and in an effective manner led to the nationwide introduction of the internet-based situation display system ELAN by which situation information and additional data and information are provided for the competent authorities and organisations connected to the system through a secured server connection.

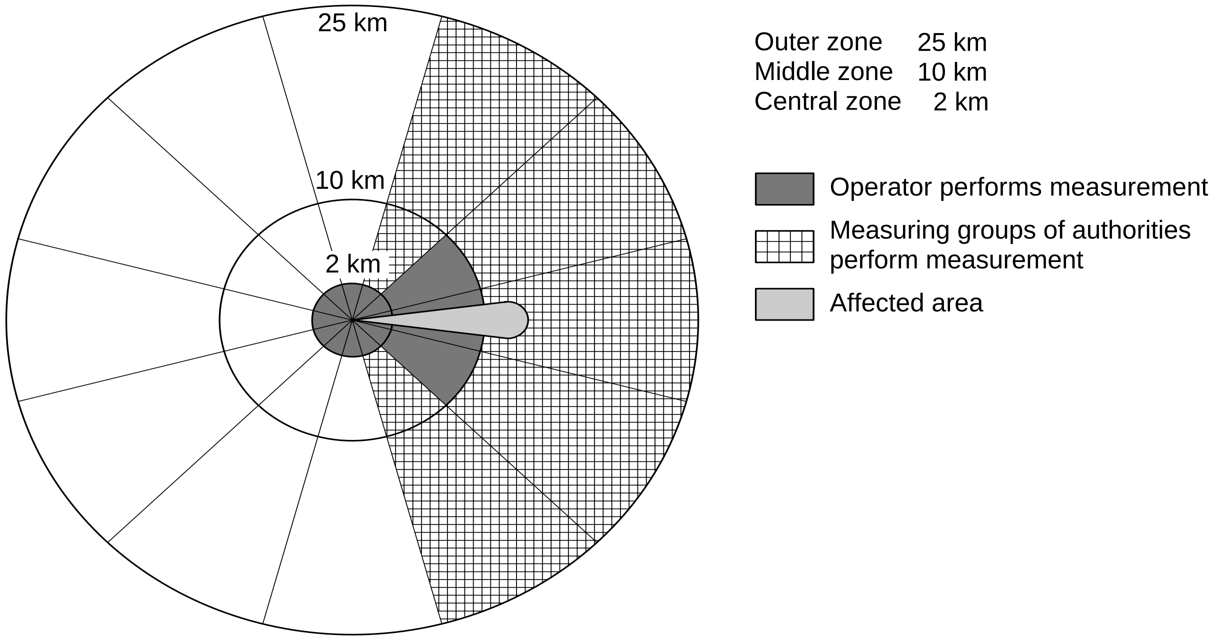


Figure 16-3 Areas of the Different Measuring and Sampling Teams

In addition to the computer-based system RODOS, two documents are available: the "Guidance for the expert advisor for radiation protection of disaster control management in case of nuclear emergencies" [4-4] with the associated explanatory report [4-4.1] and the so-called Catalogue of Measures [4-3] "Survey of Measures for the Reduction of Radiation Exposure after Events with Significant Radiological Consequences" (Vol. 1 and 2) which provide additional help and support.

The "Guidance for the expert advisor for radiation protection" especially aims at the situation assessment within the disaster control and was republished in 2004 under consideration of updates of the supporting documents and data. It was also provided as computer-based version referred to as PLUTO.

In addition to disaster control measures, the Catalogue of Measures [4-3] also deals with preventive health protection and here especially with measures in the area of agriculture. It documents, among others, derived target and reference values as decision basis. At present, the Catalogue of Measures is being revised. Besides information on decision strategies in the agricultural area, it will include information on waste management options and problems in the area of processing and commercialisation of contaminated agricultural products with regard to questions of their acceptance by the consumers [4-3, Part 3].

On-site measures

The operator of a nuclear power plant is responsible for the performance of all on-site measures for coping with emergencies. This also includes alerting of the competent authority according to the alert plans provided for it. Procedures to be taken in case of abnormal operation, incident or accident situation are explained in Article 19 (iv). Measures to reduce the probability of severe accidents (preventive accident management measures) or measures to mitigate the consequences of severe accidents with core damage (mitigative accident management measures) were taken into account during design and construction of the facilities or were subject to backfitting activities. They are explained in Article 18 (i).

Off-site measures

Criteria for protective actions

For the determination of criteria and the decision on measures of disaster control, the following objectives are applicable:

- Severe deterministic effects shall be avoided by measures for reducing the individual radiation dose to limits below the threshold doses for these effects.
- The risk of stochastic effects for individuals shall be reduced by appropriate measures.
- The measures for the persons affected shall provide more benefit than harm.

The Radiological Bases [3-15.2.] explain, in particular, the intervention reference levels (as pre-defined planning values) as thresholds for consideration of the implementation of appropriate disaster control measures to reach the objectives mentioned in case of radionuclide release after a nuclear accident. In case of an event, the intervention levels applied are derived from these reference values, taking into account the current boundary conditions and optimisation considerations.

Further, the Radiological Bases explain the transition in Germany from the bandwidth or two threshold concept recommended in ICRP Publication 63 to the initial value concept as described in the IAEA Basic Safety Standards No. 115 where the use of measure-specific intervention reference levels are recommended as initial values which already consider general optimisation considerations regarding the decision. These correspond to the lower threshold of the interval of the two threshold concept stated in the ICRP Publication 63. Potential doses are referred to as decision basis. The concept of the avoidable dose formulated by ICRP is not used as decision basis for short-term measures for reasons of practicability.

Table 16-2 includes the intervention reference levels (thresholds for investigating the initiation of protective actions) for protective actions specified in the Radiological Bases. Other criteria referred to within the framework of precautionary radiation protection are the maximum permissible levels of the EU for activity concentrations in foodstuffs [1F-30], [1F-31].

Table 16-2 Intervention Reference Levels for Protective Actions

Protective action	Intervention reference level		
	Thyroid dose	Effective dose	Explanations on integrations periods and exposure paths
Sheltering		10 mSv	Sum of effective dose from external exposure within 7 days and committed effective dose caused by the radionuclides inhaled within this period
Taking iodine tablets	50 mSv children and teenagers under age 18 and pregnant women 250 mSv persons of age 18 to 45		Thyroid dose caused by the radio-iodine inhaled within 7 days
Evacuation		100 mSv	Sum of effective dose from external exposure within 7 days and committed effective dose caused by the radionuclides inhaled within this period
Long-term resettlement		100 mSv	Effective dose from external exposure caused by radionuclides deposited on the ground and other surfaces within 1 year
Temporary resettlement		30 mSv	Effective dose from external exposure within 1 month

Specifications on radiation protection of the task forces in case of an event deployed as plant personnel, safety and rescue personnel (e.g. police, fire brigade, ambulance staff, physicians) or for specific work (e.g. measurements, transports, repairs, construction works) are included in the Radiation Protection Ordinance (Section 58 and, in particular, Section 59) and the Radiological Bases [3-15.2.] (→ Table 15-1). These are considered in the relevant fire service regulations [4-5] and the police service regulations [4-6].

<u>Protective actions in the area affected for averting of danger</u>

Off-site emergency preparedness refers to the preparation and performance of measures for protecting the public from the effects of radionuclide releases caused by incidents or accidents and leading to contaminations and increased radiation exposure.

With priority for implementation of these objectives, the short-term measures

- sheltering,
- taking iodine tablets,
- evacuation, and
- bans on the consumption of fresh, locally produced foodstuffs

are planned as part of disaster control and, if appropriate, joined by supplementary and accompanying measures (e.g. pre-distribution of iodine tablets).

While the measures “sheltering” and “evacuation” have not been modified regarding concept and implementation and will still be pre-planned for an area with a radius of up to 10 km, the concept regarding the distribution of iodine tablets has been adapted and extended in recent years. On the basis of the SSK recommendation on iodine blockage of the thyroid in case of nuclear accident “Iodblockade der Schilddrüse bei kerntechnischen Unfällen“ (1996) and the statements on the topic of iodine blockage of 1997 and 2001 in which the SSK adopted, among others, the internationally accepted recommendations of the WHO of 1989, the planning for purchase and distribution of potassium iodide tablets (iodine tablets) for the protection of the public was implemented. The instruction sheets for informing the public on the use of iodine tablets was revised for adaptation to the updated state of the recommendations and passed and republished by the SSK in 2004 [3-15.3.].

Within the special planning area up to 25 km, the following is provided for iodine tablets: Pre-distribution to households for all persons under age 45 in a radius of 0 to 5 km, pre-distribution to the households or stocks of iodine tablets at several points in the communities (e.g. town halls, schools, hospitals, businesses) in a radius of 5 to 10 km, also for task forces, and to stocks in communities or in suitable establishments in a radius of 10 to 25 km. The realisation of distribution and stockpiling is the responsibility of the *Länder*.

In a radius of 25 to 100 km, the tablets are held in stock for children and youth under age 18 and pregnant women in seven central stores. For event-related distribution from these stores for the radius of 25 to 100 km area, a concept was developed and implemented by a joint Government - *Länder* working group.

In addition to these measures, to prevent incorporation doses by ingestion of freshly harvested foodstuffs, a precautionary warning against consumption of such foodstuffs will be issued. This precaution will be adapted to the current situation as soon as corresponding data from measurements are available.

Beyond these protective actions, the Basic Recommendations [3-15.1.] include a list of further measures to be considered in the planning:

- Warning and informing the public,
- controlling, regulating and restricting road traffic,
- decontamination of the public and task personnel affected,
- medical care and treatment of the public and task personnel affected,
- initiating traffic restrictions for rail, waterway and air traffic,
- informing the water catchment bodies,
- closing contaminated water catchment points,
- warning the public against using water and against aquatic sports and fishing,
- informing waterway traffic,
- closing heavily contaminated areas,
- ensuring food supply,
- ensuring water supply,
- providing the animals with feed, in special cases relocation; where required, killing and disposal of heavily contaminated animals,
- decontaminating traffic routes, houses, equipment and vehicles, and
- preventing the putting into circulation of contaminated foodstuffs and feedstuffs.

Some of these measures also serve the purpose of precautionary radiation protection and are taken according to the Catalogue of Measures [4-3].

Protective measures of precautionary radiation protection for risk minimisation

In those areas where disaster control measures are not justified, the measures of precautionary radiation protection serve to reduce the radiation exposure of the public.

One focal point of the Catalogue of Measures [4-3] developed for this purpose are measures of precautionary radiation protection in form of recommendations for protective actions for the public and a large number of measures in the area of agriculture to prevent or reduce contamination of agricultural products and agricultural surfaces. The measures in the agricultural area are structured, as the situation demands, according to the accident phases (before and during passage of the radioactive cloud; after passage of the cloud) and, in particular, oriented to the limits of the EU [1F-30] for activity in foodstuffs. In addition, the catalogue contains information and measures for disposal as well as concretisations of the decision making philosophies and the assessment of the acceptance of measures in the agricultural area. More recent findings indicate that for acceptance reasons, the use of contaminated agricultural products will be limited and thus disposal will be of more importance than processing with the aim of decontamination.

Exercises

Training

In order to be able to perform the protective actions required in the case of an event effectively, the persons involved in coping with the crisis have to be properly, qualified and trained. Therefore, great importance is attached to on-site and off-site training of task personnel. This applies, in particular, to the preparation of the plant personnel and especially of the responsible shift personnel for coping with an emergency at the plant [3-2], [3-38], [3-39], [3-65].

For external task personnel, qualification and training are performed task-specifically in the respective organisations.

Exercises of the plant operator

The measures provided by the plant operator are trained, checked and further developed by means of exercises performed at regular intervals. Exercises involving the emergency organisation of the plant operator are generally performed once a year.

In order to be able to perform exercises as close to reality as possible, the accident scenarios on which the exercises are based are prepared generally in very detail. Typical exercise scenarios are events with loss of coolant, external events (earthquake, flood, aircraft crash, etc.), anticipated transients without scram (ATWS) and station blackout. In order to simulate dangerous situations according to the objectives of the respective exercise, these events are combined with inadequate core cooling and/or residual heat removal and/or inadequate containment isolation. In the last years, events in the field of physical protection have increasingly been included in the exercise programme of the plant operators.
The exercises are performed at the plants as realistic as possible, making increasingly use of the nuclear power plant simulators.

The annual exercises are generally limited to the nuclear power plant site. At larger intervals, the interaction between the emergency response team of the manufacturer, the Kerntechnische Hilfsdienst and the authorities responsible for off-site emergency preparedness is practised.

Exercises of the authorities at the national level

The disaster control authorities at the *Länder* level and at the regional level regularly perform large-scale disaster control exercises at the nuclear power plant sites, albeit at intervals of several years due to the considerable efforts and expenditure required. In addition to the competent authorities and the technical advisory commissions, the plant operator also participates in the exercises. Active involvement of the potentially affected population is normally not foreseen.

Objectives of these exercises are the improved interaction of the different organisations and authorities involved in emergency management and the assurance of effective co-operation in the disaster control and precautionary radiation protection. Another objective of the exercises is the practical deployment of forces within the framework of measurements and special support services, such as testing of temporarily established emergency care centres, dedicated to decontamination and medical services for the public, and the communication and co-operation of the different authorities and organisations involved.

The scenario of these exercises focussing on off-site measures is generally developed by the authority. The scenario involves the assumption of the release of radioactive material into the environment without considering the specific accident sequence within the plant, and the main tasks within the disaster control management are exercised. This includes, in particular, the assessment of the radiological situation, nature and scope of measures, command and control of the task forces and information of the public.

In currently still rare cases, site-specific integrated exercises are performed in which the plant operator and the competent authorities of potentially affected *Länder* perform an exercise with a plant-specific scenario. The most extensive exercise of this kind so far, covering several *Länder*, was performed in 2005 for the nuclear power plant Krümmel (KKK) in Northern Germany. In future, such exercises will also be performed at other sites.

To improve disaster control measures, emphasis will be placed in the next years on systems based on the use of modern information technologies, such as a joint measurement centre, a disaster control data management and information system or an electronic situation display with corresponding communication concept.

Participation in exercises at the international level

As part of international co-operation and on the basis of bilateral contracts, representatives from authorities of neighbouring countries are actively involved in exercises concerning plants near the border, or at least participate as observers.

A good example of cross-border co-operation is an INEX-3 table-top exercise, jointly executed by Germany and Austria, which took place simultaneously at the German Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) in Bonn and at the Austrian Federal Ministry of Agriculture, Forestry, Environment and Water Management (BMLFUW) in Vienna from 13 to 14. September 2005. About 60 persons from different authorities of the Federal Government and the *Länder* and expert organisations in Germany and Austria participated in the successful exercise whose aim was the co-ordinated cross-border emergency management in case of a radiological event (combustion of a Cs-137 source).

Further, Germany participated in the CONVEX-3 exercise from 11 to 12 May 2005 which was performed at the national level as command post exercise. The Federal Ministry for the Environment (BMU), in its function as national competent authority, the Federal Office for Radiation Protection (BfS), the German Meteorological Service (Deutscher Wetterdienst, DWD) and the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH participated in CONVEX-3. The intended objectives of this exercise, such as fast information exchange between BMU, IAEA, EU and Romania, improved alerting, the determination of the radiological situation and its assessment and the web-based information exchange between the different national command posts, were achieved.

In principle, BMU representatives take part - in line with their respective responsibilities - in the regular exercises of the EU (ECURIE exercises), the IAEA (CONVEX 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.

Regarding further development and harmonisation of nuclear emergency preparedness regulations at an adequate high international level, representatives of the BMU and other organisations participate for Germany in the relevant commissions at OECD/NEA, IAEA and the EU.

16 (2) Informing the General Public and Neighbouring Countries

Informing the public

The requirements of the EURATOM directive regarding the information of the public in case of a radiation emergency [1F-29] have been incorporated in Sections 51 and 53 of the Radiation Protection Ordinance [1A-8]. The main contents of the information of the public are specified in Appendix XIII of the Radiation Protection Ordinance where distinction is made between information to be issued to the public in advance as preparation for a radiological emergency and the relevant information in case of a concrete emergency according to Section 51 (2) of the Radiation Protection Ordinance.

The most important issues about which the public in the vicinity of a plant has to be informed at least every five years concern, among others,

- basic terminology and related explanation on radioactivity and its impacts on humans and the environment,
- radiological emergencies and their consequences for the public and the environment, including planned rescue and protective actions,
- 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 information is realised by means of a brochure, financed by the plant operators, which is posted to the public living in the vicinity of a nuclear installation in co-ordination with the disaster control authorities.

In case of a safety-relevant event at a nuclear installation leading to a radiological emergency in the surrounding area, the competent authorities inform the potentially affected public without any delay according to Section 51 (2) of the Radiation Protection Ordinance and give information on how to behave including specifications on health protection measures to be taken.

The information to be given to the public are summarised in Appendix XIII, Part A of the Radiation Protection Ordinance and concern, among others,

- type and characteristics of the event, in particular origin, dispersion and expected development of the situation,
- protection instructions and measures for certain groups of the population, and
- designation of the authorities responsible for disaster control.

Also in case of pre-alarm level (early warning), respective information are to be given to the public. Details and corresponding text modules are laid down in the revised Basic Recommendations. An information concept for further concretisation is being developed.

The information are given through different media, such as radio, TV or by loudspeaker announcements.

Informing the neighbouring countries

In the event of an emergency, the measurement data acquired within the monitoring programmes and the situation assessment of the plant operator will be the basis for reporting in accordance with the EU agreement on rapid information exchange [1F-33] and the Convention on Early Notification of a Nuclear Accident [1E-6]. They also serve as basis for the information exchange for fulfilling bilateral agreements. This ensures that Germany's neighbouring countries will receive timely information. The measurements routinely performed in accordance with the Guideline on Emission and Immission Monitoring [3-23] 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. Moreover, assistance agreements have been concluded with Lithuania, Hungary and the Russian Federation. Similar agreements with Italy and Bulgaria have been initialled or are in preparation. Due to such agreements, there are direct information and data exchanges at the regional level at nuclear power plant sites near the border between the respective disaster control authorities or organisations for determining the radiological situation.

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

Not applicable to Germany.

Article 16: Progress and Changes Since 2004

Within the reporting period from 2004 to 2006, numerous amendments of regulatory documents related to emergency preparedness were performed:

In the field of disaster control, the concept for the provision of iodine tablets was revised and adapted to the recent recommendation of the SSK. Likewise, tools for situation determination and assessment were harmonised (RODOS decision support system, ELAN information system) and revised, respectively (Basic Recommendations). The guidance for the expert advisor for radiation protection of disaster control management in case of nuclear emergencies [4-4] was republished by the SSK in 2004 and made available as computer-based version (PLUTO).

The Basic Recommendations for Emergency Preparedness in the Environment of Nuclear Facilities [3-15.1.] were updated in the recent years and have been finalized now. After agreement by the responsible committees, which is expected for 2007, they will be newly published. Besides the adaptations due to updates of the nuclear rules and regulations, the revision took special attention to the consideration of “events with rapid sequences“ and to the development of an improved communication and information concept. The instruction sheets for informing the public on the use of iodine tablets was revised for adaptation to the updated state of the recommendations and passed and republished by the SSK in 2004.

In the area of environmental monitoring, especially regarding emission and immission monitoring during routine operation and also in case of incidents or accident situations, the Guideline on Emission and Immission Monitoring [3-23] was revised and adapted to the state of the art in science and technology (→ Article 15). This allowed a significant further development of the resulting contributions to precautionary radiation protection and disaster control.

The SSK supplemented the Catalogue of Measures [4-3] by information and measures for disposal as well as concretisations of the decision making philosophies and the assessment of the acceptance of measures in the agricultural area [4-3.3].

Article 16: Future Activities

Further extension of national exercises under participation of several *Länder* and international exercises in areas near to the border is strived for. The experiences will be incorporated into the further development of off-site emergency preparedness. Moreover, it is intended to increase interaction of the emergency systems at the national (between Federal Government and *Länder*) and the international level by improved and more extensive information exchange in the radiological emergency management.

17 Siting

ARTICLE 17 SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented:

i) for evaluating all relevant site-related factors likely to affect the safety of a nuclear installation for its projected lifetime;

ii) for evaluating the likely safety impact of a proposed nuclear installation on individuals, society and the environment;

iii) for re-evaluating as necessary all relevant factors referred to in sub-paragraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation;

iv) for consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, upon request providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation.

In Germany, licences for the construction of new nuclear power plants pursuant to Section 7 (1) of the Atomic Energy Act are no longer granted. Therefore, the following presentation deals with the procedures as practiced in the past for the plants in operation today. Further, the design against external events and their current evaluation are addressed.

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 landscape conservation. 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 in the vicinity of the site,
- seismic hazard and geological condition of the building ground,
- 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, and
- distance to military installations.

Proceeding within the licensing procedure

After the applicant had pre-selected a site, a regional planning procedure was initiated which preceded the nuclear licensing procedure. This took into account all impacts of the individual project on the public, on traffic ways, regional development, landscape protection and nature conservation. Besides the site characteristics, the design of the nuclear installation against external events was checked in the nuclear licensing procedure (→ Article 7 (2ii)). Further, investigations were carried out as to whether general public interests oppose the selection of the site. Within the nuclear licensing procedure, the respective competent authorities analysed 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 events

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 external events. In this respect, the following has to be considered in particular:

- External natural events, as far as to be considered, such as earthquake, landslide, storm, flood, storm surge,
- external man-induced events, such as aircraft crash, impact of dangerous and, in particular, explosive substances, and
- malevolent disruptive acts or other third-party intervention.

The design requirements specified in regulatory guideline [3-33] for external events distinguish between events to be treated as design basis accidents and events which, on account of their low occurrence probability, are not considered as design basis accidents, and for which measures must be taken to minimise the risk. Accordingly, the external natural events (earthquake, flood, external fire, lightning and other natural impacts) are considered as design basis accidents (→ Article 18 (i)), whereas external events such as aircraft crash and pressure wave or impacts from dangerous materials from outside of the plant require risk minimisation.

The requirements for the design and for protective measures against external events for construction of the German nuclear power plants followed the nuclear safety regulations applicable at that time. 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 Article 17 (iii).

All nuclear installations have not only been designed taking into account external natural events, such as wind and snow, but also floods and, where there was 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 power plant, the system design resulted also in requirements important to safety for the cooling water supply. It had to be verified for the individual site conditions that the cooling water supply was ensured even under unfavourable conditions, e.g. low water in the river or failure of a river barrage.

<u>Design against flood</u>

The requirements for flood protection measures are included in nuclear safety standard [KTA 2207]. Pursuant to this standard, a permanent flood protection is to be provided. Under special boundary conditions, protection against the difference between the water levels of the flood with an exceedance probability value of 10^{-2}/a and the design basis water level may also be provided by temporary measures. The amended safety standard [KTA 2207] has been available since November 2004. The changes compared with the previous version concern, in particular the specification and determination of the design basis flood. It is now consistently based on an exceedance probability of 10^{-4}/a. The design of the plants was based on the respective applicable version of this standard.

The sites of the nuclear power plants are mostly located inland at rivers and, in some cases, at estuaries with tidal influences. In most of the cases, sites have been selected which are located sufficiently high. In all other cases, the 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 permanent protective measures should not be sufficient, mobile barriers are available to close off openings.

Design against earthquake

Since 1990, the design against earthquakes is based on a design basis earthquake (formerly called safe shut-down earthquake) in accordance with safety standard [KTA 2201.1]. The so-called operating basis earthquake, formerly considered additionally, was replaced by an inspection earthquake where only the plant condition has to be checked. The design basis earthquake has the largest intensity that, under consideration of scientific findings, could occur in a wider vicinity of the site (up to a radius of about 200 km). Depending on the site, the intensity of the design basis earthquake varies between less than VI and a maximum of VIII on the MSK scale. In the 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. Safety standard [KTA 2201.1] is currently being revised to adapt the methods for determination of the design basis earthquake and of the seismic impacts on civil structures and plant components to the state of the art in science and technology.

Protection against aircraft crash

Protection against aircraft crash concerns the accidental crash of an aircraft onto safety-relevant plant areas. 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 exceedance probability for impacts on safety-relevant buildings is about 10^{-6}/a 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 for further risk minimisation. In accordance with RSK guidelines [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 older nuclear installations, protection by system design against the consequences of an aircraft craft was improved by the design of buildings and components in interaction with additional auxiliary emergency system physically separated from the actual reactor building. The auxiliary emergency systems can ensure the safe enclosure of radioactive material in the reactor even if important plant components would be destroyed due to external events.

The spatial arrangement of the buildings is to ensure that the safety systems and equipment located in the central reactor building and in the additional auxiliary emergency systems do not become inoperative due to the postulated events at the same time. The scope of protection of these plants against aircraft crash was demonstrated by subsequent reviews of the design margins of the safety-relevant buildings and extended within the framework of backfitting measures. New buildings were designed according to the increased requirements and the measures against induced vibrations improved.

For the newer 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 event (e.g. the emergency feed-water building in newer PWRs). 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 wave 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) Evaluation of the Likely 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 Environmental Impact Assessment Act [1F-12] (→ Article 7 (2ii)). The impacts 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-16] 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. Here, the water law regulations generally prescribe more narrow limits with regard to heating of river than the safety requirements. If, under extreme weather conditions, it is foreseeable that the permissible temperature rise would be exceeded, the respective nuclear installation must reduce its power accordingly.

Reductions of power operation became necessary in the last summers. Nuclear power plants without cooling tower had to reduce their power up to 50 %. Even if there is the possibility to switch operation to closed circuit cooling (cooling exclusively with water in closed circuit cooling tower operation), the net efficiency decreases by several percent compared to fresh water cooling (condenser cooling exclusively with river water). Especially in the hot summer of 2006, the power reductions had to be maintained over a longer period.

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 Article 15. In addition to the total dose of 1 mSv per year, the permissible limit for discharges with exhaust air and waste water is 0.3 mSv each per year for specified normal operation. For design basis accidents, it is to be demonstrated in the planning that through the precautionary measures taken an effective dose of 50 mSv is not exceeded for a reference person at the most unfavourable receiving point.

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

Article 17 (i) describes the current design of German nuclear power plants against external events. The safety reviews which are scheduled to be repeated every ten years (→ Article 14 (ii)) also include a re-evaluation, taking the development in the state of the art into consideration. 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 events flood, earthquake, aircraft crash and pressure wave from explosion are described below.

<u>Flood</u>

The re-examinations on flood protection in the years 2000 to 2002, initiated by the BMU, 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 not consistent. Within the framework of the amendment (finalised in November 2004) of the safety standard [KTA 2207] on flood protection, in particular the methods for determining the design basis flood were therefore specified. The amended safety standard will be applied to all future modification licences where flood protection is concerned. Moreover, it has to be referred to as assessment criterion for any kind of safety review, e.g. the legally required safety review according to section 19a of the Atomic Energy Act (→ Article 14 (i)).

Possible protective measures at the individual sites strongly depend on the respective topographic conditions. Therefore, the individual measures planned or taken result in a heterogeneous picture. For some nuclear power plants, for example, directly located at rivers, an island situation may already occur in case of a flood expected at 10^{-3}/a for which corresponding organisational and administrative measures are provided.

Earthquake

For some nuclear installations at sites with relevant seismicity, the ongoing development of methods to determine seismic load assumptions and to verify design specifications led to a re-evaluation of seismic safety. 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 partly has considerable margins with respect to seismic loading. At plants for which a need for upgrading was identified (e.g. Philippsburg 1), comprehensive safety retrofits were performed on the basis of these re-evaluations.

Aircraft crash

For older plants, a further risk reduction regarding accidental aircraft crashes was achieved by backfitting with physically separated auxiliary emergency systems that are completely independent from other systems (→ Table 6-2). All in all, the risk contribution from accidental aircraft crash is considered as being negligible. As regards accidental aircraft crash with the above load assumptions, some nuclear installations were re-evaluated with regard to the load transfer in conjunction with probabilistic safety assessments. As results of the probabilistic assessments by the plant operators it was stated that even in the cases where the reactor building does not withstand the loads to be assumed according to the currently applicable rules and regulations, the contribution to damage states with considerable release was to be assessed as low.

Pressure wave 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 within the framework of the safety reviews. The results show that in almost every case the actual structural design is sufficient to withstand the assumed loads specified. In every case, however, the nuclear installations are sufficiently protected under general risk aspects. The certifications required in the licensing procedures for industrial complexes ensure 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 information exchange in connection with the construction of nuclear installations in the border regions. With six of the nine neighbouring countries of Germany (the Netherlands, France, Switzerland, Austria, the Czech Republic and Denmark) bilateral agreements regarding the exchange of information on those nuclear installations built in the border regions have been signed.

Among other things, the agreements include provisions for

- 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, and
- 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 European Community also ratified the agreement, however limited to the application of the provisions among the member states.

In accordance with Article 37 of the EURATOM Treaty [1F-1], 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 [1F-4]. This serves to establish the possible impacts on the other member countries. 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. Article 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.

Article 17: Progress and Changes Since 2004

The revision of safety standard [KTA 2207] (Flood Protection for Nuclear Power Plants) was finalised in November 2004. In particular, the procedures for determining the design basis flood were specified. The exceedance probability value for the design basis flood was defined at 10^{-4}/a.

Article 17: Future Activities

The safety standard [KTA 2201] (Design of Nuclear Power Plants against Seismic Events) is being revised.

18 Design and Construction

ARTICLE 18 DESIGN AND CONSTRUCTION

Each Contracting Party shall take the appropriate steps to ensure that:

i) the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur;

ii) the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis;

iii) the design of a nuclear installation allows for reliable, stable and easily manageable operation, with specific consideration of human factors and the man-machine interface.

18 (i) Safety Concept

Regulatory requirements

The Atomic Energy Act [1A-3] elevates the protection against damages as required according to the state of the art in science and technology to a major criterion for licensing and supervision (→ Article 7). For this damage precaution, a concept of defence in depth reflects today's state of the art in science and technology. The basic features of the concept are specified by the provisions of the nuclear rules and regulations (→ Article 7 (2i)). The safety criteria of 1977 [3-1] with its supplementing interpretations [3-49] cover the design for normal operation, abnormal occurrences and the control of design basis accidents.

The resulting concept is firstly based on high requirements regarding a plant operation as failure free and environmentally compatible as possible. In this respect, high demands are being placed on the design and quality of the nuclear power plant as well as on personnel qualification. Further, the concept includes prevention and control of abnormal occurrences and design basis accidents. Here, the safety criteria require sufficiently reliable technical safety systems and equipment. Section 49 of the Radiation Protection Ordinance [1A-8] defines specific planning values for the measures against release of radioactive material in case of design basis accidents. The design basis accidents considered in the design of the last licensed nuclear power plants are specified in the accident guidelines [3-33]. In the licensing procedure it is verified that the releases of radioactive material determined for all design basis accidents under conservative boundary conditions are below the planning values of Section 49 of the Radiation Protection Ordinance. In this range, the internationally accepted design principles, such as redundancy, single failure criterion, physical separation etc. are considered.

Moreover, the nuclear rules and regulations also include requirements for precautions against event beyond the design against design basis accidents. These are, among others,

– very rare events (e.g. ATWS, accidental crash of a military aircraft, gas cloud explosion),
– events with multiple failure of safety systems and equipment (e.g. station blackout), as well as
– accidents with core damages.

In this area, the RSK guidelines for pressurised water reactors [4-1] are of relevance and other RSK recommendations. For such events, damage preventing (preventive) and damage mitigating (mitigative) measures are provided.

Further, the safety criteria [3-1] stipulate that organisational and technical measures inside and outside the nuclear installation are to be provided by way of precaution to identify and mitigate the consequences of accidents.

Implementation by the plant operator

The concept for prevention and control of design basis accident is implemented at all German nuclear power plants. The main requirements of the safety criteria were already considered in the design of the first construction lines. In the early eighties, the RSK guidelines were revised and, above all, included new requirements for the separation of redundancies. These requirements could be considered in the design of the nuclear installations that were in the planning phase at that time. At the already existing nuclear installations, backfitting measures, some of them extensive, were performed to achieve this safety standard there. So, e.g., an own emergency building with diesel generators and bunkered reactor protection system was installed at several nuclear power plants to establish additional and physically separated redundancy for special external and internal event sequences (→ Table 6-2).

At German nuclear power plants, the defence in depth concept was further strengthened in two different aspects. On the one hand, the respective measures taken refer to accident prevention and the control of accidents and, on the other hand, also to measures for beyond design basis scenarios.

An example of accident prevention is the break preclusion. This concept concerns the design basis accident “double ended break of a large pipe of the reactor coolant system“. Research programmes performed since the mid seventies showed: If defined prerequisites and boundary conditions are fulfilled (concept of basic safety), double ended breaks can be precluded. The concept of break preclusion is described in Article 18 (iii) in more detail.

Another example for a design concept that also shall prevent damages within the defence in depth safety concept from the outset is the integrity concept for steam generator tube at PWRs. This integrity concept has the objective to prevent corrosion and high mechanical loads of the steam generator tubes as far as possible. To this end, since the seventies,

- a different material has been used for the steam generator tubes than internationally in use at that time,
- a special concept for installation of the tubes in the tube sheet and the holding device of the tubes has been developed, and
- chemical processes have been used to increase the pH value at the secondary side where the evaporation does not lead to increased concentrations with residue formation.

Moreover, measures for the control of the steam generator tube rupture were optimised, e.g. by increase of the response pressure of the safety valves at the secondary side.

In addition to the multi-level concept for the prevention and control of design basis accidents, measures were provided, following international developments, to prevent core melt also in case of beyond design basis scenarios or to mitigate the radiological consequences of a core meltdown.

For this reason, work has been increasingly performed since the early eighties to assess plant behaviour in situation where safety systems and equipment do not function as designed and to develop measures to mitigate the consequences of such event sequences.

Under consideration of these objectives, measures to reduce the probability of occurrence of severe accidents (preventive accident management measures) or measures to mitigate the consequences of severe accidents with core damage (mitigative accident management measures) have been included in the design or performed as backfitting measures at existing plants at the German nuclear power plants since the eighties (→ Table 6-2).

Preventive accident management measures concern, in particular, measures by which heat removal from the core can be restored at a state before developing into a large core damage. As examples for these backfitting measures, implemented at the German nuclear power plants in the last years, provisions for secondary and primary side bleed and feed may be mentioned.

Mitigative accident management measures mainly aim at preventing impermissible pressure build-up in the containment, thus preventing uncontrolled release of radioactive material. At German nuclear power plants, appropriate measures were provided or implemented as backfitting measures. For PWRs these are, above all, the use of passive catalytic recombiners in the containment, and for BWRs the inertisation of the containment with nitrogen during power operation. For both plant types, filtered containment venting and an accident sampling system were also implemented.

With these and other measures, it was possible to counteract the newly identified risks - which did not have to be considered at the time of the safety assessment - within the framework of the defence in depth concept.

18 (ii) Qualification and Proof of Incorporated Technologies

The nuclear rules and regulations contain extensive requirements regarding qualification and proof of incorporated technologies and the reliability of safety-relevant structures, systems and components. In a general form, these requirements are laid down in the safety criteria [3-1] and RSK guidelines [4-1] and their implementation in practice and verification defined in the quality assurance system according to KTA safety standard [KTA 1401]. The requirements are classified according to the principles of the defence in depth concept and their safety relevance. Details regarding the technical realisation are specified in the nuclear regulations and guidelines. 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 also made to the proven operating experience. Special requirements and, where appropriate, verifications by experiments for individual systems and components are also derived from safety analyses.

Passive systems

Passive systems are systems not requiring activation with regard to the fulfilment of 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 government 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 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 electrical components and 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. 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 (→ Article 19 (i)).

Proof of qualification

The qualification of the installed techniques is 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, and
- 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 (→ Articles 19 (vi) and (vii)).

Experience feedback has shown in particular 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 all volatile treatment (high-AVT) of 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 loads, 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

The general requirements with regard to the design of nuclear power plants, to simplicity of system design, physical separation of redundant subsystems, as well as to accessibility for inspections, maintenance and repairs are specified in the safety criteria [3-1] und [3-51]. High reliability of systems and components is already to be achieved during manufacturing by adherence to the design principles. This includes high-quality materials and comprehensive quality assurance. In combination with good maintenance, a high reliability and availability of systems and components is to be achieved.

The safety criteria also include the requirement that workplaces and work sequences are to be designed under ergonomic aspects such that they offer the prerequisites for an optimum safety-oriented behaviour of the persons employed. Detailed requirements both with regard to technical measures and to the administrative procedures of work tasks are specified, among others, in the KTA safety standards series 1200 and 3200.

Ergonomic design of control stations

Specifications on the ergonomic-technical design of control room, emergency control room and local control stations are laid down in safety standard [KTA 3904]. These also include requirements for the functional and spatial arrangement, personnel staffing, the design of works systems and equipment and environmental influences with specifications on lighting, air conditioning and acoustics. Concrete requirements for analog and digital displays of position, size, arrangement etc. of units, scale graduation marks, numbering etc. are also described precisely. In some cases, reference is made to German DIN standards for further details. These also include the procedure for the implementation of modifications. A change in the state of knowledge is taken into consideration where necessary. The procedure covers the following steps:

- Description of the tasks of new components,
- description of the tasks of the operating personnel,
- check of task performance within the construction and testing phase, and
- analysis and assessment of different concepts with regard to human performance capabilities and performance limitations.

In this respect, requirements for analysis and assessment methods applied are also made. As a tool for the support of ergonomic studies for the purpose of the items mentioned, the database system EKIDES (Ergonomics Knowledge and Intelligent Design System) was developed and implemented by the plant operators.

Design of the socio-technical overall system, man-technology-organisation (MTO)

In addition to technical measures, human and organisational measures and their interactions are also of great importance. Therefore, the Atomic Energy Act and the other legal regulations and non-mandatory guidance instruments mentioned provide that for licensing the fulfilment of requirements regarding reliability, the requisite qualification and knowledge of the groups of persons defined there is equally necessary as the fulfilment of the requirements regarding damage precaution by construction and operation of the plants. These requirements must be seen comprehensively and also extend to the economic reliability and appropriateness of the organisation.

With the development of nuclear technology and due to world-wide findings from occurrences at nuclear power plants, the requirements on the integration of the components, the technical systems and the social system interacting at a nuclear power plant were further developed.

At the end of the seventies, the development of concepts was started, going beyond the ergonomic design of work systems and equipment, for optimisation of an integrated design of the socio-technical overall system man-technology-organisation (MTO) for a reliable and appropriate plant design and the power plant process.

The governing principles of this concept are as follows:

- The optimum design of the overall system shall be the objective, not the adaptation of the social to the technical system or vice versa.
- The sharing of functions between man and technology shall be realised under consideration of the human capabilities and limitations.
- Man shall be effectively supported in his indispensable role in the fulfilment of safety-related tasks and relieve him from tasks conflicting with safety objectives.
- Human actions must be protected to the larges possible extent by a system behaviour resistant against human failure.

The concept led, in particular, to requirements for the improvement of technical support of man by further development of the defence in depth concept (→ Article 18 (i)). Particular importance is attached to the limitation systems that precede the protection system. There are three types of limitation systems that are classified according to task and requirement. Operational limitations are I&C systems with increased reliability which, for the rest, are to be comparable with the control systems. Limitations of process variables and protective limitations are I&C systems comparable with the reactor protection system and are part of the safety system. In case of operational occurrences, the limitations shall automatically

- limit the process variables to defined values in order to increase the availability of the plant (operational limitations),
- limit the process variables to defined values in order to maintain initial conditions for the accidents to be considered (limitations of process variables) und
- bring safety variable back to values at which continuation of specified normal operation is permissible (protective limitations).

The aim is to reach a high degree of automation for relief of man from short-term measures and comprehensive preventive measures to counteract development of abnormal occurrences into accidents and a high tolerance against human failures. The requirements for comprehensive, reliable and user friendly process information systems also provide technical support for the human role. The aim of these extensions of technology is to enable man with his limitations and special performance capabilities to fulfil his safety task within the overall system in an optimal manner.

The concept is also connected with extended requirements on organisational support of man in fulfilling his safety tasks. This concerns, in particular, the protection goal oriented procedure in case of design basis accidents and beyond design basis accidents (→ Article 19 (iv)).

At the newer nuclear power plants, these requirements were directly and completely implemented. For plans of older construction lines, the organisational improvements were implemented. Moreover, a number of technical retrofit measures were performed to achieve the central objectives within the framework of the respective plant concept.

Since the late nineties, the development of the requirements on a comprehensive safety management system (SMS) has been of importance for further optimisation of the integrated structuring of the socio-technical overall system MTO. The task of SMS is to ensure the continuous and systematic control and improvement of the reliability of the complex MTO system of the nuclear power plant (→ Article 10 and Article 12).

The effectiveness of the measures taken for realisation of socio-technical overall system MTO meeting the requirements is checked, also with regard to further optimisation possibilities, within the framework of the relevant inspections and reviews of plant operators and authorities, as e.g. the periodic safety review (→ Article 14 (i)), evaluation of experience feedback (→ Article 14 (ii) and Article 19 (vii)) and the comprehensive event analysis (→ Article 12).

Integrity concept

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, and
- accounting for the operating medium.

These principles were immediately applied at the newer nuclear power plants. At the older plants, they have led to post-qualifications to comply with these principles or for the assessment of identified non-compliances. In several cases, the assessments showed a need for extended safety demonstrations and measures to be implemented.

For continuous assurance of component integrity during power operation of light water reactors in Germany, that integrity concept was applied which has been developed on the basis of the safety criteria [3-1] on damage precaution, RSK guideline [4-1] and the general specification of basic safety in the last 25 years. Recent developments in this area focus on the incorporation of ageing processes and their control within the framework of an overall concept which puts all aspects of integrity demonstration into predefined interrelations (→ Appendix 4).

Of particular relevance is the integrity demonstration for piping systems. Altogether, the operating experience with pipes at German nuclear power plants fabricated according to the concept of basic safety is positive. For these systems, no indication changes or even service-induced cracks were detected by in-service inspections. Until now, the integrity concept has been proven in practice and presents an important contribution with regard to damage precaution.

The main process elements of the consistent German integrity demonstration have been incorporated in KTA safety standard [KTA 3201.4] in form of a process diagram.

Article 18: Progress and Changes Since 2004

For the support of ergonomic studies, the database system EKIDES (Ergonomics Knowledge and Intelligent Design System) was developed and implemented by the nuclear power plant operators.

Further preventive and mitigative measures for plant-internal accident management at the nuclear power plants were implemented.

Article 18: Future Activities

The general duties of the plant operators and the competent authorities according to the regulatory requirements in terms of a continuously improving safety culture and in terms of the requirements of the Convention are a guide for action and measures. Furthermore, no specific measures are provided for the next three years.

19 Operation

ARTICLE 19 OPERATION

Each Contracting Party shall take the appropriate steps to ensure that:

i) the initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning programme demonstrating that the installation, as constructed, is consistent with design and safety requirements;

ii) operational limits and conditions derived from the safety analysis, tests and operational experience are defined and revised as necessary for identifying safe boundaries for operation;

iii) operation, maintenance, inspection and testing of a nuclear installation are conducted in accordance with approved procedures;

iv) procedures are established for responding to anticipated operational occurrences and to accidents;

v) necessary engineering and technical support in all safety-related fields is available throughout the lifetime of a nuclear installation;

vi) incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the regulatory body;

vii) programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies;

viii) the generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal.

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

Licensing procedures for the construction and operation of nuclear power plants pursuant to Section 7 of the Atomic Energy Act [1A-3] were performed stepwise with partial licences for construction and operation.

The respective licensing prerequisites are to be examined for each licence step. The information required for the examination go beyond the safety analysis report and are contained in the regulatory guidelines [3-7.1] and [3-7.2]. Distinction is made between A and B information. A information had to be submitted for examination of the licensing prerequisites prior to a licence step. B information are needed in accompaniment of construction, e.g. in fulfilment of mandatory requirements specified in the corresponding licensing decision or to be submitted by the applicant/licensee for the purpose of accompanying control by the authorised experts called in.

For granting of an operating licence it is necessary that information and commissioning programme submitted fulfil the requirements according to the nuclear rules and regulations and confirm that the plant was constructed according to the granted partial licence and imposed obligations.

The following description explains the proceeding for construction and commissioning of the existing nuclear power plants. The construction of new nuclear power plants in Germany is not provided according to § 7 (1) of the Atomic Energy Act. The proceeding also applies according to the rules and regulations for modifications of plants and their operation requiring a licence permit.

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 (→ Article 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 checked whether the nuclear installation in its as-built condition meets all applicable design and safety requirements.

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.

Safety analysis

For the safety analysis within the framework of the licensing procedure, verifications are to be submitted to the licensing authority and its experts that by the design the development of abnormal occurrences into accident situations is mitigated and accidents are controlled with a high reliability. In particular, the sequences of design basis accidents and their radiological impacts are calculated in accident analyses under conservative assumptions. Proof is to be furnished that in case of an accident the radiation exposure of the general public does not exceed the planning values pursuant to Section 49 of the Radiation Protection Ordinance. The calculations of the accident sequence are based on the automatic actions of the safety systems and the measures to be taken by the plant personnel laid down in the operating manual.

Originally, this safety analysis was merely performed deterministically and has later been supplemented by probabilistic assessments. 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 verified alternate analytical methods.

Accompanying control during construction

The accompanying control during the entire manufacturing process ensures that the actual design of the systems and components important to safety meets the requisite requirements.

The accompanying control 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 reaching of 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 authorised experts called in by the supervisory authority. The authorised experts examine the commissioning programme and participate in tests chosen by them. 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

Pursuant to the Nuclear Licensing Procedure Ordinance [1A-10], a compilation is to be submitted as part of the application documents containing all the data relevant to the safety of the plant and its operation, the measures to be taken in the event of incidents or damage, and an outline plan of the in-service inspection tests provided for safety-related components of the plant (safety specifications).

The associated regulatory guideline [3-4] requires that safety specifications are to be submitted as licensing documents to the licensing authority in charge at the latest together with the application for the license to operate the relevant plant. It shall be permissible to establish "preliminary safety specifications" for the duration of trial operation; however, they shall be substituted by "final safety specifications" as soon as all relevant data are on hand and at the latest when continuous operation begins.

The safety specifications, in their capacity as licensing documents, constitute a binding and updated documentation of the licensed and, in terms of safety, reliable frame for the condition and mode of operation of the plant (operational limits and conditions for safe operation). They are the basis upon which the safety relevance of modifications of the plant or its operation will be assessed. As a matter of principle, modifications concerning data contained in the safety specifications shall require the approval of the licensing or supervisory authority in charge.

For the design of the plant, the design principles laid down in the nuclear rules and regulations are to be applied and the ability to control design basis accidents to be verified. On this basis, limits and conditions for operation and measures for the control of accidents are derived. These are collated as the so-called safety specifications in accordance with the Nuclear Licensing Procedure Ordinance [1A-10] and with the Guidelines Concerning the Requirements for Safety Specifications for Nuclear Power Plants [3-4]. They give a quick and comprehensive survey of all data, limits, conditions, requirements and measures that determine the safety of the nuclear installation. The safety specifications are a constituent part of the operating manual and the testing manual.

Objectives and status of the documents mentioned within the frame of the licensing procedure shall briefly be explained. Section 3 of the Nuclear Licensing Procedure Ordinance [1A-10] requires the collection of data and documents known as “safety specifications”. Details on implementation are given in regulatory guideline [3-4]. Accordingly, the safety specifications contain “all the data relevant to the safety of the plant and its operation, the measures to be taken in the event of incidents or damage, and an outline plan of the tests provided for safety-related components of the plant”. These safety specifications are to be submitted to the competent authority when filing the application for an operating licence and describe the safety-related framework within which the plant must be operated. The limits and conditions of safe operation are thus defined and documented.

The operating manual (→ Article 19 (iii)) is the most important working document for the plant personnel. It contains all operating and safety-related instructions required for normal operation of the plant as specified and for the control of incidents as well as plant regulations applicable for all persons working at the plant. Structure and contents of the operating manual are described in the nuclear rules and regulations in KTA safety standard [KTA 1201]. The safety specifications are included in the operating manual as separate chapter or as marked sections. An exception are those parts of the safety specifications that are not contained in the operating manual but in the testing manual. Any modifications of the safety specifications require approval by the licensing or supervisory authority. The limits and conditions of safety operation in the safety specifications prescribed by the licensing authority must be met at all times.

Those chapters of the operating manual are to be assigned to the safety specifications which deal with the following topics:

- Prerequisites and conditions for power operation, among others for startup and plant shutdown and for refuelling,
- safety-relevant limits,
- 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 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 plant operator for safe operation of his nuclear installation. They comprise, among others:

- 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 startup, 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,
- 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 safety-relevant 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,
- measured 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, and
- compilation of the accident monitoring instrumentation at the control room and the emergency control room in tabular form.

Deviations from limits and conditions are immediately detected at the control room if the permissible tolerance range is exceeded. In case of deviations from limits and conditions, the measures to be taken are laid down in the operating manual. Irrespective of how fast restoration of normal operating conditions is performed, the result is documented and is made part of the internal experience feedback as alarm notice (→ Article 19 (vii)).

In case of modifications of the plant or its operation, the conditions concerned have to be reviewed and to be changed where required early before continuation of operation. Regarding further development of the state of knowledge, the safety specifications are reviewed by the plant operator and the supervisory authorities and their experts whether amendments are required.

In case of modifications of the safety specifications, the shift personnel concerned is directly informed about the new situation through meetings or notices. In addition, the simulator training (→ Article 11 (2)) regularly required for maintenance of the technical qualification are used to practice new procedures in a targeted manner.

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

In addition to technical prerequisites, the licence of a nuclear power plant is also based on personnel and organisational prerequisites. The approved procedures during operation, including maintenance and testing, but also for the control of abnormal occurrences and accidents described in Article 19 (iv) determine the organisational and operational structure in the nuclear power plant. This structure is laid down in detail in the operating manual of the respective plant.

For the organisational structure, the following principles are of importance, among others:

- The plant manager is responsible for safe operation. In the event of his absence, this responsibility is transferred to his deputy or 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, authorisation and responsibilities of the managing personnel are clearly, without any overlap and completely specified.
- To avoid any conflict of interests, the organisational units and persons responsible for quality assurance and for radiation protection are independent of the divisions responsible for operation and maintenance.

The organisational structure is defined in the operating manual in the chapter “Personnel Organization”.

The supervisory authority and its authorised experts check within the framework of their on-site inspections (→ Articles 7 (2 iii) and 14 (ii)) whether the regulations for the organisational structure are also adhered to in practice. In addition to plant walk-downs and controls at the plant control room, especially the close supervisory accompaniment of major proceedings at the plant (e.g. modification procedures, maintenance measures, investigations in response to reportable events) provide an insight into personnel/organisational processes.

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

All nuclear power plants have an 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 (→ Article 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 accidents originating from external impacts and the related procedures to control these accidents.
- *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.

The operating manual is kept up to date through a revision service. The copy of the operating manual at the plant control room also contains all modifications in process. All modifications of the operating manual are subject to the regulatory supervision.

The fulfilment of the regulations of the operating manual is checked by the regulator and through on-site inspections performed by its authorised experts. The control of organisational processes includes, e.g., keeping a shift log, performance of prescribed walkabouts, the proceeding for the change of shift or the handling of alarms and work authorisations. In the area of radiation protection, e.g., compliance with dose limits and regulations on controlled areas and on the storage of radioactive material are inspected. Apart from that, safety-relevant measured values for plant operation or emission of radioactive material are checked within the framework of on-site inspections.

Testing manual

The testing manual regulates the number and proceeding of the in-service inspections on safety-relevant plant systems and components to be performed by the plant operator. 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 requirement), 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. In addition to a correct testing procedure, the testing instructions ensure that the limits of safe operation are also not exceeded during tests.

At specified intervals, also defined in the testing schedule, the authorised experts participate in the in-service inspections of the plant operator on behalf of the supervisory authority. The frequency of such participation depends on the safety significance of the respective inspection. The supervisory authority is informed about the results of the in-service inspections.

Modifications of the testing schedule or the testing instruction are reviewed by the supervisory authority by consultation of authorised experts.

Specification of the procedure for maintenance or modifications

The procedures for in-service inspections, maintenance measures and modifications are specified in the nuclear rules and regulations in the regulatory guideline on maintenance [3-41]. The term maintenance comprises inspection, servicing and repair. In particular, the regulatory guideline on maintenance specifies the work steps from planning of the measure (or occurrence of abnormal operation if it is a reaction to it), during its implementation up to the restoration of operational readiness and documentation. This proceeding ensures that a planned measures is assessed with regard to the actual plant condition and that aspects of plant safety, radiation protection [3-43.1] (→ Article 15) and personal protection which also go beyond merely nuclear issues (industrial safety, fire protection) are completely taken into consideration with appropriate timing. Within the operating procedures, the maintenance guideline is primarily implemented in the maintenance regulation of the operating manual.

The maintenance measures performed by the plant operator on safety-relevant equipment are subjected to on-site inspections performed by the supervisory authority and its authorised experts. An obligation to review maintenance schedules or instructions through authority and experts is not generally laid down in the nuclear rules and regulations but in many cases stipulated in the licensing conditions. Procedures for regulatory review and inspection for those modifications which are not subject to formal licensing were defined by the plant operator and the supervisory authority.

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. At the time of the construction of the plants (1969 to 1989), 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 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.

In recent years, probabilistic approaches have increasingly been used to supplement the engineering judgements. In individual cases, the provisions based on operating experience were also checked and modified under probabilistic considerations (e.g. scope of the tests and test intervals for components of the reactor coolant pressure boundary specified in KTA safety standard [KTA 3201.4]).

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

Although abnormal occurrences during specified normal operation 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 may 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 [KTA 1201].

Design basis accidents

The procedures for the control of design basis accidents are a combination of based on protection goal oriented and event based approaches.

The procedures for control of design basis accidents are based on the following types of written instructions and 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, an accident sequence diagram is available which specifies the proceeding of the shift personnel. 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), and
- 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 “protection goal oriented procedure” of the operating manual.

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

Protection goal oriented procedures in case of design basis accidents

The protection goal 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 protection goal which have to be checked.

Each description of a protection goal oriented procedure 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 protection goal criteria cannot be met, the accident management measures (→ Article 18 (1)), treated in the accident management manual, 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 protection goals 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 protection goal criteria are still met. Detecting that one of the criteria failed, the event-oriented procedures will immediately be interrupted to return to the protection goal oriented procedures in order to bring the respective plant parameters back into normal range.

Emergencies

For emergencies (beyond design basis events), the technical measures to be taken at the plant (→ Article 18 (i)), the accident management measures and auxiliary means required are contained in a separate document, the accident management manual.

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. In addition to the on-site emergency response team, another emergency response team is called in at the manufacturer’s of the plants (AREVA NP) whose task is to provide support in technical issues. A corresponding co-operation agreement exists with the Kerntechnische

Hilfsdienst GmbH, a joint service set up by all operators of the German nuclear power plants to cope with emergencies and eliminate possible consequences. Alarm procedures and organisational structures are specified in the operating manual, further technical measures and accident management procedures in the accident management manual.

19 (v) Engineering and Technical Support

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, electrical and I&C 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.

Apart from this, the plant operators have established own departments for dealing with general issues, in some cases also at the company's headquarters, in which staff from different disciplines work on generic projects.

Regarding the implementation of modification measures, it is first checked which of the above mentioned units are to be involved in view of their competencies. An application for modification is jointly prepared and submitted to the authority. If extensive analyses are required for the obligatory verification of safety, the plant operators refer to the service of the major manufacturers (AREVA NP and Westinghouse). The quality of the safety analyses (measured against the state of the art in science and technology) is ensured by close co-operation of the major manufacturers with numerous research institutes also across national borders. Usually, the plant operators award the contracts for manufacture and installation of components directly to the component manufacturers. The nuclear rules and regulations are such that only qualified manufacturers may be contracted which ensure the quality of their work by adequate quality assurance under their own responsibility (→ Appendix 2 ad [4-1]; General Specification of Basic Safety). From a safety and also economic point of view, the plant operators have a self-interest in the selection of manufacturers and suppliers and their quality assurance. In most cases, long-term contracts exist between plant operators and their suppliers. This way, planning reliability and thus maintenance of competence of the proven and qualified personnel of the suppliers are ensured. Maintenance and repair of the components are mostly included in the suppliers’ service. In order to avoid scheduling conflicts for the highly specialised companies, the plant operators co-ordinate their time schedules for the major maintenance activities and plant outages on a nation-wide scale.

The plant operators are responsible for the documentation of plant-specific data. The corresponding licensing documents (operating, testing and quality assurance manuals) are updated by own staff. One exception are the manuals for the Konvoi plants (Isar 2, Emsland

and Neckarwestheim 2). For these plants, the maintenance of the relevant document is also delegated to the manufacturer (AREVA NP).

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 *Länder* Committee for Nuclear Energy. Accordingly, the operators of German nuclear power plants are obliged to report events to the supervisory authorities in accordance with nation-wide applicable reporting criteria. Then, in 1992, with the promulgation of the Nuclear Safety Officer and Reporting Ordinance [1A-17], the obligation of the operators of nuclear installations (nuclear power plants, research reactors with a thermal power larger than 50 kW and facilities of the fuel cycle) 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 regulatory reporting procedure is embedded in the regulatory supervision of nuclear installations. It mainly serves for the information of the competent supervisory authority. 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 [1A-17] 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. For the reporting criteria, separate explanations exist for application at nuclear power plants, research reactors and facilities of the nuclear fuel cycle.

Any event that is categorised as reportable in accordance with the corresponding reporting criteria is reported by the plant operator to the competent supervisory authority. The plant operator 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 Federal Ministry for the Environment. At the same time, the Federal Office for Radiation Protection (BfS) as central registration agency and the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), an expert organisation working under contract of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety are informed by the competent supervisory authority. 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 (preliminary report). 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 information about the measures taken or planned against recurrence, and
- identifying codification of the event and the affected components.

In addition to the regulatory reporting procedure in accordance with the Reporting Ordinance, the plant operator also categorises the reportable events according to the seven levels of the INES scale. 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.

In recent years, the explanatory notes on the reporting criteria for the nuclear power plants have been revised and specified under consideration of the experiences made. The revision of the reporting criteria themselves currently takes place. For practical considerations, the reporting criteria will, in future, be subdivided into separate technical parts for nuclear power plants, for research reactors, for facilities of the nuclear fuel cycle, for all decommissioned nuclear installations as well as for the storage of spent fuel elements.

The reporting criteria are included in appendices to the Nuclear Safety Officer and Reporting Ordinance [1A-17]. Further requirements on the future organisation of the reporting procedure are currently being discussed for an amendment of the Nuclear Safety Officer and Reporting Ordinance itself.

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 30 years. The resulting feedback of operating experience has been a major contributing factor to the further development of safety in nuclear installations. In addition to targeted studies and analyses, reportable events and indications that are below the reporting threshold are the main source of experience feedback.

Further to the reporting of reportable events, Section 4 of the Nuclear Safety Officer and Reporting Ordinance [1A-17] also contains requirements for the evaluation of other operating experience. It can also be derived from the safety criteria [3-1] that the plant operators have to record and evaluate events and have to take the appropriate actions where necessary.

Reportable events are evaluated by the industry and by the authorities at several levels, i.e. by the operator 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). These multiple-level and independent analyses ensure that each reportable event is evaluated in detail and regarding all aspects.

Evaluation of operating experience by the plant operator

The feedback of experiences within the plant concerned is regulated by procedural instructions in the operating manual. All deficiencies and abnormal occurrences identified by the operating personnel are recorded and documented. A corresponding failure notice is prepared which is followed up by designated units according to degree of priority and discipline concerned. Today, this is performed mainly with a computer-based operational management system. By this, it is ensured that a defined workflow customized to the abnormal occurrence is adhered whose basic principle is defined by the maintenance guideline [3.41] (→ Article 19 (iii)).

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 the safety assessments and, moreover, for a selected evaluation of individual components as well as for generic issues, for trend analyses, ageing management or the determination of reliability parameters for plant-specific probabilistic safety assessments. The operating experience is also systematically analysed by the plant operators with regard to human errors and to possible improvements which may be derived from them (→ Article 12).

Generic information are accessible to the plant operators on an own network. The central interface is the Central Incident Reporting and Evaluation Office of VGB Power Tech (VGB-ZMA). Here, all reportings from nuclear power plants in Germany are directly entered into a database to which also some plants of the manufacturer KWU (now AREVA NP) abroad are connected. Each plant performs a daily database synchronisation. In addition to reportable events, occurrences below the reporting threshold but being of interest to other plants are also recorded.

Further, the VGB-ZMA is the interface to two other institutions, i.e. the connection to WANO reporting system at the competent centre in Paris. The VGB-ZMA collects all incoming WANO reports and checks them for relevance for German nuclear power plants. Each month, a summary in German of selected reports is submitted to the plant operators to be checked for applicability to their plants.

There is also a direct connection between the operating experience evaluation centre of AREVA NP and the VGB-ZMA. According to contractual agreements, AREVA NP has been supporting the plant operators in the evaluation of events since 1989. In addition to selected events from the VGB-ZMA, AREVA NP also evaluates GRS information notices and IRS reports. Applicability to and relevance for German plants is checked. Finally, AREVA NP reports in service information on new findings and studies concerning plant components supplied by it.

Further to the described, direct reporting channels, there are different working groups and committees within the framework of VGB in which the plant operators exchange their experiences. First to be mentioned here are the working groups “BWR” and “PWR” in which the plant managers are organised to discuss concrete events and the consequences to be drawn from them. The plant directors are organised in a committee which primarily deals with general topics. For specialists, there are dedicated working groups in which particular and narrowly defined technical topics are discussed.

Evaluation of operating experience by the supervisory authorities

For safe operation and the tasks of nuclear regulatory supervision, great importance is attached to the early identification of indications for safety-relevant problems or risk factors. Indications may be obtained, in particular, from the evaluation of operation and safety-relevant operating experience as well as from the further development of safety-relevant knowledge and requirements due to the general technical progress. The authorities follow up these indications within the framework of the nuclear regulatory procedure. Through the regular supervision, the nuclear regulatory authorities and their authorised experts are informed about the actual operating condition and the basic operating processes. Further, the operators of the nuclear power plants have to submit 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. Moreover, there are reports of the plant operators on specific topics at regular intervals.

The plant operators also inform the competent nuclear supervisory authority, to some extent irrespective of their obligation to report (→ Article 19 (vi)), about findings from their plants below the reporting threshold and about findings outside their plants that may be of relevance for safety-related issues. The nuclear supervisory authority evaluates these experiences principally with the methods also applied for reportable events with the objective to achieve, where possible, measures against recurrence of negative operating experiences in the plants of their jurisdiction. As far as these operating experiences or other findings made by the experts may also be of interest for the supervisory authorities in other *Länder*, appropriate information is made available. First, information is generally forwarded within the authorised expert organisation. The expert organisations informed this way, then check the findings for applicability to the plants for which they are competent as authorised expert and inform, where necessary, the respective nuclear authority by means of recommendations.

Against the background of all findings from regulatory supervision, however, the reportable evens are the most important basis for the evaluation of operating experience by the authorities, in particular to assess safety deficiencies and to check applicability to other plants.

The *Land* supervisory authority and its expert organisation primarily analyse a reportable event regarding safety significance and the corrective measures to be taken at the affected plant. In a second step, the *Land* authority and its expert organisation investigate the significance of the event for other plants in their area of supervision. In order to allow for an evaluation at national level beyond the borders of the *Länder*, the *Land* supervisory authority forwards information about the reported event to the BMU, the BfS and GRS (→ Article 19 (vi)).

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
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
2004	154	0	6	148	0	147	7	0
2005	135	0	2	133	0	135	0	0
2006	130	0	4	126	0	130	0	0

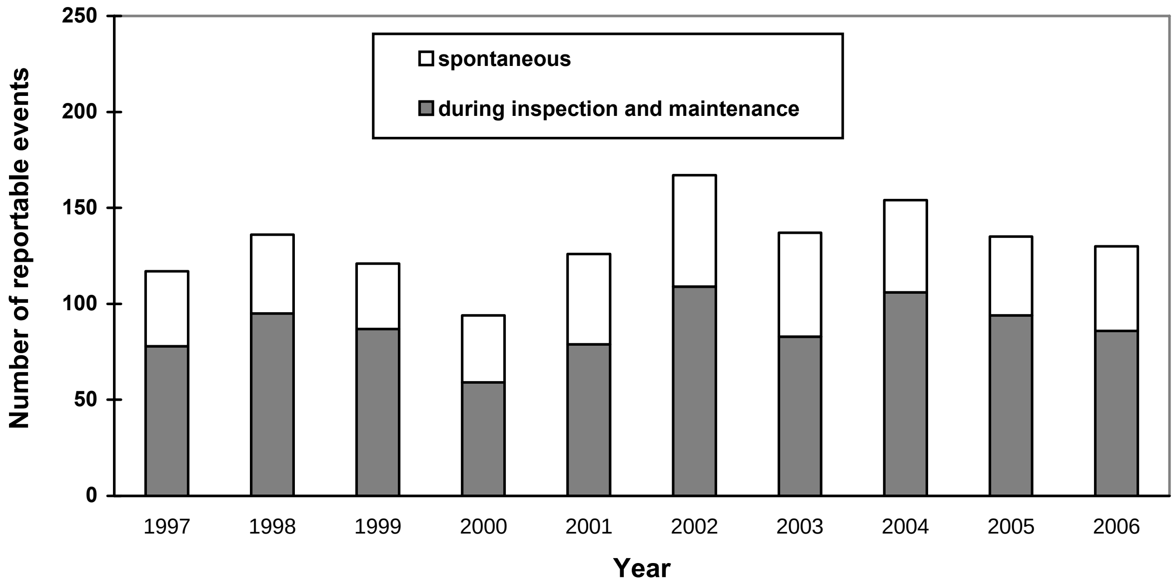


Figure 19-1 Number of Reportable Events from Nuclear Power Plants According to the Kind of Occurrence

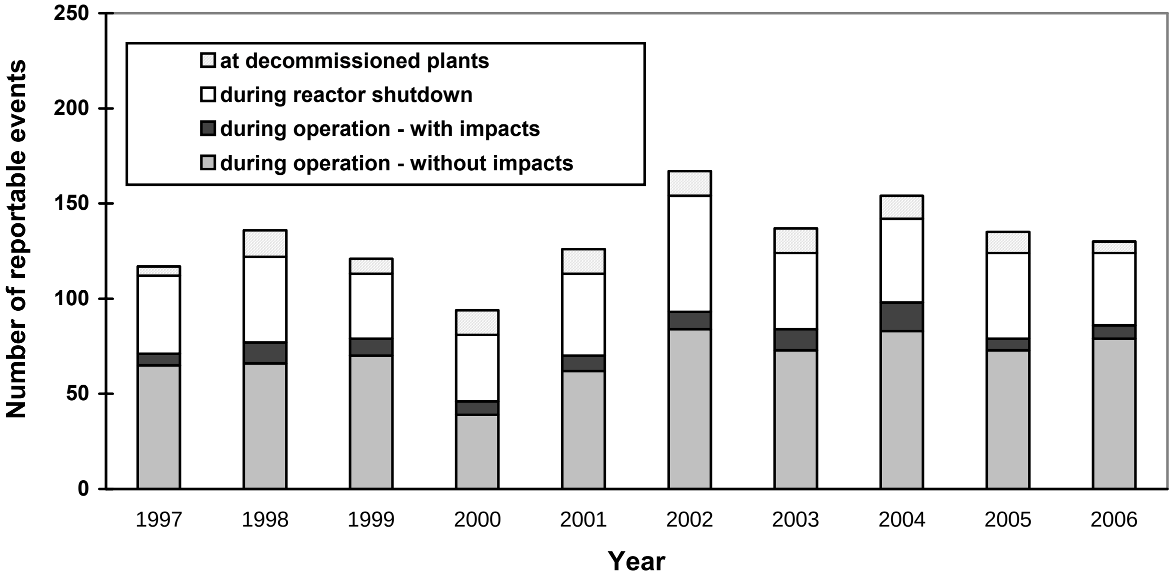


Figure 19-2 **Number of Reportable Events from Nuclear Power Plants According to Mode of and Consequence on Operation (Power Operation, Start-up and Shutdown Operation)**

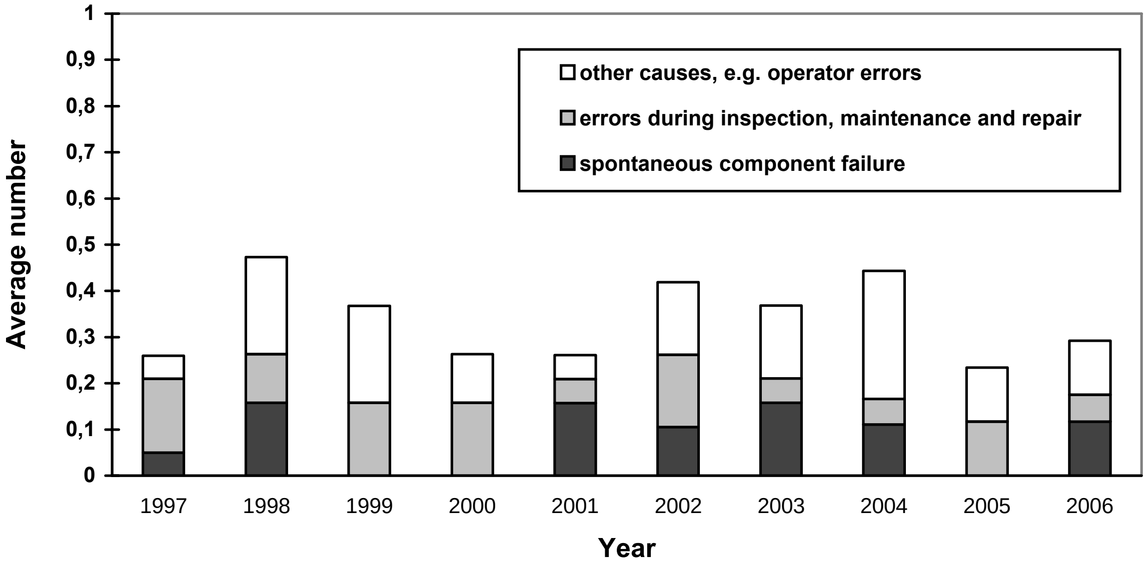


Figure 19-3 **Average Number of Unplanned Reactor Scrams per Plant and Year**

Evaluation of operating experience on behalf of the BMU

Incident Registration Centre of the BfS

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 every month and informs all nuclear authorities of the *Länder*, the expert organisations, the manufacturers and the operators of nuclear power plants as well as the general public in quarterly and annual reports which contain all reportable events in nuclear power plants, research reactors and other nuclear installations. The database of the reportable events at the BfS is accessible to the nuclear supervisory authorities of the *Länder*, the BMU and GRS. Table 19-1 lists the reportable events that occurred over the last ten years also indicating both the German and the INES reporting categories.

Figures 19-1 and 19-2 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 19-3 shows the development over the last ten years of the average number of reactor scrams, also indicating their essential causes.

Evaluation of operating experience by GRS

All reportable events from German nuclear power plants are subjected to a multi-level interdisciplinary analysis by a GRS expert team. This also includes the discussion of the events in a body of experts.

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 Incident Reporting System (IRS) of IAEA and OECD/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 particular regarding possible applicability to German plants. In quarterly reports, short descriptions for every IRS are given and commented regarding applicability to German nuclear power plants. These quarterly reports - together with the corresponding original reports by IRS - are sent to the supervisory authorities and expert organisations as well as to the operators and other competent institutions. In addition, GRS prepares annual reports containing detailed descriptions and evaluations of events of particular significance for German plants. These annual reports, comprising about 20 selected IRS reports, are distributed in the same way as the quarterly reports. The operators 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 current or potential significance and applicability to the safety of other plants. These information notices are submitted by order of the BMU to the supervisory authorities and expert organisations, the plant operators, the manufacturers and other institutions. The 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 and, as essential element, recommendations regarding investigations and, where appropriate, corrective measures to be taken at other plants. In accordance with corresponding licensing provisions, the plant operator 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 authorised experts on behalf of the competent supervisory authorities. GRS collects all comments on the information notices and prepares an annual assessment with particular regard to additional findings. These findings, again, are made available to the above addressees of the information notices in form of a report.

In case of special events at nuclear power plants abroad, GRS prepares on demand of the BMU statements in the short term on the safety significance and applicability to German nuclear power plants. In case of events that might require immediate action of the authorities, the BMU informs the authorities of the *Lander* directly.

Special events at German nuclear power plants that are, according to INES and IRS manual, also of interest for the safety of nuclear power plants in other countries, are reported to the IAEA by GRS in co-ordination with the BMU, the competent *Land* authority and plant operator. This applies, in particular, to all events classified INES Level 2 which are to be reported in the short term.

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 the generic assessments are documented in reports that are distributed in the same way as the information notices. The plant operators again perform a plant-specific evaluation of these reports and, if applicable, 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 developed a method for the performance of trend analyses of parameters important to safety which can be derived from the reportable events.

<u>Discussion of operating experience by advisory commissions and others</u>

Working groups similar to those of the plant operators 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 operators 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 RSK.

With some countries (Brazil, Czech Republic, France, The Netherlands, Spain, Switzerland etc.) there is also a direct bilateral co-operation. This also includes an intensive exchange of operating experience between the respective experts.

Overall picture of the evaluation of operating experience

The procedures implemented for recording, processing, assessing and forwarding of safety-relevant operating experience have proved to be effective and seem to be good practice at the international level. However, experiences also show that regular review and further development of the procedures is important to ensure that, in the long run, new sources of knowledge are considered in the experience feedback and knowledge gaps identified can be filled. The evaluation of international operating experience and the information of the international community about German operating experience could be improved.

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

Pursuant to 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 plant operator 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 plant operators 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 plant operators are partly supported by specialised outside contractors.

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

The clearance levels for radioactive material with minor activity and the clearance procedure are specified in the Radiation Protection Ordinance [1A-8]. For about 300 radionuclides, the Radiation Protection Ordinance prescribes mass-specific clearance levels for solid and liquid material and clearance levels for surface contamination, 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 plant operator and are subject to the supervision by the competent *Land* authority which also performs control measurements.

Pre-treatment and treatment of radioactive waste that cannot be released from regulatory control minimises its volume and converts the primary waste to intermediate products that can be handled and properly conditioned for final disposal. All arising radioactive waste is sorted according to radioactivity and type and is documented. The Radiation Protection Ordinance and the regulatory guideline on radioactive waste without heat generation [3-59] specify 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 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 elements

In order to minimise the number of transports of spent fuel elements, the plant operators have applied for the erection of local interim storage facilities for 13 sites (except Mülheim-Kärlich and Obrigheim) in the years 1998 to 2000 (→ Table 19-2). Due to the shutdown of the Stade nuclear power plant, the application for this site was withdrawn. The storage facilities are dry storage facilities for spent fuel elements in shipping and storage casks applied for, in a first step, for the Castor type. The capacity of these storage facilities is designed to accommodate all spent fuel elements accumulating until final cessation of nuclear power plant operation and to store them also after decommissioning of the respective plant until commissioning of a repository. The time of operation is limited to 40 years, beginning with the emplacement of the first cask. The applications have been approved and the 12 on-site interim storage facilities are in operation. For the Obrigheim site, an on-site interim storage facility was also applied for in 2005 to allow for the depletion of the dry storage facility within the course of plant decommissioning. In order to avoid intermittent bottlenecks in storage capacities, licences were granted for additional interim storage places with a capacity of 12 to 28 storage positions for casks for the nuclear power plants Biblis, Brunsbüttel, Krümmel, Neckarwestheim and Philippsburg. In 2006 and 2007, operation of the additional interim storage places was terminated with the beginning of interim storage on site. The granting of the storage licences for spent fuel elements at the local interim storage facilities falls within the competence of the BfS.

Waste management

Germany is member of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [1E-8]. Report on the activities on radioactive waste management in Germany was last given within the framework of the Review Meeting under this Convention in May 2006.

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

Interim Storage on Site of Nuclear Power Plant	Granting of 1st licence pursuant to Section 6 of the Atomic Energy Act	Capacity HM [Mg]	Storage Positions for Casks	Start of Construc-tion	Commis-sioning
Biblis (KWB)	22.09.2003	1400	135	01.03.2004	18.05.2006
Brokdorf (KBR)	28.11.2003	1000	100	05.04.2004	05.03.2007
Brunsbüttel (KKB)	28.11.2003	450	80	07.10.2003	05.02.2006
Grafenrheinfeld (KKG)	12.02.2003	800	88	22.09.2003	27.02.2006
Grohnde (KWG)	20.12.2002	1000	100	10.11.2003	27.04.2006
Gundremmingen (KRB)	19.12.2003	1850	192	23.08.2004	25.08.2006
Isar (KKI)	22.09.2003	1500	152	14.06.2004	12.03.2007
Krümmel (KKK)	19.12.2003	775	80	23.04.2004	14.11.2006
Lingen (KKE)	06.11.2002	1250	130	18.10.2000	10.12.2002
Neckarwestheim (GKN)	22.09.2003	1600	151	17.11.2003	06.12.2006
Philippsburg (KKP)	19.12.2003	1600	152	17.05.2004	19.03.2007
Unterweser (KKU)	22.09.2003	800	80	19.01.2004	18.06.2007
Additional storage place on Site of Nuclear Power Plant	**Granting of licence pursuant to Section 6 of the Atomic Energy Act**	**Capacity HM [Mg]**	**Storage Positions for Casks**	**Commis-sioning**	**Decom-missioning**
Biblis (KWB)	20.12.2001	300	28	07.03.2002	12.09.2006
Krümmel (KKK)	20.06.2003	120	12	05.08.2004	23.11.2006
Neckarwestheim (GKN)	10.04.2001	250	24	10.04.2001	19.12.2006
Philippsburg (KKP)	31.07.2001	250	24	31.07.2001	30.03.2007

Article 19: Progress and Changes Since 2004

Amendment of KTA safety standards: Requirements for the operating manual and the accident management manual

KTA safety standard [KTA 1201] is currently revised under consideration of the further development of knowledge and the operating requirements. Within the course of this revision, safety standard [KTA 1201] is basically restructured with additional requirements for plant startup and shutdown and plant outage (low power and shutdown state). Moreover, requirements were added on regulations for the control of design basis accident during low power and shutdown states. The requirements for the protection goal oriented part of the operating manual and transition to the accident management manual were also modified. Further, a new KTA safety standard on requirements for the accident management manual is currently also being developed.

Reporting procedure

The plant operators improved their information system regarding the faster recording and assessment of safety-relevant occurrences abroad.

Storage of spent fuel elements

For all local interim storage facilities where applications were made until 2000, licences have been issued. All are constructed and in operation. For the Obrigheim site, a new local interim storage facility was applied for in 2005.

The last transport of irradiated spent fuel for reprocessing was in the first half of 2005. Since 1 July 2005, this disposal is no longer permissible.

Article 19: Future Activities

The regulatory body in Germany will actively participate in the further development of the reporting and evaluation systems for operating experience at international organisations.

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	Constr. Line	a) Date of application b) First criticality
1	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
2	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
3	Neckarwestheim 1 (GKN 1) Neckarwestheim Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK 98,45 %	PWR 840	2	a) 02.04.1971 b) 26.05.1976
4	Brunsbüttel (KKB) Brunsbüttel Schleswig-Holstein	a) Kernkraftwerk Brunsbüttel b) AEG/KWU c) Vattenfall Europe Nuclear Energy 66,7 %	BWR 806	69	a) 10.11.1969 b) 23.06.1976
5	Isar 1 (KKI 1) Essenbach Bayern	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 100 %	BWR 912	69	a) 25.06.1971 b) 20.11.1977
6	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
7	Philippsburg 1 (KKP 1) Philippsburg Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK 100 %	BWR 926	69	a) 20.02.1970 b) 09.03.1979
8	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
9	Krümmel (KKK) Krümmel Schleswig-Holstein	a) Kernkraftwerk Krümmel b) KWU c) Vattenfall Europe Nuclear Energy 50 % E.ON Kernkraft 50 %	BWR 1402	69	a) 18.02.1972 b) 14.09.1983
10	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
11	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

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	Constr. Line	a) Date of application b) First criticality
12	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
13	Philippsburg 2 (KKP 2) Philippsburg Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK %	PWR 1458	3	a) 24.06.1975 b) 13.12.1984
14	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
15	Isar 2 (KKI 2) Essenbach Bayern	a) E.ON Kernkraft b) KWU c) E.ON Kernkraft 75 %	PWR 1475	4 Konvoi	a) 13.02.1979 b) 15.01.1988
16	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
17	Neckarwestheim 2 (GKN 2) Neckarwestheim Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) KWU c) EnKK 98,45 %	PWR 1400	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 mbH 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	ÜWR (VVER) 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 252	a) 31.01.1968 b) 05.01.1977
8	Obrigheim (KWO) Obrigheim Baden-Württemberg	a) EnBW Kernkraft (EnKK) b) Siemens	PWR 357	a) 22.09.1968 b) 11.05.2005
9	Heißdampfreaktor (HDR) Großwelzheim Bayern	a) Forschungszentrum Karlsruhe b) AEG	Super heated steam-cooled reactor 25	a) 14.10.1969 b) 20.04.1971 completely dismantled
10	Würgassen (KWW) Würgassen Nordrhein-Westfalen	a) PreussenElektra b) AEG/KWU	BWR 670	a) 22.10.1971 b) 26.08.1994
11	Niederaichbach (KKN) Niederaichbach Bayern	a) Forschungszentrum Karlsruhe Kernkraftwerkbetrieb GmbH b) Siemens	Pressure tube reactor 106	a) 17.12.1972 b) 31.07.1974 completely dismantled
12	Greifswald 1 (KGR 1) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 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
13	Greifswald 2 (KGR 2) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 03.12.1974 b) 14.02.1990
14	Greifswald 3 (KGR 3) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 06.10.1977 b) 28.02.1990
15	Kompakte natriumgekühlte Reaktoranlage (KNK II) Karlsruhe Baden-Württemberg	a) Kernkraftwerkbetriebs-gesellschaft b) Interatom	FBR 21	a) 10.10.1977 b) 23.08.1991
16	Greifswald 4 (KGR 4) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 22.07.1979 b) 02.06.1990
17	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
18	Mülheim-Kärlich (KMK) Mülheim-Kärlich Rheinland-Pfalz	a) RWE Power b) BBR	PWR 1302	a) 01.03.1986 b) 09.09.1988
19	Greifswald 5 (KGR 5) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 26.03.1989 b) 30.11.1989
	Projects stopped			
20	Greifswald 6 (KGR 6) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) Project stopped
21	Greifswald 7 (KGR 7) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) Project stopped
22	Greifswald 8 (KGR 8) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) Project stopped
23	SNR 300 Kalkar Nordrhein-Westfalen	a) Schnell-Brüter Kernkraftwerksgesellschaft 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
24	Stendal A Stendal Sachsen-Anhalt	a) Altmark Industrie b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 1000	a) b) Project stopped
25	Stendal B Stendal Sachsen-Anhalt	a) Altmark Industrie b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 1000	a) b) Project stopped

*) decommissioned or shut down

Appendix 2 Research Reactors

Appendix 2-1 Research Reactors in Operation

	Research Reactor Site	Licensee	Reactor type thermal output th. n-flow [$cm^{-2}s^{-1}$]	First criticality
1	FRG-1 Geesthacht Schleswig-Holstein	GKSS-Forschungszentrum	Swimming pool/MTR 5 MW $< 1,4 \cdot 10^{14}$	23.10.1958
2	SUR Berlin	Technische Universität Berlin Institut für Energietechnik	SUR-100 100 mW $< 5 \cdot 10^{6}$	26.07.1963 last operation 2000
3	SUR Stuttgart Baden-Württemberg	Universität Stuttgart Institut für Kernenergetik und Energiesysteme	SUR-100 100 mW $< 5 \cdot 10^{6}$	24.08.1964
4	FRMZ Mainz Rheinland-Pfalz	Universität Mainz Institut für Kernchemie	Swimming pool/ TRIGA Mark II 0,1 MW $< 4 \cdot 10^{12}$	03.08.1965
5	SUR Aachen Nordrhein-Westfalen	RWTH Aachen Institut für elektrische Anlagen und Energiewirtschaft	SUR-100 100 mW $< 5 \cdot 10^{6}$	22.09.1965
6	SUR Ulm Baden-Württemberg	Fachhochschule Ulm Labor für Strahlenmess-technik und Reaktortechnik	SUR-100 100 mW $< 5 \cdot 10^{6}$	01.12.1965
7	SUR Kiel Schleswig-Holstein	Fachhochschule Kiel	SUR-100 100 mW $< 5 \cdot 10^{6}$	29.03.1966 last operation 1997
8	SUR Hannover Niedersachsen	Universität Hannover Institut für Werkstoffkunde	SUR-100 100 mW $< 5 \cdot 10^{6}$	09.12.1971
9	SUR Furtwangen Baden-Württemberg	Fachhochschule Furtwangen	SUR-100 100 mW $< 5 \cdot 10^{6}$	28.06.1973
10	BER II Berlin	Hahn-Meitner-Institut Berlin	Swimming pool/MTR 10 MW $< 1,5 \cdot 10^{14}$	09.12.1973
11	AKR Dresden Sachsen	Technische Universität Dresden Institut für Energietechnik	SUR type 2 W $< 3 \cdot 10^{7}$	28.07.1978
12	FRM-II Garching Bayern	Technische Universität München	Swimming pool/ compact core 20 MW $< 8 \cdot 10^{14}$	02.03.2004

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 [$cm^{-2}s^{-1}$]	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 Kern-forschungstechnik 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) 01.02.2005 4. TSG[2]
3	FR 2 Karlsruhe Baden-Württemberg	Forschungszentrum Karlsruhe	Tank type/ D_2O reactor 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) 29.11.2004 4. 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 decomm. 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 16.01.2006 Licence for dismantling
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) facility released
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) 18.12.1996 c) 08.05.2006 SG[4]
10	FRJ-2 (DIDO) Jülich Nordrhein-Westfalen	Forschungszentrum Jülich	Tank type D_2O reactor 23 MW $< 2{\cdot}10^{14}$	a) 14.11.1962 b) 02.05.2006 c) 27.04.2007 AS

[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	Reactor type thermal output th. n-flow [$cm^{-2}s^{-1}$]	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
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)	Argonaut 1 kW $< 1.5 \cdot 10^{11}$	a) 16.12.1962 b) 25.09.1991
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
12	SUR Hamburg	Fachhochschule Hamburg	SUR-100 100 mW $< 5 \cdot 10^{6}$	a) 15.01.1965 b) 08/1992

Appendix 2-3 Research reactors Completely Dismantled

	Decommissioned or dismantled research reactors Site	Licensee	Reactor type thermal output th. n-flow [$cm^{-2}s^{-1}$]	a) First criticality b) Shutdown
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
22	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
23	ZLFR Zittau Sachsen	Hochschule Zittau/Görlitz Fachbereich Maschinenwesen	Tank type/WWR-M 10 W $< 2{\cdot}10^{8}$	a) 25.05.1979 b) 24.03.2005

Appendix 2-3 Research reactors Completely Dismantled

	Decommissioned or dismantled research reactors Site	Licensee	Reactor type thermal output th. n-flow [$cm^{-2}s^{-1}$]	a) First criticality b) Shutdown
24	FRF 2 Frankfurt Hessen	Johann Wolfgang Goethe Universität Frankfurt	modified TRIGA 1 MW $< 3{\cdot}10^{13}$	a) no criticality b) 1980

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	Construction Line 1	Construction Line 2	Construction Line 3	Construction Line 4
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 construction lines 1-3, 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 (after 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	Optimised welding material and enlargement of water gap in the reactor pressure vessel 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, except: closure head, bottom head
- 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; under review **)
Reduction of embrittlement from neutron radiation exposure	Special fuel element management (low leakage loading)	

*) for KRB II: only stabilised austenitic pipes are used
**) for KRB II: the break preclusion concept was approved by the competent authority with a modification licence

2. Emergency Core Cooling PWR

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

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	Construction Line 1	Construction Line 2	Construction Line 3	Construction Line 4
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	Design in accordance with the nuclear safety regulations (→ Article 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	Design in accordance with the nuclear safety regulations (→ Article 17 (i)), emergency systems integrated in the safety system

Appendix 5 Reference List of Nuclear Safety Regulations

(The selection of nuclear power plants as well as structure and order of the references are largely in accordance with the “Handbuch Reaktorsicherheit und Strahlenschutz" (*Handbook on Nuclear Safety and Radiation Protection*) www.bfs.de)

Contents

1 Legislation
1A National Legislation on Nuclear Safety and Radiation Protection

1A-1 **Grundgesetz** für die Bundesrepublik Deutschland vom 23. Mai 1949 (BGBl.I 1949, S. 1), geändert bzgl. Kernenergie durch Gesetz vom 23. Dezember 1959, betreffend §§ 74 Nr. 11a, 87c (BGBl.I 1959, S. 813), erneut geändert bzgl. Kernenergie durch Gesetz vom 28. August 2006 betreffend §§ 73, 74 und 87c (BGBl.I 2006, Nr. 41, S. 2034)

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 1985, Nr. 41, S. 1565), zuletzt geändert durch die 9. Zuständigkeitsanpassungsverordnung vom 31. Oktober 2006 (BGBl.I 2006, Nr.50, S. 2407)

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 1986, Nr. 69, S. 2610), zuletzt geändert durch die 9. Zuständigkeitsanpassungsverordnung vom 31. Oktober 2006 (BGBl.I 2006, Nr. 50, S. 2407)

1A-8 Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (**Strahlenschutzverordnung** - StrlSchV) vom 20. Juli 2001 (BGBl.I 2001, Nr. 38, S. 1714), zuletzt geändert durch Gesetz vom 1. September 2005 (BGBl.I 2005, Nr. 55, S. 2618), Dosiskoeffizienten in (BAnz 2001, Nr. 16)

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, S. 180), letzte Änderung durch das Öffentlichkeitsbeteiligungsgesetz vom 9. Dezember 2006 (BGBl.I 2006, Nr. 58, S. 2819)

1A-11 Verordnung über die Deckungsvorsorge nach dem Atomgesetz (**Atomrechtliche Deckungsvorsorge-Verordnung** - AtDeckV) vom 25. Januar 1977 (BGBl.I 1977, Nr. 8, S. 220), zuletzt geändert durch Gesetz vom 12. August 2005 (BGBl.I 2005, Nr. 49, S. 2365) und Berichtigung vom 11. Oktober 2005 (BGBI.I 2005, Nr. 64, S. 2976

1A-13 Verordnung über Vorausleistungen für die Einrichtung von Anlagen des Bundes zur Sicherstellung und zur Endlagerung radioaktiver Abfälle **Endlagervorausleistungsverordnung** - EndlagerVlV) vom 28. April 1982 (BGBl.I 1982, Nr. 16, S. 562), zuletzt geändert durch Verordnung vom 6. Juli 2004 (BGBl.I 2004, Nr. 33, S. 1476)

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, S. 1766), zuletzt geändert durch VO vom 18. Juni 2002 (BGBl.I 2002, Nr. 36, S. 1869)

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, S. 1918), zuletzt geändert durch Gesetz vom 12. August 2005 (BGBl.I 2005, Nr. 49, S. 2365)

1A-19 Verordnung für die Überprüfung der Zuverlässigkeit zum Schutz gegen Entwendung oder erhebliche Freisetzung radioaktiver Stoffe nach dem Atomgesetz (**Atomrechtliche Zuverlässigkeitsüberprüfungs-Verordnung** - AtZüV) vom 1. Juli 1999 (BGBl.I 1999, Nr. 35, S. 1525), letzte Änderung durch Gesetz vom 11. Oktober 2002 (BGBl.I 2002, Nr. 73, S. 3970)

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, S. 850), zuletzt geändert durch Gesetz vom 21. Juni 2005 (BGBl. I 2005, Nr. 39, S. 2976)

1A-21 **Kostenverordnung** zum Atomgesetz (AtKostV) vom 17. Dezember 1981 (BGBl.I 1981, S. 1457), zuletzt geändert durch Verordnung vom 15. Dezember 2004 (BGBl.I 2004, Nr. 69, S. 3463)

1A-22 Gesetz über die Errichtung eines **Bundesamtes für Strahlenschutz** - BAStrlSchG vom 9. Oktober 1989 (BGBl.I, Nr. 47, S. 1830), geändert durch Gesetz vom 6. April 1998 (BGBl.I 1998, Nr. 21, S. 694), zuletzt geändert durch Gesetz vom 3. Mai 2000 (BGBl.I 2000, Nr. 20, S. 636)

1A-23 **Organisationserlaß** des Bundeskanzlers vom 5. Juni 1986 (BGBl.I, Nr. 25, S. 864) zur Bildung des Bundesministeriums für Umwelt, Naturschutz und Reaktorsicherheit

1B Legal Provisions Also to be Applied in Nuclear Safety and Radiation Protection

1B-1 **Verwaltungsverfahrensgesetz** vom 25. Mai 1976 (BGBl.I 1976, Nr. 59, S. 1253),
Neufassung vom 23. Januar 2003 (BGBl.I 2003, Nr. 4, S. 102), zuletzt geändert durch Artikel 4 Absatz 8 des Gesetzes vom 5. Mai 2004 (BGB.I 2004, Nr. 20, S. 718)

1B-2 **Umweltinformationsgesetz** - UIG vom 22. Dezember 2004 (BGBl.I 2004, Nr. 73, S. 3704),
Hinweis: Umsetzung der RL 2003/4/EG
Umweltinformationskostenverordnung vom 7. Dezember 1994 (BGBl.I 1994, Nr. 88, S. 3732), Neufassung vom 23. August 2001 (BGBl.I 2001, Nr. 45, S. 2247), zuletzt geändert durch Artikel 4 des Gesetzes vom 22. Dezember 2004 (BGBl.I 2004, Nr. 73, S. 3704)

1B-3 **Umweltverträglichkeitsprüfungsgesetz** - UVPG vom 12. Februar 1990 (BGBl.I, Nr. 6, S. 205), Neufassung vom 25. Juni 2005 (BGBl.I 2005, Nr. 37, S. 1757), Berichtigung vom 9. September 2005 (BGBl.I 2005, Nr. 59, S. 2797), zuletzt geändert duch Artikel 2 des Gesetzes vom 21. Dezember 2006 (BGBl.I 2006, Nr. 64, S. 3316) Hinweis: Umsetzung der RL 2001/42/EG

1B-4 **Umweltauditgesetz** vom 7. Dezember 1995 (BGBl.I 1995, S. 1591),
Neufassung vom 4. September 2002 (BGBl.I 2002, Nr. 64, S. 3490) zuletzt geändert durch Artikel 8 Absatz 1 des Gesetzes vom 4. Dezember 2004 (BGBl.I 2004, Nr. 65, S. 3166)

1B-10 **Umwelthaftungsgesetz** vom 10. Dezember 1990 (BGBl.I 1990, Nr. 67, S. 2634), zuletzt geändert durch Artikel 129 des Gesetzes vom 19. April 2006 (BGBl.I 2006, Nr. 18, S. 866)

1B-11 **Strafgesetzbuch** vom 15. Mai 1871 (RGBl. S. 127) in der Fassung der Bekanntmachung vom 13. November 1998 (BGBl.I 1998, Nr. 75, S. 3322), zuletzt geändert durch Artikel 1 des Gesetzes vom 16. Juli 2007 (BGBl.I 2007, Nr. 31, S. 1327)

1B-14 **Raumordnungsgesetz** vom 18. August 1997 (BGBl.I 1997, Nr. 59, S. 2081), zuletzt geändert durch Artikel 10 des Gesetzes vom 9. Dezember 2006 (BGBl.I 2006, Nr. 58, S. 2819)

1B-16 Gesetz zum Schutz vor schädlichen Umwelteinwirkungen durch Luftverunreinigungen, Geräusche, Erschütterungen und ähnliche Vorgänge (**Bundes-Immissionsschutzgesetz** - BImSchG) in der Fassung der Bekanntmachung vom 26. September 2002 (BGBl.I 2002, Nr. 71, S. 3830), zuletzt geändert durch Artikel 3 des Gesetzes vom 18. Dezember 2006 (BGBl.I 2006, Nr. 62, S. 3180); mit diversen Verordnungen

1B-24 **Kreislaufwirtschafts- und Abfallgesetz** vom 27. August 1994 (BGBl.I 1994, Nr. 66, S. 2705), zuletzt geändert durch Artikel 2 des Gesetzes vom 19. Juli 2007 (BGBl.I 2007, Nr. 33, S. 1462)

1B-27 Gesetz zur Ordnung des Wasserhaushalts (**Wasserhaushaltsgesetz**) vom 27. Juli 1957, Neufassung vom 19. August 2002 (BGBl.I 2002, Nr. 59, S. 3245), zuletzt geändert durch Artikel 2 des Gesetzes vom 10. Mai 2007 (BGBl.I 2007, Nr.19, S. 666)

1B-29 **Bundesnaturschutzgesetz** - BNatSchG vom 25. März 2002 (BGBl.I 2002, Nr. 22, S. 1193), zuletzt geändert durch Artikel 3 des Gesetzes vom 10. Mai 2007 (BGBl.I 2007, Nr. 19, S. 666)

1B-31 Verordnung zum Schutz vor gefährlichen Stoffen (**Gefahrstoffverordnung**) vom 23. Dezember 2004 (BGBl.I 2004, Nr. 74, S. 3759), zuletzt geändert durch Artikel 4 der Verordnung vom 6. März 2007 (BGBl.I 2007, Nr. 8, S. 261)

1B-32 Verordnung über die Qualität von Wasser für den menschlicehn Gebrauch
(**Trinkwasserverordnung - TrinkwV 2001)** 21. Mai 2001 (BGBl.I 2001, Nr. 24, S. 959), zuletzt geändert durch Artikel 363 der Verordnung vom 31. Oktober 2006 (BGBl.I 2006, Nr. 20, S. 2407)

1B-33 **Geräte- und Produktsicherheitsgesetz** - GPSG - vom 6. Januar 2004 (BGBl. I 2004, Nr. 1, S. 2), zuletzt geändert durch Artikel 3 Absatz 33 des Gesetzes vom 7. Juli 2005 (BGBl.I 2005, Nr. 42, S. 1970)
Druckgeräteverordnung (14. GPSGV) vom 27. September 2002 (BGBl.I 2002, Nr. 70, S. 3777), zuletzt geändert durch Artikel 21 des Gesetzes vom 6. Januar 2004 (BGBl.I 2004, Nr. 1, S. 2),
Hinweis: "Geräte, die speziell zur Verwendung in kerntechnischen Anlagen entwickelt wurden und deren Ausfall zu einer Freisetzung von Radioaktivität führen kann" sind hier ausgenommen

1B-34 **Betriebsicherheitsverordnung** vom 27. September 2002 (BGBl.I 2002, Nr. 70, S. 3777), zuletzt geändert durch Artikel 5 der Verordnugn vom 6. März 2007 (BGBl.I 2007, Nr. 8, S. 261)
Hinweis: es bleiben "atomrechtliche Vorschriften des Bundes und der Länder unberührt, soweit in ihnen weitergehende oder andere Anforderungen gestellt oder zugelassen werden."

1B-37 **Unfallverhütungsvorschrift Kernkraftwerke** (BGV C16, bisher VBG 30) und Durchführungs-anweisung zur Unfallverhütungsvorschrift (DA zu BGV C16, bisher VGB30) vom 1. Januar 1987 in der Fassung vom 1. Januar 1997 (Berufgenossenschaftlichess Vorschriften- und Regelwerk)

1B-39 Gesetz über Betriebsärzte, Sicherheitsingenieure und andere **Fachkräfte für Arbeitssicherheit** vom 12. Dezember 1973 (BGBl.I 1973, Nr. 105, S. 1885, zuletzt geändert durch Artikel 226 der Verordnung vom 31. Oktober 2006 (BGBl.I 2006, Nr. 50, S. 2407)

1B-41 **Lebensmittel-, Bedarfsgegenstände- und Futtermittelgesetzbuch** - LFGB vom 1. September 2005 (BGBl.I 2005, Nr.55, S. 2618), Neufassung durch Bekanntmachung vom 26. April 2006 (BGBl.I 2006, Nr. 20, S. 945)
Gesetz über den Verkehr mit Lebensmitteln, Tabakerzeugnissen, kosmetischen Mitteln und sonstigen Bedarfsgegenständen (**Lebensmittel- und Bedarfsgegenständegesetz**) vom 15. August 1974 (BGBl.I 1975, Nr. 17, S. 2652), Neufassung und Umbenennung in „Vorläufiges Tabakgesetz“ vom 9. September 1997 (BGBl.I 1997, Nr. 63, S. 2296), zuletzt geändert durch Artikel 1 des Gesetzes vom 21. Dezember 2006 (BGBl.I 2006, Nr. 35, S. 3365)
Bedarfsgegenständeverordnung vom 10. April 1992 (BGBl.I 1992, Nr. 20, S. 866), Neufassung vom 23. Dezember 1997 (BGBl.I 1998, Nr. 1, S. 5), zuletzt geändert durch Artikel 1 der Verordnung vom 20. Dezember 2006 (BGBl.I 2006, Nr. 65, S. 3381)

1B-42 **Informationsfreiheitsgesetz** - IFG) vom 5. September 2005 (BGBl.I 2005, Nr. 57, S. 2722)
Informationsgebührenverordnung vom 2. Januar 2006 (BGBl. I 2006, Nr.1, S. 6)

1B-43 Gesetz über die Öffentlichkeitsbeteiligung in Umweltangelegenheiten nach der EG-Richtlinie 2003/35/EG (**Öffentlichkeitsbeteiligungsgesetz**) vom 9. Dezember 2006 (BGBl.I 2006, Nr. 58, S. 2819), berichtigt am 16. Februar 2007 (BGBl.I 2007, Nr. 6, S. 195)

1B-44 Gesetz über ergänzende Vorschriften zu Rechtsbehelfen in Umweltangelegenheiten nach der EG-Richtlinie 2003/35/EG (**Umwelt-Rechtsbehelfsgesetz**) vom 7. Dezember 2006 (BGBl.I 2006, Nr. 58, S. 2816)

1B-45 Gesetz zur Einführung der **strategischen Umweltprüfung** und zur Umsetzung der Richtlinie 2001/42/EG - SUPG vom 25. Juni 2005 (BGBl.I 2005, Nr. 37, S. 1746)

1B-46 Verordnung über die **Berufsausbildung zur Fachkraft für Schutz und Sicherheit** vom 23. Juli 2002 (BGBl.I 2002, Nr. 51, S. 2757), Hinweis: als Weiterbildung Rahmenplan des DIHK für die "geprüfte Schutz- und Sicherheitskraft", Zustimmung der Genehmigungs- und Aufsichtsbehörde im Einzelfall

1E Multilateral Agreements in the Field of Nuclear Safety and Radiation Protection Including Their National Implementing Provisions

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, in Kraft seit 10. September 1997, 1. Änderung vom Februar 2001, 2. Änderung vom Juni 2004
Gesetz dazu vom 7. Juni 2001 (BGBl.II 2001, Nr. 22, S. 1406), in Kraft für Deutschland seit 8. August 2002
2. ESPOO-Vertragsgesetz vom 17. März 2006 (BGBl.II 2006, Nr. 7, S. 224)

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
Gesetz dazu (Informationsfreiheitsgesetz) vom 5. September 2005 (BGBl.I 2005, Nr. 57, S. 2722)
Gesetz dazu (**Vertragsgesetz**) vom 9. Dezember 2006 (BGBl.II 2006, Nr. 31, S. 1251)

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, Nr. 15, S. 326), zuletzt geändert durch Artikel 4 Absatz 4 des Gesetzes vom 26. Januar 1998 (BGBl.I 1998, Nr. 6, S. 164), 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, S. 434),
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, S. 130)
in Kraft für Deutschland seit 20. April 1997 (BGBl.II 1997, Nr. 14, S. 796)

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, S. 1752)
in Kraft für Deutschland seit 18. Juni 2001 (BGBl.II 2001, Nr. 36, S. 1283)

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, Nr. 34, 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, Nr. 2, S. 17), zuletzt geändert durch Verordnung vom 29. Oktober 2001 (BGBl.I 2001, Nr. 55, S. 2785)
Zusatzprotokoll vom 22. September 1998, in Kraft seit 30. April 2004
Gesetz zum Zusatzprotokoll vom 22. September 1998 vom 29. Januar 2000 (BGBl.I 2000, Nr. 4, S. 70)
Ausführungsgesetz zum Verifikationsabkommen und zum Zusatzprotokoll vom 29. Januar 2000 (BGBl.I 2000, Nr. 5, S. 74)

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), zuletzt geändert durch Artikel 30 des Gesetzes vom 9. September 2001 (BGBl.I 2001, Nr. 47, S. 2331)
in Kraft für Deutschland seit 1. Januar 1976 (BGBl.II 1976, S. 308)
Gesetz hierzu vom 21. Mai 1985 (BGBl.II 1985, Nr. 19, S. 690)
in Kraft für Deutschland seit 7. Oktober 1988 (BGBl.I 1989, Nr. 6, S. 144)

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, S. 202)
in Kraft für Deutschland seit 13. September 2001 (BGBl.II 2001, Nr. 24, S. 786)

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), zuletzt geändert durch Artikel 30 des Gesetzes vom 9. September 2001 (BGBl.I, Nr. 47, S. 2331)
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 Legal Provisions of the European 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 vom 8. Juni 2000, Verordnung (EG) 1209/2000 (ABl.EG 2000, L138), zuletzt geändert durch Verordnung (EURATOM) 1352/2003 der Kommission (ABl.EG 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) 302/2005 der Kommission vom 8. Februar 2005 über die Anwendung der **EURATOM-Sicherungsmaßnahmen** (ABl.EG 2005, Nr. L54)

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 2003/35/EG des EP und des Rates vom 26. Mai 2003 (ABl. 2003, Nr. L156), konsolidierte Fassung 2003,
Hinweis: Umsetzung s. UVP-Gesetz
Richtlinie 2001/42/EG des EP und des Rates vom 27. Juni 2001 über die **Prüfung der Umweltauswirkungen** bestimmter Pläne und Programme (ABl.EG 2001, Nr. L 197),
Hinweis: Umsetzung s. UVP-Gesetz [1B-3]

1F-13 Richtlinie 2001/42/EG des EP und des Rates vom 27. Juni 2001 über die **Prüfung der Umweltauswirkungen** bestimmter Pläne und Programme (ABl.EG 2001, Nr. L 197),
Hinweis: Umsetzung s. UVP-Gesetz [1B-3]

1F-14 Richtlinie 2003/4/EG des EP und des Rates vom 28. Januar 2003 über den **Zugang der Öffentlichkeit zu Umweltinformationen** und zur Aufhebung der RL 90/313/EWG des Rates (ABl.EG 2003, Nr. L 41),
Hinweis: Umsetzung s. UI-Gesetz

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), mehrfach geändert, letzte konsoldierte Fassung 2006

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 01.06.1976 (ABl.EG 1976, Nr. L187),
- Richtlinie 79/343/EURATOM vom 27.03.1977 (ABl.EG 1979, Nr. L83),
- Richtlinie 80/836/EURATOM vom 15.07.1980 (ABl.EG 1980, Nr. L246),
- Richtlinie 84/467/EURATOM vom 03.09.1984 (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 18.07.1989 (ABl.EG 1989, Nr. L211),
- Kommissionsverordnung (EURATOM) 944/89 vom 12.04.1989 (ABl.EG 1989, Nr. L101),
- Kommissionsverordnung (EURATOM) 770/90 vom 29.03.1990 (ABl.EG 1990, Nr. L83)

1F-31 Ratsverordnung (EWG) 2219/89 vom 18.07.1989 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 Verordnung (EWG) 737/90 vom 22. März 1990 über die **Einfuhrbedingungen für landwirtschaftliche Erzeugnisse** mit Ursprung in Drittländern nach dem Unfall im Kernkraftwerk Tschernobyl (ABl.EG 1990, Nr. L82), zuletzt geändert durch VO (EG) 806/2003 des Rates vom 14. April 2003 (ABl.EU 2003, Nr. L128), mit diversen Verordnungen und Durchführungsbestimmungen

1F-33 Entscheidung 87/600/EURATOM des Rates vom 14. Dezember 1987 über Gemeinschaftsvereinbarungen für den **beschleunigten Informationsaustausch** im Fall einer radiologischen Notstandssituation (ECURIE) (ABl.EG 1987, Nr. L371)

1F-34 Abkommen zwischen **EURATOM und Nichtmitgliedsstaaten der EU** über die Teilnahme an Vereinbarungen in der Gemeinschaft für den schnellen Austausch von Informationen in einer radiologischen Notstandssituation (ECURIE) (ABl. 2003, Nr. C102)

2 General Administrative Provisions

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 § 40 Abs. 2, § 95 Abs. 3 Strahlenschutzverordnung und § 35 Abs. 2 Röntgenverordnung (**AVV Strahlenpass**) vom 20. Juli 2004 (BAnz. 2004, Nr. 142a)

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 zur Überwachung der Radioaktivität in der Umwelt nach dem **Strahlenschutzvorsorgegesetz** (AVV-IMIS) vom 13. Dezember 2006 (BAnz. 2006, Nr. 244a)

2-5 Allgemeine Verwaltungsvorschrift zur Durchführung der **Überwachung von Lebensmitteln** nach der Verordnung (Euratom) Nr. 3954/87 des Rates vom 22. Dezember 1987 zur Festlegung von Höchstwerten an Radioaktivität in Nahrungsmitteln und Futtermitteln im Falle eines nuklearen Unfalls oder einer anderen radiologischen Notstandssituation (AVV-Strahlenschutzvorsorge-Lebensmittelüberwachung - AW-StrahLe) vom 28. Juni 2000 (GMBl. 2000, Nr. 25)

2-6 Allgemeine Verwaltungsvorschrift zur Überwachung der **Höchstwerte für Futtermittel** nach der Verordnung (Euratom) Nr. 3954/87 des Rates vom 22. Dezember 1987 zur Festlegung von Höchstwerten an Radioaktivität in Nahrungsmitteln und Futtermitteln im Falle eines nuklearen Unfalls oder einer anderen radiologischen Notstandssituation (Futtermittel-Strahlenschutzvorsorge-Verwaltungsvorschrift - FMStrVVwV) vom 22. Juni 2000 (BAnz. 2000, Nr. 122)

3 Regulatory Guidelines Published by BMU and the Formerly Competent Ministry of the Interior

3-1 Sicherheitskriterien für Kernkraftwerke vom 21. Oktober 1977 (BAnz. 1977, Nr. 206)

3-2 Richtlinie für den Fachkundenachweis von Kernkraftwerkspersonal vom 14. April 1993 (GMBl. 1993, Nr. 20, S. 358), eine Ergänzung für das verantwortliche Kernkraftwerkspersonal zur probewiesen Anwendung für 3 Jahre ab 1. Januar 2005 liegt den Ländern vor

3-3 Richtlinie für den Fachkundenachweis von Forschungsreaktorpersonal vom 16. Februar 1994 (GMBl. 1994, Nr. 11, S. 366)

3-4 Richtlinien über die Anforderungen an Sicherheitsspezifikationen für Kernkraftwerke vom 27. April 1976 (GMBl. 1976, Nr. 15, S. 199)

3-5 Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor vom 26. Juli 07.1976 (GMBl. 1976, Nr. 26, 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. September 1976 (BAnz. 1976, Nr. 179)

3-7.1 Zusammenstellung der in atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen (ZPI) vom 20. Oktober 1982 (BAnz. 1983, Nr. 6a)

3-7.2 Zusammenstellung der zur bauaufsichtlichen Prüfung kerntechnischer Anlagen erforderlichen Unterlagen vom 6. November 1981 (GMBl. 1981, Nr. 33, S. 518)

3-8 Grundsätze für die Vergabe von Unteraufträgen durch Sachverständige vom 29. Oktober 1981 (GMBl. 1981, Nr. 33, S. 517)

3-9.1 Grundsätze zur Dokumentation technischer Unterlagen durch Antragsteller /Genehmigungsinhaber bei Errichtung, Betrieb und Stillegung von Kernkraftwerken vom 19. Februar 1988 (BAnz. 1988, Nr. 56)

3-9.2 Anforderungen an die Dokumentation bei Kernkraftwerken vom 5. August 1982 (GMBl. 1982, Nr. 26, 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. April 1983 (GMBl. 1983, Nr. 13, S. 220), in Überarbeitung

3-15
1. Rahmenempfehlungen für den Katastrophenschutz in der Umgebung kerntechnischer Anlagen vom 9. August 1999 (GMBl. 1999, Nr. 28/29, S. 538), die Überarbeitung ist abgeschlossen, Veröffent lichung erfolgt nach der abschließenden Zustimmung der Beschlussgremien
2. Radiologische Grundlagen für Entscheidungen über Maßnahmen zum Schutz der Bevölkerung bei unfallbedingten Freisetzungen von Radionukliden vom 9. August 1999 (GMBl. 1999, Nr. 28/29, S. 538)
3. Verwendung von Jodtabletten zur Jodblockade der Schilddrüse bei einem kerntechnischen Unfall, Bekanntmachung des BMU vom 20. Oktober 2004 einer Empfehlung der SSK (BAnz. 2004, Nr. 220)

3-23 Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen (REI) vom 7. Dezember 2005 (GMBl. 2006, Nr. 14-17)

3-24 Richtlinie über Dichtheitsprüfungen an umschlossenen radioaktiven Stoffen vom 20. Januar und 4. Februar 2004 (GMBl. 2004, Nr. 27, S. 530)

3-25 Grundsätze zur Entsorgungsvorsorge für Kernkraftwerke vom 19. März 1980 (BAnz. 1980, Nr. 58)

3-27 Richtlinie über die Gewährleistung der notwendigen Kenntnisse der beim Betrieb von Kernkraftwerken sonst tätigen Personen vom 30. November 2000 (GMBl. 2001, Nr. 8, S. 153)

3-31 Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 27. Dezember 1976 (GMBl. 1977, Nr. 4, S. 48), geändert durch GMBl. 1977, Nr. 30, S 664) und die REI (GMBl. 1993, Nr. 29, S. 502)

3-32 Änderung der Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 18. Oktober 1977 (GMBl. 1977, Nr. 30, 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. Oktober 1983 (BAnz. 1983, Nr. 245a)
Störfallberechnungsgrundlagen für die Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gemäß § 28 Abs. 3 StrlSchV vom 18. Oktober 1983 (BAnz. 1983, Nr. 245a), Fassung des Kapitels 4 "Berechnung der Strahlenexposition" vom 29. Juni 1994 (BAnz. 1994, Nr. 222a), Neufassung des Kapitels 4 "Berechnung der Strahlenexposition" gemäß § 49 StrlSchV vom 20. Juli 2001 verabschiedet auf der 186. Sitzung der Strahlenschutzkommission am 11. September 2003, veröffentlicht in Heft 44 in der Reihe "Berichte der Strahlenschutzkommission" 2004

3-34 Rahmenrichtlinie über die Gestaltung von Sachverständigengutachten in atomrechtlichen Verwaltungsverfahren vom 15. Dezember 1983 (GMBl. 1984, Nr. 2, S. 21)

3-37.1 Empfehlung über den Regelungsinhalt von Bescheiden bezüglich der Ableitung radioaktiver Stoffe aus Kernkraftwerken mit Leichtwasserreaktor vom 8. August 1984 (GMBl. 1984, Nr. 21, S. 327)

3-38 Richtlinie für Programme zur Erhaltung der Fachkunde des verantwortlichen Schichtpersonals in Kernkraftwerken vom 1. September 1993 (GMBl. 1993, Nr. 36, S. 645)

3-39 Richtlinie für den Inhalt der Fachkundeprüfung des verantwortlichen Schichtpersonals in Kernkraftwerken vom 23. April 1996 (GMBl. 1996, Nr. 26, S. 555)

3-40 Richtlinie über die im Strahlenschutz erforderliche Fachkunde (Fachkunde-Richtlinie Technik nach Strahlenschutzverordnung) vom 21. Juni 2004 (GMBl. 2004, Nr. 40/41, S. 799), Änderung vom 19. April 2006 (GMBl. 2006, Nr. 38, S. 795)

3-41 Richtlinie für das Verfahren zur Vorbereitung und Durchführung von Instandhaltungs- und Änderungsarbeiten in Kernkraftwerken vom 01.06.1978 (GMBl. 1978, Nr. 22, S. 342), in Überarbeitung

3-42 Richtlinie für die Physikalische Strahlenschutzkontrolle zur Ermittlung der Körperdosen
Teil 1: Ermittlung der Körperdosis bei äußerer Strahlenexposition (§§ 40, 41, 42 StrlSchV; § 35 RöV) vom 8. Dezember 2003 (GMBl. 2004, Nr.22, S. 410), anzuwenden ab 1. März 2004

3-42.1 Richtlinie für die physikalische Strahlenschutzkontrolle zur Ermittlung der Körperdosen Teil 2: „Ermittlung der Körperdosis bei innerer Strahlenexposition (Inkorporationsüberwachung) (§§ 40, 41 und 42 StrlSchV)“ vom 12. Januar 2007 (GMBl. 2007, Nr. 31/32, S. 623), anzuwenden ab 1. März 2007

3-43 Richtlinie für den Strahlenschutz des Personals bei der Durchführung von Instandhaltungsarbeiten in Kernkraftwerken mit Leichtwasserreaktor;
Teil 1: Die während der Planung der Anlage zu treffende Vorsorge vom 10. Juli 1978
(GMBl. 1978, Nr. 28, S. 418), in Überarbeitung

3-43.1 Richtlinie für den Strahlenschutz des Personals bei Tätigkeiten der Instandhaltung, Änderung, Entsorgung und des Abbaus in kerntechnischen Anlagen und Einrichtungen:
Teil 2: Die Strahlenschutzmaßnahmen während des Betriebs und der Stilllegung einer Anlage oder Einrichtung - IWRS II vom 17. Januar 2005 (GMBl. 2005, Nr. 13, S. 258)

3-44 Kontrolle der Eigenüberwachung radioaktiver Emissionen aus Kernkraftwerken vom 5. Februar 1996 (GMBl. 1996, Nr. 9/10, S. 247)

3-49 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke; Einzelfehlerkonzept - Grundsätze für die Anwendung des Einzelfehlerkriteriums vom 2. März 1984 (GMBl. 1984, Nr. 13, S. 208)

3-50 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 17. Mai 1979 (GMBl. 1979, Nr. 14, 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. November 1979 (GMBl. 1980, Nr. 5, 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 Erläuterungen zu den Meldekriterien für meldepflichtige Ereignisse in Anlagen zur Spaltung von Kernbrennstoffen (12/04)
- Zusammenstellung von in den Meldekriterien verwendeten Begriffen (Anlagen zur Spaltung von Kernbrennstoffen) (05/04)
- Meldeformular (Anlagen zur Spaltung von Kernbrennstoffen) (04/04)

3-52.3 Erläuterungen zu den Meldekriterien für meldepflichtige Ereignisse in Anlagen, die nicht der Spaltung von Kernbrennstoffen dienen (Stand 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 Spaltprodukten (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 Schichtpersonals in Forschungsreaktoren vom 14. November 1997 (GMBl. 1997, Nr. 42, S. 794)

3-54 Rahmenempfehlung für die Fernüberwachung von Kernkraftwerken vom 12. August 2005 (GMBl. 2005, Nr. 51, S. 1049)

3-54.1 Empfehlung zur Berechnung der Gebühr nach § 5 AtKostV für die Fernüberwachung von Kernkraftwerken (KFÜ) vom 21. Januar 1983 (GMBl. 1983, Nr. 8, S. 146)

3-57 Anforderungen an den Objektsicherungsdienst und an Objektsicherungs- beauftragte in kerntechnischen Anlagen der Sicherungskategorie I (OSD-Richtlinie) vom 8. April 1986 (GMBl. 1986, Nr. 14, S. 242), in Überarbeitung

3-57.3 Richtlinie für den Schutz von Kernkraftwerken mit Leichtwasserreaktoren gegen Störmaßnahmen oder sonstige Einwirkungen Dritter vom 06.12.1995 (GMBl. 1996, S. 32, Nr. 2, ohne Wortlaut)

3-59 Richtlinie zur Kontrolle radioaktiver Abfälle mit vernachlässigbarer Wärmeentwicklung, die nicht an eine Landessammelstelle abgeliefert werden vom 16. Januar 1989 (BAnz. 1989, Nr. 63a), letzte Ergänzung vom 14. Januar 1994 (BAnz. 1994, Nr. 19)

3-61 Richtlinie für die Fachkunde von Strahlenschutzbeauftragten in Kernkraftwerken und sonstigen Anlagen zur Spaltung von Kernbrennstoffen vom 10. Dezember 1990 (GMBl. 1991, Nr. 4, 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. Januar 1991 (GMBl. 1991, Nr. 9, S. 228)

3-65 Anforderungen an Lehrgänge zur Vermittlung kerntechnischer Grundlagenkenntnisse für verantwortliches Schichtpersonal in Kernkraftwerken - Anerkennungskriterien - vom 10. Oktober 1994

3-67 Richtlinie über Anforderungen an Personendosismeßstellen nach Strahlenschutz- und Röntgenverordnung vom 10. Dezember 2001 (GMBl. 2002, Nr. 6, S. 136)

3-69 Richtlinie für die Überwachung der Radioaktivität in der Umwelt nach dem Strahlenschutzvorsorgegesetz
Teil I: Meßprogramm für den Normalbetrieb (Routinemeßprogramm) vom 28. Juli 1994 (GMBl. 1994, Nr. 32, S. 930), in Überarbeitung

3-69.2 Teil II: Meßprogramm für den Intensivbetrieb (Intensivmeßprogramm) vom 19. Januar 1995 (GMBl. 1995, Nr. 14, S. 391), 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, S. 29)

3-73 Leitfaden zur Stillegung von Anlagen nach § 7 des Atomgesetzes vom 14. Juni 1996 (BAnz. 1996, Nr. 211a), in Überarbeitung

3-74 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)
3-74.3 Sicherheitsüberprüfung gemäß § 19 des Atomgesetzes
- Leitfaden Probabilistische Sicherheitsanalyse

Bekanntmachung vom 30. August 2005 (BAnz. 2005, Nr. 207)

3-79 Schadensvorsorge außerhalb der Auslegungsstörfälle, RdSchr. des BMU vom 15. Juli 2003, RS I 3 - 10100/0

3-80 Entschließung des Länderausschusses für Atomkernenergie zu Entscheidungen nach der Strahlenschutzverordnung, deren Wirkung über den Bereich eines Landes hinausgeht, RdSchr. des BMU vom 8. Dezember 2003 RS I 1 - 17031/47

3-81 Grundlagen für Sicherheitsmanagementsysteme in Kernkraftwerken, Bekanntmachung des BMU vom 29. Juni 2004 (BAnz. 2004, Nr. 138)

4 Recommendations of RSK and SSK, Other Relevant Provisions and Recommendations

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)

4-2 Kriterien für die Alarmierung der Katastrophenschutzbehörde durch die Betreiber kerntechnischer Einrichtungen, Gemeinsame Empfehlung der SSK und RSK, 1994, revidiert 2003, (BAnz 2004, Nr. 89)
Verabschiedet auf der 186. Sitzung der SSK am 11./12.09.2003
Verabschiedet auf der 366. Sitzung der RSK am 16.101.2003
Veröffentlicht in Heft 39 der Reihe „Berichte der Strahlenschutzkommission“

4-2.1 Erläuterungen zu den Kriterien für die Alarmierung der Katastrophenschutzbehörde durch die Betreiber kerntechnischer Anlagen, Stellungnahme der Strahlenschutzkommission,
Verabschiedet auf der 127. Sitzung der SSK am 12.10.1994
Veröffentlicht in Heft 3 der Reihe „Berichte der Strahlenschutzkommission“

4-3 Übersicht über Maßnahmen zur Verringerung der Strahlenexposition nach Ereignissen mit nicht unerheblichen radiologischen Auswirkungen (Maßnahmenkatalog),
Band 1 und 2, herausgegeben vom Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit, Oktober 1999
Teil 3: Behandlung und Entsorgung kontaminierter landwirtschaftlicher Produkte,
Verabschiedet auf der 200. Sitzung der SSK am 30.06./01.07.2005

4-4 Leitfaden für den Fachberater Strahlenschutz der Katastrophenschutzleitung bei kerntechnischen Notfällen, Stellungnahme der Strahlenschutzkommission
Verabschiedet auf der 182. Sitzung der SSK am 04.-06.12.2002
Veröffentlicht in Heft 37 der Reihe „Berichte der Strahlenschutzkommission“

4-4.1 Erläuterungsbericht zum Leitfaden für den Fachberater Strahlenschutz der Katastrophenschutzleitung, Stellungnahme der Strahlenschutzkommission
Verabschiedet auf der 185. Sitzung der SSK am 03./04.07.2003
Veröffentlicht in Heft 38 der Reihe „Berichte der Strahlenschutzkommission“

4-5 Feuerwehrdienstvorschrift FwDV 500 „Einheiten im ABC-Einsatz“, Stand 2003,
Die FwDV 500 wurde am 15. und 16.09.2003 vom Ausschuss Feuerwehrangelegenheiten, Katastrophenschutz und zivile Verteidigung (AFKzV) genehmigt und den Ländern zur Einführung empfohlen.
Erläuterungen zur FwDV 500 „Einheiten im ABC-Einsatz“ ,Stand: Feb. 2004

4-6 Leitfaden Polizei LF 450 "Gefahren durch chemische, radioaktive und biologische Stoffe" Ausgabe 2005, Stand: 10.11.2005, nicht veröffentlicht - nur für den Dienstgebrauch durch die Polizei

4-7 Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke:
Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-37/05
Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-38/05,
herausgegeben vom Bundesamt für Strahlenschutz, Oktober 2005

5 Standards of the Nuclear Safety Standards Commission (KTA)

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>1000 KTA-interne Verfahrensregeln</u>					
	<u>1100 Begriffe und Definitionen</u>		-		-	-
	(KTA-Begriffesammlung KTA-GS-12) *(KTA Collection of Definitions)*	01/06		6/91 1/96 1/04		
	<u>1200 Allgemeines, Administration, Organisation</u> *<u>General, Administration, Organization</u>*					
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	-	16.11.04	+
	<u>1300 Radiologischer Arbeitsschutz</u> *<u>Radiological (aspects of) industrial safety</u>*					
1301.2 *	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	-	16.11.04	+
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	+
	<u>1400 Qualitätssicherung</u> *<u>Quality Assurance</u>*					
1401 *	Allgemeine Forderungen an die Qualitätssicherung *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 aktivitätsführende 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 Fass-ung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fass-ungen	Bestäti-gung der Weiter-gültigkeit	Engl. Über-setz-ung
1408.2 *	Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitäts-führende 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 aktivitäts-führende 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 Kern-kraftwerken *Stationary System for Monitoring Area Dose Rates within Nuclear Power Plants*	11/04	35 a 19.02.05	6/91	-	+
1502.1	Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 1: Kernkraftwerke mit Leichtwasser-reaktor *Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 1: Nuclear Power Plants with Light Water Reactors*	11/05	101a 31.05.06	6/86	-	+
(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 Tempera-ture 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 2: 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	-	16.11.04	+
1503.3	Überwachung der Ableitung gasförmiger und aerosolgebundener radioaktiver Stoffe; Teil 3: 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	-	16.11.04	+
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 (Regel wurde am 16.11.04 zurückgezogen) *Measuring Local Dose Rates in Exclusion Areas of Nuclear Power Plants (withdrawn on 16.11.04)*	6/86	162 a 03.09.86 Berichtigung 229 10.12.86	-	16.11.04 zurück-gezogen	+
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*	11/06	245b 30.12.06	9/88	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	22.11.05	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
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	-	22.11.05	+
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	-	22.11.05	+
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	22.11.05	+
	<u>2200 Einwirkungen von außen</u> *<u>External Events</u>*						
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 Berichtigung 115	30.01.91 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	7.11.06	+
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	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
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*	11/04	133a	16.07.05	6/92	-	+
	2500 Bautechnik *Civil Engineering*						
2501	Bauwerksabdichtungen von Kernkraftwerken *Waterproofing of Structures of Nuclear Power Plants*	11/04	172 a	13.09.02	9/88 6/02	-	+
2502 *	Mechanische Auslegung von Brennelementlagerbecken 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	+
	3000 Systeme allgemein *Genaral Systems*						
	3100 Reaktorkern und Reaktorregelung *Reactor Core and Reactor Control*						
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	+
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	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
(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	+
(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	-	16.11.04	+
	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	11.11.03	+
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	-	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
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	-	+
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	-	7.11.06	+
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	-	20.06.00	+
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	-	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
3211.2 *	Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 2: Auslegung, Konstruktion und Berechnung *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	-	-	+
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	+
	3300 Wärmeabfuhr *Heat Removal*					
3301 *	Nachwärmeabfuhrsysteme von Leichtwasserreaktoren *Residual Heat Removal Systems of Light Water Reactors*	11/84	40 a 27.02.85	-	15.06.99 1)	+
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	+
	3400 Sicherheitseinschluss *Containment*					
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	22.11.05	+
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	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
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	7.11.06	+
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	-	16.11.04	+
3403 *	Kabeldurchführungen im Reaktorsicherheitsbehälter 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 Reaktorsicherheitsbehälter 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	-	11.11.03	+
3405 *	Integrale Leckratenprüfung des Sicherheitsbehälters 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 Reaktorsicherheitsbehälter *Pipe Penetrations through the Reactor Containment Vessel*	6/91	113 a 23.06.92	-	7.11.06	+
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	-	16.11.04	+
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	-	16.11.04	+
	3500 Instrumentierung und Reaktorschutz *Instrumentations and Reactor Protection*					
3501 *	Reaktorschutzsystem und Überwachungseinrichtungen 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	16.11.04	+
3503	Typprüfung von elektrischen Baugruppen des Reaktorschutzsystems *Type Testing of Electrical Modules for the Reactor Protection System*	11/05	101a 31.05.06	6/82 11/86	-	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
3504	Elektrische Antriebe des Sicherheits-systems in Kernkraftwerken *Electrical Drives of the Safety System in Nuclear Power Plants*	11/06	245b	30.12.06	9/88	-	-
3505	Typprüfung von Messwertgebern und Messumformern des Reaktorschutzsystems *Type Testing of Measuring Transmitters and Transducers of the Reactor Protection System*	11/05	101a	31.05.06	11/84	-	-
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	-	+
	3600 Aktivitätskontrolle und -führung Activity Control and Activity Management						
3601	Lüftungstechnische Anlagen in Kernkraftwerken *Ventilation and Air Filtration Systems in Nuclear Power Plants*	11/05	101a	31.05.06	6/90	-	+
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 2)	+
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*	11/05	101a	31.05.06	6/83	-	+
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	-	16.11.04	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr.	vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
	<u>3700 Energie- und Medienversorgung</u> *<u>Energy and Media Supply</u>*						
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	16.11.04	+
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)	22.11.05	-
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	16.11.04	+
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	16.11.04	+
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*	11/06	245b	30.12.06	9/88 6/99	-	-
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	-	22.11.05	+
	<u>3900 Systeme, sonstige</u> *<u>Other Systems</u>*						
3901	Kommunikationsmittel für Kernkraftwerke *Communication Devices for Nuclear Power Plants*	11/04	35a	19.02.05	3/77 3/81	-	11/04
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	16.11.04	6/99
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	16.11.04	+

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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) The KTA issued on its 43th meeting "Instructions for the user of KTA 3301 (11/84)".

2) In this safety standard, the HTR (high temperature reactor)-related requirements were deleted.

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