Nuclear Safety in Germany

Report
under the Convention on Nuclear Safety
by the Government
of the
Federal Republic of Germany
for
the First Review Meeting in April 1999

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Preface

The peaceful use of atomic energy goes back over four decades in Germany and over that period we have created a closely woven network of regulations and safety provisions. Compliance is ensured by strict government supervision carried out by independent authorities. Germany has always attached the highest priority to nuclear safety. An ongoing programme of backfitting and the application of new knowledge gained from operating experience at home and abroad have helped to improve the already high safety status.

As a result of these efforts, our nuclear power plants have a positive safety record - an achievement echoed in the report of the Federal Republic of Germany to the Convention on Nuclear Safety, which is hereby submitted. The report demonstrates in one article after another that Germany fully meets the requirements of the Convention.

The review of operations at German nuclear power plants shows that the annual number of nationally reportable incidents has fallen to half the number recorded ten years ago and that the proportion of incidents with safety implications is very small. Thus, since the introduction of the international INES scale in Germany, there has been no year in which more than 11 occurrences were classified as INES 1 incidents. Only a single INES 2 event has so far been recorded in a German nuclear power plant (1998). Operating disturbances have become a much rarer occurrence and there has been a considerable reduction in the already low level of radiation exposure of personnel.

In the last ten years, radiation exposure of the population caused by nuclear power plant operations has, even in the immediate vicinity of installations, generally been less than one thirtieth of the 0.3 mSv annual limit set for operating conditions.

I support all efforts to apply high safety standards worldwide. Using the ever greater scientific and technological possibilities at our disposal, we should actively further develop these standards, while not forgetting the need to make improvements to older installations wherever this is possible and feasible.

I am confident that the Convention on Nuclear Safety will make a major contribution to the harmonisation of nuclear safety standards at a high level and thus to the sustained improvement of safety in the use of atomic energy.

The Government of the Federal Republic of Germany is convinced that, for energy, environment and technology policy reasons, the peaceful use of atomic energy will remain necessary for the foreseeable future. This means using nuclear power plants as part of a balanced energy mix that includes fossil and renewable energy sources. This approach serves not only to secure national energy supplies but also to prevent resource depletion and protect the environment. The climate problem in particular underlines the importance of atomic energy as a virtually CO_2-free option for energy provision.

Safety concerns must in future continue to have priority over economic considerations. Only when a critical general public worldwide is convinced that atomic energy is safe, will nuclear power plants be able to make their contribution to world energy supplies as part of a long-term perspective.

The first real test for the Convention on Nuclear Safety is the Review Meeting in Vienna in April 1999, where the national reports will come under international public scrutiny.

Dr Angela Merkel
Federal Minister for the Environment,
Nature Conservation and Nuclear Safety

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Abbreviations

BfS	*Bundesamt für Strahlenschutz* Federal Office for Radiation Protection
BMBF	*Bundesministerium für Bildung, Wissenschaft, Forschung und Technologie* Federal Ministry of Education, Science, Research and Technology
BMU	*Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit* *- Bundesumweltministerium -* Federal Ministery for the Environment, Nature Conservation and Nuclear Safety
PWR	Pressurized Water Reactor
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
ICRP	International Commission on Radiological Protection
KTA	*Kerntechnischer Ausschuß* Nuclear Safety Standards Commission
OECD/NEA	Organisation for Economic Co-operation and Development/ Nuclear Energy Agency
RSK	*Reaktor-Sicherheitskommission* Reactor Safety Commission
SSK	*Strahlenschutzkommission* Commission on Radiological Protection
BWR	Boiling Water Reactor
WANO	World Association of Nuclear Operators

Introduction

In the Federal Republic of Germany the principles of a democratic social order, namely, the government's responsibility to protect life and health and natural sources of life, the separation of powers, the independence of licensing and supervisory authorities and the supervision of administrative actions by independent courts are established by the Constitution (Basic Law). The legislation, administrative authorities and jurisdiction created specifically for the peaceful utilisation of nuclear energy establish the framework of the system for assuring the protection of life, health and property of the directly employed and the general public from the hazards of nuclear energy and the damaging effects of ionising radiation as well as for the regulation and supervision of safety during construction and operation of nuclear installations. In accordance with the legal requirements in the field of nuclear technology, the assurance of safety receives the topmost priority. The nuclear safety regulations are in compliance with the internationally accepted safety principles as specified, for example, in the ”Safety Fundamentals” of the IAEA. A main intention of the safety policy of the German Federal Government in the field of nuclear energy is that the licensees of nuclear installations within their independent full responsibility develop a high safety culture.

In Germany, from a very early stage, a technical and scientific environment was created with federal support which favoured the further development of the light water reactors originally built under foreign licence agreements. On account of the high population density and the correspondingly narrow choice of sites, a progressive safety concept was developed in connection with the construction of the first major nuclear power plants that is characterised by a basic safety of all pressure retaining components, by the separation of the safety systems into independent redundancies, by designing the overall plant for a good accessibility during inspection, servicing and repair, as well as by protection against external impacts with low probabilities of occurrence and by the introduction of on-site accident management.

In the different stages of its development, the safety concept was verified by large-scale technical experiments and by independently developed computer codes for the analysis of accidents. Even if the current stagnation in the construction of nuclear installations thins out the technical, scientific and industrial resources, an efficient infrastructure will remain - taking into account the entire European Market - to assure safe operation of the nuclear installations in the future.

The Federal Republic of Germany meets the requirements of the Convention on Nuclear Safety and of other international conventions in the field of nuclear energy and radiation protection. Of major importance is the maintenance and further development of the internationally acclaimed already high standard of nuclear safety in German nuclear power plants. Germany pursues this objective by the continuous improvement of its nationally applied measures. The further development of nuclear safety means that even in the future existing nuclear power plants will be backfitted in so far as the actual state of the art in science and technology makes a safety improvement appear possible and reasonable. Furthermore, Germany has established an additional safety objective for future nuclear power plants: namely, that, even in case of a severe beyond-design-basis accident with core meltdown - which has been practically excluded as a result of the safety precautions in existing nuclear power plants - the effects of the accident will essentially be restricted to the plant; this objective was also embedded in the Atomic Energy Act in 1994.

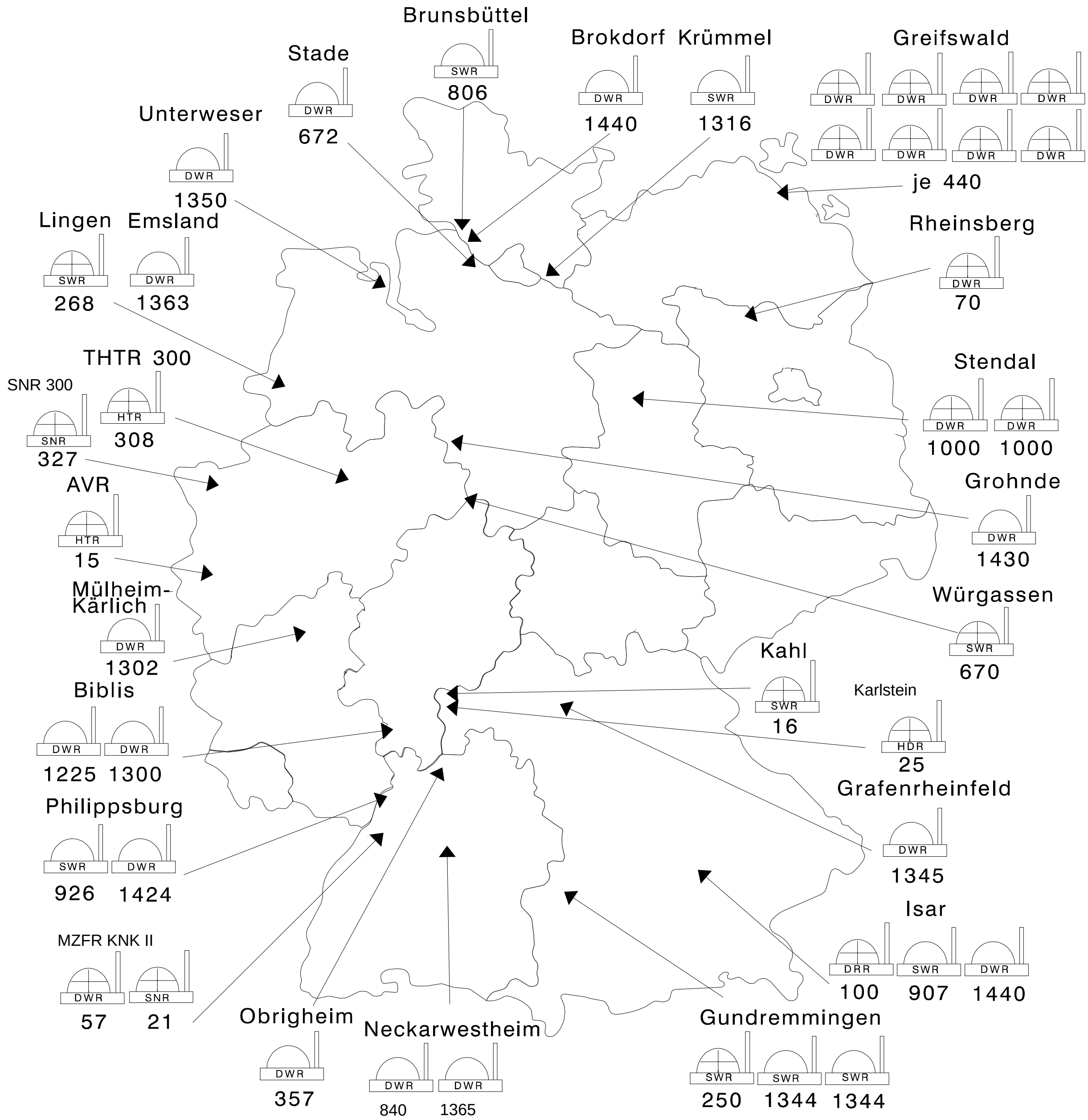
Furthermore, the German Federal Government sees a special responsibility for Germany within the framework of international co-operation and seizes upon this responsibility by supporting efforts in nuclear safety in Central and Eastern Europe.

The national report required in accordance with Article 5 of the Convention on Nuclear Safety is outlined in the following.

Both in structure and content the report closely follows the Convention and the associated Guidelines Regarding National Reports. The numbering of the chapters corresponds to the numbering of the articles in the Convention. Each commitment is individually commented on. As suggested in the Guidelines, statements made in the report are prevalently generic in nature, however, plant specific details are presented wherever necessary to support the statement that requirements of the Convention are being met. The tables in Chapter 6 list the operating nuclear power plants and those that have been decommissioned in the meantime.

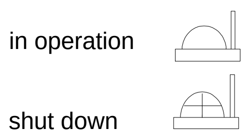
As proof for the fulfilment of commitments, the pertinent national laws, ordinances and standards are commented on, and it is described how the essential safety requirements are being met. In this national report special emphasis is put on describing the licensing procedure and regulatory surveillance as well as the applied measures by the licensees within their independent full responsibility for maintaining an appropriate safety level.

The Appendix to this report contains a compilation of design characteristics important to safety for the operating nuclear power plants (nuclear installations as defined by the Convention), sorted according to type and design generation. It also contains a comprehensive list of the legal and administrative provisions, of the nuclear safety standards and regulatory guides that have to be considered for the safety of nuclear installations as defined by the Convention.



Legend:

DWR	Pressurized Water Reactor
SWR	Boiling Water Reactor
SNR	Fast Breeder Reactor
HTR	High Temperatur Reactor
DRR	Pressure Tube Reactor
HDR	Superheated Steam-Cooled Reactor



Numbers indicate Gross Capacity [MWe]

Figure 6-1 Nuclear Power Plants in Germany

Table 6-1 Nuclear Power Plants in Operation

	Nuclear power plants in operation Site	a) licensee b) manufacturer c) major shareholder	Type Gross capacity MWe	Design generation / construction line	a) date of application b) first criticality
1	Obrigheim (KWO) Obrigheim Baden-Württemberg	a) Kernkraftwerk Obrigheim GmbH b) Siemens c) Energie Baden-Württemberg AG 63%	PWR 357	**1st**	a) 16.07.64 b) 22.09.68
2	Stade (KKS) Stade Niedersachsen	a) Kernkraftwerk Stade GmbH b) KWU c) PreussenElektra AG 66 2/3%	PWR 672	**1st**	a) 28.07.67 b) 08.01.72
3	Biblis A (KWB A) Biblis Hessen	a) RWE Energie AG b) KWU c) RWE Energie AG 100%	PWR 1225	**2nd**	a) 11.06.69 b) 16.07.74
4	Biblis B (KWB B) Biblis Hessen	a) RWE Energie AG b) KWU b) RWE Energie AG 100%	PWR 1300	**2nd**	a) 03.05.71 b) 25.03.76
5	Neckarwestheim 1 (GKN 1) Neckarwestheim Baden-Württemberg	a) Gemeinschaftskernkraftwerk Neckar GmbH b) KWU c) Neckarwerke 70%	PWR 840	**2nd**	a) 02.04.71 b) 26.05.76
6	Brunsbüttel (KKB) Brunsbüttel Schleswig-Holstein	a) Kernkraftwerk Brunsbüttel GmbH b) AEG/KWU c) HEW 66 2/3%	BWR 806	**69**	a) 10.11.69 b) 23.06.76
7	Isar 1 (KKI 1) Essenbach Bayern	a) Bayernwerk Kernenergie GmbH b) KWU c) Bayernwerk AG 50%	BWR 907	**69**	a) 25.06.71 b) 20.11.77
8	Unterweser (KKU) Esenshamm Niedersachsen	a) Kernkraftwerk Unterweser GmbH b) KWU c) PreussenElektra AG 100%	PWR 1350	**2nd**	a) 07.04.71 b) 16.09.78
9	Philippsburg 1 (KKP 1) Philippsburg Baden-Württemberg	a) EnBW Kraftwerke GmbH b) KWU c) Energie Baden-Württemberg AG 100 %	BWR 926	**69**	a) 20.02.70 b) 09.03.79
10	Grafenrheinfeld (KKG) Grafenrheinfeld Bayern	a) Bayernwerk Kernenergie GmbH b) KWU c) Bayernwerk AG 100%	PWR 1345	**3rd**	a) 07.06.73 b) 09.12.81
11	Krümmel (KKK) Krümmel Schleswig-Holstein	a) Kernkraftwerk Krümmel GmbH b) KWU c) HEW 50% PreussenElektra AG 50%	BWR 1316	**69**	a) 18.02.72 b) 14.09.83
12	Gundremmingen B (KRB B) Gundremmingen Bayern	a) Kernkraftwerke Gundremmingen Betriebsgesellschaft mbH b) KWU c) RWE Energie AG 75%	BWR 1344	**72**	a) 15.03.74 b) 09.03.84

Table 6-1 Nuclear Power Plants in Operation

	Nuclear power plants in operation Site	a) licensee b) manufacturer c) major shareholder	Type Gross capacity MWe	Design generation / construction line	a) date of application b) first criticality
13	Grohnde (KWG) Grohnde Niedersachsen	a) Gemeinschaftskernkraftwerk Grohnde GmbH b) KWU c) PreussenElektra AG 50% Gemeinschaftskraftwerk Weser 50%	PWR 1430	**3rd**	a) 03.12.73 b) 01.09.84
14	Gundremmingen C (KRB C) Gundremmingen Bayern	a) Kernkraftwerke Gundremmingen Betriebsgesellschaft mbH b) KWU c) RWE Energie AG 75%	BWR 1344	**72**	a) 15.03.74 b) 26.10.84
15	Philippsburg 2 (KKP 2) Philippsburg Baden-Württemberg	a) EnBW GmbH b) KWU c) Energie Baden-Württemberg AG 100 %	PWR 1424	**3rd**	a) 24.06.75 b) 13.12.84
16	Mülheim-Kärlich (KMK) Mülheim-Kärlich Rheinland-Pfalz (shut down by court order for unlimited time)	a) RWE Energie AG b) BBR c) RWE Energie AG 100%	PWR 1302	**4th**	a) 22.12.72 b) 01.03.86
17	Brokdorf (KBR) Brokdorf Schleswig-Holstein	a) Kernkraftwerk Brokdorf GmbH b) KWU c) PreussenElektra AG 80%	PWR 1440	**3rd**	a) 12.03.74 b) 08.10.86
18	Isar 2 (KKI 2) Essenbach Bayern	a) Bayernwerk Kernenergie GmbH b) KWU c) Bayernwerk AG 40%	PWR 1440	**4th Konvoi**	a) 13.02.79 b) 15.01.88
19	Emsland (KKE) Lingen Niedersachsen	a) Kernkraftwerke Lippe-Ems GmbH b) KWU c) VEW Energie AG 75%	PWR 1363	**4th Konvoi**	a) 28.11.80 b) 14.04.88
20	Neckarwestheim 2 (GKN 2) Neckarwestheim Baden-Württemberg	a) Gemeinschaftskernkraftwerk Neckar GmbH b) KWU c) Neckarwerke 70%	PWR 1365	**4th Konvoi**	a) 27.11.80 b) 29.12.88

6 Existing Nuclear Installations

Nuclear installations as defined by the Convention

Currently, 19 nuclear power plant units are in operation at 14 different sites producing a total of 22,194 MWe. Table 6-1 presents an overview of the nuclear power plants and Figure 6-1 shows the geographical location of the individual sites.

The Mülheim-Kärlich nuclear power plant with a rated power of 1,302 MWe has been temporarily shut down by court order since September 9, 1988. The controversy centres essentially on seismic safety.

Table 6-2 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 GmbH b) AEG/General Electric	BWR 16	a) 13.11.60 b) 25.11.85
2	Mehrzweckforschungs-reaktor (MZFR) Karlsruhe Baden-Württemberg	a) Kernkraftwerk Betriebsgesellschaft mbH b) Siemens/KWU	HWPWR 57	a) 29.09.65 b) 03.05.84
3	Rheinsberg (KKR) Rheinsberg Brandenburg	a) Energiewerke Nord GmbH b) VEB Kernkraftwerksbau Berlin	PWR (VVER) 70	a) 06.05.66 b) Nov. 90
4	Gundremmingen A (KRB A) Gundremmingen Bayern	a) Kernkraftwerk RWE-Bayernwerk GmbH b) AEG/General Electric	BWR 250	a) 14.08.66 b) Jan. 77
5	Atomversuchskraftwerk (AVR) Jülich Nordrhein-Westfalen	a) Arbeitsgemeinschaft Versuchsreaktor GmbH b) BBC/Krupp Reaktorbau GmbH (BBK)	HTR 15	a) 26.08.66 b) 31.12.88
6	Lingen (KWL) Lingen Niedersachsen	a) Kernkraftwerk Lingen GmbH b) AEG/KWU	BWR 268	a) 31.01.68 b) 05.01.77
7	Heißdampfreaktor (HDR) Großwelzheim Bayern	a) Forschungszentrum Karlsruhe b) AEG	super heated steam-cooled reactor 25	a) 14.10.69 b) 20.04.71
8	Würgassen (KWW) Würgassen Nordrhein-Westfalen	a) PreussenElektra b) AEG/KWU	BWR 670	a) 20.10.71 b) 26.08.94
9	Niederaichbach (KKN) Niederaichbach Bayern	a) Forschungszentrum Karlsruhe Kernkraftwerkbetriebs GmbH b) Siemens	pressure tube reactor 100	a) 17.12.72 b) 21.07.74
10	Greifswald 1 (KGR 1) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 15.12.73 b) 18.12.90
11	Greifswald 2 (KGR 2) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 03.12.74 b) 14.2.90
12	Kompakte natriumgekühlte Reaktoranlage (KNK II) Karlsruhe Baden-Württemberg	a) Kernkraftwerkbetriebs GmbH b) Interatom	FBR 21	a) 10.10.77 b) 23.08.91

Table 6-2 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 3 (KGR 3) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 06.10.78 b) 28.02.90
14	Greifswald 4 (KGR 4) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 22.07.79 b) 02.06.90
15	Thorium-Hochtemperatur-reaktor (THTR 300) Hamm-Uentrop Nordrhein-Westfalen	a) Hochtemperatur Kernkraftwerk GmbH b) BBC/HRB/NUKEM	HTR 308	a) 13.09.83 b) 20.09.88
16	Greifswald 5 (KGR 5) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) 24.04.89 b) 24.11.89
17	Greifswald 6 (KGR 6) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) project abandoned
18	Greifswald 7 (KGR 7) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) project abandoned
19	Greifswald 8 (KGR 8) Lubmin Mecklenburg-Vorpommern	a) Energiewerke Nord GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 440	a) b) project abandoned
20	SNR 300 Kalkar Nordrhein-Westfalen	a) Schnell-Brüter Kernkraftwerks-gesellschaft mbH b) INTERATOM /BELGONUCLEAIRE / NERATOOM	FBR 327	a) b) project abandoned 20.03.91
21	Stendal A Stendal Sachsen-Anhalt	a) Altmark Industrie GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 1000	a) b) project abandoned
22	Stendal B Stendal Sachsen-Anhalt	a) Altmark Industrie GmbH b) VEB Kombinat Kraftwerksanlagenbau	PWR (VVER) 1000	a) b) project abandoned

Corresponding to the time of their construction, the nuclear power plants with pressurised water reactors reflect four design generations whereas those with boiling water reactors belong to two different construction lines. The design generations and construction lines of the plants are noted in Table 6-1 and will be used throughout the report in presenting the results. Several of the basic plant characteristics important to safety and with respect to this classification are presented in Appendix 1. These also illustrate the continuous development in safety technology.

Since 1988 nuclear energy supplies about one third of the electric power and about 12 % of the entire primary power in Germany. In 1997 (1996) the electricity generated by German nuclear power plants amounted to 170,392 (161,702) GWh.

In 1997, as in previous years, the nuclear power plant units in operation demonstrated a high availability and high utilisation (Table 6-3).

Table 6-3: Average availability of German nuclear power plants

Year	Time availability %	Energy availability %	Energy utilisation %
1993	82.5	82.6	77.4
1994	81.2	81.1	76.8
1995	83.3	83.3	79.6
1996	88.1	87.0	82.8
1997	92.9	92.3	87.3

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

Careful planning of work procedures and manpower resulted in the fact that in 1997 the planned outages for major revisions and refuelling lasted on the average no more than 29 days (Figure 15-2).

In 1997 the Grohnde nuclear power plant (KWG) with its 12,529 GWh had the largest energy production world-wide. Second place within Germany was achieved by the Brokdorf nuclear power plant (KBR) with 11,837 GWh and third the Philippsburg 2 nuclear power plant unit (KKP-2) with 11,707 GWh.

If the planned outages for refuelling and major revisions are taken into account, the nuclear power plants in 1997 spent only 0.3 % of their time in unplanned shutdowns. Figure 6-2 shows the energy availability of the individual nuclear power plants over the last five years in their groups of design generations (PWR) and construction lines (BWR) referred to above. High availabilities were also achieved by the older nuclear power plants which is partly due to the high quality standards applied in the different areas of operation and maintenance.

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

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

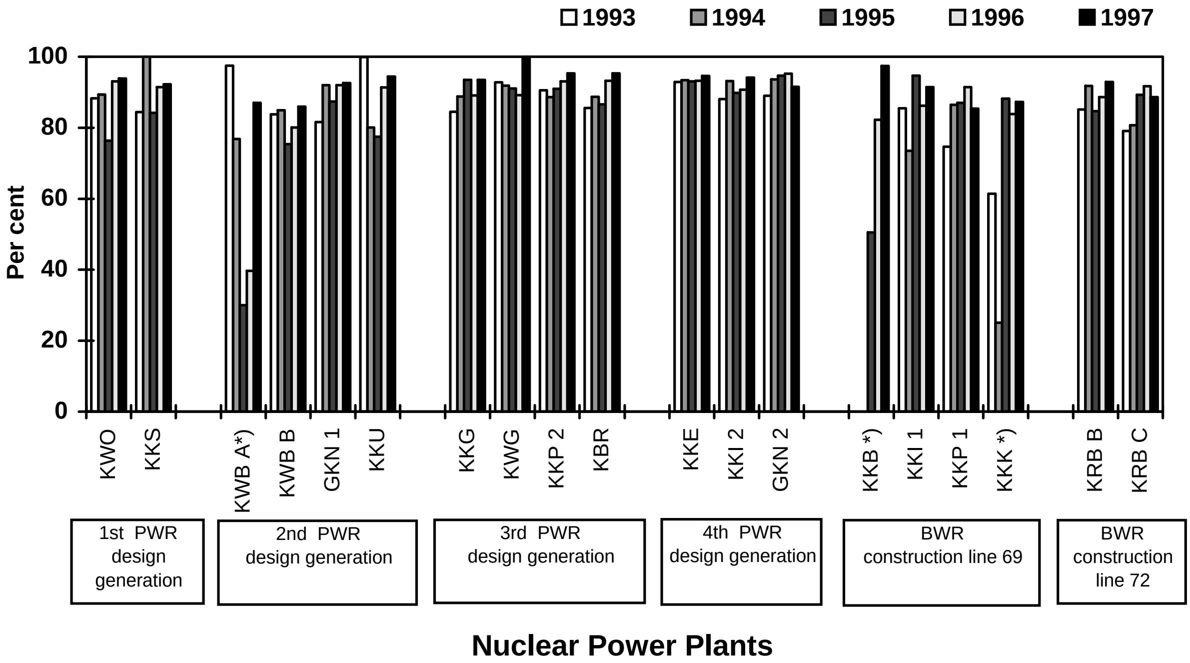


Figure 6-2 Energy Availability of German Nuclear Power Plants
*) The nuclear power plants KWB A, KKB and KKK were temporarily shutdown due to legal administrative proceedings

Other nuclear installations

To complete the picture of the utilisation of nuclear energy in Germany a short survey on the other nuclear installations outside the scope of the Convention will be presented.

Altogether 16 nuclear reactor units with an overall power production of 4,000 MWe have been shutdown for decommissioning, i.e. their decommissioning is either planned, applied for, started or completed (Table 6-2). A number of these are small reactors from the initial days of the utilisation of nuclear energy. However, also included is the THTR 300, a prototype plant for the high-temperature pebble bed reactor. Furthermore, several nuclear reactor units are included for which no utility could be found that was willing to take on the costs for probable backfitting measures and the continuation of operation (units 1 to 5 of the Greifswald nuclear power plant). The latter also applies to the Würgassen nuclear power plant. In case of the nuclear power plant at Niederaichbach - a small pressure tube reactor in operation for only a short time - the complete dismantling of the plant and the recultivation of the site were completed as early as August 1995.

Work on the construction of the nuclear reactor units 6 to 8 at Greifswald and units A and B at Stendal has been discontinued. In both cases it was intended to install Russian type VVER reactors. Since work had not proceeded far enough to have allowed fuel element loading, the dismantling of these units can be performed in a conventional manner. Likewise in 1991, work on the SNR 300 - a prototype of a fast breeder reactor - was discontinued prior to its initial loading with nuclear fuel.

The other nuclear facilities are research reactors and facilities of the nuclear fuel cycle and for the treatment and final disposal of radioactive waste. A uranium enrichment plant at Gronau and the fuel fabrication plant at Lingen are in operation. The pilot nuclear fuel 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 and, thus, prepare them for final disposal. 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. A pilot plant is under construction intended for conditioning and preparing spent fuel elements for their direct final disposal. With respect to the final disposal of radioactive waste in geological deep formations, the licensing procedures for the Schacht Konrad and the Gorleben repository projects have been initiated. The Morsleben repository (ERAM) for low and intermediate radioactive waste is in operation.

Results from the safety review of nuclear installations

Within the framework of the regulatory system concerned with the utilisation of nuclear energy and, especially, of regulatory supervision (→ Chapter 7), safety assessments are performed both on a continuous as well as on a discontinuous or periodical basis. The corresponding results are put into practice (→ Chapter 14). This ensures that a review of the safety of the nuclear installations within the meaning of Article 6 of the Convention is carried out.

All but one (KMK) of the 20 nuclear power reactor units listed in Table 6-1 have been issued an unlimited operating license. The required protection from the damages of radiation effects is achieved jointly by the plant design on which the licences are based and by the surveillance of their operation within the framework of regulatory supervision. In order to even further reduce the remaining very low risk, additional equipment and procedures were introduced in all German nuclear power plants. Thus, any damages to the general public have been made practically impossible.

The licensees of nuclear power plants - within their independent full responsibility for plant safety - adjust the safety level of the nuclear power plants in correspondence with the state of the art in science and technology over the entire operating life of the plant. Whenever new safety relevant findings are available, the necessity and adequacy of possible improvements are checked. Thus, a progressive improvement of plant safety is achieved. Over the past years this has lead to numerous improvements (→ Chapter 14 (ii)), especially in the area of beyond-design-basis accidents (→ Chapter 16 (1)). As a result, a high level of safety has been achieved even in the older nuclear power plants.

On the basis of present safety assessments the German Federal Government believes that no urgent backfitting measures are currently required in any of the nuclear power plants that would require a restriction on the continuation of their operation. In particular, not a single nuclear power plant is currently charged with regulatory obligations regarding immediately required safety improvements. These would have been imposed, had serious safety deficiencies been detected.

In summary, the German Federal Government ascertains that nothing stands in the way of continued safe operation of German nuclear power plants. This report shows that all obligations under the Convention of Nuclear Safety are being fulfilled.

7 Legislative and Regulatory Framework

7 (1) Legislative and Regulatory Framework

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

The Atomic Energy Act [1A-3] was promulgated December 23, 1959, right after the Federal Republic of Germany had officially renounced any use of atomic weapons. Originally its scope of application was restricted to the Federal Republic of Germany within the boundaries prior to 1990 and to the State of Berlin.

In Germany the legislation and its execution must also take into account any binding requirement from regulations of the European Union. With respect to radiation protection there are, e.g., the EURATOM Basic Safety Standards [1F-15] for the protection of the health of workers and the general public against the dangers arising from ionising radiation. These were issued on the basis of Article 30 ff. of the EURATOM Treaty [1F-1]. In accordance with Article 77 ff. of the EURATOM Treaty any utilisation of ores, source material and special fissile material is subject to surveillance by the European Atomic Energy Community.

7 (2i) Nuclear Safety Regulations

Acts and ordinances, in particular, the Atomic Energy Act

Following its introductory part which specifies the legally binding scope of its application, the Atomic Energy Act comprises, essentially, all general administrative regulations for the regulatory licensing and supervision regarding the utilisation of nuclear energy including regulations authorising the promulgation of legal ordinances. An additional part deals with the corresponding liability regulations. In the past, the Atomic Energy Act has been revised a number of times to adjust to the technical and regulatory development.

With respect to nuclear safety, the Atomic Energy Act is the central core of national regulations in Germany. Its primary purpose is to protect life, health and property against the hazards of nuclear energy and the detrimental effects of ionising radiation and, furthermore, to provide for the compensation for any damage and injuries incurred. It also has the purpose of preventing the internal or external security of the Federal Republic of Germany from being endangered by the utilisation of nuclear energy. Another purpose of the Atomic Energy Act is to ensure that the Federal Republic of Germany meets its international obligations in the field of nuclear energy and radiation protection.

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

A special aspect of the prerequisites for obtaining a licence in Germany is that the measures regarding precaution against damage must correspond to the state of the art in science AND technology. This is a tightening of requirements considering the general practice of German technical safety regulations which require conformity merely with the state of the art in technology. Therefore, the licensing of a nuclear installation requires a degree of precaution against damages that is considered necessary also on the basis of latest assured scientific findings.

The Atomic Energy Act is supplemented by the Precautionary Radiation Protection Act of 1989 [1A-5] which came about in the wake of the reactor accident at Chernobyl. This Act specifies the tasks of environmental monitoring also in the case of events with significant radiological effects (→ Chapter 15 and 16).

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

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

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

General administrative provisions

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

– the calculation of radiation exposure during normal operation of nuclear power plants [2-1],
– the radiation passport [2-2],
– the environmental impacts assessment [2-3] and
– the environmental monitoring [2-4].

Regulatory guidelines

The Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU), after consulting the Länder and, generally, with their consent, issues regulatory guidelines regarding technical and administrative questions arising from the licensing and supervisory procedure (→ Chapter 8 (1)). These guidelines specify the administrative practice which, generally, is followed verbatim by the competent Länder authorities in the individual case. Currently, about 50 such regulatory guidelines exist in the field of nuclear technology (see Appendix 2 Chapter 3 ”Announcements ...” [3-...]). These regulatory guidelines pertain to:

– general safety requirements (”Safety Criteria”),
– details on the design basis accidents to be considered in the design,
– dispersion calculations,
– accident management measures to be planned by the licensee with regard to the postulated severe beyond-design-basis accidents,
– measures regarding disaster control in the environment of nuclear installations,
– measures against disruptive actions or other interference by third parties,
– radiation protection during maintenance work,
– general documentation,

– documents to be supplied with the application for a license,
– trustworthiness and qualification of the personnel in nuclear installations.

RSK-Guidelines and recommendations of the RSK and the SSK

An important role in the development of the regulatory system and the preparation of technical opinions for licensing and supervision is played by the recommendations of the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK). Both of these independent expert commissions advise the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety in questions of nuclear safety and radiation protection (→ Chapter 8 (1)).

The RSK Guidelines [4-1] are of particular importance. The RSK has used these guidelines to collate all safety requirements which in its view must be met by the design, construction and operation of a nuclear power plant. They also take into consideration and make reference to other existing nuclear safety standards and regulatory guidelines. The RSK uses these guidelines as a basis of its consultations and recommendations.

KTA Safety Standards

Detailed and concrete technical requirements are contained in the safety standards of the Nuclear Safety Standards Commission (KTA). The KTA plays the role of intermediary between the regulatory and the scientific and technical positions (→ Chapter 8 (1)). In accordance with its statutes, the KTA specifies requirements wherever ”experience leads to a uniform opinion of the experts within the groups of manufacturers, construction companies, and licensees of nuclear installations, and of the expert organisations and the authorities.” Therefore, the nuclear safety standards thus created will surely meet the requirement of conforming to the state of the art in technology but may actually be behind with respect to the state of the art in science and technology (cf. above). The issued safety standards are regularly reviewed in intervals of no more than five years and eventually will be 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 even in legal proceedings. Currently (as of 06/97) the KTA has issued a total of 86 safety standards and 5 standard drafts; an additional 6 standard drafts are in preparation and 16 safety standards are in the process of being revised.

The KTA safety standards pertain to
– administrative provisions,
– industrial safety (specific requirements with respect to nuclear technology),
– civil engineering,
– nuclear and thermal-hydraulic design,
– issues regarding materials,
– instrumentation and control,
– monitoring of radioactivity and
– other requirements.

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

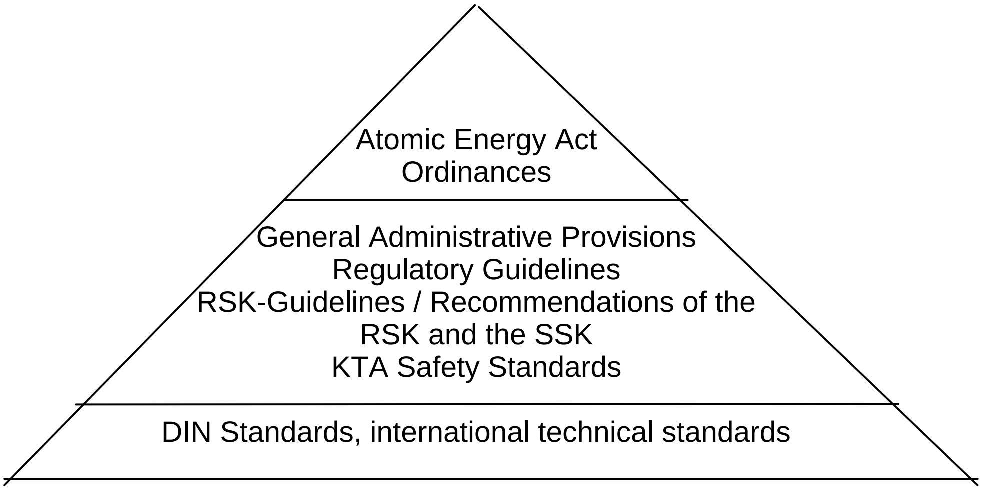
Historically, the KTA safety standards have been developed on the basis of applicable German technical standards and regulations and on the American nuclear safety standards. The ASME-Code (Section III) was used as a model for specifying the requirements regarding the design and construction of components.

Conventional technical standards

Furthermore, conventional technical standards, and here especially the national standards of the German Institute for Standardisation (DIN), are applied just as they are in the design and operation of all technical installation; an steadily growing importance is also attributed to the international standards of ISO and IEC.

Overall picture of nuclear safety regulations

The German nuclear safety regulations may be seen as hierarchically structured in the form of a pyramid with the Atomic Energy Act at the top and the other regulations with their ever increasing degree of detail at lower levels down to the base.



In this report, reference will be made to the contents of the individual regulations as the corresponding articles of the Convention are dealt with. The "Reference List of Nuclear Safety Regulations" of Appendix 2 lists the current regulations applicable to nuclear installations in the mentioned hierarchical order. All of the listed regulations are publicly available. They are published in official publications of the Federal Government.

The general structure and content of the safety provisions and regulations described herein were essentially developed in the seventies. Since then they have been applied in all legal proceedings of nuclear licensing and supervision and have been further developed in accordance with the state of the art in science and technology.

In accordance with the Atomic Energy Act as amended in 1994, any nuclear installation to be licensed in the future must, already in its design stage, fulfil the requirements of Section 7 (2a), namely, that "even incidents, the occurrence of which is practically excluded as a result of the precautions to be taken against damage, would not necessitate drastic actions for protection against the detrimental effects of ionising radiation outside the enclosed site of the installation." The incidents indicated here will be specified in a future regulatory guideline.

7 (2ii) System of Licensing and Supervision

The requirements regarding the licensing of nuclear installations are regulated in the Atomic Energy Act [1A-3]. In accordance with Section 7 of this Act, a licence is required for the construction, operation or any other holding of a stationary installation for the production, treatment, processing or fission of nuclear fuel, or for essentially modifying such installation or its operation. The applicant may only be granted a licence if he meets the individual requirements that are spelled out in Section 7 Atomic Energy Act as licence prerequisites:

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

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

When issuing a licence for a nuclear power plant, obligations may be imposed if it seems necessary to ensure safety.

Any act of operating, otherwise holding, essentially modifying or decommissioning a nuclear installation without the required corresponding licence permit is punishable by law [1B-1].

The licensing of nuclear installations lies within the responsibility of the individual Länder. The Länder have ministries that are responsible for licensing of construction, operation, essential modification and decommissioning of nuclear power plants (Table 8-1). In order to achieve a nationally uniform licensing practice, the Federal Government exercises its right to supervise the Länder. This also includes the right to issue binding directives.

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 licenses). It deals, furthermore, with the assessment of environmental impacts [1F-13] and with the consideration of other licensing requirements (e.g. regarding the possible release or discharge of non-radioactive pollutants into air or water.
(→ Chapter 17 (ii))

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

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

The high safety standards already applied make it highly improbable that serious damage would be caused by nuclear power plants. Nevertheless and with due respect to the potential magnitude of such damage, it has always been an essential licensing prerequisite in Germany that sufficient financial security is provided for covering possible claims for damage compensation. Current liability regulations account for the Paris Convention on nuclear liability [1E-11] amended by the Brussels Supplementary Convention [1E-12]. Both conventions have, in the meantime, been incorporated into the Atomic Energy Act. The corresponding details are regulated by the Nuclear Financial Security Ordinance [1A-11]. In Germany this means that the licensees are required to take out liability insurance policies for a maximum financial sum that is specified in the individual nuclear licensing procedure. The Federal Government and the *Land* issuing the licence jointly carry an additional indemnity which may be claimed by the damaged party. Currently, the maximum required financial security from liability insurances is limited to DM 500 million and that of the additional (federal) indemnity to twice this amount.

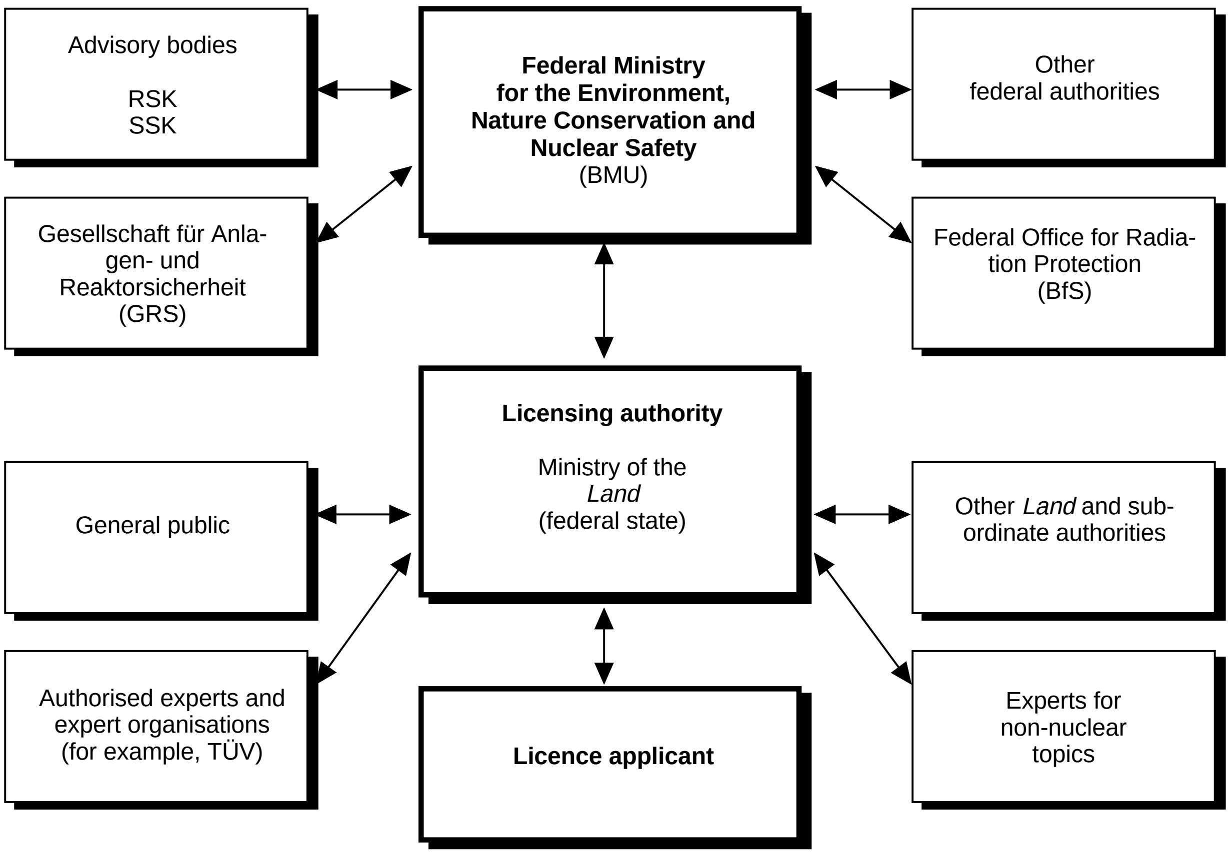


Figure 7-1 Participants in the Nuclear Licensing Procedure

Details of the nuclear licensing procedure

Licence application

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

– the protection of the plant against disruptive actions or other interference by third parties,
– the personal data on the licence applicant and those holding responsible positions, including their qualification,
– the necessary knowledge of the personnel otherwise engaged during plant operation,
– the safety specification,
– the financial security,
– the type of residual radioactive materials and their disposal,
– the intended environmental protection measures.

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

Evaluation of the application

On the basis of the submitted documents the licensing authority evaluates whether or not the licence prerequisites have been met. This authority initiates the participation of every authority in the individual Land, the communities in the vicinity of the plant site and, especially, of any technical authority and institution whose area of competence is directly involved. These are, in particular, those authorities responsible under the building code, the water code, for regional planning and for off-site disaster control. Because of the large effort involved in this evaluation it is common practice to engage expert organisations in the evaluation and examination of the application documents. The role of the expert organisations is strictly advisory in nature. In their expert analysis reports they explain whether or not the requirements regarding nuclear safety and radiation protection have been met.

Within the framework of federal executive administration, the licensing authority of the individual Land also involves the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU). In performing its function of federal supervision, the BMU is advised by, and receives technical support from, the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK) as well as the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS); the BMU states its position to the competent licensing authority. This federal position is binding for the decision of the licensing authority.

Participation of the general public

The licensing authority also involves the general public in the licensing procedure, for direct protection of the citizens who might be affected by the planned nuclear installation. In accordance with the Nuclear Licensing Procedure Ordinance [1A-10], the following requirements must be met in this respect:

– the public announcement of the project and public disclosure of the licence application with its documents at a suitable location near the site of the project for a period of two months and, furthermore,
– the holding of a public hearing where the submitted 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.

Environmental impact assessment

The Act on the Assessment of Environmental Impacts [1F-13] in conjunction with the Nuclear Licensing Procedure Ordinance specify the requirement for an environmental impact assessment and its procedure within the nuclear licensing procedure for the construction, operation and decommissioning of a nuclear power plant or for an essential modification of the plant or its operation. The competent authority performs a final evaluation of the environmental impact 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.

The Act on the Assessment of Environmental Impacts [1F-13] and the corresponding amendment of the Nuclear Licensing Procedure Ordinance came into effect in 1990 and 1995, respectively. For licences issued at an earlier date, the assessment of environmental impacts of nuclear installation was carried out on the basis of the nuclear and radiation protection regulations and of the other environmental laws.

Licensing decision

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

Licences for the operation of nuclear power plants are issued with no time limitation. Actions against the decision of the licensing authority can be brought forth in the administrative courts.

Upon a corresponding application, nuclear regulatory licensing may be carried out in a number of partial steps. It is sensible, with regard to the size and construction times of nuclear power plant projects, that the technical details of its construction and operation are evaluated and licensed in separate steps. The advantage is that the individual licensing steps can always be based on the most recent state of the art in science and technology and that partial construction can already be performed before the overall evaluation has been finished. Such partitioning into a number of partial licences has been performed for all German nuclear power plants. Prerequisite for a licensing in partial steps is, however, that the overall safety concept of the installation has already been positively evaluated before the first partial licence is issued. This positive overall evaluation is binding for all further licensing steps.

Typical licensing steps have been:

– the siting and construction of essential civil structures,
– the construction of the systems and components important to safety,

- the handling and storage of fuel elements as well as the initial fuel loading of the reactor core,
- the nuclear commissioning of the plant,
- the continuous operation of the installation.

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

Over their entire lifetime, from the start of construction to the end of decommissioning with the corresponding licenses, nuclear installations are subject to continuous regulatory supervision in accordance with the Atomic Energy Act and accessory nuclear ordinances. However, the Länder perform this supervisory procedure on behalf of the Federal Government (→ Chapter 7 (2ii), i.e., Federal Government supervises the Länder and has the right to issue binding directives. Just as in the licensing procedure, the Länder are assisted by independent authorised experts.

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

The supervisory authority pays particular attention to

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

The supervisory authority - if necessary with help of authorised experts or other authorities - monitors in particular:

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

In accordance with the Atomic Energy Act, the authority officials as well as the authorised experts working on behalf of the supervisory authority have access to the nuclear installation at all times and are authorised to perform necessary examinations and to demand pertinent information.

The licensees of nuclear power plants have to supply written operating reports to the supervisory authorities in regular intervals. Any events that are relevant to safety must be reported to the authorities [1A-17]. The regulations and procedures regarding reportable events and their evaluation are described in Chapter 19 (vi)-(vii). With respect to federal supervision, the licensees are required to submit annual reports regarding operation and radiation protection.

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-1], in the Atomic Energy Act [1A-3] and the nuclear regulatory ordinances in case of any violations. Any violation that must be considered as a criminal offence is dealt with in the Penal Code. The Atomic Energy Act and the accessory ordinances deal with administrative offences and provide for the imposition of fines on the persons responsible for the actions.

Under certain conditions, obligations for ensuring safety may be decreed even after a final licence permit has been granted. In case a considerable hazard is suspected from the nuclear installation endangering the persons engaged in the plant or the general public, and cannot be removed within a reasonable time by appropriate means, then the licensing authority has to revoke the issued licence permit. 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.

In the case of non-compliance with respect to legal provisions or to requirements of the licence permit and also if it must be suspected that the life, health or property of third parties is endangered, the competent supervisory authority of the Land is authorised by Section 19 Atomic Energy Act to issue orders stating

- that protective measures must be applied and, if so, which ones,
- that radioactive materials must be stored at a place prescribed by the authority and
- that the handling of radioactive materials, the construction and operation of nuclear installations must be interrupted or temporarily or - in case of a revocation of the licence - permanently be suspended.

Administrative offences

The Atomic Energy Act and the accessory ordinances contain provisions for the case of administrative offences incurred by the violation of, or non-compliance with, valid regulations. An administrative offence is committed by anyone who

- erects a nuclear facility without a valid licence permit,
- acts in violation of a regulatory order or provision,
- handles radioactive materials without a valid licence permit,
- as the ultimately responsible person fails to see to it that the protective and supervisory regulations of the Radiation Protection Ordinance are fulfilled.

The Atomic Energy Act and the accessory ordinances require that the persons are named who are ultimately responsible for the handling of radioactive materials, for the operation of nuclear installations or for their supervision. A person committing an administrative offence is personally liable for a fine up to DM 100 000. A legally valid penalty notice against this person may put in question the personal trustworthiness that was a prerequisite for the licence and may, therefore, require the replacement of this person in his position of responsibility (→ Chapter 9).

Criminal offences

Imprisonment or fines are imposed on anyone who, for example,

– operates, otherwise holds, changes or decommissions a nuclear installation without the required license,
– knowingly constructs a defective nuclear installation,
– handles nuclear fuel without the required license,
– releases ionising radiation or causes nuclear fission processes that can damage life and limb of other persons,
– procures or manufactures nuclear fuel, radioactive materials or other equipment for himself with the intent of performing a criminal offence.

Experience

The licensees of nuclear power plants fulfil the safety provisions and regulations, and the supervisory authorities perform the surveillance function in this respect. Because of the intense regulatory supervision carried out in Germany in the course of design, erection, commissioning, operation and decommissioning of nuclear installations (→ Chapter 7 (2iii)), any inadmissible condition is usually detected at an early stage before the possible legal actions such as imposed obligations, orders, administrative offence procedures and criminal proceedings have to be taken.

8 Regulatory Body

8 (1) Authorities, Committees and Organisations

Nuclear licensing and supervisory authorities

The nuclear licensing and supervisory authorities are administrative authorities, generally state ministries (Table 8-1), of those Länder in which the site of the nuclear installation is located (→ Chapter 7 (2ii) and (2iii)). To preserve the legal uniformity for the entire region of the Federal Republic of Germany, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) supervises the licensing and supervisory activities of the Länder authorities (so-called ”federal executive administration”)

Table 8-1 The *Länder* Licensing and Supervisory Authorities for Nuclear Installations According to Section 7 Atomic Energy Act

Land (federal state)	**Licensing authority for nuclear installations according to Section 7 Atomic Energy Act**	**Supervisory authority according to Section 19 in conjunction with Section 7 Atomic Energy Act**
Baden-Württemberg	Wirtschaftsministerium *after consultation with* Ministerium für Umwelt und Verkehr und Innenministerium	Ministerium für Umwelt und Verkehr
Bayern	Staatsministerium für Landesentwicklung und Umweltfragen, *in matters of energy in agreement with* Staatsministerium für Wirtschaft Verkehr und Technologie	Staatsministerium für Landesentwicklung und Umweltfragen
Berlin	Senatsverwaltung für Stadtentwicklung, Umweltschutz und Technologie	
Brandenburg	Ministerium für Umwelt, Naturschutz und Raumordnung	
Bremen	Senator für Frauen, Gesundheit, Jugend, Soziales und Umweltschutz *after consultation with* Senator für Arbeit	
Hamburg	Umweltbehörde	
Hessen	Ministerium für Umwelt, Energie, Jugend, Familie und Gesundheit	
Mecklenburg-Vorpommern	Innenministerium *in agreement with* Wirtschaftsministerium	Innenministerium
Niedersachsen	Umweltministerium	
Nordrhein-Westfalen	Ministerium für Wirtschaft, Mittelstand, Technologie und Verkehr	
Rheinland-Pfalz	Ministerium für Umwelt und Forsten	
Saarland	Ministerium für Umwelt, Energie und Verkehr	
Sachsen	Staatsministerium für Umwelt und Landesentwicklung	
Sachsen-Anhalt	Ministerium für Raumordnung, Landwirtschaft und Umwelt	
Schleswig-Holstein	Ministerium für Finanzen und Energie	
Thüringen	Ministerium für Landwirtschaft, Naturschutz und Umwelt	

Authorised experts

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

– training,
– professional knowledge and skills,
– trustworthiness and
– independence.

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

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

Germany has a long tradition with regard to the institution of the authorised expert. The beginnings reach back to the last century when private pressure vessel surveillance associations introduced independent surveillance and, thereby, increased the quality, safety and reliability of such installations.

The BMU will consult with further national and international experts as the need arises.

Federal Office for Radiation Protection

In performing its federal supervision, the BMU is supported by the Federal Office for Radiation Protection (BfS) in all matters concerning nuclear safety and radiation protection.

Reactor Safety Commission, Commission on Radiological Protection

The BMU receives further advisory support from the Reactor Safety Commission (RSK) and Commission on Radiological Protection (SSK). The Reactor Safety Commission was founded in 1958, the Commission on Radiological Protection in 1974. Both commissions have one mutual statute and a membership, each, of usually 17 or 18 experts from different fields of expertise. The members are appointed by the BMU and are independent and not bound by any directives. Their main activity lies in advising the BMU on questions of basic importance, but they also initiate developments directed at furthering safety technology. The results of the discussions of the individual commissions are formulated as more general recommendations and as statements on individual cases. All recommendations are published.

Gesellschaft für Anlagen- und Reaktorsicherheit

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

Nuclear Safety Standards Committee

The Nuclear Safety Standards Committee (KTA) takes up the position of mediator between the regulatory and the scientific and technical opinions. It was established in 1972 and is made up of five interest groups of representatives of the manufacturers, the utilities, the federal and *Länder* authorities, the expert organisations and parties of general concerns - e.g. unions, industrial safety, liability insurers. In accordance with its statute the KTA formulates detailed safety standards (→ Chapter 7 (2i)) in those areas where "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 working groups and sub-committees and are then passed onto 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.

Federal States Committee for Nuclear Energy

The Federal States (*Länder*) Committee for Nuclear Energy was founded as joint committee of the *Länder* and the Federal Government to help in co-ordinating the respective activities of the licensing and supervisory authorities of the *Länder* and those of the BMU in supervising the *Länder* activities. This committee is made up of representatives of the BMU and of the competent *Länder* authorities. The BMU chairs the committee. The committee reaches its decisions, usually, by mutual consent.

Personnel and financing

No limits are set for personnel costs involved with licensing and supervisory activities. These costs strongly depend on whether, and how many, nuclear power plants are subject to licensing and supervision in each individual *Land*. Regulatory supervision, including the activities of authorised experts, requires an annual personnel effort of 30 to 40 man years for each nuclear power plant unit. The funds available to the authorities are allotted by the federal and *Länder* parliaments in their respective annual budgets.

The licensee of a nuclear power plant is liable for the costs of the licence permits issued and for the associated supervisory activities. These costs are payable to the public treasury. The overall costs of the licences for construction and operation are set at 2 per mil of the construction costs. A modification requiring a licence permit will cost between DM 1,000 and DM 1,000,000. The fees for supervisory activities are charged on the basis of the individual activity and cost to anywhere between DM 50 and DM 500,000. The licence applicant or licensee also carries the costs charged as reimbursements for the authorised experts.

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

In Germany, those organisations involved with the utilisation and promotion of nuclear energy are effectively separated from those responsible for licensing and supervising the nuclear power plants.

The licensees of nuclear installations are economic enterprises under private law and are either power utilities themselves or are made up of shareholders from German power utilities.

The power utilities are also economic enterprises under private law, usually stock corporations (→ Chapter 11 (1)).

The licensing and supervisory authorities - both, at the federal and the Länder level - are administrative authorities that are bound in their actions by laws and regulations. Their topmost commitment is derived from the Atomic Energy Act, namely, to ascertain that 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 nuclear installation.

To support the administrative authorities in technical issues, experts - organized under private law - may be involved who are, by themselves, obliged to deliver impartial and technically qualified statements (→ Chapter 7 (2ii and 2iii) and Chapter 8(1)).

An additional adjusting tool is the possibility for anybody concerned to go to court if he believes a regulatory decision (→ Chapter 7 (2ii)) to be faulty, in particular with regard to whether or not the precautionary measures required in accordance with the Atomic Energy Act actually comply with the state of the art in science and technology.

In the area of research on nuclear reactor safety, the promotion of a peaceful utilisation of nuclear energy in accordance with Section 1 Atomic Energy Act is achieved by applying research funds provided by the Federal Ministry of Education, Science, Research and Technology (BMBF). The general and lasting objective of the reactor safety research funded by BMBF is that it should continuously contribute to furthering the development of safety technology and that it continuously delivers improved know-how and procedures for a realistic safety assessment of nuclear installations (→ Chapter 11 (1)).

9 Responsibility of the Licence Holder

The licensee has the primary responsibility for the safety of a nuclear power plant. He may be issued a licence only if he fulfils all prerequisites for a licence as specified in Chapter 7 (2ii). Regarding his overall responsibility, one prerequisite of particular importance is the trustworthiness of the licensee and his personnel. They must also give certified proof that they possess the required qualifications. These facts provide the basis for responsible performance under the license.

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

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

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

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

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

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

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

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

The actual structure of the plant organisation is at the sole discretion of the licensee, provided it accounts for the requirements of the above mentioned responsible persons as well as for the general requirements regarding quality assurance (→ Chapter 7 (2i), KTA Safety Standards). The plan of the organisational structure showing the task distribution and the names of the responsible persons must be submitted to the licensing and supervisory authority.

The intense interaction in all questions of nuclear safety between the responsible plant personnel, the members of the licensing and supervisory authorities and their authorised experts effectively supports the licensee in converting his basic responsibility into actual technical solutions. However, this support in no way whatsoever relieves the licensee of his primary and all encompassing responsibility for the safety of the nuclear power plant transferred to him by the license.

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

10 Priority to Safety

Nuclear safety has always been considered as the primary objective of the Atomic Energy Act that must at all times be considered in its application. As early as 1972, the supreme administrative court of Germany decreed that nuclear safety has priority over any of the other objectives of the Atomic Energy Act. This decree has always been upheld in later court decisions. The principle of ”safety first” has been the guiding theme in any administrative action in the field of nuclear energy.

This principle has been concretised with respect to the individual licence by the following licensing prerequisite (Section 7 Atomic Energy Act):
”A licence may only be granted if 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.”

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

An important basis for implementing the safety-first principle is the independent full responsibility of the licensee as the party ultimately responsible for safety (→ Chapter 9). The willingness of the licensee to employ an all-encompassing safety management is of crucial importance. This must comprise all measures required for ensuring the achievement of a sufficient safety level.

In conjunction with the requirements regarding the qualification and trustworthiness of personnel (→ Chapter 11 (2)), this creates for all whose activities are directly connected with nuclear installations a frame for a safety culture that affects the personal attitude of each individual. The high safety culture developed in Germany has been reflected in the past, for example, by the willingness of the licensees to provide measures for the prevention and mitigation of severe beyond-design-basis accidents (→ Chapter 16 (1)) and to perform periodic safety reviews in reasonable intervals (→ Chapter 14 (ii)), even without a direct regulatory requirement.

However, in recent years a development has been observed that gives rise to concern that this mutual consent is in danger of disintegrating. Therefore, in 1997 the RSK released a ”Memorandum on Safety Culture” which points out the changed frame of reference and identifies the emerging deficits. In particular, it refers to the fact that

– despite discrepancies in current political thinking, be it for further utilisation or for a withdrawal from the nuclear energy programme, it is important to continue to strive for a consensus in supervisory philosophy, supervisory practice and the operation of nuclear installations,
– aside from the job-oriented training by the utilities it is important to train and employ qualified personnel in sufficient numbers by the manufacturers, the authorities and the expert and research organisations, and that
– the safety research required for the further development of safety culture must continue to be supported to the necessary extent by funds from the government and the utilities.

11 Financial Means and Human Resources

11 (1) Financial Means

Expenditures by the licensees

The licensees of nuclear power plants adjust the safety levels of their installations to account for the development of the state of the art in science and technology during the entire operating life of the plant including decommissioning and dismantling. To this end, they invest in updating the safety provisions of their installations to account for any new safety findings. Over the past years a great number of measures have been implemented which have continually adapted the technical equipment of nuclear power plants to the increasing safety requirements. A total of DM 3 billion is spent annually on operating installations alone for maintenance, inservice inspections, replacement of components, and for backfitting measures. In addition, the licensees spend DM 250 million each year for the inspection activities of the nuclear licensing and supervisory authorities and their authorised experts.

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

The licensees build up financial reserves to be prepared for the follow-up costs connected with the operation of a nuclear power plant such as the decommissioning and dismantling of the installations, and the treatment and disposal of radioactive materials including spent fuel elements. One quarter of the financial reserves is earmarked for decommissioning and dismantling and three quarters for waste management.

According to a cost estimate prepared by the Association of German Power Utilities (VdEW) in 1995, the costs for decommissioning and for dismantling buildings and the equipment in restricted access areas of a nuclear power plant will average out to an amount of about DM 650 million for a pressurised water reactor, and to about DM 770 million for a boiling water reactor. Additional costs are anticipated for the post-operating phase, for the dismantling of conventional buildings, and to cover possible risks. To current knowledge, the financial reserves will be sufficient to cover the costs incurred after the end of commercial operation.

The Association of Major Power Utilities (VGB), in which all German and several foreign licensees of nuclear power plants are members, spends between approximately DM 4 and 5 million annually for the evaluation and feed-back of operating experience (→ Chapter 19 (vii)). In addition, VGB has financed about 250 projects over the past ten years, three-quarters of which - for a total amount of about DM 140 million - were directly aimed at improving safety.

Governmental expenditures

The cost of personnel needed by the Länder to perform licensing and supervisory activities are included in the annual budgets of the Länder; the project-related costs of licensing and supervision, however, are charged to the applicants and licensees (→ Chapter 8 (1)).

The BMU finances the federal supervisory activities in reactor safety to the tune of about DM 47 million each year. The activities include evaluation of operating experience, safety investigations, further development of requirements for nuclear installations, and specific issues regarding the licensing and supervision of nuclear power plants. Also included is the development of safety requirements to be applied to new reactor concepts (→ Chapter 7 (2i)). About 20 % of overall expenditures are used for safety investigations and evaluations of and assistance to nuclear power plants in the countries of central and eastern Europe.

With respect to governmental responsibility and provision it is imperative that the Federal Republic of Germany participates in the world-wide efforts to further develop the safety of nuclear power plants by performing independent safety research on a national level. Therefore, the Federal Ministry for Education, Science, Research and Technology (BMBF) sponsors research and development projects on generic questions, the answers to which are of governmental concern. In the past year BMBF allocated DM 52 million of its annual budget to reactor safety research.

The research activities regarding light water reactors included experimental and analytical investigations on plant behaviour during accidents, on the safety of pressurised components, on core meltdown, on human behaviour, on non-destructive examination methods for early detection of faults in materials difficult to inspect, and on methods development for probabilistic safety analyses. With their orientation into the future and to innovative technology, these R&D projects also further the state of the art in science and technology.

The Federal Republic of Germany in its federal budget also provides the financial means for the decommissioning of those nuclear installations for which the Federal Government has taken responsibility (pilot-plants, experimental and research reactors). This requires an annual expenditure of approximately DM 230 million. An additional amount of about DM 65 million is required for the associated legally required investments in the final disposal.

11 (2) Human Resources

German nuclear power plants currently in operation are staffed with personnel that has a long experience in the operation of nuclear power plants. Single-unit plants are staffed with about 350 people, double-unit plants with about 600. Additional personnel - partly at the headquarters of the licensee - is engaged in project planning, project management, licensing, and technical support. Supplementing its own personnel, the licensees of nuclear power plants extensively use personnel from contracted external firms, particularly for maintenance work during the yearly inspection outages, during refuelling and system modifications. This also includes personnel of the manufacturer of the nuclear power plants and other external specialists for specific tasks, e.g. contractually required maintenance and inspection of specific components.

Regulations regarding the qualification of personnel

Section 7 of the Atomic Energy Act [1A-3] specifies the prerequisite that a licence for the construction and operation of a nuclear power plant shall only be granted if the persons responsible for construction and operation have the necessary qualification. Likewise, the personnel otherwise engaged during operation must have the necessary knowledge with respect to safe operation, possible risks, and relevant protection measures. Furthermore, there shall be no doubts as to the trustworthiness of the personnel. Accordingly, proof of the qualification of the responsible personnel as well as of the necessary knowledge of the personnel otherwise engaged during operation must already be included in the licence application for construction, operation or essential modifications [1A-10]. The trustworthiness of the personnel is evaluated by the licensing authority according to the relevant regulatory guideline [3-57-1]. The qualification certificates and regular training measures to maintain qualification are checked within the framework of regulatory supervision (→Chapter 7 (2iii)).

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

The German public vocational system ensures that the licensees of nuclear power plants can find skilled workers, fore-men, technicians, engineers and scientists who received relevant occupational training that is documented by an officially recognised certificate. Generally, craftsmen and engineers in the fields of mechanical and electrical engineering, process engineering, physics and chemistry are already qualified before they begin employment in a nuclear power plant. Engineers can specialise in nuclear engineering during their course of study. To supplement the public vocational system, in 1970 the licensees of power plants founded a power plant school which offers special courses to provide the specific skills required for work at power plants. The different courses lead to a degree as skilled power plant worker and as power plant fore-man in the fields of mechanical and electrical technology, instrumentation and control, and nuclear technology.

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

The nuclear safety regulations define the following as responsible personnel:

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

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

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

These groups also apply to personnel from external firms.

The training manager of the licensee of the nuclear power plant is responsible for planning, performance, follow-up and documentation of the training activities. On the basis of training objectives given in [3-39] he establishes a programme specific to the plant and task to acquire and maintain qualification. The training of the responsible shift personnel is performed at a nuclear training facility, at the manufacturer, at a nuclear power plant itself, and at a full-scope training simulator suited to the respective nuclear power plant.

Training of shift personnel

Newly hired shift personnel first attends a three-months external course on basic nuclear technology which must be recognised by the competent authorities on the basis of standardised criteria [3-65]. This course is completed with an examination at the different levels of training. Within the framework of customer training, the manufacturer provides courses on specific topics (e.g. thermal-hydraulics, instrumentation and control, pumps) and a number of systems courses each with a duration of several weeks on the functions and operation of all essential systems of the nuclear power plant. The initial plant-specific training at the installation itself consists of theoretical instructions, of on-the-job training in various divisions, and of a longer participation as a shift member in the control room. A simulator training course of at least seven weeks (boiling water reactor) or eight weeks (pressurised water reactor) is mandatory. The initial simulator training covers all operating procedures from normal operation, abnormal operation to the control of design basis accidents and includes measures for the control and mitigation of beyond-design-basis accidents.

Qualification of the responsible shift personnel is completed with a written and oral examination. The oral examination is held before a board composed of representatives of the supervisory authority, independent experts, members of the training institutions (test on basic nuclear technology), and of the licensee (test on plant-specific knowledge). Unanimity of the board is required for passing the oral examination.

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

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

Table 11-1 Simulators for Nuclear Power Plants

	Nuclear power plant	Type Gross capacity MWe	Identification and site of the simulator	a) Manufacturer of the simulator b) Number of signals sent to control room	Start of training
1	Obrigheim KWO	PWR 357	D56 KSG/GfS	a) Thomson b) 10 600	1997 (until 1996 at D1)
2	Stade KKS	PWR 672	D1 KSG/GfS	a) Singer b) 12 900	1977
			Simulator KKS	a) CAE b)	planned for 1998
3	Biblis A KWB A	PWR 1225	D1 KSG/GfS	a) Singer b) 12 900	1977
4	Biblis B KWB B	PWR 1300	D1 KSG/GfS	a) Singer b) 12 900	1977
5	Neckarwestheim 1 GKN 1	PWR 840	D52 KSG/GfS	a) Thomson b) 11 100	1997 (until 1996 at D1)
6	Brunsbüttel KKB	BWR 806	S1 KSG/GfS	a) Singer b) 14 800	1978
7	Isar 1 KKI 1	BWR 907	S31 KSG/GfS	a) Atlas Elektronik b) 18 000	1997 (until 1996 at S1)
8	Unterweser KKU	PWR 1350	D51 KSG/GfS	a) Thomson b) 16 000	1997 (until 1996 at D1)
9	Philippsburg 1 KKP 1	BWR 926	S32 KSG/GfS	a) Atlas Elektronik b) 16 600	1997 (until 1996 at S1)
10	Grafenrheinfeld KKG	PWR 1345	D3 KSG/GfS	a) Krupp Atlas Elektronik b) 26 500	1988
11	Krümmel KKK	BWR 1316	S1 KSG/GfS	a) Singer b) 14 800	1978 until 1997
			Simulator KKK Krümmel	a) Siemens/S3T b) 27 000	1997
12	Gundremmingen B KRB B	BWR 1344	S2 KSG/GfS	a) Siemens b) 21 800	1993
13	Grohnde KWG	PWR 1430	D3 KSG/GfS	a) Krupp Atlas Elektronik b) 26 500	1988
14	Gundremmingen C KRB C	BWR 1344	S2 KSG/GfS	a) Siemens b) 21 800	1993
15	Philippsburg 2 KKP 2	PWR 1424	D42 KSG/GfS	a) Siemens/S3T b) 26 700	1997 (until 1997 at D1,D3)
16	Mülheim-Kärlich KMK	PWR 1302	D2 KSG/GfS	a) EAI/Singer b) 23 400	1986
17	Brokdorf KBR	PWR 1440	D43 KSG/GfS	a) Siemens/S3T b) 28 700	1996 (until 1995 at D3)
18	Isar 2 KKI 2	PWR 1440	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 (until 1995 at D3)
19	Emsland KKE	PWR 1363	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 (until 1995 at D3)
20	Neckarwestheim 2 GKN 2	PWR 1365	D41 KSG/GfS	a) Siemens/S3T b) 23 000	1996 (until 1995 at D3)

Training at simulators and models

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

The specifications by the licensees ensure a uniform minimum standard for the capabilities of the simulators, and ensure the qualification of the instructors and an adequate course programme. With respect to maintaining qualification, the following courses must be attended within a three-year cycle: a minimum of 20 days of instructions with at least 80 hours of simulator training (PWR) and 15 days of instructions with at least 60 hours of simulator training (BWR). The training is focused on normal operation, abnormal operation, design basis accidents, and severe beyond-design-basis accidents. The training programme is regularly reviewed by authorised experts on behalf of the BMU.

Since 1990 an additional simulator has been operated by Siemens, a manufacturer of nuclear power plants, initially at Karlstein and since 1997 at its company owned training centre in Offenbach. This simulator is a nuclear function trainer and is capable of simulating the most important safety procedures in a pressurised water reactor of recent design (fourth generation, Konvoi).

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

Maintaining qualification, advanced training

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

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

Certain training measures are directed at maintaining the qualifications and competence of the plant manager and of the division and subdivision heads. In these cases, participation in technical meetings and special courses is counted as training measure. The training measures actually performed for these personnel are likewise contained in the annual report to the supervisory authority.

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

Assessment of personnel qualification

The shift personnel of all nuclear power plants, generally, has many years of practical experience in the operation of nuclear power plants. The technical personnel - during initial training and repeatedly during advanced training - is regularly made aware of the importance of safety-oriented actions. Here, the findings from evaluating operation and operational events are of particular value. The good results of this continuously optimised training and advanced training is shown by the small number of incidents caused by human failure.

In the case of decommissioned nuclear power plants the number of personnel is reduced in accordance with the actual needs. The plant organisational structure and personnel necessary to assure the safety of the decommissioned plant is specified in the decommissioning licence required under the Atomic Energy Act.

12 Human Factors

Safe operation of a nuclear power plant depends not only on the technical equipment but also on the reliable and safety-oriented actions of the personnel that operates and maintains the technical equipment. In this respect the ergonomic design of equipment and work procedures, proper supervision, and plant organisational structure are just as important as the proper qualification of the personnel (→ Chapter 11 (2)).

German nuclear power plants are highly automated. This relieves the personnel from many manual actions. In addition to the extensive instrumentation and control systems available for operation, many of the more complex procedures can be activated by automatic controls. Of particular importance in this respect are the high-quality limitation systems. They prevent any physical operating parameters from exceeding the set control range and, thus, usually prevent actions by the reactor protection system. The limitation systems also ensure that the limiting conditions used in accident analyses are not exceeded. In the case of abnormal operation or design basis accidents, sufficient time is available to diagnose the situation and to take appropriate actions. The reactor protection system is designed to automatically control design basis accidents for a period of at least 30 minutes without the need for any manual action. Only in exceptional cases, namely where the accident situation is truly unambiguous and the required measures can be performed in time, is it permissible to call for earlier manual actions. Should the control room lose its functional capability it is ensured that independent auxiliary emergency systems will bring the plant, normally without the need for any manual actions, into a safe shutdown state in which it can remain for at least 10 hours. The limitation systems and the reactor protection system have absolute priority over manual actions and automatic operational controls.

German nuclear power plants are controlled and operated from a central control room. The control room is equipped with all means for information, activation and communication necessary for normal operation and for coping with abnormal operation and design basis accidents. Should the control room not be available, the nuclear power plant can be brought into a permanent safe shutdown state from a remote shutdown station [KTA 3904].

The ergonomic arrangement and grouping of the indicators and controls on the consoles and panels in the control room have a positive effect on overview and handling procedures. Indicators and controls are arranged along flow diagrams which schematically represent the structures and interrelationships of the systems. Release buttons prevent inadvertent actuations. Computerised information systems supplement the information base at all nuclear power plants.

With regard to maintenance and here, especially, the inservice inspections, extensive technical measures are provided in order to prevent faulty actions or to minimise their effects [KTA 3201.4; KTA 3211.4; 3-41; 3-43]. These measures reach from permanently installed and unambiguously identifiable testing devices to testing computers and the automatic resetting of safety devices by the reactor protection system should a major abnormal condition occur in the course of the inservice inspection. Valve positions of the safety systems are continuously monitored via annunciation loops and key-switch systems, to quickly detect and rectify faulty positions.

Apart from the corresponding design of the equipment, a comprehensive and complete documentation of operation is necessary to assure reliable and safety-oriented actions of the operating personnel [KTA 1404]. The actions necessary for operating conditions as well as for coping with design basis accidents are laid down in an operating manual [KTA 1201] (→ Chapter 19 (ii)-(iv)). On the one hand it comprises the plant regulations specifying tasks, authorisation and responsibilities of the personnel and, on the other, detailed instructions for the operation of the entire plant and the individual systems, as well as for the control of abnormal and accident conditions. Compliance with the safety relevant instructions of the operating manual is a mandatory provision. Deviations are permissible only in exceptional cases. The instructions for inservice inspections are laid down in an individual testing manual [KTA 1202] (→ Chapter 19 (iii)). The procedures to be followed in any maintenance and modification work are specified in a special maintenance regulation of the operating manual in accordance with the Regulatory Guideline on Maintenance [3-41] (→ Chapter 19 (iii)).

In addition to the documents in paper form, many nuclear power plants have or are in the process of introducing an ”integrated operation management system”. This enables the computer-aided description and control of work sequences and, to a certain extent, also the automatic surveillance of boundary conditions to be met.

Measures regarding the control of design basis accidents are laid down in the operating manual (→ Chapter 19 (iv)). The corresponding measures regarding severe beyond-design-basis accidents are laid down in an accident management manual specifying the actions to be taken and the procedures to be followed (→ Chapter 16 (1)).

The plant organisational structure is of great importance for a sound performance of the personnel. At all German nuclear power plants individual duties and responsibilities are clearly defined; the tasks of operation, maintenance, and supervision are all separated by organisational means from each other. The overall management concept relies on technical knowledge, understanding of the interrelationships of processes important for the safety of the plant, good working conditions, and on the responsibilities with respect to safety.
(→ Chapter 9).

The operational experience is systematically analysed with regard to human actions and to possible improvements in this field. In addition, the licensees have installed their own ”Human Factors Programme” aimed at detecting human actions, plant organisational or technical weak points. The feedback of experience has led to a large number of optimising measures both in the field of technology as in operating procedures, general organisation, and in training. The procedure for benefiting from operational experience is described in Chapter 19 (vii).

13 Quality Assurance

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

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

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

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

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

– Before erection of a nuclear power plant, but also before any material alterations or modifications, it is specified which component parts, components, systems and structures have an influence on plant safety and must, therefore, be classified as important to safety. In these individual cases quality characteristics must be specified and measures to assure that the quality characteristics are actually achieved.
– Persons charged with the task of implementing and auditing the quality assurance system must be authorised to have access to all relevant information, to monitor compliance with the quality assurance measures, and to propose solutions to possible problems. They must personally be independent from those persons or organisational units they monitor.
– All persons charged with the performance of certain duties are individually responsible for meeting the corresponding quality requirements.
– Those persons charged with the tasks of independent quality inspections must not themselves have been responsible for, or involved in, the manufacture of the product or the activity to be inspected.
– If it is essential for achieving the required quality characteristics, the qualification of the performing personnel is specified; the personnel qualification and its maintenance must be verifiable.

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

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

Ageing

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

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

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

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

This practice is being supplemented by appropriate research and development work.

The results achieved with respect to reliable and sound operation confirm the effectiveness of measures taken (→ Chapter 19 (iii)).

14 Assessment and Verification of Safety

14 (i) Assessment of Safety

The safety assessment during construction, commissioning and essential modifications of a nuclear power plant is performed within the licensing process (→ Chapter 7 (2ii)). Continuous safety evaluation during operation is performed within the scope of regulatory supervision.

Safety assessment in the licensing process

To be granted a licence for the construction, operation, essential modifications, or decommissioning of a nuclear power plant, an application must be filed with the competent authority. The safety assessment is then performed on the basis of the application and the documents to be submitted (→ Chapter 7 (2ii)).

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

– a safety analysis report which enables a conclusion to be made as to whether the rights of third parties could be violated by the operation of the nuclear installation (see below),
– supplementing plans, technical drawings, and descriptions of the nuclear installation and its parts,
– details on protective measures against disruptive actions or other 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 personnel employed in other areas at the installation,
– a safety specification comprising all important details on the safety of the nuclear installation and its operation (see below),
– information on compliance with legal liability provisions,
– description of the accumulating radioactive residual materials and of the intended measures for their treatment,
– description of the antipollution measures regarding water, air, and soil.

<u>Safety analysis report</u>

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

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

– the site,
– the nuclear power plant itself,
– the organisational structure and responsibilities,
– the radioactive materials located in the plant and the corresponding protective measures taken,

- protection against internal and external impacts,
- the operation of the nuclear power plant,
- the analyses of design basis accidents.

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

Additional information necessary for safety evaluation

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

Information is presented on the following subjects:

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

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

- concept,
- erection of civil structures,
- manufacturing of product forms,
- manufacturing of components,
- pressure test at the manufacturing plant,
- installation of components,
- pressure and leak rate tests at the construction site,
- commissioning of systems,

- delivery of fuel elements,
- initial core loading of the reactor,
- nuclear start-up of the facility, and
- refuelling.

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

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

Safety specifications

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

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

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

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

Involvement of authorised experts

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

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

Safety assessment in the supervisory procedure

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

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

<u>Accompanying inspections during construction</u>

In the course of their assessment of the documents submitted by the applicant, the authorised experts will also perform inspections during the construction phase. These accompanying inspections are performed independently of those by the manufacturer. They are required to verify the values, dimensions, or functions specified in the submitted documents. This includes, e.g. verification of materials composition, checking the assembling of components, and the performance of functional tests at the manufacturing plant. Similar inspections are also carried out at the construction site.

<u>On-site inspections during operation</u>

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

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

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

If deficiencies are found, the supervisory authority requests a corresponding correction by the licensee. In extreme cases where a situation might pose a high safety risk, this could result in an order for an immediate shutdown of the plant (→ Chapter 7 (2iv)).

<u>Assessment of reportable events</u>

The competent authority is notified by the licensee of any safety relevant event that occurs at his nuclear power plant. The reporting procedure and criteria are specified in regulatory guideline [1A-17]. In addition, the events will also be classified in accordance with the international INES evaluation scale. As a rule, the supervisory authority consults authorised experts and requests them to assess the events as well as the remedial measures taken or planned by the licensee. The central collection of reported events and a first assessment is performed by the Federal Office for Radiation Protection (→Chapter 19 (vi)-(vii)).

Specific safety reviews

Reportable events, modifications to the plant or its operation, maintenance tasks or new findings relevant to safety can lead to specific safety reviews of certain systems, components or items, on which the supervisory authority may request detailed documentation. Such a safety review may also be carried out systematically for the nuclear power plant as a whole taking probabilistic safety analyses into account (→ Chapter 14 (ii)). These reviews and analyses are generally also evaluated by the authorised experts consulted.

Reporting

Each licensee of a nuclear power plant reports regularly to the supervisory authority on the operation of his plant. He also submits an annual report to inform the Reactor Safety Commission (RSK).

14 (ii) Verification of Safety

Within his independent full responsibility for plant safety each licensee adjusts the safety level of the nuclear power plant to be in correspondence with the state of the art in science and technology over the entire operating life of the plant(→ Chapter 6). If new safety relevant findings come to light, the need for and appropriateness of improvements is evaluated. In addition safety assessments are continuously performed as part of the regulatory supervisory procedure, and discontinuously or periodically as a specific safety review (e.g. probabilistic safety reviews) or risk studies.

The main findings of these safety reviews and risk studies and the resulting safety relevant modifications to German nuclear power plants are summarised below.

Furthermore, the safety evaluations performed by OSART missions of the IAEA on German nuclear power plants are addressed.

Routine verification of safety by the licensee

The licence applicant submits safety verifications for the first time with the application for construction of a nuclear power plant. These must show that the plant will be in conformity with the valid nuclear safety regulations and will have the necessary safety characteristics (→ Chapter 14 (i)).

During operation a regularly repeated verification is required to show that the system functions important to plant safety are executed properly and, also, that the quality characteristics have not deteriorated below acceptable levels. To this end the systems are subjected to inservice inspections graded according to their individual safety relevance. Functional tests are performed to verify that the systems are in functioning order after a shutdown period, (e.g. for maintenance work). The licensee plans and performs regular preventive maintenance of all plant systems during plant operation, and he evaluates the operational experience. In planning and executing quality assurance, a distinction is generally made between inservice inspections of systems and components important to safety and other quality assurance activities.

The inservice inspections of systems important to safety are performed strictly in accordance with the requirements specified in the testing manual (→ Chapter 19 (iii)). The testing schedule contained therein specifies the test object, the nature, extent, and interval of the tests, the operating state of the nuclear installation at which they have to be performed, the identification and name of the test procedure, and which of the tests require the participation of the 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, methods of indirect verification are required. The specified required tests are reviewed regularly considering operating experience and new findings from safety research, and are adapted if necessary. Intended modifications of the testing manual are submitted to the supervisory authority for approval. Table 14-1 lists the nature and number of the mentioned inservice inspections, which can be deemed typical for a nuclear power plant with a pressurised water reactor (PWR).

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

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

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

In connection with the inservice inspections and the evaluation of operational experience special attention is paid to the early detection of cause for failures due to ageing. The causes of such failures are often systematic phenomena. There are specific regulatory requirements regarding ageing of certain plant components (e.g., fatigue analyses as part of component design, or type tests of instrumentation and control equipment in accordance with [KTA 3503] or [KTA 3504]). Due to the high frequency of inspections of the safety equipment in German nuclear power plants, ageing phenomena are usually detected at an early stage and remedies are taken. This is why failures due to ageing caused by systematic phenomena have so far been rarely observed. A particular case is the neutron irradiation 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 induced embrittlement, surveillance samples of the identical material as the reactor pressure vessel are tested in a number of intervals over the entire operating lifetime of the nuclear power plant. The test results deliver actual fracture mechanical parameters on which an assessment of the integrity of the reactor pressure vessel can then be based. In addition, the licensee performs the legally required tests and inspections on components in accordance with the conventional standards and regulations (e.g. the Steam Boiler Ordinance).

Inspections under governmental supervision

The supervisory activities of the *Länder* under nuclear legislation include the performance of safety assessments on a continuous as well as discontinuous and periodic basis, both as specific safety reviews and probabilistic safety analyses. They lead to remedial measures wherever appropriate. These continuous supervisory activities require about 30-40 man-years per year and power plant unit and assure an intensive assessment of plant safety. Federal supervision, on the other hand, is engaged in the analysis of more general safety aspects.

The safety reviews performed so far did not reveal the need for any immediate action. However, the plant-specific inspections during operation and the analysis of national and international operating experience have resulted in manifold improvements that have affected specific components and maintenance measures. These individual changes are not contained in this report. Moreover, there were changes that affected a large number of the nuclear power plants. These modifications are described below together with the backfitting measures and improvements important to safety.

Specific safety reviews

Against the background of the Chernobyl accident the Reactor Safety Commission (RSK), performed a safety review of all German nuclear power plants between 1986 and 1988 (see below). In addition, following a recommendation by the RSK, periodic safety reviews have been proposed and will be repeated in intervals of approximately ten years in order to take new safety findings into consideration. The periodic safety reviews supplement the continuously performed safety assessments as part of the nuclear regulatory supervision. The safety reviews of older plants performed during the past years can already be seen as components of a periodic safety review.

<u>Safety review by the RSK</u>

The Chernobyl accident prompted the Federal Government to ask the RSK for a safety review of all German nuclear power plants in 1986. This review was to take the following points into consideration:

– operating experience from domestic and foreign plants,
– the continued development of safety technology,
– the results of research in the field of nuclear safety, and
– risk studies.

The design basis accidents were analysed under the aspect of achieving the protection goals. Another priority was given to the realm of beyond-design-basis accidents and led to proposals of further measures for on-site severe accident management measures (→ Chapter 16 (1)).

All in all, these investigations revealed no deficiencies which would have required any immediate action. The engineered safety features of the individual nuclear power plants are different due to the different requirements valid at the time of their licensing; however, individual backfitting measures taken in the course of time sufficiently adapted the safety characteristics of the older nuclear power plants to the new developments in safety technology.

Periodic safety reviews

The licensees of German nuclear power plants committed themselves voluntarily to conduct periodic safety reviews at regular intervals of about ten years. In the case of a number of nuclear power plants this is already a licensing condition. Deterministic safety status analyses have been performed for ten nuclear power plants. Probabilistic safety analyses have been completed for 14 nuclear power plants, another five are in progress (Table 14-2).

In the future, the deterministic safety status analyses and probabilistic safety reviews will be performed in accordance with corresponding federal regulatory guidelines [3-74] recently published.

Table 14-2 Comprehensive Safety Reviews of Nuclear Power Plants
Listed is the year of submittance to the competent authority Status 12/97

	Nuclear power plant		**Type**	**Submitted to the authority** probabilistic safety analysis	safety status analysis
1	Obrigheim *)	KWO	PWR	1998	1992
2	Stade	KKS	PWR	1997	1987, 2000
3	Biblis A	KWB A	PWR	1991	1991
4	Biblis B	KWB B	PWR	1989	1998 **)
5	Neckarwestheim 1	GKN 1	PWR	1996	1996
6	Brunsbüttel	KKB	BWR	1997	---
7	Isar 1	KKI 1	BWR	1994	1994
8	Unterweser *)	KKU	PWR	1995	1990, 2000
9	Philippsburg 1	KKP 1	BWR	1995	1995
10	Grafenrheinfeld	KKG	PWR	1996	1998
11	Krümmel *)	KKK	BWR	1997	1997
12	Gundremmingen B	KRB B	BWR	1993	1998
13	Grohnde	KWG	PWR	1998	2000
14	Gundremmingen C	KRB C	BWR	1993	1998
15	Philippsburg 2	KKP 2	PWR	1998	1998
16	Mülheim-Kärlich	KMK	PWR	---	---
17	Brokdorf *)	KBR	PWR	1996	1996
18	Isar 2 *)	KKI 2	PWR	1998	1998
19	Emsland *)	KKE	PWR	1998	1998
20	Neckarwestheim 2 *)	GKN 2	PWR	1998	1998

*) licensing provision for periodic safety review
**) date planned by the authorities

Risk studies, probabilistic safety analyses

After publication of the American risk study WASH 1400 a German risk study was conducted in two phases for a pressurised water reactor (PWR). Phase A (1976-79) followed the methodology used in WASH 1400 and also considered radiological effects on the general public and environment in accordance with the knowledge base of that time. Phase B (1985-89) continued this work, this time with advanced methodologies. This later work limited itself to evaluating the damage frequencies of different plant systems and of the core. In the nineties an additional safety study with probabilistic evaluation was conducted, this time for a more modern boiling water reactor (BWR). These risk studies were carried out on behalf of the Federal Government with two main purposes; firstly, of furthering the development of the methods of probabilistic safety review and, secondly , of quantifying the safety level of the analysed reference plants. They also yielded important insights into the effects of the technical design on the safety level. Currently studies are in progress investigating specific safety aspects, e.g. event sequences during specific modes of operation (start-up, shutdown, safe shutdown of the plant).

An important result of these risk studies is that the significance of the event sequence “double ended rupture of a main coolant pipe“ had been grossly overestimated in view of its extremely low frequency of occurrence. On the other hand, the sequences “small leak in the coolant pressure boundary“, “loss of coolant from connecting pipes outside of the containment“, and “transients“, are now accounted a much higher significance due to a frequency of occurrence which is higher by several orders of magnitude than that of a large rupture, although the reliability of their control is not essentially different.

The event sequence analyses of a small leak in the reactor coolant pressure boundary have revealed that different measures are needed to control the situation depending on leak size and the leak location (leakage in piping, leakage in the steam space of the pressuriser, leakage of a defective steam generator tube). These findings were considered in subsequent safety evaluations.

Probabilistic safety analyses have shown the superior significance of a highly reliable steam generator feed-water supply and a controlled main steam pressure relief for controlling transients and, thus, for the safety of a plant. This resulted in safety improvements in a number of PWR plants.

Backfitting and safety improvements

The findings of the safety evaluations and the resultant backfitting and safety improvements show that the licensed safety status of the plants have at least been successfully maintained but, also, that newer safety findings were given appropriate consideration during the time of licensed operation. Thus, the safety of nuclear power plants has been successfully adapted to the state of the art in science and technology.

Table 14-3 gives an overview of the main technical modifications important to safety, broken down with regard to the four design generations of PWRs and the two construction lines of BWRs.

In the following, a compilation of the essential backfitting and safety improvements is given which result from the mentioned safety reviews and risk studies described and were relevant for a considerable number of nuclear power plants.

<u>Improvement of accident resistance</u>

In particular the instrumentation, transducers, cables (including distribution boxes, ducts, connection boxes) and actuators of the newer plants have a significantly improved accident resistance with respect to the pressure, temperature and humidity conditions expected after a loss-of-coolant accident. The respective equipment of older plants was thoroughly checked and technically improved wherever necessary.

<u>Evaluation and expansion of the considered spectrum of design basis accidents</u>

Each of the subsequently performed licensing procedures of the ever developing designs of nuclear power plants has led to an increasing number and increasing differentiation of the design basis accidents that are subject to an accident analysis. This includes the specification of representative radiological accidents outside of the containment vessel (instrumentation pipe in the annulus, open for 30 minutes after rupture) and precise specifications for the steam generator tube leakage. Subsequently, older plants also had to be verified with regard to their ability to cope with these accidents and were technically improved where necessary.

Exchange of materials of main steam and feed-water lines, and of high-energy pressure vessels

Initiated by earlier discussions about possible bursting of large vessels and pipes, and after cracks were detected in the main steam and feed-water lines of BWRs, the RSK developed a concept to preclude rupture of the reactor coolant pressure boundary. This concept comprises requirements with respect to the basic safety of materials, design, manufacturing, inspections, as well as the respective quality assurance that, taken together, ensure that large ruptures can be excluded. Correspondingly, an exchange was performed in all BWRs where the pipes out to the second isolation valve did not meet this new requirement regarding basic safety. A similar exchange was performed on vessels with high-energy content at nuclear power plants with PWR and BWR where this would result in a significant reduction of risk.

Extended verification of seismic design

Due to progress in methods developed for the determination of the characteristics for seismic design, and due to the development of better dynamic model analyses with respect to the design of structures and components, a supplementary evaluation was conducted for those nuclear installations where the seismic design was still based on older methods. In individual cases this may require technical improvements.

Investigation of event sequences affecting more than one system or redundancy

Following a suggestion by the RSK, special investigations were performed on older plants where the physical separation of redundant safety equipment was not realised in the same way and to the same extent as in newer plants. In these investigations particular emphasis was placed on the impact and consequences of fire, flooding due to pipe rupture, and faulty maintenance work. The proper functional de-coupling of redundant instrumentation and control systems was also investigated. The findings resulted in a general improvement in fire protection. In individual cases additional plant-specific protective measures were taken against flooding. For all but one nuclear power plant (where the investigations are still in progress) it has been shown that a sufficient functional de-coupling exists for the instrumentation and control equipment.

Improvement of fire protection measures

Due to the development of fire protection concepts towards predominantly structural protective means as well as to more stringent quality and inspection requirements placed on the technical fire protection equipment, older plants were subjected to comprehensive improvement measures. Apart from a general improvement of structural protective means, in particular for the protection of cables, the improvement measures comprise additional or expanded fire fighting systems, and, where not already in existence, the formation of a plant fire brigade having the same qualification as a professional fire brigade.

Table 14-3 Backfitting and Safety Improvements in Nuclear Power Plants - According to Design Generation (PWR) and Construction Line (BWR)

X improvement through backfitting measures
● already covered by the design

Purpose of improvement	PWR design generation				BWR construction line	
Improvement measures	1	2	3	4	69	72
1. Enhanced reliability of normal operation						
Additional off-site power supplies	X	X	●	●	X	●
2. Enhanced effectiveness and reliability of safety equipment						
Additional emergency diesel generators	X	X	●	●	X	●
Additional high pressure and low pressure emergency core cooling systems (PWR)	X	●	●	●		
Extension of emergency core cooling systems / additional injection lines (PWR)	X	X	●	●		
Technical improvement of the high-pressure/low-pressure interfaces	X	X	X	X	X	X
Self-supporting emergency core cooling systems/ new diversified emergency core cooling system (BWR)					X	X
Additional emergency feed water systems	X	X	●	●	●	●
Technical improvement of components important to safety to withstand design-basis accidents	X	X	●	●	X	●
Additional valves for containment isolation (BWR)					X	●
Diversified pilot valves for safety and pressure relief valves (BWR)					X	●
Diversified pressure relief valves (BWR)					X	X
3. Control of specific emergency situations						
Emergency systems	X	X	●	●	X	●
4. Mitigation of fire consequences						
Physical separation by installing new systems in separate buildings	X	●	●	●	X	●
Additional fire fighting systems	X	●	●	●	●	●
Backfitting of fire fighting systems	X	●	●	●	●	●
Technical improvement of fire dampers and fire partitions	X	X	●	●	●	●
Additional fire dampers	X	●	●	●	X	●
5. Improvement of barriers						
New pipes of improved materials for main steam, feed water, 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. Emergency preparedness						
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

Investigation of the high-to-low-pressure transition in primary coolant systems

Initiated by an event which resulted in the release of a small amount of reactor coolant outside of the containment, the pressure relief of systems connecting to the primary coolant pressure boundary as well as the quality of the isolation valves were subjected to a thorough investigation. In individual cases this led to a plant-specific improvement of system technology and an improved design of the valves.

Backfitting of an independent residual heat removal system in one BWR

Probabilistic analyses indicated the need for a higher reliability of the residual heat removal function in one particular BWR. Sufficiently high reliability was achieved by installing a new diverse system that was independent of the existing systems.

Improvement of the steam generator feed and the main steam relief in PWRs

At a number of plants additional systems were installed to improve the reliability of emergency feed-water supply. It could generally be shown, however, that, taking the emergency feed-water systems into consideration, the additionally installed emergency systems are sufficiently reliable to cope with transients. The measures with respect to the controlled main steam relief concerned, in particular, the plant-specific technical improvements of the main steam relief control valves.

Optimised control of steam generator tube leakage in PWRs

In phase B of the German Risk Study, in-depth investigations were conducted with respect to the different event sequences following a steam generator tube leak. These revealed fundamental scope for optimising both the system technology and the operating procedures. This concerned, in particular, detection of the affected steam generator, raising of the actuation pressure limits for main steam relief to the atmosphere, lowering the primary circuit pressure to reduce release of coolant and utilisation of shutdown procedures to avoid actuation of the high pressure safety injection system. A corresponding optimisation was implemented in all PWRs under consideration of the individual plant design.

Improvement of the off-site power supplies and the emergency power supply

To reliably cope with a long-term emergency power situation, all nuclear power plants are now equipped with two off-site power supplies (main and reserve grid connection). Furthermore, all plants now also have an emergency power supply that is protected even against rare external impacts.

Extended automation

Analyses have shown the benefits of a more extensive use of preceding limits to avoid actuation of safety systems as much as possible, and of a further automation of procedures for controlling abnormal occurrences. This can significantly reduce the frequency of occurrence of impermissible thermal-hydraulic conditions and of transients. Corresponding plant-specific improvement have been applied in the meantime.

Furthermore, as a result of phase A of the German Risk Study, measures have been taken in PWRs to increase the reliability in controlling a small-leak loss-of-coolant accident by introducing an automated load reduction in these cases.

Decoupling of operational and safety systems

Reliability analyses have shown that those safety systems functioning independently from operational systems are more reliable than those, where the systems are intermeshed. At the same time, the failure susceptibility of the operational systems is also reduced. Therefore, following a recommendation of the RSK, the de-coupling of these systems was considered to a large extent in the design of newer plants. As far as possible, corresponding technical improvements have been carried out also at the older plants, or the respective impact on safety was evaluated.

Significance of common-cause failures

All probabilistic assessments revealed the particular significance of the possibility for simultaneous failure of several redundancies of the safety equipment due to a common-cause failure. Remedial measures are the diversification of equipment, of functions and of procedures. To properly verify the evaluation results of common cause failures or possible remedies, improved evaluation methods are needed and are currently under development. In a number of plants diversified equipment has been backfitted, e.g. diversified pilot valves for the safety and relief valves of BWRs.

Improved reliability of operational and safety functions during plant shutdown

As a result of recent investigations of the safety relevance of plant conditions during reactor shutdown, improvement measures have been or are being taken in those nuclear power plants where this had not already been considered in the design in order to increase the reliability of residual heat removal, e.g. at mid-loop operation of the cooling systems.

Improvement of the accident monitoring instrumentation

To adapt to current nuclear regulations, the accident monitoring instrumentation [KTA 3502] of older plants has been improved, both, with respect to extent and quality. Following a recommendation of the RSK, all nuclear power plants have also been equipped with additional instrumentation to facilitate accident management measures.

Control of hydrogen release in the event of a loss-of-coolant accident

Thermal recombiners have generally been provided in all nuclear power plants for the long-term limitation of hydrogen concentration in the containment to below 4 % vol. In further investigations the effects of local concentrations were analysed. These have demonstrated that due to given circumstances inside the containment and with additional means for mixing no impermissible high local H_2 concentrations can occur. Improvement measures, therefore, concentrated on measures for an improved mixing of the containment atmosphere, and on equipment for measuring the H_2 concentrations.

Backfitting of emergency systems

Following the design of more recent plants against aircraft crash and pressure waves from external explosion, the older plants were backfitted with an emergency system that is physically and functionally independent of the existing safety system. This emergency system is designed and constructed in such a way that in the case of external impacts, including disruptive actions or other interference by third parties, it could take over possibly failed functions of the safety system, such as reactor shutdown, steam generator feed-water supply, and residual heat removal. An emergency control room which is independent of the control room was also part of these backfitting measures.

Utilising plant margins for accident management measures

Investigations have shown that in the event of unavailability of safety equipment, the utilisation of plant margins for (severe) accident management can significantly reduce the probability of conditions leading to core damage. Furthermore, specific on-site accident management measures can mitigate the consequences of a severe beyond-design-basis accident with core melt (→ Chapter 16 (1)).

Equipment of control room and emergency control room for emergency situations

Measures were taken in connection with the introduction of on-site accident management measures to enable extensive controls and operator actions also under conditions prevailing under severe beyond-design-basis accidents, both in the control room and in the emergency control room. This included, in particular, independent ventilation of the control room with air filtration for radioactive materials, and an improved emergency power supply from batteries.

Improving pressure limitation and pressure relief of the primary coolant circuit of a PWR

To improve function and reliability of the primary side pressure limitation during an ATWS, and of the pressure relief to enable “bleed and feed“ as a preventive accident management measure, the pressuriser relief valves have been technically improved wherever this had not already been part of the original design.

Limiting the hydrogen concentration during severe accidents with core damage in PWR

Investigations have shown that a large amount of hydrogen will be generated in the case of severe accidents with core damage or complete core melt. Due to the prevailing boundary conditions it can be assumed that early uncontrolled hydrogen combustion will occur in the containment at a time that would not be critical to its integrity. The RSK has requested investigations and developments for the early removal and reduction of hydrogen. The developments and preparations for implementing corresponding measures are nearly completed, as already mentioned in Chapter 16 (1). First implementation of these measures is expected for 1998.

Inertisation the containment vessels of BWRs

To control the release of hydrogen during a severe beyond-design-basis accident in a BWR of the construction line 69, the containment vessel is inertised even during commercial operation. This measure also completely covers unfavourable conditions during loss-of-coolant accidents.

Measures for filtered venting of the containment after severe beyond-design-basis accidents with core damage

In order to maintain containment integrity even during severe beyond-design-basis accidents with core damage, the RSK has requested the installation of a filtered venting system to enable the relief of containment pressure while retaining radioactive aerosols and iodine. The RSK has also specified the corresponding design requirements and modes of operation. These measures are part of accident management measures described in Chapter 16 (1) and have already been implemented to a large extent.

International safety evaluations, OSART Missions

Upon invitation, the IAEA has so far conducted four OSART Missions at the following German nuclear power plants:

- Biblis A (PWR) 1986,
- Krümmel (BWR) 1987,
- Philippsburg 2 (PWR) 1987, and
- Grafenrheinfeld (PWR) 1991 (mission) and 1993 (follow-up visit).

The Missions evaluated the following areas of nuclear power plant operation

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

and in none of these areas found any significant deficiencies. On the contrary, it was specifically pointed out that all four plants were well designed and that the operating personnel were committed to safety. The recommended improvements of individual items were implemented, which was demonstrated by the follow-up visit to Grafenrheinfeld. The recommendations that had not been realised at the time were those where the licensing procedure was still in progress or where a standardised Federal regulation was expected to be issued.

15 Radiation Protection

Fundamentals

The Radiation Protection Ordinance [1A-8] is the legal basis for the handling of radioactive materials. Its objective is to protect persons, property and the environment from damage from ionising radiation when handling radioactive materials, from nuclear fuel and from nuclear installation in accordance with Section 7 Atomic Energy Act as well as from installations for the generation of ionising radiation. It achieves this objective by means of specifying requirements and limits. The ordinance has been amended and revised a number of times over the years, and has been adapted to individual EURATOM Basic Safety Standards [1F-15]. Currently, an amendment to bring it into line with the EURATOM Basic Standards of 1996 is in progress. The EURATOM Basic Safety Standards are the framework for radiation protection in the European Union. The Radiation Protection Ordinance specifies the requirements regarding licensing and reporting obligations for the handling of radioactive materials, for their import, export and their transport, and it specifies requirements for administrative and physical/ technological protective measures and for medical surveillance.

The following principles of radiation protection specified in Section 28 Radiation Protection Ordinance are decisive for any activity involving radiation protection:

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

Together with the principle of proportionality - a constitutional right to be accounted for in all cases - these principles lead to an obligation to minimise radiation exposure that is similar to the internationally accepted ALARA principle (as low as reasonably achievable).

The essential dose limits specified in the Radiation Protection Ordinance and addressed below are listed in Table 15-1.

Occupationally exposed persons

The limit for the body dose of occupationally exposed persons that was specified by the Radiation Protection Ordinance is a maximum effective dose of 50 mSv per year. In the future, the maximum effective dose will be limited to 20 mSv per year because of the incorporation of the EURATOM Basic Safety Standards [1F-15] into German law. Other limits apply to partial body doses, to persons who do not regularly work inside restricted access areas and to special groups of persons.

The body doses may not exceed one half of the annual limits in any three consecutive calendar months. In the case of occupationally exposed persons, the effective doses summed up over all calendar years shall not exceed an overall life dose of 400 mSv.

Exposures to radiation exceeding these limits may be allowed in order to combat the consequences of accidents or danger to persons. The body doses received in these cases may not exceed twice the annual limit in a given calendar year and five times the annual limit over the life-span of the person.

Table 15-1 Maximum Permissible Dose Limits Specified in the Respective Sections of the Radiation Protection Ordinance

Section	Scope of Applicability	time period	limit value [mSv]
	Design and operation of nuclear installations		
28	**Design-basis accident limit values**		
	1 Effective dose: partial body dose for gonads, uterus, red bone marrow	event	50
	2 Partial body dose: all organs and tissues unless specified under 1, 3, 4 or 5	event	150
	3 Partial body dose: bone surface, skin unless specified under 5	event	300
	4 Thyroid	event	150
	5 Partial body dose: hands, arms, forearms, feet, lower legs, ankles including the associated skin	event	500
44	**Environment of nuclear facilities**		
	Direct radiation including radiation exposure from discharges	calendar year	1.5
45	**Limit values for the discharges with exhaust air or waste water during specified normal operation**		
	1 Effective Dose: partial body dose for gonads, uterus, red bone marrow	calendar year	0.3
	2 Partial body dose: all organs and tissues unless specified under 1or 3	calendar year	0.9
	3 Partial body dose: bone surface, skin	calendar year	1.8
	Dose limit values for occupationally exposed persons		
49	Occupationally exposed persons in Category A		
	1 Effective dose: partial body dose for gonads, uterus, red bone marrow	calendar year	50
	2 Partial body dose: all organs and tissues unless specified under 1, 3 or 4	calendar year	150
	3 Partial body dose: thyroid, bone surface, skin unless specified under 4	calendar year	300
	4 Partial body dose: hands, arms, forearms, feet, lower legs, ankles including the associated skin	calendar year	500
	Occupationally exposed persons in Category B		
	5 Effective Dose: partial body dose for gonads, uterus, red bone marrow	calendar year	15
	6 Partial body dose: all organs and tissues unless specified under 5, 7 or 8	calendar year	45
	7 Partial body dose: thyroid, bone surface, skin unless specified under 8	calendar year	90
	8 Partial body dose: hands, arms, forearms, feet, lower legs, ankles including the associated skin	calendar year	150
	Body dose in three successive months	quarter year	50 % of 1-8
	Body dose for persons under 18 years	calendar year	10 % of 1-4
	Partial body dose at the uterus of women of childbearing age	month	5
	Effective Dose	entire life	400
50	Removal of consequences of accidents or of pending danger to persons (only Category A above 18 years)	calendar year	200 % of 1-4
	Removal of consequences of accidents or of pending danger to persons (only Category A above 18 years)	entire life	500 % of 1-4

The body doses are determined for all persons spending any time in the controlled access area. This is usually done by measuring the personal dose. All occupationally exposed persons are examined by authorised physicians. The Radiation Protection Ordinance also regulates the required documentation of personnel doses and of the results of the medical examination as well as the obligations with regard to reports to the supervisory authority. Data on the radiation exposure of occupationally exposed persons are recorded by the Federal Office for Radiation Protection in a central register.

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

Radiation exposure of the general public during operating conditions

The dose limits and requirements applying to the radiation exposure of the general public during specified normal operation of the nuclear installation are specified in Sections 44 to 46 Radiation Protection Ordinance.

It specifies an effective dose limit of 1.5 mSv per calendar year including direct radiation and radiation exposure from discharges. This will be reduced to 1 mSv in accordance with the EURATOM Basic Safety Standards [1F-15]

The radiation protection supervisor - i.e. the holder of the licence - is required to plan the technical design and the operation of the nuclear installation in such a way that the radiation exposure of a person caused by a discharge of radioactive materials with air or water from this nuclear installation will not exceed the effective dose limit of 0.3 mSv per calendar year. Additional limits are specified for partial body doses. The specification of these limits is guided by the risk assessments of the ICRP and by the regional variations of natural radiation exposure in Germany.

In the nuclear licensing procedure upper limits are set for the discharge of radioactive materials from nuclear installations with air or water by specifying maximum permissible amounts for annual discharges and short-term emissions that ensure that the mentioned dose limits are not exceeded.

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

Radiation exposure of the general public during design basis accidents

Central issues evaluated during the licensing procedure of a nuclear power plant are the planned structural and other technical measures to protect against design basis accidents (→ Chapter 18 (i)). In accordance with Section 28 Radiation Protection Ordinance it has to be shown - notwithstanding the obligation to minimise radiation exposure - that the effective dose in the vicinity of the nuclear installation will not exceed the planning value of 50 mSv in a design basis accident (integrated over all exposure paths as 50-year dose commitment). Further planning values apply to partial body doses. Regulatory guideline [3-33] specifies the analytical models and assumptions to be applied for these verifications.

Emission monitoring

In the event of a discharge of radioactive materials to air, water or soil it must be ensured that

– an uncontrolled release is avoided,
– the discharged radioactivity is as low as practicable,
– the discharge is monitored and is reported to the competent authority at least once a year, specifying its nature and activity.

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

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

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

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

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

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

Monitoring immissions

In addition to the immission measurements performed by the licensee in the vicinity of the nuclear installation, the competent authority will, in accordance with Section 48 Radiation Protection Ordinance, employ an independent organisation to also monitor the vicinity for immissions. These measurements are performed to supplement the emission monitoring with the purpose:

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

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

In order to be able to detect an increase of radioactivity with respect to the natural background in the vicinity of nuclear installations, a measurement of background radiation is required for a two-year period prior to commissioning. The extent of this programme depends on the measurements to be performed during operating conditions. In addition, regulatory guideline [3-23] specifies at what time and to what extent which monitoring measures can be discontinued and which must continue to be performed after a final shutdown of the plant and its long-term safe storage.

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

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

Remote monitoring of nuclear power plants

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

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

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

Integrated measurement and information system

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

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

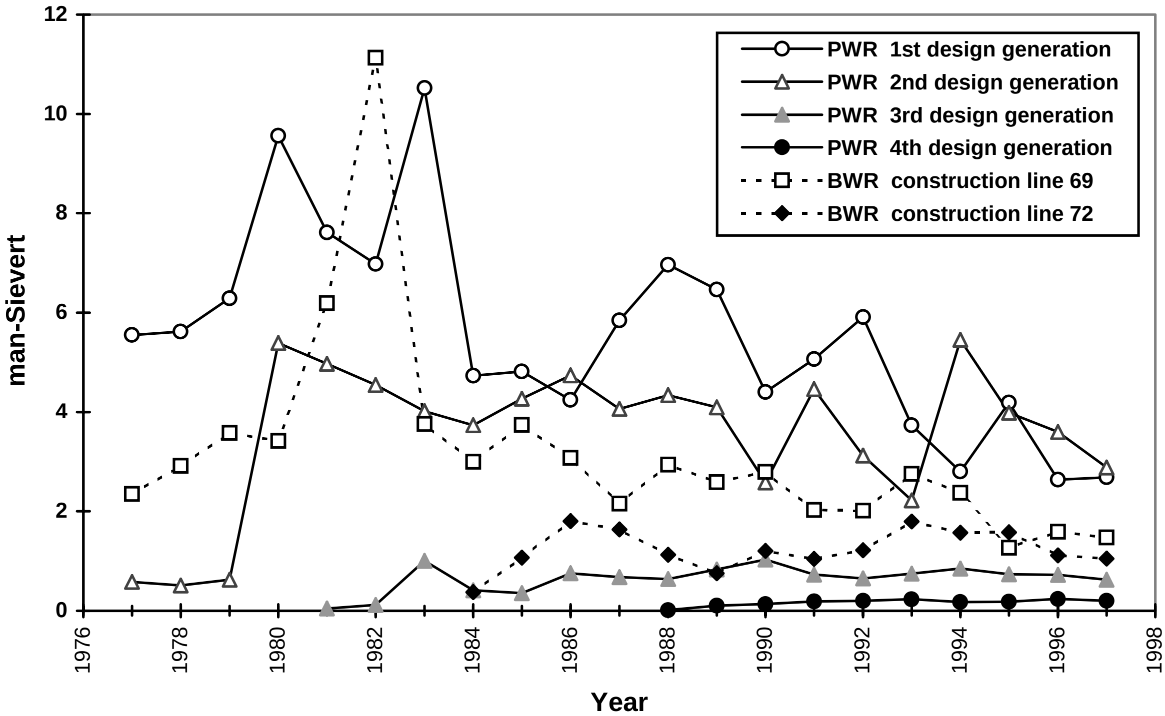


Figure 15-1 Average Annual Collective Dose in Nuclear Power Plants

Results of radiation protection in nuclear power plants

Data on the discharge of radioactive materials with air and water from German nuclear installations and on the resulting radiation exposure are published by the Federal Government in its annual report “Environmental Radioactivity and Radiation Exposure“ to the *Bundestag* (the German Parliament), and in a further more detailed annual report with the same name issued by the BMU. Generally, the nuclide-specific annual discharges stay considerably below the emission limits individually licensed for each nuclear power plant. In the majority of cases they reach just a few percentage points of these limits. In turn, the values calculated for the radiation exposure of the general public is far less than 10 % of the maximum allowed dose limits specified in the Radiation Protection Ordinance. The nuclide-specific discharges of radioactive materials from German nuclear power plants are listed in Tables 15-2 and 15-3 for the year 1996. With respect to tritium, there are technical reasons for the fact that its discharge with the waste water from PWR plants reaches a level up to 35 % of the licensed limits.

Table 15-2 Discharge of Radioactive Materials with Vent Air in 1996 from Nuclear Power Plants

	Noble gases [Bq]	Aerosols [Bq]	Iodine 131 [Bq]	Tritium [Bq]	Carbon-14 [Bq]
PWR					
Obrigheim	3.3 E+11	9.2 E+06	6.3 E+03	1.5 E+11	6.1 E+10 [2)]
Stade	1.9 E+12	1.0 E+06	2.0 E+06	3.3 E+11	1.6 E+11 [3)]
Biblis A	8.3 E+10	1.4 E+06	1.4 E+07	1.0 E+11	4.5 E+10 [4)]
Biblis B	2.5 E+12	6.2 E+05	1.6 E+07	1.2 E+11	2.7 E+11 [5)]
Neckarwestheim 1	7.1 E+11	3.1 E+06 [10)]	3.6 E+05	2.6 E+11	7.4 E+09
Unterweser	3.5 E+12	1.5 E+06	9.7 E+04	5.6 E+11	5.1 E+10
Grafenrheinfeld	1.6 E+11	2.5 E+06	1.5 E+05	5.5 E+11	1.0 E+11
Grohnde	2.5 E+13	9.6 E+05	8.2 E+06	6.8 E+11	5.2 E+10
Philippsburg 2	1.1 E+12	1.5 E+05	4.3 E+05	9.7 E+11	1.9 E+11 [6)]
Mülheim-Kärlich	< MDL *)	< MDL	< MDL	8.0 E+10	4.9 E+09
Brokdorf	8.0 E+11	< MDL	6.0 E+05	3.7 E+11	2.1 E+11 [7)]
Isar 2	1.7 E+11	1.8 E+06	< MDL	1.3 E+12	4.7 E+11
Emsland	1.2 E+11	6.6 E+05	< MDL	2.0 E+12	1.8 E+11 [8)]
Neckarwestheim 2	3.9 E+12	1.3 E+05	3.5 E+05	1.9 E+11	1.8 E+11 [9)]
BWR					
Brunsbüttel	7.2 E+12	3.4 E+07	1.2 E+07	4.0 E+10	7.0 E+10
Isar 1	1.5 E+11	1.6 E+07	2.3 E+07	5.6 E+10	2.0 E+11
Philippsburg 1	5.2 E+11	2.1 E+07	4.7 E+07	7.1 E+10	5.8 E+11
Krümmel	1.4 E+13	8.6 E+07	2.2 E+08	4.6 E+10	8.2 E+10
Gundremmingen B+C	< MDL	7.4 E+04	1.4 E+05	2.2 E+12	1.6 E+12

*) < MDL : less than minimum detectable limit
2) of the total value 7.5 E+09 Bq are due to CO_2
3) of the total value 2.2 E+10 Bq are due to CO_2
4) of the total value 8.0 E+09 Bq are due to CO_2
5) of the total value 2.0 E+10 Bq are due to CO_2
6) of the total value 4.3 E+10 Bq are due to CO_2
7) of the total value 9.0 E+10 Bq are due to CO_2
8) of the total value 1.7 E+11 Bq are due to CO_2
9) of the total value 1.2 E+11 Bq are due to CO_2
10) of the total value 2.6 E+05 Bq are due to Sb 122 (half-life < 8 days)

Table 15-3 Discharge of Radioactive Materials with Waste Water in 1996 from Nuclear Power Plants

	Fission and activation products (without Tritium) [Bq]	Tritium [Bq]	α-emitters [Bq]
PWR			
Obrigheim	3.6 E+08	5.7 E+12	< MDL *)
Stade	1.8 E+08	2.9 E+12	3.0 E+05
Biblis A	2.4 E+07	3.6 E+12	< MDL
Biblis B	5.0 E+08	1.1 E+13	< MDL
Neckarwestheim 1	4.7 E+06	1.3 E+13	9.3 E+04
Unterweser	2.0 E+08	1.2 E+13	< MDL
Grafenrheinfeld	1.1 E+07	1.6 E+13	< MDL
Grohnde	1.1 E+08	1.0 E+13	1.1 E+05
Philippsburg 2	2.9 E+08	1.5 E+13	< MDL
Mülheim-Kärlich	8.9 E+06	4.9 E+10	< MDL
Brokdorf	2.6 E+07	1.4 E+13	< MDL
Isar 2	2.9 E+05	2.0 E+13	< MDL
Emsland	9.8 E+03	1.2 E+13	< MDL
Neckarwestheim 2	9.9 E+07	2.1 E+13	9.6 E+04
BWR			
Brunsbüttel	1.1 E+08	3.5 E+11	4.1 E+04
Isar 1	1.6 E+08	1.0 E+12	2.2 E+06
Philippsburg 1	8.4 E+08	5.4 E+11	3.0 E+06
Krümmel	1.4 E+07	6.8 E+11	< MDL
Gundremmingen B+C	6.4 E+08	1.1 E+13	< MDL

*) < MDL : less than minimum detectable limit

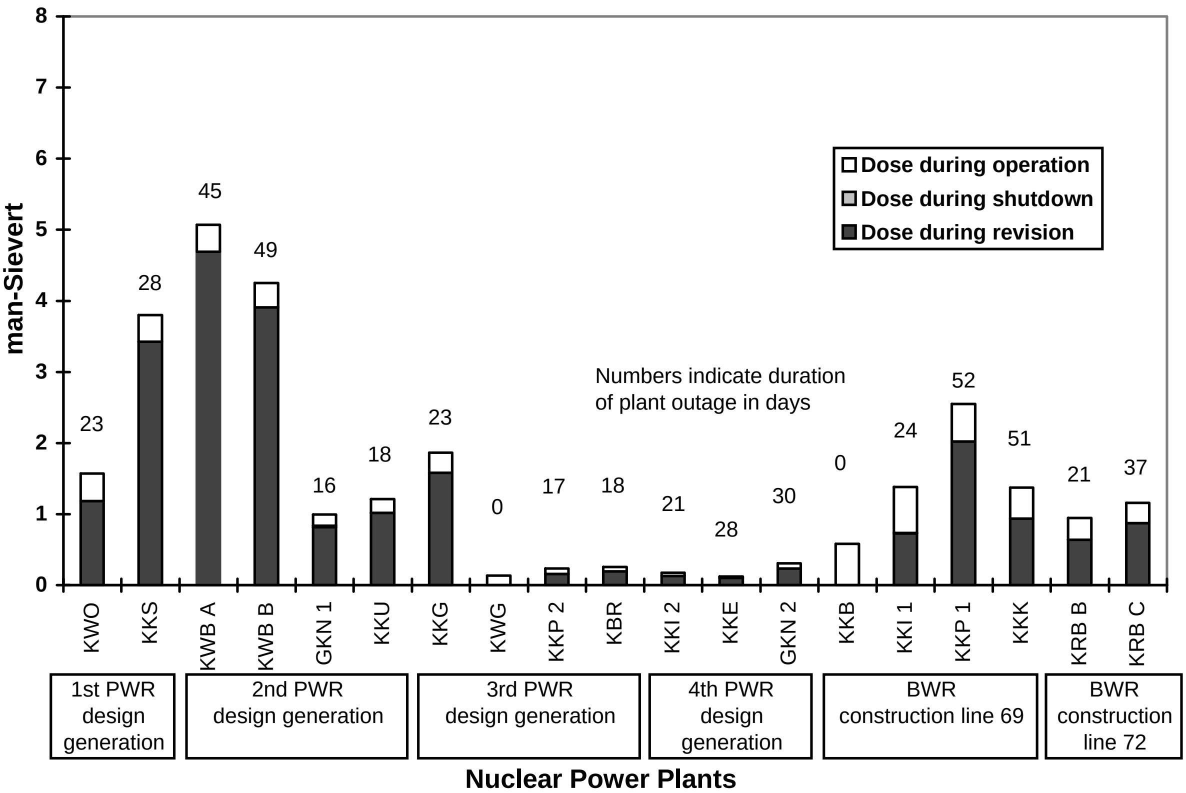
The personal dose of the workers in German nuclear power plants has decreased continuously over the past years. Figure 15-1 shows the annual average collective dose in nuclear power plants of different design generations and construction lines. The peaks shown for the nuclear power plants with BWRs in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

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

With respect to BWRs, the backfitting measures over the last two or three years that contributed most in reducing the personal dose of plant and external personnel are, in particular:

– reconstruction of the forced circulation pumps by introducing hydrodynamic bearings for the pump shafts in the BWRs of construction line 69 (KKB, KKP 1, KKI 1); this replaced the pressurised bearing water system that had required a high maintenance effort at high radiation levels;
– introduction of the special austenitic materials inspection programmes in all BWR plants which became necessary after cracks were detected in austenitic pipes. These special inspections identified a number of faulty pipes which were, then, either repaired or, in many cases exchanged entirely. In the case of the exchanged pipes, the number of weld seams could be considerably reduced which meant that the number of required inservice inspections could be reduced (→ Chapter 14 (ii)) and, in turn, led to a further reduction of the radiation exposure of the personnel.

In the ten-year period from 1988 to 1997, with up to 22 operating nuclear power plants - together grossing 196 reactor operating years - a total of 2030 events were reported in accordance with the criteria specified in the nuclear reporting ordinance [1A-17] (→ Chapter 19 (vi)). None of these reportable events caused any of the limits to be exceeded, be it with respect to discharges with exhaust air or waste water, with respect to personal doses or with respect to uncontrolled releases of radioactive materials to the vicinity of the enclosed sites of the nuclear power plants.



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

Figure 15-2 Annual Collective Dose in Nuclear Power Plants in 1997 - According to Mode of Operation

16 Emergency Preparedness

16 (1) Emergency Preparedness, Emergency Plans

In Germany, the emergency preparedness with respect to severe accidents (events beyond the design basis) is directed to prevent any impacts on the environment and, if that is no longer possible, to reduce these impacts. This involves, on the one hand, measures planned to be taken inside the nuclear power plant (on-site accident management) and, on the other hand, measures planned to be taken outside of the plant (off-site disaster control). It should be emphasised that emergency preparedness refers exclusively to severe beyond-design-basis accidents. The design basis accidents have to be controlled by automatic procedures (→ Chapter 12), and their radiation impacts must stay below specified planning limits (→ Chapter 15).

On-site accident management only concerns the nuclear power plant itself and is planned and performed under the responsibility of the licensee. Although no legal obligations exist, the licensees have taken extensive measures in this respect.

The German Constitution (Basic Law) specifies that in times of peace the competency for disaster control lies in the hands of the *Länder*. The corresponding measures are planned by the competent authorities and are co-ordinated with the licensee of the nuclear power plant. The legal framework for disaster control is laid down, firstly, in the Atomic Energy Act [1A-3] in conjunction with the Radiation Protection Ordinance [1A-8] and, secondly, in the disaster control regulations of the individual *Länder* and in the Precautionary Radiation Protection Act [1A-5]. These are supplemented by regulatory guidelines and by recommendations of the RSK and SSK.

On-site accident management

On-site accident management comprises all measures to be taken in a nuclear power plant in order to detect severe accidents in a reliable way and in time, to keep them under control and to bring them to an end with as little damage as possible. Most of the technical measures of on-site accident management were developed in the eighties, and - on the suggestion of the RSK after the accident at Chernobyl - were increasingly established in the operating nuclear power plants. The preventive measures shall avoid serious core damage. The main goal is to maintain or restore cooling of the reactor core and to convey the nuclear power plant to a safe condition. Mitigating measures, on the other hand, shall reduce serious radiation impacts on the plant site and in the environment. Here the main goal is the support of barriers retaining activity and to reach long-term controlled conditions of the plant, both with a view toward the protection of the environment. Essential to this entire endeavour is, firstly, the flexible utilisation of the safety and operating systems even beyond design usage and, furthermore, the utilisation of additional systems.

Extensive technical and administrative measures are taken in the German nuclear power plants in order to be able to perform effective on-site accident management should an event actually occur.

In the case of <u>pressurised water reactors</u> the preventive measures are:

– secondary side bleed and feed,
– primary side bleed (pressure reduction) and feed,

and the mitigating measures:

– assured containment isolation,

– primary side bleed,
– filtered pressure relief of the containment vessel,
– H_2 countermeasures,
– supply-air filtering for the main control room.

In the case of boiling water reactors the preventive measures are:
– an independent injection system,
– additional possibility for injection and refilling of the reactor pressure vessel,

and the mitigating measures:
– assured containment isolation,
– (diversified) pressure relief of the reactor pressure vessel,
– filtered pressure relief of the containment vessel,
– inertisation of the atmosphere of the containment vessel or of the pressure suppression pool air volume,
– supply-air filtering for the main control room.

The auxiliary measures supporting the preventive and mitigating measures in both reactor types comprise:
– emergency power supply from neighbouring plant unit (if applicable),
– sufficient capacity of the batteries,
– possibilities for a prompt restoration of the off-site power supply,
– an additional off-site power supply (underground cable),
– sampling system in the containment vessel.

The technical development of the H_2 countermeasures for pressurised water reactors are almost completed and the RSK has recommended installation of catalytic recombiners in these plants. The sampling system for the atmosphere in the reactor containment vessel is in the planning stage. All of the other on-site accident management measures have been realised in almost all nuclear power plants and, where not, are in the planning stage.

In accordance with Section 7 (2a) Atomic Energy Act, the measures for the prevention and mitigation of impacts of certain severe beyond-design-basis accidents must already be taken into account as early as the construction phase of future nuclear power plants.
(→ Chapter 7 (2i))

The administrative measures taken in all nuclear power plants to cope with emergencies include the institution of an emergency response team which will be supported by personnel from the operating staff. It should be possible for the emergency response team to be assembled and working within one hour. Appropriate working rooms as well as working and communication equipment are kept in readiness at all times. Pre-arranged contracts assure the assistance of external institutions such as the plant manufacturer and the Nuclear Emergency Service Company, a permanent organisation jointly installed by the licensees of German nuclear power plants to help in coping with emergencies and the removal of possible consequences. Corresponding alarm plans and organisational structures are specified in the operating manual. The descriptions of the individual technical measures to be applied are contained in a separate document, the accident management manual. Chapter 19 (iv) describes in detail how the transition from operating to accident management manual is handled.

Alarming in the transition from design basis accident to emergency

To have definite evaluation criteria in the transition from a design basis accident to a severe beyond-design-basis accident, the operating manual contains plant specific criteria and radiation limits regarding discharges with air and water which, if exceeded, lead to off-site information and may initiate corresponding countermeasures. The action is specified in the alarm procedure which is part of the operating manual. The off-site alarms concern, essentially, the supervisory authority and the disaster control authority which, in turn, will then forward the information and, if necessary, will initiate specific measurements and measures.

Disaster control

In the event of an emergency, the disaster control authority - depending on the *Land* and the site, more than one authority may be involved - installs a disaster control task force and, if necessary, a technical task force at the plant site [3-15 (1), 3-15 (2)]. The disaster control task force will be supported by a knowledgeable liaison person deployed by the licensee to help in the exchange of information and the co-ordination of measures taken. The supervisory authority will also deploy a liaison person. The disaster control authority is responsible for deciding whether to declare a disaster situation, the licensee for supplying the necessary information.

The *Länder* and federal authorities perform independent measurements during operating conditions and during an emergency. This supplements the precautionary measurement programmes performed by the licensee with respect to environmental monitoring, the extent and details of which are specified in the licence permit. Extent, type and frequency of the measurements by the authorities are closely related to the requirements of the individual situation. Decisive for the initiation of emergency measures are the emission and immission measurements in the vicinity of the nuclear power plants (→ Chapter 15). Depending on the individual situation, the remote monitoring system for nuclear power plants, KFÜ (→ Chapter 15), can help with further information on the condition of the plant.

The reference doses for the initiation of disaster control measures such as "staying indoors“, "taking iodine tablets“ or "evacuation“ (Table 16-1) as well as reference values for contamination (Table 16-2) are specified in regulatory guideline [3-15 (1)]. These reference doses follow the ICRP recommendation (ICRP Publication 40).

Table 16-1 Reference Doses for Initiating Disaster Control Measures

Measure	Whole body (external irradiation and inhalation) [mSv]		Thyroid (inhalation) [mSv]		Lung or any preferred irradiated individual organ (external irradiation and inhalation) [mSv]	
	lower value	upper value	lower value	upper value	lower value	upper value
Staying indoors	5	50	50	250	50	250
Taking iodine tablets	-	-	200 *)	1000 -	-	-
Evacuation	100	500	300	1500	300	1500

*) changes in planning:
in the age group 0-12 years, and for pregnant and breastfeeding women: > 50 mSv
in the age group 13-45 years > 250 mSv

Table 16-2 Reference Values for Initiating Measures in Case of Contamination of Skin or Clothing

Level	I	II	III	IV
Contamination ranges *) [kBq/cm^2]	0.4 - 4	4 - 40	40 - 400	> 400
Decontamination measures	to be considered	recommended	necessary	urgent need
Gamma dose rate in 1 m distance [µSv/h]	≤ 1	1 - 10	10 - 100	> 100

*) below a value of 0.4 kBq/cm^2 no decontamination measures are required

On the basis of the Precautionary Radiation Protection Act [1A-5] it is possible at a Federal level to specify dose and contamination values for the initiation of disaster control measures and to issue prohibitions and restrictions for the entire country regarding the use of foodstuffs and feedstuffs. Special behaviour patterns can be recommended to the general public in order to keep the radiation impacts as low as possible.

As mentioned already, national limits will always reflect the related regulations of the European Union which has specified maximum radiation limits that are binding for all member states. With regard to emergencies, European regulations under [1F-30] specify maximum limits for radioactivity in foodstuffs and feedstuffs. [1F-31] deals with conditions for export, and the various regulations under [1F-32] with conditions for import in the particular post-accident situation of Chernobyl.

In Germany, a nation-wide and site-independent surveillance of environmental radiation is performed by the Integrated Measurement and Information System (IMIS) already described in Chapter 15.

Training exercises

Severe accident management measures can only be performed effectively if the participating personnel and parties are properly qualified and prepared for this task. Therefore, corresponding training exercises are of particular importance. In accordance with [3-2] the basic and advanced training of the responsible shift personnel also involves auxiliary measures required in the case of unforeseen events.

In addition, the accident management measures of the licensee are practised regularly at the plant, in particular, in co-ordination with the on-site emergency response team. Likewise, the procedures of disaster control are practised by the disaster control authority itself. This will also include simulated accident situations in the plant performed together with the on-site emergency response team. The extent of these training exercises range from simple alarm drills, up to command post exercises, and on to complex scenarios involving the authorities and the plant manufacturer. In recent years realistic scenarios have increasingly been carried out using plant simulators. Training exercises involving the plant manufacturer are performed every three years. They are part of the contractual arrangements between the licensee and the manufacturer which also extend to the installation of emergency response centres at the manufacturer's premises with technical equipment as well as to other supporting measures.

In the case of nuclear installations close to the border, training exercises are also performed jointly with the neighbouring countries. Furthermore, staff of the BMU together with supporting organisations and the disaster control authorities of the *Länder* participate in regular training exercises of the European Union (ECURIE exercises) and of the OECD/NEA (INEX exercises).

16 (2) Emergency Preparedness, Informing the General Public and Neighbouring Countries

The requirements of the EURATOM directive regarding the information of the public in case of a radiation emergency [1F-29] have been incorporated into German regulations by an amendment of Section 38 Radiation Protection Ordinance. Accordingly, the general public must be informed at least every five year of the planned safety measures and of appropriate behaviour in such a situation. The most important facts to be included in this information are:

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

In the event of an emergency, the measurement data accumulated within of the above-mentioned surveillance programmes will be the basis for the reports required in accordance with the EU agreement on rapid information exchange with EURATOM [1F-28] and the corresponding agreement with IAEA [1E-5] which was signed by Germany in 1989, and in fulfilling any of the bilateral agreements. This ensures that Germany's neighbouring countries will receive timely information. The measurements routinely performed in accordance with [3-69] are also used for the reports to the EU in accordance with Article 36 of the EURATOM Treaty [1F-1].

Germany has signed bilateral agreements regarding mutual help in the case of an emergency with seven of its nine neighbouring countries . These are the Netherlands, Belgium, Luxembourg, France, Switzerland, Austria and Denmark. Additional agreements with the two remaining neighbouring countries, the Czech Republic and Poland, have already been signed. This has also led to a co-operation on a local level wherever sites are located close to the borders. In addition, corresponding assistance agreements have been made with Lithuania and the Russian Federation; similar agreements with Italy, Bulgaria and Hungary have already already signed.

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

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

not applicable to Germany

17 Siting

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 nuclear licensing procedure and, in addition, those criteria pertaining to the suitability of the site with respect to regional planning as well as to nature conservation and care for the countryside. With respect to nuclear safety the following points must be taken into account:

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

Siting and regional planning

After the applicant has pre-selected a site on the basis of the points listed above, a regional planning procedure is initiated which precedes the nuclear licensing procedure. These procedures are prescribed for all industrial facilities and take into account all impacts of the individual project on the public, on traffic ways, regional development, protection of the countryside and nature conservation.

Nuclear licensing procedure

In the nuclear licensing procedure (→ Chapter 7 (2ii)) the characteristics of the site are checked with regard to the site evaluation data specified in [3-12] and, furthermore, the design of the nuclear installation with regard to external impacts. In addition, investigations are carried out as to whether general public interests oppose the selection of the site. Within the nuclear licensing procedure the respective competent authorities analyse if the requirements regarding water utilisation, immission protection and nature conservation are met.

Design against external impacts

With respect to external impacts, safety criteria [3-1] require that all plant components necessary to safely shutdown the reactor, to remove residual heat or to prevent uncontrolled release of radioactive materials shall be designed to be able to perform their function even in the case of external impacts caused by nature or man-made. The following external impacts are considered:

– external impacts caused by nature such as earthquakes, flooding, extreme weather conditions, and
– man-made external impacts such as aircraft crash, impacts of dangerous and explosive materials, and

– disruptive actions or other interference by third parties.

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

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

All nuclear installations have been designed taking into account natural external impacts - wind and snow - and also floods and - where there is a risk of this kind - against earthquakes. In this context, both nuclear safety standards (flood protection [KTA 2207] and design for earthquakes [KTA 2201]) and conventional civil engineering standards were applied. Depending on the overall cooling concept for the nuclear installation, the system design results in requirements important to safety for the cooling water supply. It has to be verified for particular conditions on the site that the cooling water supply will function even under unfavourable conditions, e.g. low water in the receiving water or failure of a river barrage.

The design against earthquakes is based on a design basis earthquake (formerly called safety earthquake) in accordance with safety standard [KTA 2201.1]. This design basis earthquake has the largest intensity that, under consideration of scientific findings, could occur in a wider vicinity of the site (up to a radius of about 200 km). Depending on the site, the intensity of the design basis earthquake varies less than 6 and maximum 8 on the MSK scale. Recent probabilistic assessments assume that any stronger earthquake will occur with a frequency of less than 10^{-5} per year.

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

The requirements for protecting nuclear power plants against pressure waves from chemical reactions in the course of accidents outside of the plant arose in the seventies due to the specific situation of riverbank sites on rivers with corresponding ship traffic and transported explosive materials. The load assumptions specified in regulatory guideline [3-6] are based on an initial overpressure of 0.45 bar and, since its publication, have been applied independently of the individual site.

17 (ii) Evaluating of Impacts

With the impacts that an operating nuclear power plant has or could have on its environment and on the general public in its vicinity, a distinction is to be made between conventional impacts which would also emanate from other industrial facilities and radiation impacts both during operating conditions of the plant or 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, the fulfilment of these requirements is assessed explicitly on the basis of the Act on the Assessment of Environmental Impacts [1F-13] (→ Chapter 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 impacts during operation of a nuclear installation as low as possible. In this respect, the provisions of the Federal Immission Control Act [1B-3] must be observed together with its individual ordinances.

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

An individual licensing procedure 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.

Radiation impacts during operation and design basis accidents

The Radiation Protection Ordinance [1A-8] specifies limits that shall not be exceeded during operating conditions (normal operation and abnormal operating conditions) and during design basis accidents. These limits serve, specifically, to protect the general public in the vicinity of nuclear installations from the effects of ionising radiation.

The most important limits for the effective dose are:

– during operating conditions:
  0.3 mSv per year from a discharge with air or water,
– for protection against design basis accidents:
  50 mSv per event. This limit applies to the integral dose taking all exposure paths into account and a time period of 50 years.

Additional limits from the Radiation Protection Ordinance are listed in Table 15-1; the fulfilment of the corresponding requirements is checked within the licensing procedure (→ Chapter 15).

In accordance with [1A-8] it must be ensured that an uncontrolled release of radioactive materials is prevented, that the maximum permissible amount of radioactive discharge is specified such that the limits for radiation exposure are not exceeded, that the discharge of radioactivity is as low as practical and that the discharge is monitored and the amount specified according to type and radioactivity is reported to the supervisory authority (→ Chapter 15).

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

Chapter 17 (i) describes the current design of German nuclear power plants against external impacts.

Within the safety reviews extensive analyses and evaluations have been performed with regard to the actual protective condition of the nuclear installations. Also recent findings concerning safety and developments of the nuclear safety regulations have been included. The periodic safety reviews which are scheduled to be repeated every ten years (→ Chapter 14 (ii)) also include a re-evaluation of the protective measures of the particular nuclear installation with regard to external impacts. These take site-specific factors into consideration and the development in the state of the art regarding the effects and behaviour of civil structures and components under the assumed loads resulting from individual external impacts. As a result of the reviews of older plants, measures as far as necessary were initiated such that after their introduction there will be no essential difference to the safety level of the more modern German nuclear power plants.

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

<u>Floods</u>

In most cases, the single most protective measure against floods is the sufficiently high elevation of the site. In other cases, the civil structures important to safety were insulated for sufficient water tightness and were built with water impermeable concrete. Furthermore, the openings (e.g. doors) are located above the level of the highest expected flood. In individual cases mobile barriers have been provided to close off openings. The re-evaluations have indicated individual weak points which were improved by taking suitable remedial actions.

<u>Earthquakes</u>

All nuclear power plants are designed against earthquakes in accordance with the particular local seismic risk. On account of the large area that would be affected by an earthquake, the load assumptions specified in the licensing procedure for earthquakes are applied not only to the radioactivity retaining systems and structures of the nuclear installation but basically to all systems and structures important to safety including the control room. In the older nuclear installations, simplified (quasi-static) procedures were applied to the seismic analysis of civil structures, components and plant sections. These analyses delivered the basic values for the corresponding design specifications.

In more recent nuclear installations the newly developed dynamic analyses were also applied. The progressive development of methods in determining seismic load assumptions and of the verification of design specifications led to a re-evaluation of the older nuclear installations. This re-evaluation was performed on the basis of probabilistically determined intensity specifications correlated to site-specific ground response spectra, and on the basis of recent results from scientific experiments such as the shaker tests. These were performed at the shutdown superheated steam reactor (HDR) at Karlstein and led to a deeper understanding of the seismic response of civil structures and plant components.

An essential result of this re-evaluation is that the seismological situation at the sites of German nuclear power plants does not require a more unfavourable rating than at the time of construction of the older plants. Furthermore, these re-evaluations that were performed on the basis of more precise seismic parameters revealed that the technological components and equipment of the plants have considerable margins with respect to seismic loading. Only in individual cases did the re-evaluation lead to the necessity for backfitting of individual components. These measures have since been carried out.

Aircraft crashes

In more recent nuclear installations, the design for protection of civil structures and components against aircraft crash was performed using the load assumptions described in Chapter 17 (i). Aside from the reactor building all those buildings were designed accordingly that contain systems required for the control of this external impact (e.g. the emergency feed-water building in more recent plants with pressurised water reactors). Furthermore, protective measures were taken to counter the pressure waves from an aircraft crash, e.g. by uncoupling the ceilings and inner walls from the outer wall, so that no vibrations would be induced in the contained components and internals. The older nuclear installations were re-evaluated with regard to the transfer of the respective loads in conjunction with the probabilistic safety assessments. The results from these assessments showed that even if a reactor building would not withstand the specified load assumptions there is still a sufficiently low risk from aircraft crashes, particularly, due to other buildings in the flight path of the aircraft that would protect the reactor building. A further risk reduction was achieved by backfitting the older plants with physically separated auxiliary emergency systems that are completely independent from other systems (→ Chapter 14 (ii)). All in all, the risk contribution from aircraft crash is seen as being negligibly small.

Pressure waves from explosion

The more recent nuclear power plants are designed against pressure waves from external explosion in accordance with the load assumptions described in Chapter 17 (i). Furthermore, with respect to possibly larger peak pressures at the accident location itself, a sufficient safety distance is kept from potential sources of explosions (e.g. traffic routes, industrial complexes). 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 on account of the site conditions, corresponding analyses where performed in the course of the safety reviews. The results show that in almost every case the actual structural design will withstand the specified assumed loads. In every case, however, the nuclear installations are sufficiently protected under general risk aspects. The certifications required in the licensing procedures for industrial complexes ensures that new industries settling in the vicinity of nuclear power plants will not entail any unconsidered events that could endanger the nuclear power plant.

17 (iv) Consultations with Neighbouring Countries

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

– taking the interests of the neighbouring country into consideration when selecting the site,
– making licensing documents available,
– the area of obligatory mutual information, and

– the framework of regular 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 the field of radiation protection,
– regulatory development of the safety requirements especially with regard to accident management measures in the case of severe accidents.

The legal obligation in Europe for a cross-border participation of the competent authorities [1F-13] was transposed into German law by a corresponding amendment of the Nuclear Licensing Procedure Ordinance [1A-10]. Accordingly, the competent authorities of the neighbouring country participate in the licensing procedure if a project could considerably affect the other country.

In accordance with Article 37 of the EURATOM Treaty, the European Commission will be informed of any plan for discharging radioactive materials of any sort. For this purpose, general information on the planned discharge, on the site and the essential characteristics of the nuclear installation are reported to the Commission six months before the competent authority issues a licence permit for the discharge in question. This serves to establish the possible impacts on the other member countries [see also 1F-12]. 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 the nuclear installations in border regions will have on the safety of their own country. Chapter 16 (2) has already dealt with the joint agreements with neighbouring countries regarding information exchange and mutual assistance in the case of emergencies, and with the further agreements entered into with other countries, the IAEA and the EU.

18 Design and Construction

18 (i) Safety Concept

Protection goals

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

In order to ensure this protection the radioactive materials - essentially concentrated in the reactor core - are retained by a number of barriers. These are the fuel-rod cladding, the pressure retaining boundary of the reactor coolant and the containment vessel. To ensure integrity of these barriers the following preceding protection goals must be observed at all times: the reactivity of the reactor core must be kept within close limits, and the fuel elements must always be sufficiently cooled.

All in all, four protection goals apply to nuclear power plants that must be observed at all times in order to assure the protection required by the Atomic Energy Act:

- control of reactivity,
- cooling of fuel elements,
- confinement of radioactive materials and
- limitation of radiation exposure.

To achieve these protection goals, the following basic requirements must be met:

<u>Control of reactivity</u>

- Reactivity changes stay within permissible values,
- it is possible to safely shutdown the reactor core and keep it in a subcritical condition on the long term,
- the fuel elements always stay subcritical during handling both in the storage area for fresh fuel elements as well as in the spent fuel pool.

<u>Cooling of fuel elements</u>

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

<u>Confinement of radioactive materials</u>

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

Limitation of radiation exposure

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

Defence-in-depth concept

The protection goals and, in turn, the integrity of the barriers are ensured by applying a concept of progressive safety at several levels of safety (the "defence-in-depth concept“). This concept comprises a well balanced combination of measures for the prevention of abnormal occurrences and design basis accidents, of measures for their control and, finally, of measures for the mitigation of the consequences from any severe accident that might occur.

At the first level of safety - the operating level - the high quality of design and manufacturing and careful operational management ensure a high availability of the nuclear installation; at the same time, this helps to prevent abnormal occurrences.

The following basic principles are observed:

- sufficient safety margins in the design of systems and plant components,
- careful choice of materials, comprehensive materials testing,
- comprehensive quality assurance during manufacturing, construction and operation,
- independent verification of the quality achieved,
- quality surveillance by inservice inspections corresponding with the operational loads,
- ease of maintenance of the systems taking the possible radiation exposure of the personnel into account,
- reliable monitoring of the operating conditions,
- taking operating experience into account,
- comprehensive training of the operating personnel, and
- prevention of inadvertent operation, e.g. by interlocks.

At the second safety level, any abnormal occurrence is limited in its effects by inherent safety characteristics and by active systems in such a way that the nuclear power plant stays within the design limits of operating conditions. The following preventive measures belong to this category:

- the reactor core is designed such that even in case of failure of control equipment, the temperature and pressure in the core will reach stable conditions without need for manual actions,
- the control room is equipped with status indicators and failure alarms to inform the operating personnel and to enable any required manual actions,
- control and limitation equipment will keep the nuclear power plant within permissible design limits, and prevent abnormal occurrences from developing into accident situations.

At the third level of safety the safety equipment will keep accidents under control in such a way that the radiation exposure of the public and environment stays below the specified limits. The individual accidents for the design of the safety equipment (the design basis accidents) are chosen to be representative for a number of similar accidents. It is subject to the licensing procedure to prove for the most affected persons that the radiologically representative design basis accidents will not cause a radiation exposure exceeding the maximum limits specified in Section 28 (3) Radiation Protection Ordinance (→ Chapter 15).

The design of the safety equipment is based on the following principles:

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

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

Safety Level			**Measures**	**Objectives**
1	operating conditions	normal operation	quality of the operating systems and procedures as well as safety consciousness at work	prevention of abnormal occurrences
2		abnormal operation	inherently safe plant behaviour; limitation systems	prevention of design basis accidents
3	design basis accidents		inherently safe plant behaviour; passive and active safety equipment	control of design basis accidents
4	(beyond-design-basis) severe accidents	specific very rare events	specific precautionary measures	control of specific very rare events
		severe beyond-design-basis conditions / emergencies	on-site accident management measures	prevention of core damage and, if this is not possible, limitation of the impacts to the environment

At the fourth level of safety the defence-in-depth concept is extended even further by taking precautionary measures against events which, due to their low probability of occurrence, are not part of the design basis. These comprise specific measures to take account of very rare events such as aircraft crashes, external pressure waves and ATWS (anticipated transients without scram) and, since the eighties, on-site severe accident management measures. Even in the event of a highly improbable beyond-design-basis accident, the main objective is to avoid serious core damage and, if this will no longer be possible, to limit the radiation impacts to the environment of the nuclear power plant. In this case, the systems and components of the nuclear power plant will possibly be operated beyond design specifications; the negative effects on the regular function or even a damaging of these systems and components will be accepted to achieve the superior objectives in such an extreme situation. These possibilities have been supplemented by on-site accident management measures listed in Chapter 16 (1).

This defence-in-depth concept with its four levels of safety, as shown in Table 18-1, has been realised in all German nuclear power plants.

Of course, the original equipment important to safety in the individual nuclear power plants was different due to the stages of development at the time of licensing. However, specific, and in a number of cases, quite extensive backfitting was carried out in the course of their operating life (→ Chapter 14 (ii)). So the safety level of the nuclear power plants were adjusted to the state of the art in safety relevant findings.

The evaluation of the safety of current nuclear power plants within the periodic safety reviews is carried out on the basis of the design basis accidents (Table 18-2 for PWR and Table 18-3 for BWR). Furthermore, a spectrum of extremely improbable beyond-design-basis conditions are covered (likewise in Tables 18-2 and 18-3). In view of the special equipment and measures available in the nuclear power plants for these events, practically no radiation impacts that would endanger the general public may occur.

Table 18-2 Design Basis Accidents and Severe (Beyond-design-basis) Accidents to be Considered in a Periodic Safety Review of PWR

Level 3, design basis accidents
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, severe (beyond-design-basis) accidents	
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	

Table 18-3 Design Basis Accidents and Severe (Beyond-design-basis) Accidents to be Considered in a Periodic Safety Review of BWR

Level 3, design basis accidents
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 cm^2 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
– 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, severe (beyond-design-basis) accidents
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

18 (ii) Qualification and Proof of Incorporated Technologies

The requirements with respect to qualification and proof of the incorporated technologies and to the reliability of the structures, components and systems important to safety are in accordance with the principles of the defence-in-depth concept. In a general form, these requirements are specified in the Safety Criteria [3-1]. In detail, the requirements are derived from safety analyses. Details regarding the technical realisation are specified in the nuclear safety regulations. The corresponding KTA safety standards are listed in Appendix 2, in particular the series 1400, 3200, 3400, 3500, 3700 and 3900. In these standards reference is always made to the employment of proven technologies.

With respect to proven technologies, the passive and active equipment must meet the following requirements:

Passive equipment

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

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. radiation, this is accounted for in the corresponding requirements regarding materials and qualification certifications. In the case of components with barrier functions, the influence of quality reducing factors during manufacturing is examined on the basis of conservative assumptions.

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. reactor coolant pressure boundary, secondary systems, containment vessel, lifting equipment. The qualification proof of a manufacturing process is carried out for each manufacturer individually and is repeated at specified time intervals. An independent authorised expert will participate in certain manufacturing steps that are important with respect to the qualification of the materials, the manufacturing process and the components. The results of tests are documented and the evaluations of the authorised experts will be submitted to the licensing authority.

Active equipment

The majority of components and their operating hardware are series-produced items for which extensive industrial experience is available. This applies in particular to the instrumentation and control equipment, such as electric motors, controller drives, switch gear, 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 used in the secondary coolant and auxiliary systems and within the range of the turbine, however, not those used within the reactor coolant pressure boundary. These items are deployed in large numbers 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 so than in the case of passive components (→ Chapter 13). Type and extent of the qualification proof are specified both in nuclear and in conventional standards, in accordance with the individual safety significance. Wherever specific nuclear influences are expected, e.g. with respect to 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 for an evaluation by the licensing authority.

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

Proof of qualification

The qualification of the installed techniques are proven in various ways. These are:

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

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

The test programmes are submitted to the licensing and supervisory authority and are checked by the authorised expert. 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 checks all aspects subject to the licensing and supervisory procedure with regard as to whether additional requirements are necessary beyond those specified in applicable standards and guidelines.

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

Experience feedback from individual failures or from general technical findings has shown in a number of cases that certain technical equipment is or would seem to be ill-suited to long-term operation. It is part of the safety culture in Germany, and has proven very effective, that all parties 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 replacements of pipes in the main steam and feed-water systems of boiling water reactors both inside and outside of the containment vessel, or the backfitting of diverse pilot valves in the overpressure protection system of boiling water reactors. Other examples are the conversion of all pressurised water reactors to a high-AVT-operation with respect to the secondary-loop water chemistry, or the fabrication of weld seams for better testability with ultrasonic procedures either by machining the weld surfaces or by re-welding the seams on components and pipes in pressurised and boiling water reactors. Furthermore, the instrumentation needed for a more exact determination of local loading, e.g. due to thermal stratifications and fluctuations, was increased in all nuclear installations. The results from these measurements are used both for optimising operating procedures as well as in ageing assessments for a more reliable determination of the utilisation factor of components.

18 (iii) Design for Reliable and Easily Manageable Operation

General requirements with the objective of ensuring a reliable and sound operation of the nuclear power plant are already specified in the Safety Criteria [3-1, 3-51]. These pertain to simplicity of system design, physical separation, as well as to the accessibility for inspections, maintenance and repairs. They also contain further requirements regarding workplace design, work procedure and working environment. Detailed requirements both with regard to technical measures and to the administrative procedures of work tasks are specified in safety standards [4-1.1, KTA safety standard series 3200 and 1200].

The implementation of these requirements led to design and construction characterised by the following features:

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

In the late seventies the concept of basic safety was developed. In addition to the general requirements mentioned above for all barriers, basic safety is applied to the reactor coolant pressure boundary and other pressure retaining components. It comprises a catalogue of detailed technical requirements with the special objective of preventing catastrophic failure of plant components due to manufacturing defects. 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 inspection techniques,
– awareness of any possible fault conditions and their evaluation,
– accounting for the operating medium.

These principles were immediately applied in the newer nuclear power plants and have led to post-qualifications in the older plants. The effects of basic safety are seen in the design characteristics of the reactor coolant pressure boundary of the different design generations of PWRs and construction lines of BWRs (→ Appendix 1).

The development of the materials for the containment vessel of pressurised water reactors led from the different high strength steels used in the first through the third design generation of PWRs to the present optimised steel type characterised by a lower strength but a higher fracture toughness and better workability (15 MnNi 6 3) (→ Appendix 1).

The following facts characterise the positive results achieved with respect to a reliable and sound operation; they are derived from operating experience:

– The barrier ‘fuel element cladding’ is very reliable. The number of defects that have led to leakages average out to between 1 and 2 per year and plant.
– The barrier ‘reactor coolant pressure boundary’ is very reliable. The frequency of occurrences of minute leakages average out to 10^{-1} per year and plant. In the last twenty years not a single leakage occurred at this barrier in German nuclear power plants that would have led to an actuation of the safety system. In all cases where general indications for cracks due to reduced quality and to operating influences were found, the

respective components were repaired or exchanged. The countermeasures applied were effective as is seen by the continuously low number of damage reports on pipe failures in the nuclear auxiliary systems and the reactor coolant pressure boundary out to the primary isolation valve. Figure 18-1 shows all reports on damage that has occurred in the systems for nuclear power production and in the nuclear auxiliary systems. These reports comprise both the through-wall cracks with leakages as well as the detected incipient cracks without leakages.

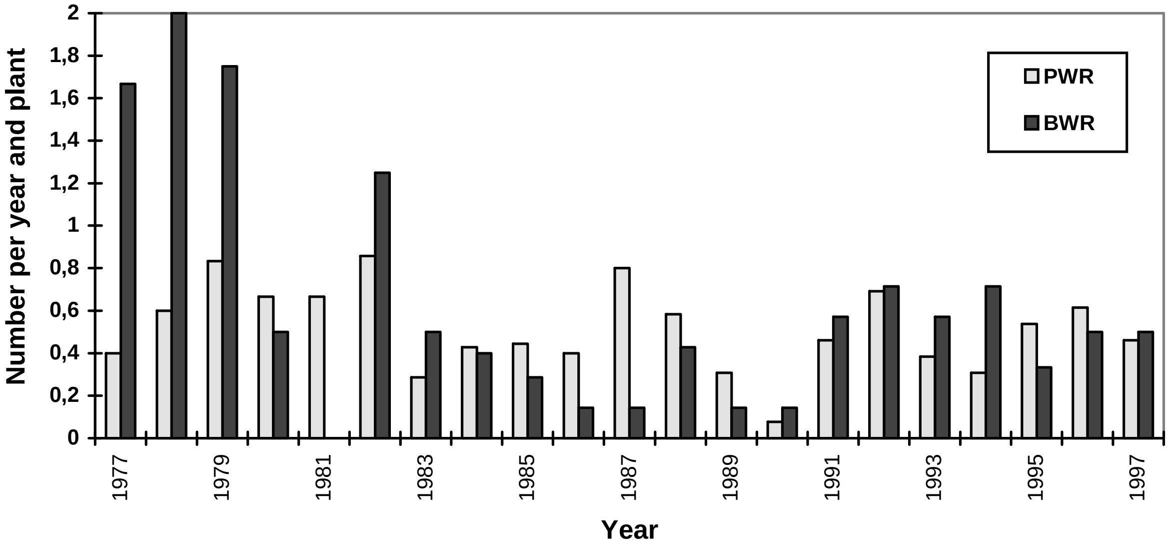


Figure 18-1 Reported Pipe Damages in Nuclear Auxiliary Systems and the Primary Coolant System up to the First Isolation Valve

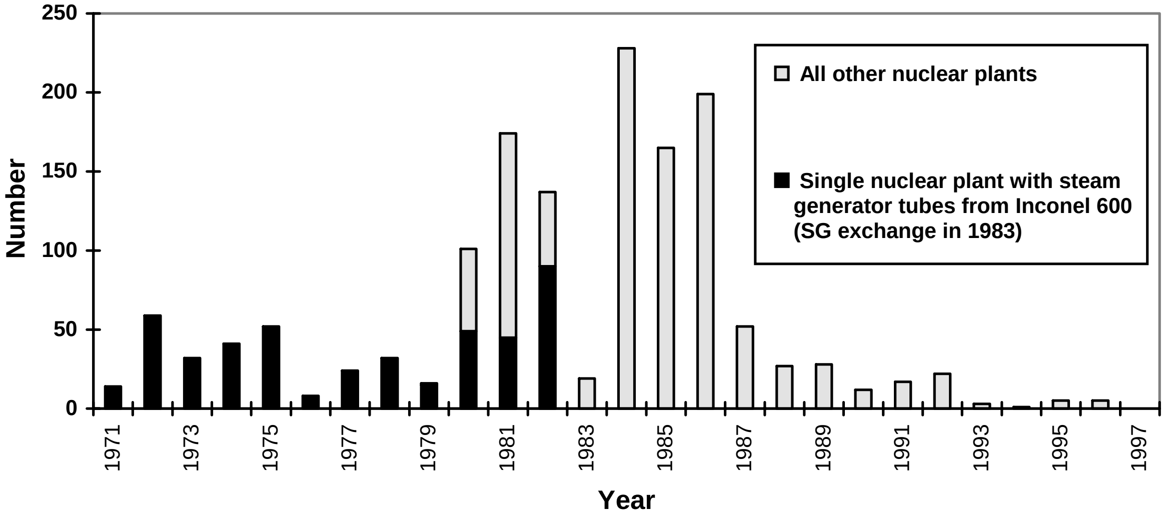


Figure 18-2 Number of Steam Generator Tubes Annually Plugged in PWR

- The barrier ‘containment vessel’ is very reliable. The requirements regarding leaktightness have been certified in corresponding tests. Functional restrictions occurred only in isolated cases, e.g. in case of rupture of an instrumentation pipe without stop valve.
- The frequency of leakages between the reactor coolant pressure boundary and the connected systems is very low. The measures taken for an optimisation of water chemistry, which were finalised in 1987, have been particularly effective in pressurised water reactors together with the materials being employed in the steam generator tubes that are insensitive to stress-corrosion cracking (Figure 18-2). Ever since, the number of steam generator tubes that have to be plugged due to a wall thickness reduction has been reduced to only a few tubes per year, all pressurised water reactors taken together.
- The inservice inspections (→ Chapter 14 (ii) and 19 (iii)) show that the safety system is very reliable. Although functional failures were observed, the defence-in-depth concept has never been put in question.
- All of the deterministic assumptions for the accident analyses have been taken into account. These are, in particular, the assumption of two physically different initiation criteria, the assumption of sequential failures simultaneous with the initiating event, the assumption of a single failure together with a failure from non-availability due to maintenance. This procedure has been very effective as is also shown by the high time availability of the German nuclear power plants and by the low number of defects detected by inservice inspections.

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

19 Operation

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

The construction and commissioning of a nuclear power plant is usually performed by the manufacturer as a general contractor. The turnkey plant is turned over to the operating utility by the manufacturer after a successful commissioning operation. The responsibility for commissioning stays with the commissioning management of the manufacturer as licensee until his official handover to the operating utility as future licensee. The personnel required for commissioning is supplied by the manufacturer. It has to demonstrate the required qualification [3-2]. The personnel of the future licensee of the nuclear installation 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 was based, firstly, on the results of a safety analysis and its detailed evaluation by an authorised expert organisation, carried out on behalf of the competent authority (→ Chapter 14 (i)), secondly, on accompanying inspections during erection and, thirdly, on the results of a comprehensive commissioning programme, subject to approval by the competent authority. Special emphasis is put throughout on verifying that all applicable safety requirements specified in nuclear safety regulations are fulfilled at the time the permit for initial operation is granted. It is generally ascertained that the nuclear installation in its as-built condition meets all applicable design and safety requirements at this time.

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

Safety analysis

The deterministic safety analysis covers accident analysis and systems analysis. In the course of time, probabilistic assessments have been increasingly included in the safety analyses, e.g. reliability analyses of the reactor safety system and of the emergency core cooling systems.

In an accident analysis, the behaviour of the nuclear power plant is analysed using extensive computer simulations for all operating transients and design basis accidents to be assumed in accordance with the nuclear safety regulations. The analytical methods take into account all important physical aspects. As far as possible the analytical models have been verified by experiments. The results being conservative is ensured by using unfavourable assumptions and conditions as input data.

The systems analysis is performed to verify that the available systems for operation and surveillance are designed so that accidents due to abnormal operation are prevented with a high reliability. Furthermore, these analyses must prove that reliable technical safety equipment is available for the control of design basis accidents. In particular, the systems analysis is aimed at ensuring that the safety equipment meets the following design principles in accordance with [3-1]:

– redundancy; diversity; largely unmeshed partial systems; physical separation of redundant partial systems;
– fail-safe behaviour in the event of malfunction of partial systems or plant components.

The safety analysis is submitted to the competent authority and is subject to a detailed evaluation by the authorised experts. In his evaluation the authorised expert, to a considerable extent, uses independently developed computer codes or generally accepted alternate analytical methods.

Accompanying inspection during manufacturing and erection

The accompanying inspections during the entire manufacturing process ensure that the actual design of the systems and components important to safety meets the requisite requirements. The accompanying inspection is split up 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 materials testing and construction tests, pressure tests and assembly 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,
- warm test operation phase 1,
- warm test operation phase 2,
- zero power and partial load 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 warm test operation 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. Warm test operation phase 2 is performed after initial fuel loading of the reactor. It covers those commissioning activities that it is not possible or sensible to perform before the core is loaded. Its objective is to verify the functionality and the safety of the plant as a whole before starting nuclear operation. The final step of commissioning begins after first criticality and covers comprehensive tests at zero and partial power levels. The levels are chosen to be most suitable for the technical or physical verification of satisfactory functioning.

19 (ii) Operational Limits and Conditions for Safe Operation

All data, limits and measures important with respect to safe operation and to the controlling of design basis accidents are collated as the so-called safety specifications in accordance with the Nuclear Licensing Procedure Ordinance [1A-10] and with the Regulatory Guideline on the Requirements for Safety Specifications of Nuclear Power Plants [3-4]. The safety specifications enable a quick and comprehensive survey of all limits, conditions and measures that determine the safety of the nuclear installation. It is a constituent part of the operating manual.

Part 2 of the operating manual contains the following chapters that are part of the safety specifications:

- prerequisites and provisions for operation including the permissible unavailability and repair times,
- safety system settings,
- specified actions with respect to abnormal operation (e.g., load rejection to auxiliary station supply, turbine trip, failure of a primary coolant pump),
- reporting procedure and criteria for reportable events.

As early as when he applies for a licence to erect a nuclear power plant, an applicant submits for evaluation the limits and conditions for safe operation in the form of the safety specifications. The prescribed limits and conditions for safe operation must be met at all times. Any modifications of the prescribed limits and conditions require approval by the licensing authority.

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

Limits for safe operation

The limits for safe operation comprise the following:

- limits for an automatic actuation of the reactor protection system including the overpressure protection of the steam generator in pressurised water reactors,
- limits for the limitation systems activation preceding the reactor protection system,
- limits for discharge of radioactivity.

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

- values on the effectiveness of a reactor scram, of the residual heat removal, of the overpressure protections, and of the radioactivity confinement,
- values on the conditions inside the containment vessel, inside the reactor building and the turbine building,
- measured values of emission monitoring,
- measured values for the assessment of design limits of the activity barriers in the event of severe accidents,
- alarms important to safety and short descriptions of the actions to be taken in the event of, e.g., 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.

Conditions for safe operation

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

– 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 period of documents, procedures for technical modifications to the plant and for changes to operating procedures, as well as conditions regarding the discharge of radioactive materials with exhaust air or waste water;
– prerequisites and conditions for start-up, power operation, plant shutdown and refuelling;
– conditions for maintenance during power operation including the specification of permissible maximum unavailability times of safety equipment; provisions for preventive maintenance.

The specifications concerning the extent of limits and conditions for safe operation and their compilation in marked sections of the operating manual have turned out well.

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

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

– The plant manager or his deputy is responsible for safe operation. In the event of their absence, this responsibility is transferred to the shift supervisor on duty. Only the plant manager - or the shift supervisor on duty - has the right to take decisions regarding the access to protected areas, the release of work orders, the procedures in the course of accidents and for imminent danger.
– The functional tasks of the managing personnel are clearly and completely specified, so that concurrent instructions of more than one person are avoided.
– To avoid any conflict of interests, the organisational units for quality assurance and for surveillance are independent of the division responsible for operation.

The organisational procedures required for a safe and licence-conform operation of the plant are laid down in written instructions in the operating manual and the testing manual. Their essential contents are outlined below.

Operating manual

Structure and contents of the operating manual of a nuclear power plant follow 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. In particular, detailed instructions for the shift personnel with additional information regarding the particular plant conditions involved are laid down. All parts of the operating manual that belong to the safety specifications are marked. 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. They contain precise and complete instructions approved by the authorities.

– Plant Operation
This part contains the prerequisites and conditions for operation and the safety system settings (→ Chapter 19 (ii)), the criteria for the reporting of events to the supervisory authority and detailed instructions for normal and abnormal operation of the plant. All cases of abnormal operation treated in the licensing procedure are presented both in a short form (strategy paper) and in a long form. The short form contains the detection criteria, automatic measures, required manual actions and the plant conditions to be achieved, as well as the plant parameters to be monitored in particular. The long form contains the sequential order of actions in the form of step programmes.
– Design basis accidents
The procedures required for the control of incidents are presented in the same way. The procedures regarding accidents are treated in Chapter 19 (iv).
– Systems Operation
This part covers the actions to be taken by the shift personnel for all systems as step programmes.
– Alarms
This is a complete listing by systems of all alarm signals from failures or dangerous conditions together with corresponding counteractions and possible alternatives.

Testing manual

Structure and contents of the testing manual of a nuclear power plant follow KTA safety standard [KTA 1202]. The testing manual comprises general instructions, the testing schedule and corresponding testing instructions for all inservice 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 with all inservice inspections important to safety is clearly arranged . It covers the test object, extent of test, test interval, required plant conditions under which the test is performed and a clear notation testing instruction. The testing schedule is part of the safety specifications.

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

Specifying procedures and intervals for tests, inspections and maintenance

The procedure employed by the licensee to verify that those systems important to the safety of the plant are in functioning order and that the corresponding quality characteristics remain within permissible limits, were described in Chapter 14 (ii). The individual functions important to safety and all systems and components important to safety are identified on the basis of this procedure - initially during construction and recurrently with every plant modification. In addition, the required qualification proofs, the inservice inspections, the preventive maintenance measures as well as the permissible operating procedures for systems and components are reviewed according to their safety relevance. The regulatory guideline on maintenance [3-41] is also taken into account. All this considered, the following maintenance and testing measures are performed during operation:

- inservice inspections in accordance with the testing manual; basically, the principle of overlapping partial tests is applied (e.g. the actuation of the safety valves is tested at intervals different to those of the actual function of the valves),
- preventive maintenance regularly scheduled and performed under the independent responsibility of licensee,
- functional tests of systems and components following maintenance,
- periodic evaluation of the documentation from operation and testing,
- feedback of operating experience to operational practice.

Since the nuclear power plants were built, the verification methods on which these procedures are based have been developed against the background of operating experience and of findings from safety research. Deterministic verifications were dominant at the time of the construction of the plants (1964 to 1989). Thus, the classification of systems important to safety, components and other plant equipment as well as the specification of the extent 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 inservice 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 inservice inspection concept a number of shortcomings caused by inaccessibility, technical restrictions, or an insufficient representativeness of the tests with regard to the conditions of required operation of the component were revealed, which have been overcome to a large extent by appropriate modifications of the components, of the testing techniques, or of the testing procedures. With respect to the component reliability achieved, see Chapter 18 (iii).

In recent years, probabilistic safety assessments are increasingly used to supplement the (deterministic) engineering judgements. In the past in individual cases, the results from the deterministic approach were already checked by probabilistic means, e.g. with regard to the reactor protection system and the emergency core cooling systems. Probabilistic methods are used in determining the balance within the plant concept, and in evaluating the interaction of plant technology, plant operation and tests and with regard to the defence-in-depth concept (→ Chapter 18 (i)). If the results so indicate, corrections and optimisations of the operating instructions, operating procedures, of tests and system technology will be carried out. However, any modification will still be decided upon on the individual case, taking into account all plant-specific circumstances.

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

Although abnormal occurrences during normal operation will cause operational restrictions (e.g. reduction of reactor power in case of a failure of one reactor coolant pump) there will be no safety reasons to discontinue operation. In the case of accidents, on the other hand, plant operation must be discontinued for safety reasons. Despite the similarity of basic procedures for controlling abnormal occurrences and accidents, the individually realised protective measures are different. Detailed procedural instructions are specified for the shift personnel covering the individual operating procedures for each of the abnormal occurrences or design basis accidents dealt with in the licensing procedure.

The procedures for control of design basis accidents are a combination of protection goal oriented (or symptom based) and event oriented procedures. The protection goal oriented procedures supplement the event oriented procedures, the latter being summarised in typical groups of accidents (e.g. loss-of-coolant accidents, failure of heat removal without loss of coolant, external impacts). The procedures for control of abnormal occurrences and design basis accidents are based on the following types of written instructions and visual aids:

– accident sequence diagram,
– accident decision tree,
– control of protection goal criteria,
– protection goal oriented handling of accidents,
– event oriented handling of accidents.

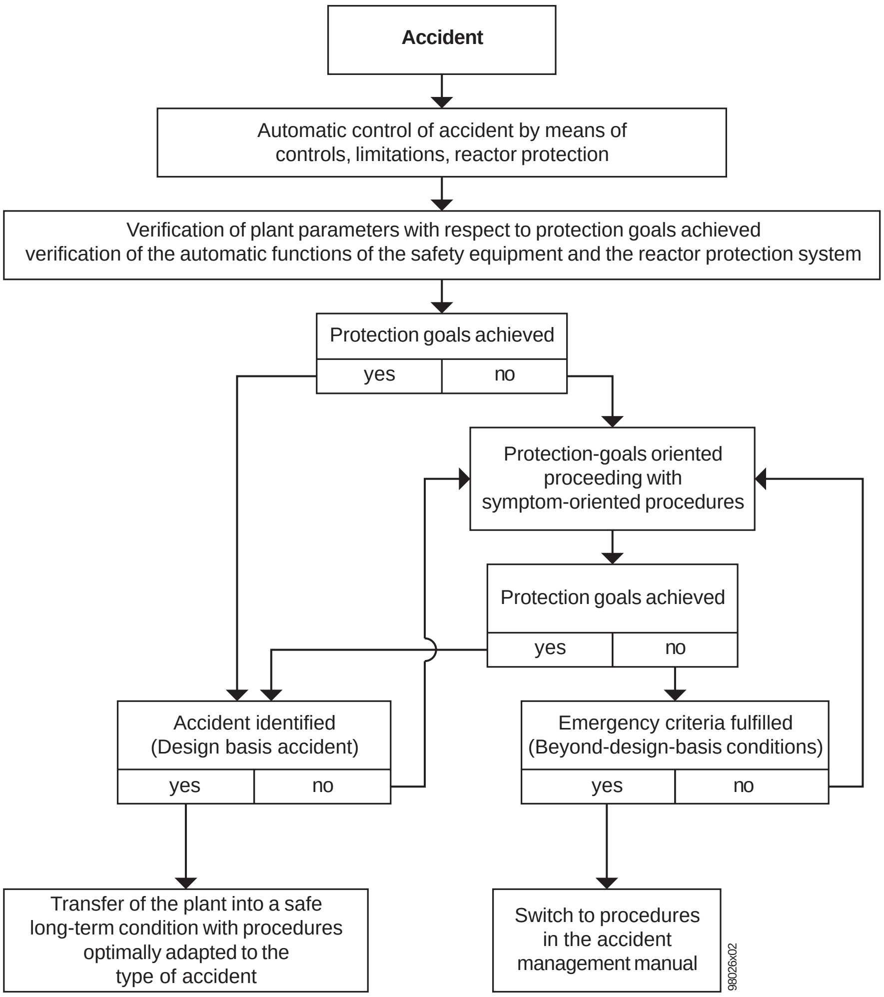


Figure 19-1 Typical Example of an Accident Sequence Diagram

In the case of an abnormal event that may lead to a reactor scram, the accident sequence diagram (Figure 19-1) serves as first orientation for the shift personnel with respect to further proceeding. As a next step, the shift personnel should control the protection goal criteria to determine whether or not

– control of reactivity (subcriticality),
– cooling of fuel elements (cooling medium inventory, heat transport, pressurisation of the primary side, heat sink, and steam generator feed of the secondary side),
– confinement of radioactive materials (in particular, integrity of the reactor containment vessel)

have been achieved and thus effectively limit any radioactive discharge to the vicinity of the installation. In case, a violation of a protection goal criterion is detected then the protection goal oriented procedures are used to bring the plant parameters back into their normal range. If a violation of protection goal criteria is detected and the reason may be assigned to a specific 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 procedures. The transition from design basis accident procedures to accident management procedures is described in the section "Protection Goal Oriented Procedures“ of the operating manual. The on-site severe accident management measures are mentioned in Chapter 16 (1).

Protection goal oriented procedures

The protection goal oriented procedures do not require the identification of the actual event but are rather guided by the observable condition (symptom) of the plant. The operating manual lists the corresponding plant parameters for every protection goal. These are used to ascertain that the required protection goal criteria have been achieved. If the recovery of the plant condition fails then a transition to the accident decision tree is performed in accordance with specified criteria. The accident decision tree identifies the corresponding on-site accident management measures. These are dealt with in the accident management manual (→ Chapter 16 (1)).

Each of the descriptions of the protection goal oriented procedures is structured as follows:

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

Event oriented procedures

If none of the protection goals are endangered or if the application of protection goal oriented procedures has successfully brought the plant parameters back into the normal range, then the accident decision tree (Figure 19-2) is used to appoint the type of accident. The protection goal criteria are continuously monitored in parallel. Detecting that one of the protection goal criteria failed, the event oriented procedures will immediately be interrupted to return to the protection goal oriented procedures in order to bring the plant parameters concerned back into the normal range. Likewise, if it turns out to be impossible to identify the type of accident involved, the protection goal oriented procedures are used to bring the plant into a long-term safe condition. Whenever the accident type can be identified by the accident decision tree the optimally suited event oriented procedures will be used to bring the plant into its long-term safe condition.

Every event oriented procedure has two major parts, the short form (strategy paper) and the so-called long form. The strategy paper is part of the safety specifications and contains, in particular,

– the criteria for determining the plant condition or for identifying the event,
– the sequence of automatic procedures including a description of the corresponding plant dynamics (operational automatisms, limitations and actions of the reactor protection system),
– protection goals of the plant that might be endangered,
– the long-term plant condition to be reached in the end,
– the measures for achieving this plant condition,
– important plant parameters.

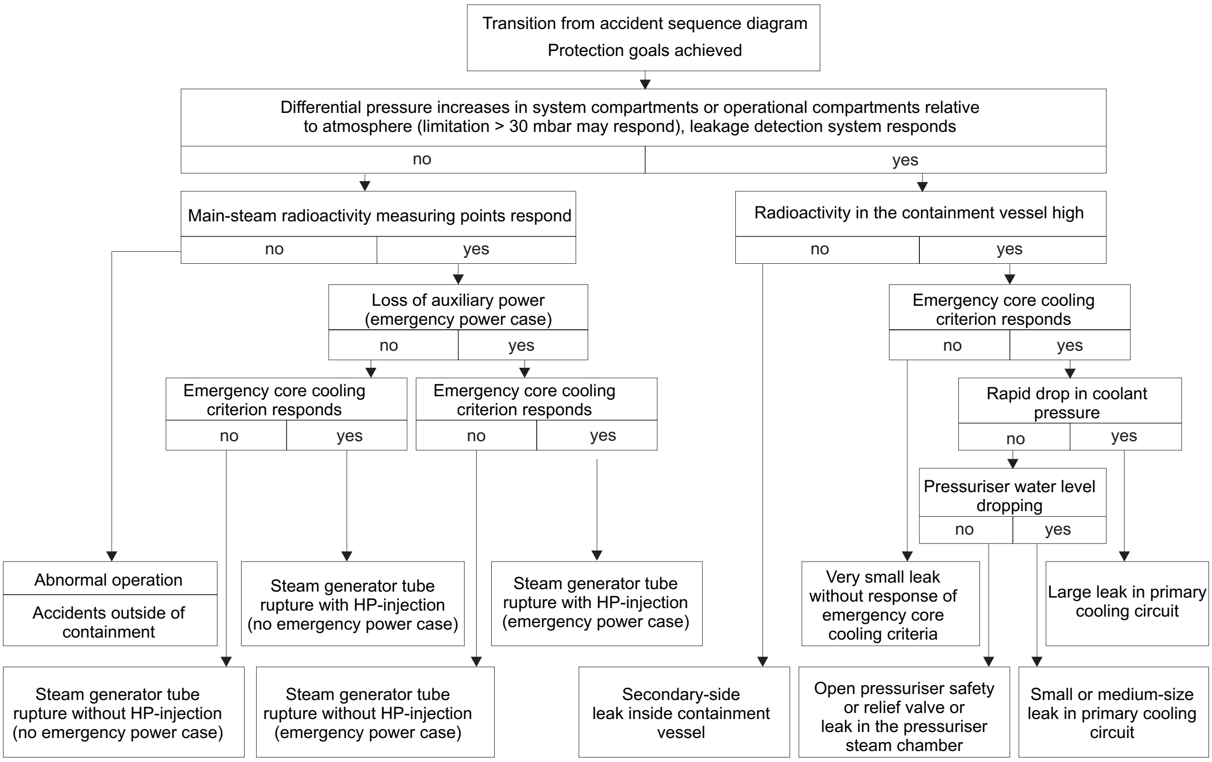


Figure 19-2 Typical Accident Decision Tree (PWR, in Principle)

The strategy paper is used by the shift supervisor as a decision making tool during the course of the accident. All essential displays, alarms and activities associated with controlling the individual event are specified in the form of step programmes. These step programmes quote the plant identification code of the individual component and, depending on the particular situation, indicate possible alternative actions to be taken, and they draw attention to characteristic details. In case of partial steps, the displays, limits and the expected tendencies of these values are quoted. Appendices to these procedures contain background information which is suited to increase basic understanding of the event sequence and its controls. Possible alternative event sequences are accounted for by an appropriate branch-off within the step programmes or by splitting it up into two or more procedures.

A prerequisite for the application of the event oriented procedures is that the individual criteria of the protection goals are met and that the type of event can be unambiguously identified. The fulfilment of this prerequisite is ensured by applying the accident sequence diagram (Figure 19-1) and the accident decision tree (Figure 19-2) , both of which are part of the safety specifications of the nuclear power plant.

The long form of the event oriented procedures is the working basis for the reactor operator and other plant operators in their handling of the accident situation. Its outline corresponds to that of the strategy paper. The long form contains the detailed step programmes of the event sequence showing every operational step independent of whether it is automatically initiated or based on manual actions. Pertinent chapters of the operating manual are incorporated by cross references.

19 (v) Engineering and Technical Support

In Germany, the engineering and technical support is based on its proven educational system for the technical professions and on the experience accumulated in almost four decades of industrial utilisation of nuclear energy. Depending on the individual activities, experience is concentrated in those companies and organisations predominantly concerned with the design and construction, the safety assessment and licensing or the operation of nuclear power plants, and furthermore, in the nuclear research institutes.

In accordance with the Atomic Energy Act [1A-3] the licensee is required, among other things, to present proof that he has sufficient personnel with the required qualification at his disposal. This proof is furnished by competent operation of the nuclear installation and in the training sessions at the plant simulators (→ Chapter 11 (2)).

In accordance with the organisational structure in German nuclear power plants, the production division which is directly responsible for plant operation is supported in its activities by the organisational units Technology, Maintenance and Surveillance. These organisational units have well-defined service tasks and keep at their disposal the necessary technical expertise:

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

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

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

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

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
1988	292	0	10	279	3	-	-	-
1989	301	0	10	289	2	-	-	-
1990	224	1	8	214	1	-	-	-
1991	243	0	10	233	0	232	11	0
1992	224	0	3	221	0	216	8	0
1993	179	0	2	177	0	172	7	0
1994	161	1	1	159	0	158	3	0
1995	152	0	2	150	0	151	1	0
1996	137	0	2	135	0	131	6	0
1997	117	0	3	114	0	114	3	0

19 (vi) Reporting of Events, Regulatory Reporting Procedure

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

The regulatory reporting procedure is embedded in the regulatory supervision of nuclear installations. The event reports are timely indicators of possible deficiencies. 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 correlated with one of the individual reporting categories. Each of these reporting categories is associated with different obligations for preventive actions by the authorities as follows:

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, generally, 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. These would be only slightly different from routine operational events while plant conditions and operation would 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 (pre-core-loading - reporting deadline: within 10 days)
This category V is used for events occurring during erection and commissioning of the nuclear power plant of which the supervisory authority should be informed with regard to the later safe operation of the plant.

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

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

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

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

The BfS is responsible for the central collection and documentation of the reported events. The BfS performs an initial evaluation of the reported events and informs the *Länder* nuclear authorities, the expert organisations, the manufacturers and the licensees of nuclear power plants as well as the general public in quarterly reports which contain all reportable events in nuclear power plants and research reactors. Table 19-1 lists the reportable events that occurred over the last ten years also indicating both the German and the INES (see below) reporting categories. Figure 19-3 shows these events according to whether they happened spontaneously or during inspections and maintenance, and Figure 19-4 according to the prevalent operating condition and whether they had any impacts on operation. All events are included in these presentations, even those reported or re-classified at a later date.

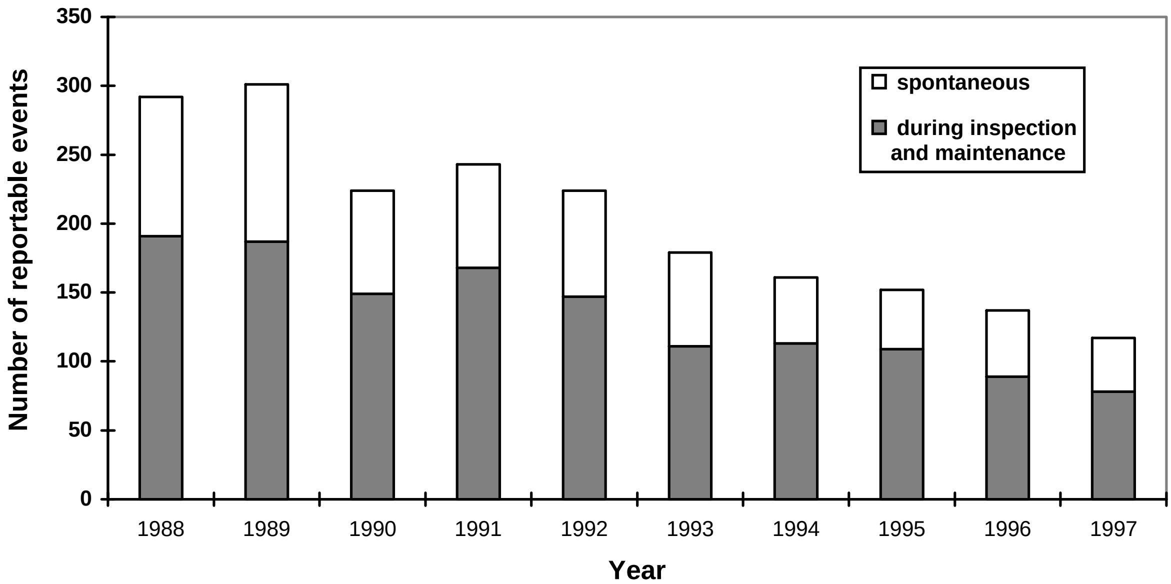


Figure 19-3 Number of Reportable Events from Nuclear Power Plants - According to the Kind of Occurrence

Figure 19-5 shows the development over the last ten years of the average number of reactor scrams also indicating their essential causes.

The detailed evaluation of those reported events with particular importance to safety is performed by GRS. If the results indicate that they would also be applicable to other nuclear installations, an information notice is issued to inform the *Länder* nuclear authorities, the expert organisations, the manufacturers and the licensees of nuclear installations (→ Chapter 19 (vii)).

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

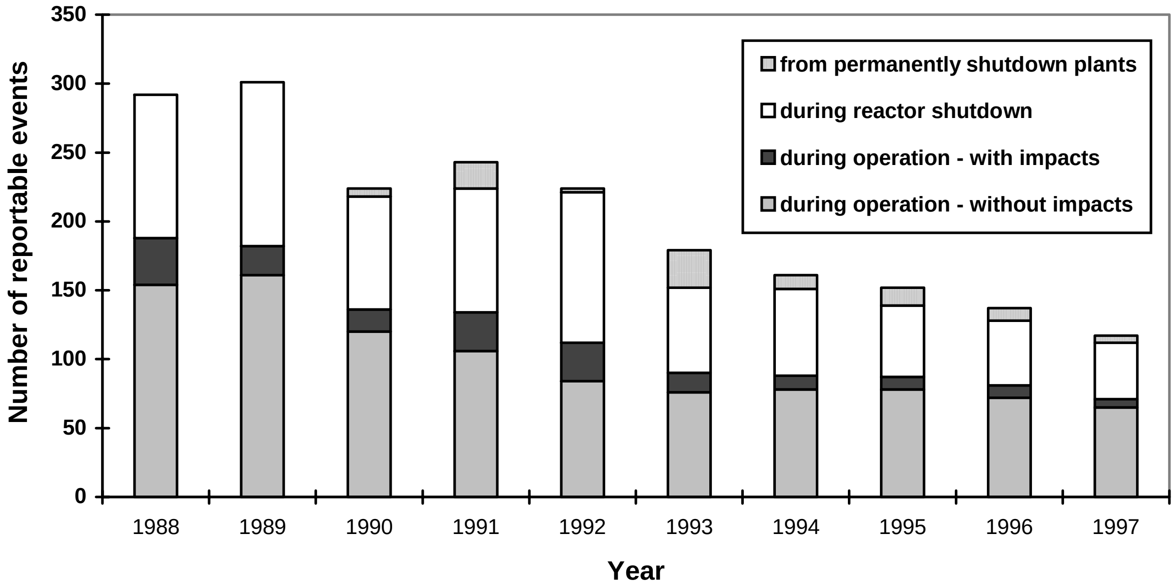


Figure 19-4 Number of Reportable Events in Nuclear Power Plants - According to Mode of and Consequence on Operation (Power Operation, Start-up and Shutdown Operation)

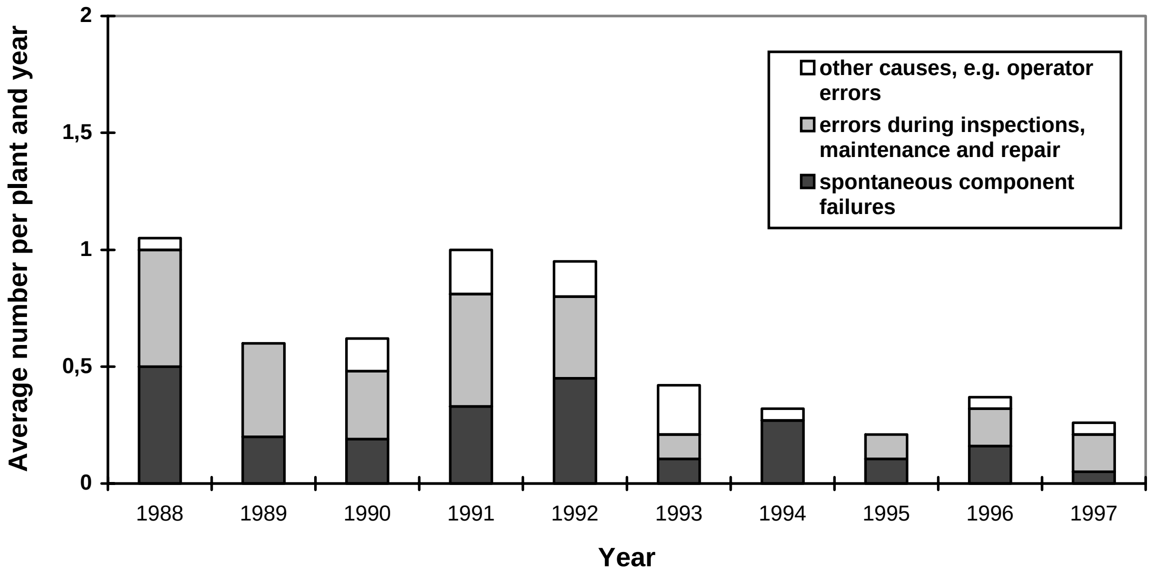


Figure 19-5 Average Number of Unplanned Reactor Scrams - per Plant and Year

19 (vii) Collecting, Analysing and Exchanging Operating Experience

From a very early stage in utilising nuclear energy in Germany a system was established for collecting and exchanging operating experience from nuclear installations. This system has been improved over more than 25 years. The resulting feedback of operating experience has been a major contributing factor to the further development of safety in nuclear installations and to the high degree of maturity and high safety level achieved today. Participating parties are the licensees of the nuclear power plants themselves and the manufacturers, and the licensing and supervisory authorities of the federation and the *Länder* and their respective expert organisations.

Important reportable events and operating experience are regularly discussed by the RSK and are evaluated with regard to their applicability and relevance to other nuclear power plants. This also includes operating experience and reportable events from foreign nuclear installations and their applicability and relevance to German nuclear power plants.

Events not required to be reported

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

Reportable events

The utilisation of the feedback from plant specific experience by other nuclear installations is essentially based on the system for handling reportable events (→ Chapter 19 (vi)). Parallel to submitting the report to the competent authority, the licensee also informs the Association of Major Power Utilities (VGB). VGB collects these reports and distributes them among its members independently of the reporting path via the authorities. The manufacturers participate in the information exchange via VGB and via the authorities.

Reportable events are evaluated at several levels: by the licensee of the affected nuclear installation, by the *Länder* authorities and their expert organisations and, at a federal level, by BfS and GRS. The licensee and the competent *Länder* authority with its expert organisation analyse the event, primarily to find a remedy for the affected installation. However, in a second step the *Länder* authority together with its expert organisation also investigates the significance of the event to other nuclear installations. The licensees, on the other hand, are obliged to additionally evaluate the reportable events from other nuclear installations with respect to possible conclusions for their own installation. They issue regular reports on these results to the supervisory authority (e.g. within their monthly reports) and include these results in their annual report to the RSK.

At a federal level, an initial evaluation of the reportable events is carried out by BfS. The in-depth analysis of all reportable events is carried out by GRS regarding both the behaviour of technical components and systems as well as the manual actions of the plant personnel.

These multiple-level and independent analyses ensure that each event is evaluated in sufficient detail and that the required remedies are taken.

In addition to this experience, feedback through the system for handling reportable events, the licensees have installed a number of working groups which regularly meet for detailed discussion of operating experience. Also mutual programmes are established for investigation and research on issues important to safety and on optimising the operation of nuclear power plants. Working groups similar to those of the licensees have also been installed by the expert organisations and the *Länder* authorities 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.

International reporting systems

In addition to the German experience, another important source for operating experience is found at the international level. Even since the initial days of nuclear energy, the Germany has intensively utilised internationally available operating experience. With respect to an efficient feedback from operating experience on an international level, two systems are of importance: the Incident Reporting System (IRS) of IAEA and OECD/NEA and the reporting system of WANO. The purpose of both systems is to present detailed information on important events - both with respect to the technical circumstances and the manual actions of personnel - that would be significant also to other countries. Germany has been an active member in both systems from the very beginning.

International exchange of experience

In addition to the international reporting systems, Germany has entered a number of bilateral agreements the major purpose of which includes the exchange of operating experience (→ Chapter 17 (iv)). The licensees, likewise, exchange experience feedback beyond that of the reporting systems. For instance, some licensees are connected to the respective manufacturer system on experience feedback. Also, several licensees of foreign nuclear power plants are members of the VGB and, thereby participate in the exchange of experience.

Analysis of operating events in foreign nuclear installations

The IRS is the essential source of safety findings from international operating experience. All events reported here are systematically evaluated by GRS. In its quarterly reports, GRS presents short descriptions for every IRS event and a comment regarding applicability and relevance to German nuclear power plants. These quarterly reports - together with the corresponding reports by IRS - are sent to the supervisory authorities and expert organisations as well as to the licensees and other competent institutions. In addition, GRS prepares annual reports containing detailed descriptions and evaluations of the most important events. These annual reports are distributed in the same way as the quarterly reports. The licensees perform an independent evaluation of the IRS reports with special regard to their own nuclear installations.

Information notices

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

GRS also performs a generic assessment of German and international operating experience. Safety problems that tend to be associated with a group of events (event collective) are subject to in-depth analysis. The results and conclusions from these generic assessments are presented in reports that are distributed in the same way as the information notices. The licensees again perform a plant-specific evaluation of these reports and possibly implement the issue.

Experience feedback from technical modifications and backfitting

In addition to the experience from abnormal occurrences and deficiencies, the experience feedback also covers technical modifications and backfitting. The licensees report to the supervisory authorities on the realised modifications and backfitting in their monthly, maintenance outage and annual reports. The corresponding exchange of information takes place in the different working groups within the VGB. In addition, GRS performs an evaluation of the annual reports submitted by the licensees to the RSK. The results of this evaluation are discussed by the RSK.

Precursor and trend analyses

GRS performs systematic precursor analyses for reportable events in Germany. The purpose is the identification of weak points by probabilistic methods and trend analyses of the safety status. These analyses have been performed since 1993 and confirm the high safety level of German nuclear power plants.

The licensees perform a trend analysis with indicators of the WANO reporting system. Furthermore, the reportable events are systematically evaluated with special attention to determine or verify the frequencies of initiating events (including reactor scrams), to the reliability of systems and components and to the frequency of common cause failures.

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

In accordance with Section 9a Atomic Energy Act [1A-3] anyone who creates residual radioactive materials shall make provisions to ensure that they are utilised without detrimental effects or are lawfully disposed of as radioactive waste.

Generation, treatment, conditioning and disposal of radioactive waste

Any activities concerning the management of radioactive waste are subject to regulatory surveillance by the respective *Länder* nuclear authorities. The licensee submits a conceptual waste programme to the competent supervisory authority; it accounts for all waste accumulated in the restricted access area during operation of the nuclear power plant. By careful operational management and corresponding planning for major plant revisions (refuelling outage), the licensee keeps radioactive waste to a minimum. Treatment, conditioning and disposal of radioactive waste are often supported by specialised outside contractors.

From the time of its generation, the accumulated radioactive material is sorted by radioactivity and type. After clearance measurement and decontamination if necessary, as much of the material as possible is sent for recycling with or without prerequisites. However, if the prescribed criteria for this cannot be met an attempt is made to at least meet the corresponding criteria for disposal as conventional waste. The clearance criteria are specified in the licence permit of the nuclear power plant. The clearance is performed by the licensee and is subject to surveillance by the competent *Länder* authority which also performs control measurements.

The treatment of radioactive waste minimises its volume and converts the primary waste to intermediate products that can be handled and properly conditioned for reposition. From the time of its generation all radioactive waste is documented by type, content and radioactivity. The regulatory guideline on radioactive waste without heat generation [3-59] specifies the sorting criteria and the requirements regarding determination of radioactivity and recording. The waste producer will always be able to give information on the amount of activity and the storage place of the radioactive waste.

The treatment and conditioning of the radioactive waste, is carried out with qualified procedures as far as possible and practicable. Treatment and conditioning is always performed with regard to the requirements of subsequent disposal. Pretreatment 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, disposal-conform conditioning 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 is, then, either returned to the nuclear power plant or transported to an interim storage or to the repository (currently Morsleben).

Table 19-2 Accumulated Radioactive Waste on 31 December 1996

	Waste Volume [m^3]	
	negligible heat generation	heat generating
accumulated primary waste	6 176	390
accumulated conditioned waste *)	5 926	-
waste that was conditioned *) in 1996	3 174	-

*) stated in m^3 of packaging

Data acquisition regarding accumulated waste from nuclear power reactors

The Federal Office for Radiation Protection (BfS) performs an annual survey on the accumulated radioactive waste. This survey includes the overall volume and newly accumulated radioactive waste from nuclear power plants. In addition to the differentiation by source of waste, and with the final repository in mind, BfS also differentiates between radioactive waste that produces heat and such whose heat generation is negligible. Table 19-2 shows data on waste accumulation and conditioning as reported by the licensees of nuclear power plants.

Storage of spent fuel

The spent fuel elements are stored in the fuel pool of the nuclear power plant. Subcriticality and cooling of the fuel elements as well as their protection against external impacts in the fuel pool are ensured. For safety reasons, the spent fuel pool must always have free capacity of one complete core loading [KTA 3602], this permits the core to be unloaded in the event of a design basis accident. The free capacity for fuel storage in one nuclear power plant can not be used by another plant. In 1997 the on-site free storage capacities for spent fuel elements reached the values shown in Table 19-3; the heavy metal (HM) share of the fuel elements is different from plant to plant.

In case of short capacity in the fuel pool of the nuclear power plant, spent fuel may be shipped to off-site interim storage - which may also be used by other nuclear power plants - or may be shipped to France or the United Kingdom for reprocessing. The shipping casks are loaded inside the fuel pool of the plant, wet transports leave for France and the United Kingdom, dry transports for France.

Table 19-3 On-site Storage Capacity for Spent Fuel Elements in all Nuclear Power Plants on 31 December 1997

On-site storage capacity	Number of fuel elements	Weight of fuel [t HM]
overall capacity	20 843	6 575
used storage	8 386	2 816
free capacity *)	5 970	1 839

*) not counting the required space for one core loading nor the locations used for operational purposes

Waste management

Safe waste management at nuclear power plants is an indispensable prerequisite for the use of nuclear energy. Its legal basis is the Atomic Energy Act and the Provisions Regarding Waste Management for Nuclear Power Plants [3-25] published in 1980. The waste management of nuclear power plants comprises

– interim storage for spent fuel elements on-site or off-site interim storage, and, if possible, direct disposal of the fuel elements,
– reprocessing of spent fuel elements, and the utilisation of recovered material as well as the lawful disposal of waste,

– conditioning, interim storage and disposal in a final repository of the radioactive waste from operation and decommissioning of the nuclear power plants.

According to the Provisions Regarding Waste Management, each nuclear power plant has to prove its ability for safe and lawful disposal of spent fuel elements and radioactive waste for the coming six year period.

Planned Activities for Improvement of Safety

Apart from the implementation of on-site accident management (→ Chapter 16 (1)), no need for specific requirements to further improve the safety of the operating of nuclear power plants in Germany is indicated for the near future. Special attention will be given to the phenomenon of ageing which is of concern world-wide.

The German nuclear regulatory authorities - in their continuous efforts to provide a dynamic protection of basic rights - pay particular attention to the development of the state of the art in science and technology and the corresponding enhancement of the safety of nuclear power plants. In this way, the Federal Government, considering the allocation of responsibilities of federal and *Länder* authorities, will continue to ensure that nuclear power plants are backfitted. This should not fail because of exaggerated regulatory requirements in an individual licensing procedure.

The nuclear licensing and supervisory authorities ensure that safety issues will continuously have priority over economic considerations. The liberalisation of the energy market, the increasing competition in global trading and the resulting pressure to reduce costs shall not jeopardise the principle of safety priority.

The nuclear regulatory authorities of Germany and France have agreed on further developing common safety ideas for future nuclear power plants. These will consider that the radiation impacts, even in case of a severe beyond-design-basis accident, - an extremely improbable event even in existing nuclear power plants - are essentially restricted to the nuclear power plant itself. In Germany this was already embedded in the Atomic Energy Act in 1994 (→ Chapter 7 (2i)).

Appendix 1 Design Characteristics Important to Safety

1. Reactor Coolant Pressure Boundary PWR

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

1. Reactor Coolant Pressure Boundary BWR

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

2. Emergency Core Cooling PWR

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

2. Emergency Core Cooling BWR

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

3. Containment Vessel PWR

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

3. Containment Vessel BWR

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

4. Limitations and Reactor Protection System PWR

4.1 Limitations

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

4.2 Reactor Protection System

Design Characteristics	1st Design Generation	2nd Design Generation	3rd Design Generation	4th Design Generation
Actuation criteria derived from accident analysis	Largely, yes	Yes		
Different physical actuation criteria for reactor protection system	Yes, or higher-grade redundancy	Yes, or diverse actuation channels		
Failure combinations	Random failure, systematic failure, consequential failures, non-availability due to maintenance			
Testing of reactor protection system is possible during power operation	Yes, largely by automatic self-monitoring (of functional readiness)			
Safety actuation system	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 Reactor Protection System 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 Reactor Protection System

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)	
Safety actuation system	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. Electrical Power Supply PWR

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

5. Electrical Power Supply BWR

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

6. Protection against External Impacts PWR

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

6. Protection against External Impacts BWR

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

Appendix 2 Reference List of Nuclear Safety Regulations Status 12/97

(A selection concerning nuclear power plants,
order as in the "Handbuch Reaktorsicherheit und Strahlenschutz")

Gliederung

1 Rechtsvorschriften
- 1A Nationales Atom- und Strahlenschutzrecht
- 1B Rechtsvorschriften, die im Bereich der Sicherheit kerntechnischer Anlagen anzuwenden sind
- 1E Multilaterale Vereinbarungen über nukleare Sicherheit und Strahlenschutz mit nationalen Ausführungsvorschriften
- 1F Recht der Europäischen Union

2 Allgemeine Verwaltungsvorschriften

3 Bekanntmachungen des Bundesumweltministeriums und des vormals zuständigen Bundesinnenministeriums

4 Empfehlungen

5 Regeln des Kerntechnischen Ausschusses (KTA)

1 Rechtsvorschriften

1A Nationales Atom- und Strahlenschutzrecht

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

1A-3 Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz - AtG) vom 23. Dezember 1959, Neufassung vom 15. Juli 1985 (BGBl.I, Nr. 41), zuletzt geändert durch Gesetz vom 29. April 1997 (BGBl.I 1997, Nr. 28)

1A-4 Fortgeltendes Recht der Deutschen Demokratischen Republik aufgrund von Artikel 9 Abs. 2 in Verbindung mit Anlage II Kapitel XII Abschnitt III Nr. 2 und 3 des Einigungsvertrages vom 31. August 1990 in Verbindung mit Artikel 1 des Gesetzes zum Einigungsvertrag vom 23. September 1990 (BGBl.II, S. 885, 1226), soweit dabei radioaktive Stoffe, insbesondere Radonfolgeprodukte, anwesend sind:
- Verordnung über die Gewährleistung von Atomsicherheit und Strahlenschutz vom 11. Oktober 1984 und Durchführungsbestimmung zur Verordnung über die Gewährleistung von Atomsicherheit und Strahlenschutz vom 11. Oktober 1984 (GBl.(DDR) I 1984, Nr. 30, berichtigt GBl.(DDR) I 1987, Nr. 18)
- Anordnung zur Gewährleistung des Strahlenschutzes bei Halden und industriellen Absetzanlagen und bei Verwendung darin abgelagerter Materialien vom 17. November 1990 (GBl.(DDR) I 1990, Nr. 34)

1A-5 Gesetz zum vorsorgenden Schutz der Bevölkerung gegen Strahlenbelastung (Strahlenschutzvorsorgegesetz - StrVG) vom 19. Dezember 1986 (BGBl.I, S. 2610), zuletzt geändert durch das Gesundheitseinrichtungen-Neuordnungsgesetz vom 24. Juni 1994 (BGBl.I 1994, Nr. 39)

1A-8 Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 13. Oktober 1976, Neufassung vom 30. Juni 1989 (BGBl.I, S. 1321), Berichtigung vom 16. Oktober 1989 (BGBl.I, S. 1926), zuletzt geändert durch Verordnung vom 18. August 1997 (BGBl.I 1997, Nr. 59)

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)

1A-11 Verordnung über die Deckungsvorsorge nach dem Atomgesetz (Atomrechtliche Deckungsvorsorge-Verordnung - AtDeckV) vom 25. Januar 1977 (BGBl.I 1977, S. 220), zuletzt geändert durch das 6. Überleitungsgesetz vom 25. September 1990 (BGBl.I 1990, S. 2106)

1A-12 Kostenverordnung zum Atomgesetz (AtKostV) vom 17. Dezember 1981 (BGBl.I, S. 1457), zuletzt geändert durch VO vom 18. Dezember 1992 (BGBl.I 1992, Nr. 57)

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, S. 562), zuletzt geändert durch das 6. Überleitungsgesetz vom 25. September 1990 (BGBl.I 1990, S. 2106)

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)

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

1B-1 Strafgesetzbuch vom 15. Mai 1871 (RGBl. S. 127) in der Fassung der Bekanntmachung vom 10. März 1987 (BGBl.I 1987, S. 945+1160), zuletzt geändert (Kernenergie betreffend) durch Gesetz vom 26. Januar 1998 (BGBl.I 1998, Nr. 6)

1B-2 Bau- und Raumordnungsgesetz 1998 vom 18. August 1997 (BGBl.I 1997, Nr. 59), in Kraft ab 1. Januar 1998

1B-3 Gesetz zum Schutz vor schädlichen Umwelteinwirkungen durch Luftverunreinigungen, Geräusche, Erschütterungen und ähnliche Vorgänge (Bundes-Immissionsschutzgesetz - BImSchG) in der Fassung der Bekanntmachung vom 14. Mai 1990 (BGBl.I 1990, S. 880), zuletzt geändert durch Gesetz vom 19. Juli 1995 (BGBl.I 1995, Nr. 37), mit diversen Verordnungen

1B-5 Gesetz zur Ordnung des Wasserhaushalts (Wasserhaushaltsgesetz) vom 27. Juli 1957, Neufassung vom 12. November 1996 (BGBl.I 1996, Nr. 58)

1B-6 Gesetz über Naturschutz und Landschaftspflege (Bundesnaturschutzgesetz) vom 12. März 1987 (BGBl.I 1987, S. 889)

1B-7 Gesetz über technische Arbeitsmittel (Gerätesicherheitsgesetz) vom 24. Juni 1968, Neufassung vom 23. Oktober 1992, (BGBl.I 1992, Nr. 49) zuletzt geändert durch Gesetz vom 2. August 1994 (BGBl.I 1994, Nr. 52)

1B-8 Verordnung über Dampfkesselanlagen (Dampfkesselverordnung) vom 27. Februar 1980 (BGBl.I 1980, S. 173), zuletzt geändert am 22. Juni 1995 (BGBl.I 1995, S. 836) (Verzeichnis der technischen Regeln für Dampfkessel (TRD)

1B-9 Verordnung über Druckbehälter, Druckgasbehälter und Füllanlagen (Druckbehälterverordnung) in der Neufassung vom 21. April 1989 (BGBl.I 1989, S. 843), zuletzt geändert am 22. Juni 1995 (BGBl.I 1995, S. 836)
Verordnung zum Gerätesicherheitsgesetz und zur Änderung der Druckbehälterverordnung vom 25. Juni 1992 (BAnz. 1992, Nr. 29)

1B-10 Unfallverhütungsvorschrift Kernkraftwerke (VBG 30) und Durchführungsanweisung zur Unfallverhütungsvorschrift vom 1. Januar 1987

1B-11 Gesetz über den Verkehr mit Lebensmitteln, Tabakerzeugnissen, kosmetischen Mitteln und sonstigen Bedarfsgegenständen (Lebensmittel- und Bedarfsgegenständegesetz) vom 15. August 1974 (BGBl.I 1975, S. 2652), Neufassung vom 9. September 1997 (BGBl.I 1997, Nr. 63), mit diversen Verordnungen

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

Nukleare Sicherheit und Strahlenschutz

1E-1 Übereinkommen Nr. 115 der Internationalen Arbeitsorganisation vom 22. Juni 1960 über den Schutz der Arbeitnehmer vor ionisierenden Strahlen,
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-2 Ratsbeschluß der Organisation für Wirtschaftliche Zusammenarbeit und Entwicklung (OECD) vom 18. Dezember 1962 über die Annahme von Grundnormen für den Strahlenschutz (OECD-Grundnormen),
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), s. auch EURATOM-Grundnormen

1E-4 Übereinkommen vom 26. Oktober 1979 über den physischen Schutz von Kernmaterial,
Gesetz hierzu vom 24. April 1990 (BGBl.II 1990, S. 326), zuletzt geändert durch das Strafrechtsänderungsgesetz vom 27. Juni 1994 (BGBl.I 1994, Nr. 40),
in Kraft für Deutschland seit 6. Oktober 1991 (BGBl.II 1995, Nr. 11)

1E-5 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,
Gesetz zu den beiden IAEA-Übereinkommen vom 16. Mai 1989 (BGBl.II 1989, Nr. 18),
in Kraft für Deutschland seit 15. Oktober 1989 (BGBl.II 1993, Nr. 34)

1E-6 Übereinkommen über nukleare Sicherheit vom 20. September 1994,
Gesetz dazu vom 7. Januar 1997 (BGBl.II 1997, Nr. 2)
in Kraft für Deutschland seit 20. April 1997 (BGBl.II 1997, Nr. 14)

1E-9 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),
Gesetz hierzu vom 4. Juni 1974 (BGBl.II 1974, S. 794),
in Kraft für alle Vertragsparteien seit 21. Februar 1977 (BGBl.II 1980, S. 102),
Ausführungsgesetz hierzu vom 7. Januar 1980 (BGBl.I 1980, S. 17), zuletzt geändert durch Gesetz vom 27. Dezember 1993 (BGBl.I 1993, S. 2378)

Haftung

1E-11 Übereinkommen vom 29. Juli 1960 über die Haftung gegenüber Dritten auf dem Gebiet der Kernenergie (Pariser Atomhaftungs-Übereinkommen),
Gesetz dazu vom 8. Juli 1975 (BGBl.II 1975, S. 957), geändert durch Gesetz vom 9. Juni 1980 (BGBl.II 1980, S. 721),
in Kraft für Deutschland seit 30. September 1975 (BGBl.II 1976, S. 308),
Protokoll vom 16. November 1982 zur Änderung des Übereinkommens vom 29. Juli 1960 in der Fassung des Zusatzprotokolls vom 28. Januar 1964,
Gesetz dazu vom 21. Mai 1985 (BGBl.II 1985, S. 690),
in Kraft für Deutschland seit 7. Oktober 1988 (BGBl.II 1988, S. 144)
letzte Neufassung des Pariser Atomhaftungs-Übereinkommens vom 15. Juli 1985 (BGBl.II 1985, S. 963)

1E-12 Zusatzübereinkommen vom 31. Januar 1963 zum Pariser Übereinkommen vom 29. Juli 1960 (Brüsseler Zusatzübereinkommen), ergänzt durch das Protokoll vom 28. Januar 1964 (BGBl.II 1976, S. 310),
Gesetz dazu vom 8. Juli 1975 (BGBl.II 1975, S. 957), geändert durch Gesetz vom 9. Juli 1980 (BGBl.II 1980, S. 721),
in Kraft für Deutschland seit 1. Januar 1976 (BGBl.II 1976, S. 308),
Protokoll vom 16. November 1982 zur Änderung des Zusatzübereinkommens vom 31. Januar 1963 in der Fassung des Zusatzprotokolls vom 28. Januar 1964,
Gesetz dazu vom 21. Mai 1985 (BGBl.II 1985, S. 690),
in Kraft für seit 1. August 1991 (BGBl.II 1995, S. 657),
Neufassung des Brüsseler Zusatzübereinkommens vom 15. Juli 1985 (BGBl.II 1985, S. 963)

1E-15 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)

F Recht der Europäischen Union

Verträge, Allgemeines

1F-1 Vertrag vom 25. März 1957 zur Gründung der Europäischen Atomgemeinschaft (**EURATOM**) in der Fassung des Vertrages über die **Europäische Union** vom 7. Februar 1992, geändert durch den Beitrittsvertrag vom 24. Juni 1994 in der Fassung des Beschlusses vom 1. Januar 1995 (BGBl.II 1957, S. 753, 1014, 1678; BGBl.II 1992, S. 1251, 1286; BGBl.II 1993, S. 1947; BGBl.II 1994, S. 2022; ABl.EG 1995, Nr. L1), 1F-2 Verifikationsabkommen siehe [1E-9]

1F-3 Verordnung (EURATOM) 3227/76 der Kommission vom 19. Oktober 1976 zur Anwendung der Bestimmungen der EURATOM-Sicherungsmaßnahmen (ABl.EG 1976, Nr. L363), geändert durch Verordnung EURATOM 2130/93 der Kommission vom 27. Juli 1993 (ABl.EG 1993, Nr. L191)

1F-4 Bekanntmachung über die Meldung an die Behörden der Mitgliedsstaaten auf dem Gebiet der Sicherungsmaßnahmen gemäß Artikel 79 Abs. 2 des EURATOM-Vertrages vom 12. August 1991 (BAnz. Nr. 158)

1F-7 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
Hinweis: Laufzeit 30 Jahre, Nachfolgevereinbarung für ein entsprechendes Abkommen, das 35 Jahre in Kraft war, Basis für den Handel mit Nuklearmaterial und Ausrüstung

1F-12 Empfehlung 91/4/EURATOM der Kommission vom 7. Dezember 1990 betreffend die Anwendung von Artikel 37 des EURATOM-Vertrages (ABl.EG 1991, Nr. L6)

1F-13 Richtlinie 85/337/EWG des Rates vom 27. Juni 1985 über die Umweltverträglichkeitsprüfung bei bestimmten öffentlichen und privaten Projekten, Gesetz hierzu vom 12. Februar 1990 (BGBl.I, S. 205), zuletzt geändert durch das 6. Überleitungsgesetz vom 25. September 1990 (BGBl.I 1990, S. 2106)

1F-14 Richtlinie 90/313/EWG des Rates vom 7. Juni 1990 über den freien Zugang zu Informationen über die Umwelt (ABl.EG 1990, Nr. L158)
Gesetz hierzu vom 8. Juli 1994 (BGBl.I 1994, Nr. 42)
- Verordnung über Gebühren für Amtshandlungen der Behörden des Bundes beim Vollzug des Umweltinformationsgesetzes (Umweltinformations-gebührenverordnung) vom 7. Dezember 1994 (BGBl.I 1994, Nr. 88)

Strahlenschutz

1F-15 Richtlinien des Rates, mit denen die Grundnormen für den Gesundheitsschutz der Bevölkerung und der Arbeitskräfte gegen die Gefahren ionisierender Strahlungen festgelegt wurden (EURATOM-Grundnormen)
- Richtlinie vom 2. Februar 1959 (ABl.EG 1959, Nr. 11),
- Richtlinie vom 5. März 1962 (ABl.EG 1962, S. 1633/62),
- Richtlinie 66/45/EURATOM (ABl.EG 1966, Nr. 216),
- Richtlinie 76/579/EURATOM vom 1.6.1976 (ABl.EG 1976, Nr. L187),
- Richtlinie 79/343/EURATOM vom 27.3.1977 (ABl.EG 1979, Nr. L83),
- Richtlinie 80/836/EURATOM vom 15.7.1980 (ABl.EG 1980, Nr. L246),
- Richtlinie 84/467/EURATOM vom 3.9.1984 (ABl.EG 1984, Nr. L265),
- Neufassung mit Berücksichtigung der ICRP 60 in Richtlinie 96/29/EURATOM vom 13. Mai 1996 (ABl.EG 1996, Nr. L159)

Hinweis: gemäß Artikel 55 der Richtlinie 96/29/EURATOM haben die Mitgliedstaaten die erforderlichen Rechts- und Verwaltungsvorschriften zur Erfüllung dieser Richtlinie bis zum 13. März 2000 zu erlassen. Die aufgeführten Richtlinien von 1959 bis 1984 werden gemäß Artikel 56 der Richtlinie 1996 mit Wirkung vom 13. Mai 2000 aufgehoben.

1F-16 Mitteilung der Kommission zur Durchführung der Richtlinien des Rates 80/836/EURATOM und 84/467/EURATOM (ABl.EG 1985, Nr. C347)

1F-19 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)

Radiologische Notfälle

1F-28 Entscheidung 87/600/EURATOM des Rates vom 14. Dezember 1987 über Gemeinschaftsvereinbarungen für den beschleunigten Informationsaustausch im Fall einer radiologischen Notstandssituation (ABl.EG 1987, Nr. L371)

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 Gesundheitsschutzmaßnahmen (ABl.EG 1989, Nr. L357)
- Mitteilung der Kommission betreffend die Durchführung der Richtlinie 89/618/EURATOM (ABl.EG 1991, Nr. C103)

1F-30 Verordnungen zur Festlegung von Höchstwerten an Radioaktivität in Nahrungsmitteln und Futtermitteln im Fall eines nuklearen Unfalls oder einer anderen radiologischen Notstandssituation:
- Ratsverordnung (EURATOM) 3954/87 vom 22.12.1987 (ABl.EG 1987, Nr. L371) geändert durch Ratsverordnung (EURATOM) 2218/89 vom 18.7.1989 (ABl.EG 1989, Nr. L211),
- Kommissionsverordnung (EURATOM) 944/89 vom 12.4.89 (ABl.EG 1989, Nr. L101),
- Kommissionsverordnung (EURATOM) 770/90 vom 29.3.1990 (ABl.EG 1990, Nr. L83)

1F-31 Ratsverordnung (EWG) 2219/89 vom 18.7.1989 über besondere Bedingungen für die Ausfuhr von Nahrungsmitteln und Futtermitteln im Falle eines nuklearen Unfalls oder einer anderen radiologischen Notstandssituation (ABl.EG 1989, Nr. L211)

1F-32 Ratsverordnung (EWG) 3955/87 vom 22. Dezember 1987 über die Einfuhrbedingungen für landwirtschaftliche Erzeugnisse mit Ursprung in Drittländern nach dem Unfall im Kernkraftwerk Tschernobyl (ABl.EG 1987, Nr. L371),
- Verordnung (EWG) 1983/88 der Kommission vom 5. Juli 1988 mit Durchführungsbestimmungen zu der Verordnung (EWG) 3955/87 (ABl.EG 1988, Nr. L174),
- Verordnung (EWG) 4003/89 des Rates vom 21. Dezember 1989 zur Änderung der Verordnung (EWG) 3955/87 (ABl.EG 1989, Nr. L382),
- Verordnung (EWG) 737/90 des Rates vom 22. März 1990 zur Ergänzung der Verordnung (EWG) 3955/87 (ABl.EG 1990, Nr. L82),
- Verordnung (EG) 686/95 des Rates zur Verlängerung der Verordnung (EWG) 737/90 (ABl.EG 1995, Nr. L71),
- Verordnungen der Kommission zur Festlegung einer Liste von Erzeugnissen die von der Durchführung der Verordnung (EWG) 737/90 des Rates über die Einfuhrbedingungen für landwirtschaftliche Erzeugnisse mit Ursprung in Drittländern nach dem Unfall im Kernkraftwerk Tschernobyl ausgenommen sind ,
  - Verordnung (EWG) 146/91 vom 22.1.1991 (ABl.EG 1991, Nr. L17),
  - Verordnung (EWG) 598/92 vom 9.3.1992 (ABl.EG 1992, Nr. L64),
  - Verordnung (EWG) 1518/93 vom 21. Juni 1993 (ABl.EG 1993, Nr. L150),
  - Verordnung (EG) 3034/94 vom 13. Dezember 1994 (ABl.EG 1994, Nr. L321)

2 Allgemeine Verwaltungsvorschriften

2-1 Allgemeine Verwaltungsvorschrift zu § 45 Strahlenschutzverordnung: Ermittlung der Strahlenexposition durch die Ableitung radioaktiver Stoffe aus kerntechnischen Anlagen oder Einrichtungen vom 21. Februar 1990 (BAnz. 1990, Nr. 64a)

2-2 Allgemeine Verwaltungsvorschrift zu § 62 Abs. 2 Strahlenschutzverordnung (AVV Strahlenpaß) vom 3. Mai 1990 (BAnz. 1990, Nr. 94a)

2-3 Allgemeine Verwaltungsvorschrift zur Ausführung des Gesetzes über die Umweltverträglichkeitsprüfung (UVPVwV) vom 18. September 1995 (GMBl. 1995, Nr. 32)

2-4 Allgemeine Verwaltungsvorschrift zum Integrierten Meß- und Informationssytem nach dem Strahlenschutzvorsorgegesetz (AVV-IMIS) vom 27. September 1995 (BAnz. 1995, Nr. 200a)

3 Bekanntmachungen des Bundesumweltministeriums und des vormals zuständigen Bundesinnenministeriums

3-1 Sicherheitskriterien für Kernkraftwerke vom 21.10.1977 (BAnz. 1977, Nr. 206)

3-2 Richtlinie für den Fachkundenachweis von Kernkraftwerkspersonal vom 14.4.1993 (GMBl. 1993, Nr. 20)

3-4 Richtlinien über die Anforderungen an Sicherheitsspezifikationen für Kernkraftwerke vom 27.4.1976 (GMBl. 1976, S. 199)

3-5 Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor vom 26.7.1976 (GMBl. 1976, S. 418)

3-6 Richtlinie für den Schutz von Kernkraftwerken gegen Druckwellen aus chemischen Reaktionen durch Auslegung der Kernkraftwerke hinsichtlich ihrer Festigkeit und induzierten Schwingungen sowie durch Sicherheitsabstände vom 13.9.1976 (BAnz. 1976, Nr. 179)

3-7-1 Zusammenstellung der in atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen (ZPI) vom 20.10.1982 (BAnz. 1983, Nr. 6a)

3-7-2 Zusammenstellung der zur bauaufsichtlichen Prüfung kerntechnischer Anlagen erforderlichen Unterlagen vom 6.11.1981 (GMBl. 1981, S. 518)

3-8 Grundsätze für die Vergabe von Unteraufträgen durch Sachverständige vom 29.10.1981 (GMBl. 1981, S. 517)

3-9-1 Grundsätze zur Dokumentation technischer Unterlagen durch Antragsteller/Genehmigungsinhaber bei Errichtung, Betrieb und Stillegung von Kernkraftwerken vom 19.2.1988 (BAnz. 1988, Nr. 56)

3-9-2 Anforderungen an die Dokumentation bei Kernkraftwerken vom 5.8.1982 (GMBl. 1982, S. 546)

3-12 Bewertungsdaten für Kernkraftwerksstandorte vom 11. Juni 1975 (Umwelt 1975, Nr. 43)

3-15 (1) Rahmenempfehlungen für den Katastrophenschutz in der Umgebung kerntechnischer Anlagen vom 1.12.1988 (GMBl. 1989, S. 71)
(2) Radiologische Grundlagen für Entscheidungen über Maßnahmen zum Schutz der Bevölkerung bei unfallbedingten Freisetzungen von Radionukliden vom 11.5.1989 (GMBl. 1989, S. 71) z.Z. in Überarbeitung

3-23 Richtlinie zur Emissions- und Immissionsüberwachung kerntechnischer Anlagen (REI) vom 30.6.1993 (GMBl. 1993, Nr. 29)
3-23-2 ergänzt um die Anhänge B und C vom 20.12.1995 (GMBl. 1996, Nr. 9/10)

3-24 Richtlinie über Dichtheitsprüfungen an umschlossenen radioaktiven Stoffen vom 20.8.1996 (GMBl. 1996, Nr. 35)

3-25 Grundsätze zur Entsorgungsvorsorge für Kernkraftwerke vom 19.3.1980 (BAnz. 1980, Nr. 58)

3-27 Richtlinie über die Gewährleistung der notwendigen Kenntnisse der beim Betrieb von Kernkraftwerken sonst tätigen Personen vom 30.10.1980 (GMBl. 1980, S. 652)

3-31 Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 27.12.1976 (GMBl. 1977, S. 48)

3-32 Änderung der Empfehlungen zur Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken vom 18.10.1977 (GMBl. 1977, S. 664)

3-33 Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV (Störfall-Leitlinien) vom 18.10.1983 (BAnz. 1983, Nr. 245a)

Störfallberechnungsgrundlagen für die Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gemäß § 28 Abs. 3 StrlSchV vom 18.10.1983 (BAnz. 1983, Nr. 245a), Neufassung des Kapitels 4 "Berechnung der Strahlenexposition" vom 29. Juni 1994 (BAnz. 1994, Nr. 222a) (zu § 45 StrlSchV: siehe Abteilung 2, Allgemeine Verwaltungsvorschrift)

3-34 Rahmenrichtlinie über die Gestaltung von Sachverständigengutachten in atomrechtlichen Verwaltungsverfahren vom 15.12.1983 (GMBl. 1984, S. 21)

3-36 Leitsätze für die Unterrichtung der Öffentlichkeit über die Katastrophenschutzplanung in der Umgebung von kerntechnischen Anlagen vom 10.2.1978 (Umwelt Nr. 61, 1978)

3-37-1 Empfehlung über den Regelungsinhalt von Bescheiden bezüglich der Ableitung radioaktiver Stoffe aus Kernkraftwerken mit Leichtwasserreaktor vom 8.8.1984 (GMBl. 1984, S. 327)

3-38 Richtlinie für Programme zur Erhaltung der Fachkunde des verantwortlichen Schichtpersonals in Kernkraftwerken vom 1.9.1993 (GMBl. 1993, Nr. 36)

3-39 Richtlinie für den Inhalt der Fachkundeprüfung des verantwortlichen Schichtpersonals in Kernkraftwerken vom 23.4.1996 (GMBl. 1996, Nr. 26)

3-40 Richtlinie über die Fachkunde im Strahlenschutz vom 17.9.1982 (GMBl. 1982, S. 592)

3-41 Richtlinie für das Verfahren zur Vorbereitung und Durchführung von Instandhaltungs- und Änderungsarbeiten in Kernkraftwerken vom 1.6.1978 (GMBl. 1978, S. 342)

3-42 Richtlinie für die physikalische Strahlenschutzkontrolle zur Ermittlung der Körperdosen (§§ 62, 63, 63a StrlSchV; §§35, 35a RöV) vom 20.12.1993 (GMBl. 1994, Nr. 7)

3-42-1 Richtlinie für die Ermittlung der Körperdosen bei innerer Strahlenexposition gemäß den §§ 63 und 63a der Strahlenschutzverordnung (Berechnungsgrundlage) vom 13. März 1997 (BAnz. 1997, Nr. 122a)
Richtlinie für den Strahlenschutz des Personals bei der Durchführung von Instandhaltungsarbeiten in Kernkraftwerken mit Leichtwasserreaktor;

3-43 Teil I: Die während der Planung der Anlage zu treffende Vorsorge vom 10.7.1978 (GMBl. 1978, S. 418)

3-43-1 Teil II: Die Strahlenschutzmaßnahmen während der Inbetriebsetzung und des Betriebs der Anlage vom 4.8.1981 (GMBl. 1981, S. 363)

3-44 Kontrolle der Eigenüberwachung radioaktiver Emissionen aus Kernkraftwerken vom 5.2.1996 (GMBl. 1996, Nr. 9/10)

3-45 Genehmigungen gemäß § 3 Abs. 1 StrlSchV zur ortsveränderlichen Verwendung und Lagerung radioaktiver Stoffe im Rahmen der zerstörungsfreien Materialprüfung vom 14. November 1991 (GMBl. 1992, Nr. 6)

3-46 Genehmigung gemäß § 8 Abs. 1 StrlSchV zur Beförderung radioaktiver Stoffe für Durchstrahlungsprüfungen im Rahmen der zerstörungsfreien Materialprüfung vom 29.5.1978 (GMBl. 1978, S. 334)

3-46-1 Merkblatt für die Beförderung radioaktiver Stoffe für Durchstrahlungsprüfungen im Rahmen der zerstörungsfreien Materialprüfung vom 20.11.1981 (GMBl. 1982, S. 22)

3-49 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke; Einzelfehlerkonzept - Grundsätze für die Anwendung des Einzelfehlerkriteriums vom 2.3.1984 (GMBl. 1984, S. 208)

3-50 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 17.5.1979 (GMBl. 1979, S. 161)
zu Sicherheitskriterium 2.6: Einwirkungen von außen
zu Sicherheitskriterium 8.5: Wärmeabfuhr aus dem Sicherheitseinschluß

3-51 Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 28.11.1979 (GMBl. 1980, S. 90)
zu Sicherheitskriterium 2.2: Prüfbarkeit
zu Sicherheitskriterium 2.3: Strahlenbelastung in der Umgebung
zu Sicherheitskriterium 2.6: Einwirkungen von außen
zu Sicherheitskriterium 2.7: Brand- und Explosionsschutz
ergänzende Interpretation zu Sicherheitskriterium 4.3: Nachwärmeabfuhr nach Kühlmittelverlusten

3-54 Rahmenempfehlung für die Fernüberwachung von Kernkraftwerken vom 6.10.1980 (GMBl. 1980, S 577)

3-54-1 Empfehlung zur Berechnung der Gebühr nach § 5 AtKostV für die Fernüberwachung von Kernkraftwerken (KFÜ) vom 21.1.1983 (GMBl. 1983, S. 146)

3-57 Anforderungen an den Objektsicherungsdienst und an Objektsicherungsbeauftragte in kerntechnischen Anlagen der Sicherungskategorie I vom 8.4.1986 (GMBl. 1986, S. 242)

3-57-1 Richtlinie für die Überprüfung der Zuverlässigkeit der in kerntechnischen Anlagen, bei der Beförderung und Verwendung von Kernbrennstoffen und Großquellen tätigen Personen vom 4.6.1996 (GMBl. 1996, Nr. 29)

3-57-3 Richtlinie für den Schutz von Kernkraftwerken mit Leichtwasserreaktoren gegen Störmaßnahmen oder sonstige Einwirkungen Dritter vom 6.12.1995 (GMBl. 1996, Nr. 2) (ohne Wortlaut)

3-59 Richtlinie zur Kontrolle radioaktiver Abfälle mit vernachlässigbarer Wärmeentwicklung, die nicht an eine Landessammelstelle abgeliefert werden vom 16.1.1989 (BAnz. 1989, Nr. 63a), letzte Ergänzung vom 14.1.1994 (BAnz. 1994, Nr. 19)

3-61 Richtlinie für die Fachkunde von Strahlenschutzbeauftragten in Kernkraftwerken und sonstigen Anlagen zur Spaltung von Kernbrennstoffen vom 10.12.1990 (GMBl. 1991, S. 56)

3-65 Anforderungen an Lehrgänge zur Vermittlung kerntechnischer Grundlagenkenntnisse für verantwortliches Schichtpersonal in Kernkraftwerken - Anerkennungskriterien - Stand 18.4.1989 (nicht veröffentlicht), Aktualisierung vom 10. Oktober 1994

3-66 Meldung an die Behörden der Mitgliedstaaten auf dem Gebiet der Sicherungsmaßnahmen gemäß Artikel 79 Abs. 2 des EURATOM-Vertrages vom 12.8.1991 (BAnz. 1991, Nr. 158)

3-67 Richtlinie über Anforderungen an Personendosismeßstellen nach Strahlenschutz- und Röntgenverordnung vom 26. April 1994 (GMBl. 1994, Nr. 33)

Richtlinie für die Überwachung der Radioaktivität in der Umwelt nach dem Strahlenschutzvorsorgegesetz

3-69 Teil I: Meßprogramm für den Normalbetrieb (Routinemeßprogramm) vom 28. Juli 1994 (GMBl. 1994, Nr. 32)

3-69-2 Teil II: Meßprogramm für den Intensivbetrieb (Intensivmeßprogramm) vom 19. Januar 1995 (GMBl. 1995, Nr. 14)

3-72 Richtlinie über Anforderungen an Inkorporationsmeßstellen vom 30. September 1996 (GMBl. 1996, Nr. 46)

3-73 Leitfaden zur Stillegung von Anlagen nach § 7 des Atomgesetzes vom 14. Juni 1996 (BAnz. 1996, Nr. 211a)

3-74 Leitfäden zur Durchführung von Periodischen Sicherheitsüberprüfungen (PSÜ) für Kernkraftwerke in der Bundesrepublik Deutschland, Bekanntmachung vom 18. August 1997 (BAnz. 1997, Nr. 232a)

4 Empfehlungen

4-1 RSK-Leitlinien für Druckwasserreaktoren
3. Ausgabe vom 14. Oktober 1981 (BAnz. 1982, Nr. 69a)
Leitlinie 21.2 in der Neufassung der 181. RSK-Sitzung am 15.12.1982 (BAnz. 1983, Nr. 106)
Leitlinie 21.1 in der Neufassung der 194. RSK-Sitzung am 21.03.1984 (BAnz. Nr. 104)
Kapitel 7 in der Neufassung der 298. RSK-Sitzung am 20. März 1996 (BAnz. 1996, Nr. 158a), Berichtigung (BAnz 1996, Nr. 214)
Anhänge zu Kapitel 4.2 vom 25. April 1979
1. Auflistung der Systeme und Komponenten
2. Rahmenspezifikation Basissicherheit
(BAnz. Nr. 167a)

5 Regeln des Kerntechnischen Ausschusses (KTA)

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
	<u>1000 KTA-interne Verfahrensregeln</u> (siehe KTA-Handbuch)					
	<u>1100 Begriffe und Definitionen</u> (siehe Begriffe-Sammlung der KTA-Geschäftsstelle, KTA-GS-12)	1/96	-	6/91	-	-
	<u>1200 Allgemeines, Administration, Organisation</u> *<u>General, Administration, Organization</u>*					
1201	Anforderungen an das Betriebshandbuch *Requirements for the Operating Manual*	12/85	33 a 18.02.86	2/78; 3/81	12.06.90	+
1202	Anforderungen an das Prüfhandbuch *Requirements for the Testing Manual*	6/84	191 a 09.10.84 Beilage 51/84	-	14.06.94	+
	<u>1300 Radiologischer Arbeitsschutz</u> *<u>Radiological (aspects of) industrial safety</u>*					
1301.1	Berücksichtigung des Strahlenschutzes der Arbeitskräfte bei Auslegung und Betrieb von Kernkraftwerken; Teil 1: Auslegung *Radiation Protection Considerations for Plant Personnel in the Design and Operation of Nuclear Power Plants; Part 1: Design*	11/84	40 a 27.02.85	-	14.06.94	+
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	14.06.94	+
	<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		+
1404	Dokumentation beim Bau und Betrieb von Kernkraftwerken *Documentation During the Construction and Operation of Nuclear Power Plants*	6/89	158 a 24.08.89	-	14.06.94	+
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	-	11.06.96	+

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 aktivitätsführende Komponenten in Kernkraftwerken; Teil 2: Herstellung *Quality Assurance for Weld Filler Materials and Weld Additives for Pressure and Activity Retaining System in Nuclear Power Plants; Part 2: Manufacturing*	6/85	203 a 29.10.85 Berichtigung 229 10.12.86	-	11.06.96	+
1408.3	Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitätsführende Komponenten in Kernkraftwerken; Teil 3: Verarbeitung *Quality Assurance for Weld Filler Materials and Weld Additives for Pressure and Activity Retaining System in Nuclear Power Plants; Part 3: Processing*	6/85	203 a 29.10.85	-	11.06.96	+
	1500 Strahlenschutz und Überwachung *Radiological Protection and Monitoring*					
1501	Ortsfestes System zur Überwachung von Ortsdosisleistungen innerhalb von Kernkraftwerken *Stationary System for Monitoring Area Dose Rates within Nuclear Power Plants*	6/91	7 a 11.01.92	10/77	11.06.96 1)	-
1502.1	Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 1: Kernkraftwerke mit Leichtwasserreaktor *Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 1: Nuclear Power Plants with Light Water Reactors*	6/86	162 a 03.09.86 Berichtigung 195 15.10.88	-	11.06.96	+
(1502.2)	Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 2: Kernkraftwerke mit Hochtemperaturreaktor *Monitoring Radioactivity in the Inner Atmosphere of Nuclear Power Plants; Part 2: Nuclear Power Plants with High Temperature Reactors*	6/89	229 a 07.12.89	-	-	+
1503.1	Überwachung der Ableitung gasförmiger und aerosolgebundener radioaktiver Stoffe; Teil 1: Überwachung der Ableitung radioaktiver Stoffe mit der Kaminfortluft bei bestimmungsgemäßem Betrieb *Monitoring and Assessing of the Discharge of Gaseous and Aerosolbound Radioactive Substances; Part 1: Monitoring and Assessing of the Stack Discharge of Radioactive Substances during Specified Normal Operation*	6/93	211 a 09.11.93	2/79	-	-
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	-	-
1506	Messung der Ortsdosisleistung in Sperrbereichen von Kernkraftwerken *Measuring Local Dose Rates in Exclusion Areas of Nuclear Power Plants*	6/86	162 a 03.09.86 Berichtigung 229 10.12.86	-	11.06.96	+

Regel-Nr. KTA	Titel	Letzte Fassung	Veröffentlichung im Bundesanzeiger Nr. vom	Frühere Fassungen	Bestätigung der Weitergültigkeit	Engl. Übersetzung
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*	3/84	125 a 07.07.84 Beilage 36/84 Berichtigung 136 24.07.84	-	27.06.89	+
1508	Instrumentierung zur Ermittlung der Ausbreitung radioaktiver Stoffe in der Atmosphäre *Instrumentation to Determine Atmospheric Diffusion of Radioactive Substances*	9/88	37 a 22.02.89	-	15.06.93	+
	<u>2100 Gesamtanlage</u> *<u>Plant</u>*					
2101.1	Brandschutz in Kernkraftwerken; Teil 1: Grundsätze des Brandschutzes (siehe auch 2.4.2) *Fire Protection in Nuclear Power Plants;* *Part 1: Basic Principles* *(Safety standard revision initiated)*	12/85	33 a 18.02.86	-	-	+
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)*	6/89	229 a 07.12.89	-	14.06.94 1)	+
	<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	13.06.95	+
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	13.06.95	+
2201.4	Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 4: Anforderungen an Verfahren zum Nachweis der Erdbebensicherheit für maschinen- und elektrotechnische Anlagenteile *Design of Nuclear Power Plants against Seismic Events;* *Part 4: Requirements for Procedures for Verifying the Safety of Mechanical and Electrical Components against Earthquakes*	6/90	20 a 30.01.91 Berichtigung 115 25.06.96	-	13.06.95	+
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	-	+

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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	-	10.06.97	+
2206	Auslegung von Kernkraftwerken gegen Blitzeinwirkungen *Design of Nuclear Power Plants against Lightning Effects (Safety standard revision initiated)*	6/92	36 a	23.02.93	-	-	-
2207	Schutz von Kernkraftwerken gegen Hochwasser *Flood Protection for Nuclear Power Plants (Safety standard revision initiated)*	6/92	36 a	23.02.93	6/82	-	+
	2500 Bautechnik Civil Engineering						
2501	Bauwerksabdichtungen von Kernkraftwerken *Waterproofing of Structures of Nuclear Power Plants*	9/88	37 a	22.02.89	-	14.06.94	+
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	-	13.06.95	+
	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	-	13.06.95	+
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 Beilage 23/78	06.10.78	-	20.09.88	+

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(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 Beilage 47/83	-	20.09.88	+
(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 Beilage 24/81	-	11.06.91	+
(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	-	27.06.89	+
(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	-	11.06.91	+
3103	Abschaltsysteme von Leichtwasserreaktoren Shutdown Systems of Light Water Reactors	3/84	145 a 04.08.84 Beilage 39/84	-	14.06.94	+
3104	Ermittlung der Abschaltreaktivität *Determination of the Shutdown Reactivity*	10/79	19 a 29.01.80 Beilage 1/80	-	14.06.94	+
	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*	6/90	53 a 16.03.91 Berichtigung 129 15.07.92 111 17.06.94	2/79; 11/82	-	+
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	-	+

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3201.3	Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 3: Herstellung *Components of the Reactor Coolant Pressure Boundary of Light Water Reactors;* *Part 3: Manufacture*	12/87	92 a 18.05.88	10/79	23.06.92	-
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*	6/90	53 a 16.03.91	6/82	-	+
3203	Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren *Monitoring Radiation Embrittlement of Materials of the Reactor Pressure Vessel of Light Water Reactors*	3/84	119 a 29.06.84 Beilage 33/84	-	13.06.95	+
3204	Reaktordruckbehälter-Einbauten *Reactor Pressure Vessel Internals*	3/84	205 a 27.10.84 Beilage 52/84	-	12.06.90	-
3205.1	Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen; Teil 1: Komponentenstützkonstruktionen mit nichtintegralen Anschlüssen für Primärkreiskomponenten 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* *(Safety standard revision initiated)*	6/91	118 a 30.06.92 Berichtigung 111 17.06.94	6/82	-	+
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	-	13.06.95	+
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	-	14.06.94	+
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*	6/91	118 a 30.06.92	-	11.06.96	-

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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 (Safety standard revision initiated)*	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*	6/90	41 a 28.02.91	-	10.06.97	-
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	-	-	-
	3300 Wärmeabfuhr *Heat Removal*					
3301	Nachwärmeabfuhrsysteme von Leichtwasserreaktoren 2) *Residual Heat Removal Systems of Light Water Reactors*	11/84	40 a 27.02.85	-	14.06.94	+
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	-	13.06.95	+
	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	15.06.93	-
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	13.06.95	+
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	+

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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	11.06.96	-
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	-	14.06.94	+
3403	Kabeldurchführungen im Reaktorsicherheitsbehälter von Kernkraftwerken *Cable Penetrations through the Reactor Containment Vessel*	10/80	44 a 05.03.81 Beilage 6/81	11/76	11.06.96	+
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		15.06.93	+
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 Beilage 27/79	-	14.06.94	+
3407	Rohrdurchführungen durch den Reaktorsicherheitsbehälter *Pipe Penetrations through the Reactor Containment Vessel*	6/91	113 a 23.06.92	-	11.06.96	+
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	-	14.06.94	+
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	-	14.06.94	+
	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	13.06.95	+
3502	Störfallinstrumentierung Incident Instrumentation	11/84	40 a 27.02.85	11/82	27.06.89	+
3503	Typprüfung von elektrischen Baugruppen des Reaktorschutzsystems Type Testing of Electrical Modules for the Reactor Protection System	11/86	93 a 20.05.87	6/82	10.06.97	+

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3504	Elektrische Antriebe des Sicherheitssystems in Kernkraftwerken Electrical Drives of the Safety System in Nuclear Power Plants	9/88	37 a	22.02.89	-	15.06.93	-
3505	Typprüfung von Meßwertgebern und Meß-umformern des Reaktorschutzsystems Type Testing of Measuring Transmitters and Transducers of the Reactor Protection System	11/84	40 a	27.02.85	-	10.06.97	+
3506	Systemprüfung der leittechnischen Einrich-tungen des Sicherheitssystems in Kernkraft-werken Tests and Inspections of the Instrumentation and Control Equipment of the Safety System of Nuclear Power Plants	11/84	40 a	27.02.85	-	10.06.97	+
3507	Werksprüfungen, Prüfungen nach Instand-setzung und Nachweis der Betriebsbewäh-rung für leittechnische Einrichtungen des Sicherheitssystems Factory Tests, Post-Repair Tests and Demonstra-tion of Successful Service for the Instrumentation and Control Equipment of the Safety System	11/86	44 a	05.03.87	-	11.06.96	+
	<u>3600 Aktivitätskontrolle und -führung</u> <u>Activity Control and Activity Management</u>						
3601	Lüftungstechnische Anlagen in Kernkraft-werken *Ventilation and Air Filtration Systems in Nuclear Power Plants*	6/90	41 a	28.02.91	-	13.06.95 1)	-
3602	Lagerung und Handhabung von Brennele-menten, Steuerelementen und Neutronen-quellen in Kernkraftwerken mit Leichtwas-serreaktoren *Storage and Handling of Nuclear Fuel Assemblies, Control Rods and Neutron Sources in Nuclear Power Plants with Light Water Reactors*	6/90	41 a	28.02.91	6/82; 6/84	13.06.95	-
3603	Anlagen zur Behandlung von radioaktiv kontaminiertem Wasser in Kernkraftwerken *Facilities for Treating Radioactively Contami- na-ted Water in Nuclear Power Plants*	6/91	7 a	11.01.92	2/80	11.06.96 1)	+
3604	Lagerung, Handhabung und innerbetrieb-licher Transport radioaktiver Stoffe (mit Aus-nahme von Brennelementen) in Kern-kraftwerken Storaging, Handling and On-Site Transportation of Radioactive Substances (other than Fuel Ele-ments) in Nuclear Power Plants (Safety standard revision initiated)	6/83	194 Beilage 47/83	14.10.83	-	14.06.94	+
3605	Behandlung radioaktiv kontaminierter Gase in Kernkraftwerken mit Leichtwasser-reaktoren Treatment of Radioactively Contaminated Gases in Nuclear Power Plants with Light Water Reactors	6/89	229 a	07.12.89	-	14.06.94	+

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	3700 Energie- und Medienversorgung *Energy and Media Supply*					
3701	Übergeordnete Anforderungen an die elektrische Energieversorgung in Kernkraftwerken General Requirements for the Electrical Power Supply in Nuclear Power Plants	6/97	187 a 08.10.97	KTA 3701.1 (6/78) KTA 3701.2 (6/82)	-	-
3702.1	Notstromerzeugungsanlagen mit Dieselaggregaten in Kernkraftwerken; Teil 1: Auslegung Emergency Power Generating Facilities with Diesel-Generator Units in Nuclear Power Plants; Part 1: Design	6/80	185 a 03.10.80 Beilage 37/80	-	23.06.92	+
3702.2	Notstromerzeugungsanlagen mit Dieselaggregaten in Kernkraftwerken; Teil 2: Prüfungen Emergency Power Generating Facilities with Diesel-Generator Units in Nuclear Power Plants; Part 2: Tests and Inspections	6/91	7 a 11.01.92	11/82	11.06.96	+
3703	Notstromanlagen mit Batterien und Gleichrichtergeräten in Kernkraftwerken Emergency Power Generating Facilities with Batteries and Rectifier Units in Nuclear Power Plants	6/86	162 a 03.09.86	-	23.06.92	+
3704	Notstromanlagen mit Gleichstrom-Wechselstrom-Umformern in Kernkraftwerken Emergency Power Facilities with Rotary Converters and Static Inverters in Nuclear Power Plants	6/84	191 a 09.10.84 Beilage 51/84	-	14.06.94	+
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	9/88	37 a 22.02.89	-	15.06.93	+
	3900 Systeme, sonstige Other Systems					
3901	Kommunikationsmittel für Kernkraftwerke Communication Devices for Nuclear Power Plants	3/81	136 a 28.07.81 Beilage 24/81 Berichtigung 155 22.08.81	3/77	11.06.96	+
3902	Auslegung von Hebezeugen in Kernkraftwerken Lifting Equipment in Nuclear Power Plants (Safety standard revision initiated)	6/92	36 a 23.02.93 Berichtigung 111 17.06.94	11/75; 6/78; 11/83	-	+
3903	Prüfung und Betrieb von Hebezeugen in Kernkraftwerken *Inspection, Testing and Operation of Lifting Equipment in Nuclear Power Plants (Safety standard revision initiated)*	6/93	211 a 09.11.93 Streichung 115 25.06.96	11/82	-	+
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	-	15.06.93	+

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3905	Lastanschlagpunkte an Lasten in Kernkraftwerken Load Attaching Points on Loads in Nuclear Power Plants	6/94	238 a 20.12.94	-	-	+

( ) HTR-Regel, die nicht mehr in die Überprüfung gemäß Abschnitt 5.2 der Verfahrensordnung des KTA einbezogen und nicht mehr über die Carl Heymanns Verlag KG beziehbar ist.
1) In dieser Regel wurden gleichzeitig die HTR-Festlegungen gestrichen.
2) Der KTA hat auf seiner 43. Sitzung am 27.06.89 "Hinweise für den Benutzer der Regel KTA 3301 (11/84)" beschlossen.