Gesellschaft für Anlagen-
und Reaktorsicherheit
(GRS) mbH

# Safety Releated Assessment of the Stendal Nuclear Power Plant, Unit A, of the Type WWER-1000/ W-320

November 1994

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- Ingenieurbüro Eibl, Karlsruhe
- Technischer Überwachungs-Verein Bayern e.V., München
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## Preface

The German Federal Office for Radiation Protection (BfS) commissioned the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH to assess the safety of nuclear power plants, of the WWER-1000/W-320 type, the Stendal Unit A being the reference plant. This safety assessment was conducted jointly in cooperation with the French Institut de Protection et de Sûreté Nucléaire (ISPN). A joint report is currently being prepared by ISPN and GRS. This report will be based on ISPN's own report and GRS's present safety assessment.

Within the framework of cooperation in the fields of reactor safety and radiation protection relating to the present safety assessment, there were consultations with Russian and Ukrainian experts. On the Russian side, the following institutions participated in the safety assessment: Kurchatov Institute for Atomic Energy, Atomenergoprojekt (project engineer), OKB Gidropress (chief designer) and the All-Union Institute for Nuclear Power Plants. On the Ukrainian side, the Ukrainian State Comittee for Reactor Safety and Radiation Protection took part in the consultations.

There was broad agreement between GRS and IPSN on the essential results of their assessments and the upgrading measures proposed. For a number of technical questions, additional investigations are to be performed. For this purpose, further joint projects carried out by GRS and IPSN are intended, in cooperation with partners from the countries where nuclear power plants of the WWER-1000 type are operated or built.

The transferability of the results of this safety assessment to other plants of the WWER-1000 type is restricted by the fact that there is no uniform execution. The same also applies to the transferability of the assessment results to other plants of the WWER-1000/W-320 type, as these plants are also designed differently and are furthermore only partially realized in the reference plant Stendal. The assessment is therefore based on incomplete design documents and evidence of completion. Every transfer of a conclusion drawn from the assessment of the Stendal plant to another reactor therefore requires a thorough examination.

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# 1 Introduction

The Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH conducted safety assessments of nuclear power plants of the WWER-1000/W-320 type on behalf of the German Federal Office for Radiation Protection (BfS), taking the Stendal plant, Unit A, as the reference plant. These assessments determined to what extent the required safety standards and technical codes of the Federal Republic of Germany were met by the engineered safeguards design of the plant.

Four pressurised water reactors of the WWER-1000/W-320 type were being built at the Stendal site. This location for a nuclear power plant of the WWER-1000 type was agreed in 1979. The first construction permit was granted by the former Office for Nuclear Safety and Radiation Protection of the German Democratic Republic in 1982.

After the agreement referring to the creation of a currency, economic and social unit between the Federal Republic of Germany and the German Democratic Republic had come into force on July 1, 1990, the validity of the existing permits was protected for five years. An operating license, however, already would have had to be applied for on the basis of the Atomic Law of the Federal Republic of Germany. Examinations to produce the evidence required for this purpose were started by the plant vendor.

The assessments of the Stendal NPP, Unit A, carried out by GRS were started in January 1991. The objectives were an expert assessment of the safety design of the plant and the identification of upgrading measures to remove safety deficiencies. The assessments were carried out on the basis of the existing safety guidelines, nuclear codes and engineered safeguards practice in the Federal Republic of Germany.

At the beginning of 1991, the construction of the project was suspended as no operator was prepared to continue the licensing procedure. The investigations conducted by GRS were nevertheless continued, to present an independent safety assessment of the nuclear power plants of the WWER-1000 type according to Western requirements, similar to the previous investigations for nuclear power plants of the WWER-440 type at Greifswald. Plants of this type are being built or operated in several countries in Middle and Eastern Europe, respectively (see Table 1-1).

As documents available were incomplete, only a restricted expert assessment of the Stendal plant was possible. For a final assessment of WWER-1000 type nuclear power plants it will be necessary to produce further evidence.

With respect to accident management measures no investigations were carried out.

In the course of these investigations, GRS awarded a number of subcontracts to other institutions:

- Hosser, Haß + Partner
  Ingenieurgesellschaft für Bauwesen und Brandschutz mbH
  Braunschweig

- Ingenieurbüro Eibl
  Karlsruhe

- Technischer Überwachungs-Verein Bayern e.V.
  München

- Technischer Überwachungs-Verein Norddeutschland e.V.
  Hamburg

The investigations were further supported by Kraftwerks- und Anlagenbau AG (K.A.B.), Energiewerke Nord AG and Bauakademie Berlin.

In the course of the investigations of the WWER reactors, GRS cooperated with various foreign institutions. In particular, there was close cooperation with the French Institut de Protection et de Sûreté Nucléaire (IPSN), Paris, so that, technical expert discussions on various topics took place between GRS and IPSN in the course of the investigations of the Stendal nuclear power plant.

The extent to which nuclear power plants of the WWER-1000/W-320 type fulfil the requirements of the French regulating body for nuclear power plants was examined by IPSN. A report on the results of the examination is being prepared by IPSN /DES

92/. In addition, a joint GRS/IPSN report combining the results of the German and French investigations will be prepared.

The German results of the investigations of Stendal A were passed on to the Kurchatov Institute. The essential points were discussed with French, Russian and Ukrainian experts.

The terminology of the former Soviet Union (SU) and the former German Democratic Republic (GDR) is used to describe buildings, systems and components. For a better understanding terms of the terminology used in the Federal Republic of Germany were added in brackets. It is to be noted that owing to the different design of the WWER plants and nuclear power plants in the Federal Republic of Germany, a clear assignment of the terms used is not always possible. Furthermore, in some cases the system limits for nuclear power plants in the Federal Republic of Germany differ from WWER plants so that the Federal German synonyms only vaguely describe the state of affairs.

Appendix 1 shows a list of terms used in the Federal Republic of Germany as opposed to the terms used in the SU/GDR. Terms in brackets are also contained in the documents on which this list is based. These are, however, not used in the present analysis.

To explain the contents of the technical sections (Sections 4 to 8) the technical features of the installations and systems of the Stendal Nuclear Power Plant, Unit A, are described in Section 2, where the most important engineered safeguards are referred to.

In Section 3, an overview is given of the most important German codes and regulations in the field of nuclear engineered safeguards and their application to the Stendal plant.

Sections 4 to 8 summarize and assess the results of the technical investigations. Section 4 contains an assessment of the core design and the pressurised components. Section 5 deals with accident studies, including analyses of the effectiveness of engineered safeguards, and calculations of the radiological consequences of accidents. In Section 6 the safety system is analysed. Section 7 features findings referring to civil engineering aspects, the spread of impacts and radiological protection of the workers. Section 8 provides a summarized evaluation of the operating experience of WWER-1000 type nuclear power plants being operated.

In Section 9 the results of the investigations are summarized. Sections 10 contains the up-grading measures derived from the investigations and the recommendations for further investigations.

## References, Section 1


/DES 92/ Institut de Protection et de Sûreté Nucléaire (IPSN),Department d'Evaluation de Sûreté (DES),
A Partial Assessment of the Safety of Stendal WWER-1000/320DES Report No. 74

**Table 1-1 WWER-1000 Type Reactor (Status at July 1992)**

| Location | Unit No. | Type | Status | Completion | Start of Construction | 1st Synchronisation with grid |
|---|---|---|---|---|---|---|
| **Bulgaria** | | | | | | |
| Belene | 1 | W-320 | construction stopped | 40 % | 1984 | |
| Belene | 2 | W-320 | construction stopped | 10 % | 1986 | |
| Belene | 3 | | design stopped | | | |
| Belene | 4 | | design stopped | | | |
| Kosloduj | 5 | W-320 | in operation | | 1980 | 11/87 |
| Kosloduj | 6 | W-320 | in operation | | 1984 | 03/89 |
| **CSFR** | | | | | | |
| Temelin | 1 | W-320 | in construction | 50 % | 1982 | (1994) |
| Temelin | 2 | W-320 | in construction | 10 % | 1985 | (1996) |
| **Germany** | | | | | | |
| Stendal | 1 | W-320 | construction stopped in 1991 | 40 % | 1982 | |
| Stendal | 2 | W-320 | construction stopped in 1991 | 15 % | 1982 | |
| **Hungary** | | | | | | |
| Paks | 5 | | being planned | | | |
| Paks | 6 | | being planned | | | |
| **Poland** | | | | | | |
| Klempicz | 1 - 2 | | design stopped | | | |
| Kujawy | 1 - 4 | | design stopped | | | |
| Samter | 1 - 4 | | design stopped | | | |
| Warta | 1 - 4 | | design stopped | | | |
| **Russia** | | | | | | |
| Balachovo | 1 | W-320 | in operation | | 1980 | 12/85 |
| Balachovo | 2 | W-320 | in operation | | 1981 | 10/87 |
| Balachovo | 3 | W-320 | in operation | | 1982 | 12/88 |
| Balachovo | 4 | W-320 | in construction | 98 % | 1984 | (1992) |

( ) planned

| Location | Unit No. | Type | Status | Completion | Start of Construction | 1st Synchronisation with grid |
|---|---|---|---|---|---|---|
| Balachovo | 5 | W-320 | in construction (preserved) | 75 % | 1987 | |
| Balachovo | 6 | W-320 | in construction (preserved) | 25 % | 1988 | |
| **Russia** | | | | | | |
| Bashkiria | 1 | W-320 | construction stopped in 1990 | | 1983 | |
| Bashkiria | 2 | W-320 | construction stopped in 1990 | | 1983 | |
| Bashkiria | 3 | W-320 | design stopped | | | |
| Bashkiria | 4 | W-320 | design stopped | | | |
| Kalinin | 1 | W-338 | in operation | | 1977 | 05/84 |
| Kalinin | 2 | W-338 | in operation | | 1982 | 12/86 |
| Kalinin | 3 | W-320 | in construction | 70 % | 1985 | (1992) |
| Kalinin | 4 | W-320 | in construction | 10 % | 1986 | (1995 |
| Kola | 1 | W-392 | being planned | | | |
| Kola | 2 | W-392 | being planned | | | |
| Kostroma | 1 | | construction not started | | | |
| Kostroma | 2 | | construction not started | | | |
| Novo-Voronesh | 5 | W-187 | in operation | | 1974 | 05/80 |
| Novo-Voronesh | 6 | W-392 | being planned | | | |
| Novo-Voronesh | 7 | W-392 | being planned | | | |
| Rostov | 1 | W-320 | construction stopped in 1991 | 90 % | 1981 | |
| Rostov | 2 | W-320 | construction stopped in 1991 | 10 % | 1983 | |
| Rostov | 3 | W-320 | construction stopped in 1991 | | 1989 | |
| Rostov | 4 | W-320 | being planned | | | |
| Rostov | 5 | | design stopped | | | |
| Rostov | 6 | | design stopped | | | |
| Tartary | 1 | W-320 | construction stopped | | 1987 | |
| Tartary | 2 | W-320 | construction stopped | | 1988 | |

( ) planned

| Location | Unit No. | Type | Status | Com-pletion | Start of Con-struc-tion | 1st Syn-chroni-sation with grid |
|---|---|---|---|---|---|---|
| **Russia** | | | | | | |
| Tartary | 3 | | design stopped | | | |
| Tartary | 4 | | design stopped | | | |
| Tartary | 5 | | design stopped | | | |
| Tartary | 6 | | design stopped | | | |
| **Ukraine** | | | | | | |
| Kmelnitzki | 1 | W-320 | in operation | | 1981 | 12/87 |
| Kmelnitzki | 2 | W-320 | in construction | 90 % | 1985 | (1992) |
| Kmelnitzki | 3 | W-320 | in construction | | 1986 | (1996) |
| Kmelnitzki | 4 | W-320 | in construction | | 1987 | (1996) |
| Crimea | 1 | W-320 | construction stopped in 1986 | | | |
| Crimea | 2 | W-320 | design stopped | | | |
| Rovno | 3 | W-320 | in operation | | 1981 | 12/86 |
| Rovno | 4 | W-320 | in construction | 80 % | 1986 | (1993) |
| Rovno | 5 | | being planned | | | |
| Saporoshje | 1 | W-320 | in operation | | 1980 | 12/84 |
| Saporoshje | 2 | W-320 | in operation | | 1981 | 07/85 |
| Saporoshje | 3 | W-320 | in operation | | 1982 | 12/86 |
| Saporoshje | 4 | W-320 | in operation | | 1984 | 12/87 |
| Saporoshje | 5 | W-320 | in operation | | 1984 | 08/89 |
| Saporoshje | 6 | W-320 | in construction | 95 % | 1986 | (1992) |
| South Ukraine | 1 | W-302 | in operation | | 1977 | 12/82 |
| South Ukraine | 2 | W-338 | in operation | | 1979 | 01/85 |
| South Ukraine | 3 | W-320 | in operation | | 1985 | 09/89 |
| South Ukraine | 4 | W-320 | in construction | 50 % | 1987 | (1993) |

( ) planned

# 2 Description of the Nuclear Power Plant

## 2.1 Location, Layout and Design of Buildings

The Stendal nuclear power plant is located on the left bank of the River Elbe, 15 km north-east of the town of Stendal. It is designed for four parallel units of the WWER-1000/W-320 type with their main axes in the east/west direction. The level of the nuclear power plant is about 10 m above the mean level of the River Elbe. The prevailing wind is from the west.

The layout of the buildings can be derived from Fig. 2.1-1 and 2.1-2. Unit A of Stendal NPP is a single unit. The main facilities are installed in the reactor building as well as in the turbine building, with its extension for the electrical switchboard plant. The emergency diesel buildings are arranged with physical separation, east and west of the reactor building. They contain train-related emergency power plants and the pumps for the service cooling water system A. In the north, there are the three buildings for the diesel generators of the unit. The service cooling water A is cooled down in three spray ponds, east of unit A. The natural draught cooling towers of the circulating cooling water supply of unit A are located east of the turbine hall. South of the reactor building are the radioactive service buildings. The electrical output is directed to the Schwarzholz substation 3.5 km away in a north-westerly direction.

The design of the building depends on the systems located in it, the accident-dependent loads, as well as the requirements of nuclear safety and radiation protection. The systems installed in the buildings are subdivided into operational facilities and engineered safeguards. In the Technical Project /TEP 81/ the operational facilities are subdivided into four groups according to the consequences following their breakdown or failure, respectively:

- Group 1.1 - after breakdown the ambient load, despite regular functioning of the engineered safeguards, is exceeded (e.g. reactor pressure vessel); this group is subject to increased requirements to be met by quality assurance and in-service testing.

- Group 1.2 - after breakdown the boundaries of safe operation are exceeded (accident).

- Group 1.3 - the power plant equipment is damaged by breakdown.
- Group 1.4 - breakdown has no immediate consequences for the safety of the nuclear power station.

The Stendal Nuclear Power Plant is designed against the effects of earthquakes, pressure waves and airplane crash.

Referring to the effects of earthquakes, the buildings of the Stendal Nuclear Power Plant are designed in accordance with the Russian design guidelines, which also correspond to the general technical regulations of the Comecon "Norms of Designing Earthquake-proof Nuclear Energy Plants" (NTD 04.01.50). They are subdivided into three groups:

- Category I: Maximum calculated earthquake, corresponding to a vibration of the maximum intensity at the location over a period of 10,000 years.
- Category II: Design earthquake, corresponding to a vibration of the maximum intensity at the location over a period of 100 years or as predetermined nationally.
- No consideration of earthquake loads.

The buildings of the Stendal NPP were designed against external events on the basis of the following parametres:

| | |
|---|---|
| - Maximum calculated earthquake (category I) | Intensity 7 on the MSK-64-scale |
| - Design earthquake (category II) | Intensity 5 on the MSK-64-scale |
| - Pressure wave | |
| • Excess pressure at the wave front | 0.03 MPa |
| • Maximum excess pressure of the reflected wave when colliding with an obstacle | 0.067 MP |
| • Dynamic correction value to determine statistical pressure for even wall surfaces | 1.7 |
| • Overall impact time of the pressure wave up to | 1 s |

- Airplane crash
  - Weight of the airplane 10 Mg
  - Impact speed 750 km/h
  - Impact area 7 $m^2$.

The buildings are designed as follows:

- design against earthquakes/category I and pressure wave
  - reactor building
  - exhaust stack
  - emergency power building
  - building for service cooling water system A
  - intermediate store for radioactive residues
  - store for fresh fuel elements
  - spray pond
- design against earthquakes/category II
  - turbine hall
  - extension for electrotechnical purposes
  - extension for ventilation
- design against airplane crash
  - structural design: only cupola of the containment
  - protection by surrounding walls: cylindrical part of the containment
  - protection by layout: The safety installations outside the containment are designed with threefold redundancy (3 x 100 %) and their layout is such that there is at least one redundant safety installation after an airplane crash.

## 2.2 Reactor Plant and Main Equipment of the Primary System

The reactor type WWER-1000/W-320 is a lightwater-moderated and lightwater-cooled pressurised water reactor with an electrical capacity of 1000 MW. Its location in the reactor building is shown on Fig. 2.2-1 and 2.2-2.

The main parameters of the reactor plant are:

| | | |
|---|---|---|
| thermal power | 3000 | MW |
| pressure at the reactor core | 15.7 | MPa |
| nominal pressure (calculated pressure) | 17.7 | MPa |
| coolant flow rate through the reactor | 84 800 | $m^3/h$ |
| coolant temperature at reactor core inlet | 289.8 | °C |
| coolant temperature at reactor core outlet | 320.1 | °C |

The reactor pressure vessel is made of a low-alloy ferretic steel and lined with a 7 to 8 mm austenitic coating.

The primary system consists of four main coolant loops with one steam generator and one main coolant pump (Fig. 2.2-3) each. The connecting primary main coolant pipes (DN 850) consist of low-alloy perlite steel with a 5 mm internal coating of austenite.

The steam generator (SG), of the type PGW 1000, is a horizontal cylindrical shell with horizontal austenitic heating tubes (11000 tubes) in the form of a double U-shaped bank of tubes. The overall heat exchanger suface is 6115 $m^2$. The primary coolant enters and leaves the steam generators from below via two collectors. The steam collector above the steam generator is connected with the steam plenum of the steam generator through 2 x 5 nozzles. The feedwater for the secondary side of the steam generator is supplied through a pipe of DN 400. The supply of emergency feedwater (DN 150) is connected separately. At the steam generator there are connecting branches for continuous and periodic desalination.

The main parameters of a steam generator are:

| | | |
|---|---|---|
| thermal power | 750 | MW |
| nominal steam flow rate | 408 | kg/s |
| maximum steam flow rate permitted | 437 | kg/s |
| steam pressure | 6.3 | MPa |
| feedwater temperature | 220 | °C |
| temperature of the emergency feedwater | 5-50 | °C |

The main coolant pump (MCP), of the GZN-195M type, is a vertical single-stage pump. It consists of the hydraulic part of the pump, the detachable electric motor of the type WAS 215/109-6AM05 and the auxiliary systems. An additional balance weight at the electric motor ensures a slow decrease of coolant flow in case of power failure.

The main parameters of a main coolant pump are:

| | | |
|---|---|---|
| pump capacity | 20000 - 27000 | $m^3/h$ |
| head of pump | 0.74 - 0.54 | MPa |
| pressure on suction side | 15.3 | MPa |
| coolant temperature | 290 | °C |
| power draw during normal operation | 5.3 | MW |

The pressuriser (P) is connected to the hot leg of one loop via a connecting pipe of DN 350. The pressuriser spray line is connected to the cold leg of a loop. Coolant from the make-up system (volume control and coolant cleaning system) can be sprayed directly into the steam space of the pressuriser through an auxiliary spray line. Electric heaters are used to increase and maintain the pressuriser pressure. The pressuriser is equipped with three safety valves which blowdown into a relief tank, protected by a rupture membrane against excess pressure. Lockable relief valves are not provided.

The main parameters of the pressuriser are:

| | | |
|---|---|---|
| overall volume | 79 | $m^3$ |
| water volume during nominal operation | 55 | $m^3$ |
| overall capacity of electric heating | 520 | kW |

The make-up system to guarantee water quality, the drainage system (collection and feedback of leakages and drainages), parts of the special water treatment, the steam generator desalination system and the exhaust system are essential auxiliary systems of the primary system.

## 2.3 The Secondary System

The secondary system is shown on Fig. 2.3-1 as an elementary diagram. It can be divided into the feedwater system, the main steam system, the turbogenerator and the condensate system.

Two feedwater tanks (working pressure 0.658 MPa) and two speed controlled turbo injection pumps belong to the main feedwater system. Two electrically operated auxiliary injection pumps for startup and shutdown belong to the auxiliary feedwater system. The feedwater tank is connected with the two turbo injection pumps and the two auxiliary injection pumps via a collector. The feedwater temperature is 220 °C.

The four steam generators, the main steam pipes, the engineered safeguards and control facilities, as well as the 1000 MW-turbogenerator, constitute to the main steam system. Within the containment, the four main steam pipes (DN 600) are routed to the turbine control valves and fact-acting isolating valves by two physically separated routes. Outside the containment, the four main steam pipes converge into one common route. The main steam lines are connected with each other for pressure equalisation. In each main steam line, between the steam generator and the fast-acting main steam isolation valves, there are two medium-operated safety valves (100% each), each with one control valve, and an atmospheric steam dump station (relief valves) BRU-A (flow rate 900 t/h). Downstream of the fact-acting isolating valve and one check valve in each main steam line, there are connections to the four-train steam bypass system BRU-K (opening pressure 6.67 MPa, closing pressure 5.98 MPa, flow rate 900 t/h each) and to the two-train station service reduction subsystem, BRU-SN, (flow rate 150 t/h each). The steam bypass system ist used during unit startup and shutdown to dump excess steam from the steam generators into the turbine condensers. The station service reduction subsystem, BRU-SN, can be used for residual heat removal after reactor scram, the steam relieved being cooled by two speparate so-called technological condensers.

The turbogeneator consists of the turbine, of the K-1000-60/3000 type, and the generator of the TBB-1000-2 type. It has the following main parameters:

| | | |
|---|---|---|
| generator putput | 1000 | MWe |
| maximum main steam flow rate | 5870 | t/h |
| steam pressure at turbine inlet | 5.89 | MPa |
| steam temperature at turbine inlet | 274.3 | °C |

The main condensate system serves for discharging the condensate from the condensers via the main condensate clean-up system and five low pressure preheaters into the deaerator/feedwater tanks. The condensate route has two pressure levels. The first pressure level is reached by three condensate pumps (two operational pumps and one standby pump; working pressure 0.93 MPa) and the second pressure level by five condensate pumps (four operational pumps and one standby pump; working pressure 2.15 MPa), located downstream of the second low pressure preheater.

## 2.4 Cooling Water Systems

Two natural draught cooling towers are assigned to the 1000-MW-turbogenerator as the main heat sink of the Stendal NPP. The cooling water is carried via tubes (DN 2600), with the help of four circulating cooling water pumps, from the cooling towers to the condensers in the turbine hall and back to the cooling towers. The main parameters of the turbine condensers (on the cooling water side) are:

| | | |
|---|---|---|
| flow rate of circulating cooling water | 170000 | $m^3/h$ |
| cooling surface | 8800 | $m^2$ |

Because of the poor water quality of the River Elbe and the circulating cooling water, a component cooling system (Group "C") for the users of the secondary system is provided. The component cooling system is cooled by the circulating cooling water via a unit-related heat exchanger system, positioned next to the turbine hall. Additionally, a central heat exchanger system for central users, especially in the area of the radioactive service buildings, is provided. The heat exchangers are fed by water from the River Elbe, which (after treatment) is partially used as additional water for the recooling system. The service cooling water system A supplies cooling positions of the operational and safety installations. The recooling of the service cooling water is achieved by spray ponds, the additional water for which is taken from the River Havel.

## 2.5 Engineered Safeguards Design

In the following section the objectives of the engineered safeguards design of the Stendal NPP, as stated in the safety volume of the Technical Project (status of 1981) /TEP 81/ by the Soviet project engineer, are described.

The design basis for the Stendal NPP was the Soviet guideline "General Safety Principles of Nuclear Power Plants during Design, Construction and Operation" /OPB-73/ which provides a multi-stage system of safety precautions.

According to this guideline, quality assurance during design, manufacture, erection, start of operation and operation shall represent the first stage of the safety precautions. The second stage shall comprise the technical installations and the organizational measures, compensating deviations from the intended operation. The third step of the safety precautions shall be the equipment of the nuclear safety installations.

The Soviet guideline OPB-73 and its subsequent versions claim the achievement of the protective aims of sub-criticality, core cooling and long-term residual heat removal, as well as enclosing radioactive materials according to the barrier principle. The barriers for retention of the radioactive fission products are the fuel matrix, fuel rod cladding, the reactor pressure vessel and pressurised enclosure as well as the full pressure containment.

The following accidents and initiating events were considered for the engineered safeguards design of the Stendal NPP:

- loss-of-coolant accidents
- transient accidents, like, for example, secondary side leakages (break of main steam line break), loss of off-site power, reactivity accidents
- External events initiating off accidents, e.g. earthquakes, airplane crashes, etc.

Essential parts of the safety system are located in the reactor building outside the containment, where the protection against the loads of an airplane crash shall be guaranteed by a physical separation (cf. section 2.1).

The maximum design basis accident is the spontaneous break of the primary coolant pipe with coolant escaping on both sides, assuming loss of off-site power. Loss-of-coolant accidents, according to OPB-73, are considered to be controlled, if the fuel-rod-cladding temperature is < 1200 °C, the oxidation depth < 18 % of the initial thickness of the cladding tubes and the proportion of the reacting zirconium <1% by mass of the claddings.

There is a three-train design of the safety installations, each train having a capacity of 100 %. The trains are largely independent and physically separated. Each train of the engineered safeguards is supplied by its own emergency power supply. The unit station service is supplied by a fourth diesel generator. Instrumentation and control (I & C) are designed redundantly, partially with electronic modules and partially with relay connections with an open-circuit mode. I & C are subdivided into the emergency protection system (reactor protection system - initiation of the reactor scram) and the protection system for the control of the safety system (reactor protection system without initiation of the reactor scram). To initiate the reactor scram, the emergency protection system is subdivided into two independent trains, each with three channels in 2-out-of-3 selection mode. In the protection system, each process train of the safety installations is provided with an independent instrumentation and control train working with 2-out-of-4 selection mode on the activation level and with 1-out-of-2 selection mode on the logic level.

The systems in the containment are protected against mechanical loads due to pipe ruptures, such as, jetforces, pressure waves and flying parts. Containment integrity during accidents is ensured by the isolating valves of the building.

The most important safety installations are summarized in Table 2.5-1. The safety installations are described and examined in Section 6.

## 2.6 Electrical Energy Supply

Unit A is connected to the supply system via the 220/380 kV switching centre of the Schwarzholz substation about 3.5 km north-west of the power plant. The generator output is exported via a power breaker and two unit transformers. Each of the two unit transformers is designed for 750 MVA. The station service supply is provided from four 6-kV unit distributions fed by two station service transformers. In addition, there are two standby transformers. A general station service supply, to which a station service diesel is assigend per unit, is provided for the supply of the service buildings and the auxiliary systems. Each of the three trains of the safety system is supplied by an independent emergency power system.

## 2.7 Remarks on the Concept for Controlling External Impacts

In section 2.1, the design concept for controlling the influence of external events is described. Apart from the statements listed there, no further information is known, especially with respect to plant measures for controlling the effects of external events including conesquential damage. A final evaluation of the protection against external events is therefore not possible. For this reason the presentation of a consistent concept for controlling external events is considered necessary (R 2.7-1).

The concept of protection against loads caused by an airplane crash by physicial separation within a building (cf. Section 2.5) needs to be supported by relevant evidence for its effectiveness. For the respective safety installations in the reactor building outside the containment, it has to be proved that they are not damaged to an inadmissible degree by an airplane crash, especially by the resultant vibrations (R 2.7-2).

**References, Section 2**


/OPB 73/ Ministry of Energy Management and Electrification of the USSR, General Guideline for Ensuring Safety of Nuclear Energy Plants during Design, Construction and Operation, OPB-73

/TEP 81/ Teploenergoproject, KKAB et al., NPP Stendal, Technical Project (Status 1981)

**Table 2.5.-1** **Engineered Safeguards of the Stendal NPP**

| **Engineered Safeguards** | | |
|---|---|---|
| **Protective Aim/ Safety Function** | **Degree of Redundancy** | **Components per Train** |
| **1. Reactor Scram System** | | |
| Subcriticality | Control rods I & C:<br>1 x 100 %<br>control and protection I & C<br>2 x (2 of 3) | control rods with drives<br><br>initiation level<br>logic level, control level |
| **2. HP Emergency Boron Injection System (Additional Boron Treatment System)** | | |
| Subcriticality | 3 x 100 % | emergency storage tank (15 $m^3$)<br>HP-emergency boron injection pump (15.7 MPa, 6.3 $m^3$/h) |
| **3. HP-Emergency Core Cooling System (HP-Injection System)** | | |
| Core Cooling, Subcriticality | 3 x 100 % | storage tank (15 $m^3$)<br>HP-emergency cooling pump<br>(10.8 MPa, 160 $m^3$/h)<br>switch-over to recirculation mode common 630 $m^3$ emergency boron tank for three trains emergency cooler |
| **4. LP-Emergency Core Cooling System (LP-Injection System)** | | |
| Core Cooling | 3 x 100 % | LP-emergency cooling pump (2.25 MPa, 750 $m^3$/h)<br>common 630 $m^3$ emergency boron tank<br>(identical with Item 3)<br>emergency cooler<br>(identical with Item 3) |
| **5. Core Flooding System (Accumulator)** | | |
| Core Cooling, Subcriticality | 2 x 100 % | 2 core flooding tanks<br>(60 $m^3$ each) |

**Table 2.5.-1 Engineered Safeguards of the Stendal NPP**

| Engineered Safeguards | | |
|---|---|---|
| **Protective Aim/ Safety Function** | **Degree of Redundancy** | **Components per Train** |
| **6. Containment-Spray System** | | |
| Activity retention, pressure suppression in the containment | 3 x 100 % | containment spray pump, (1.5 - 0.75 MPa, 210 - 975 $m^3$/h) common 630 $m^3$ emergency boroncontainment tank (identical with Item 3) |
| **7. Pressuriser Safety Valves** | | |
| Pressure protection primary system | 3 x 100 % | valve 1 (50 kg/s, 17.9 MP) valve 2 and 3 (50 kg/s, each 18.3 MPa) |
| **8. BRU-A (Atmospheric Main-Steam Dump Station)** | | |
| Core cooling residual heat removal | 4 x 100 % | BRU-A (opening 6.67 MPa, closing 5.98 MPa, flow rate 900t/h) |
| **9. Steam Generator Safety Valves** | | |
| Core cooling, residual heat removal, pressure protection secondary system | 4 x (2 x 100 % ) | 1st safety valve (opening 8.34 MPa, closing 6.97 MPa, flow rate 900 t/h) 2nd safety valve (opening 8.44 MPa,closing 6.97 MPa, flow rate 900 t/h) |
| **10. Main-Steam Shutoff Valves (SSA)** | | |
| Protection against under-cooling of the primary system and recriticality | | 1 x in every main steam line |
| **11. Emergency-Feedwater System** | | |
| Residual heat removal | 3 x 100 % | emergency feedwater pump (9.56 MPa,150 $m^3$/h) emergency feedwater tank (3 x 500 $m^3$) |
| **12. Emergency Power Diesel System** | | |
| Emergency power supply | 3 x 100 % | emergency power diesel (6.2 MW, run-up time 10 s) |

## Figures, Section 2

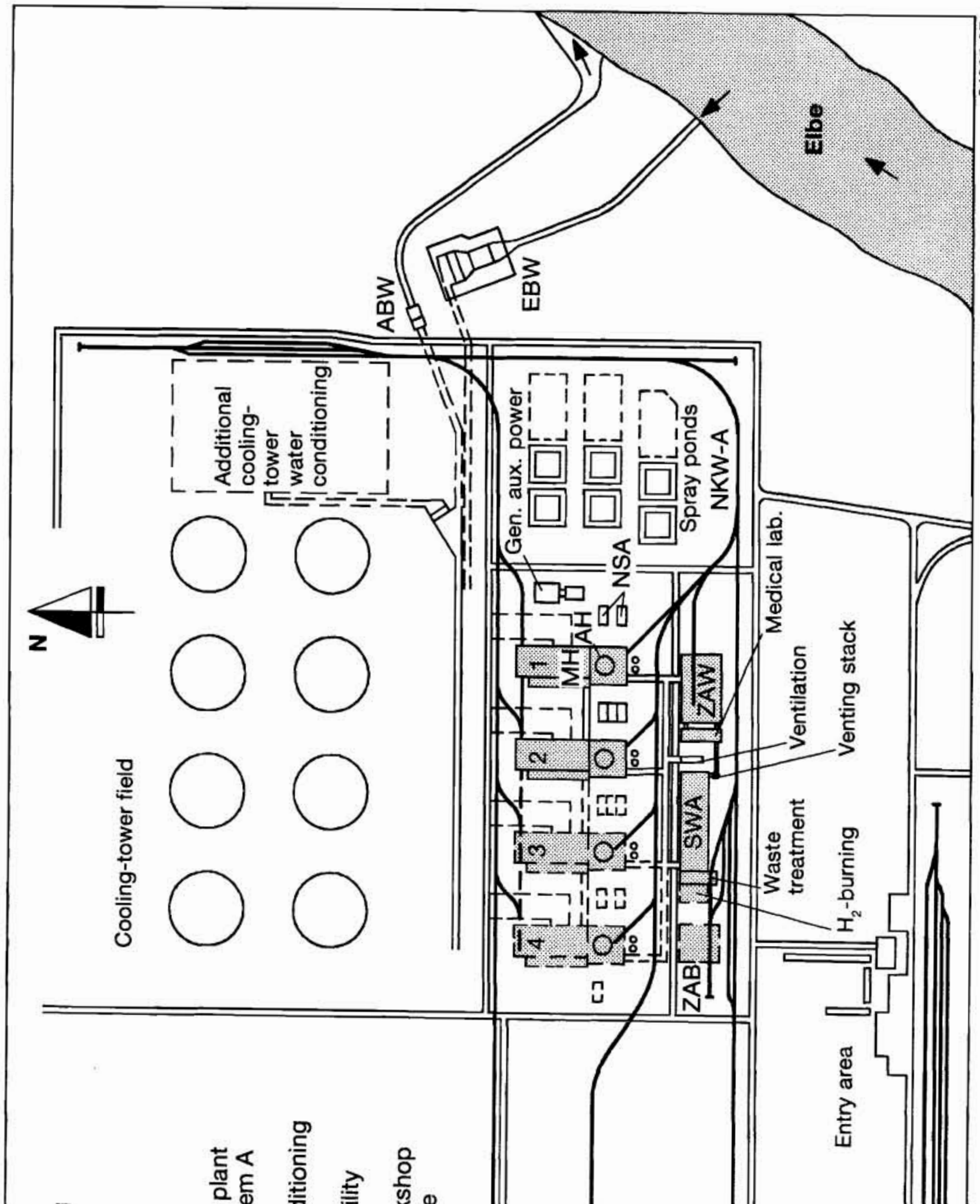



**Fig. 2.1-1** Site plan, scale 1:15000

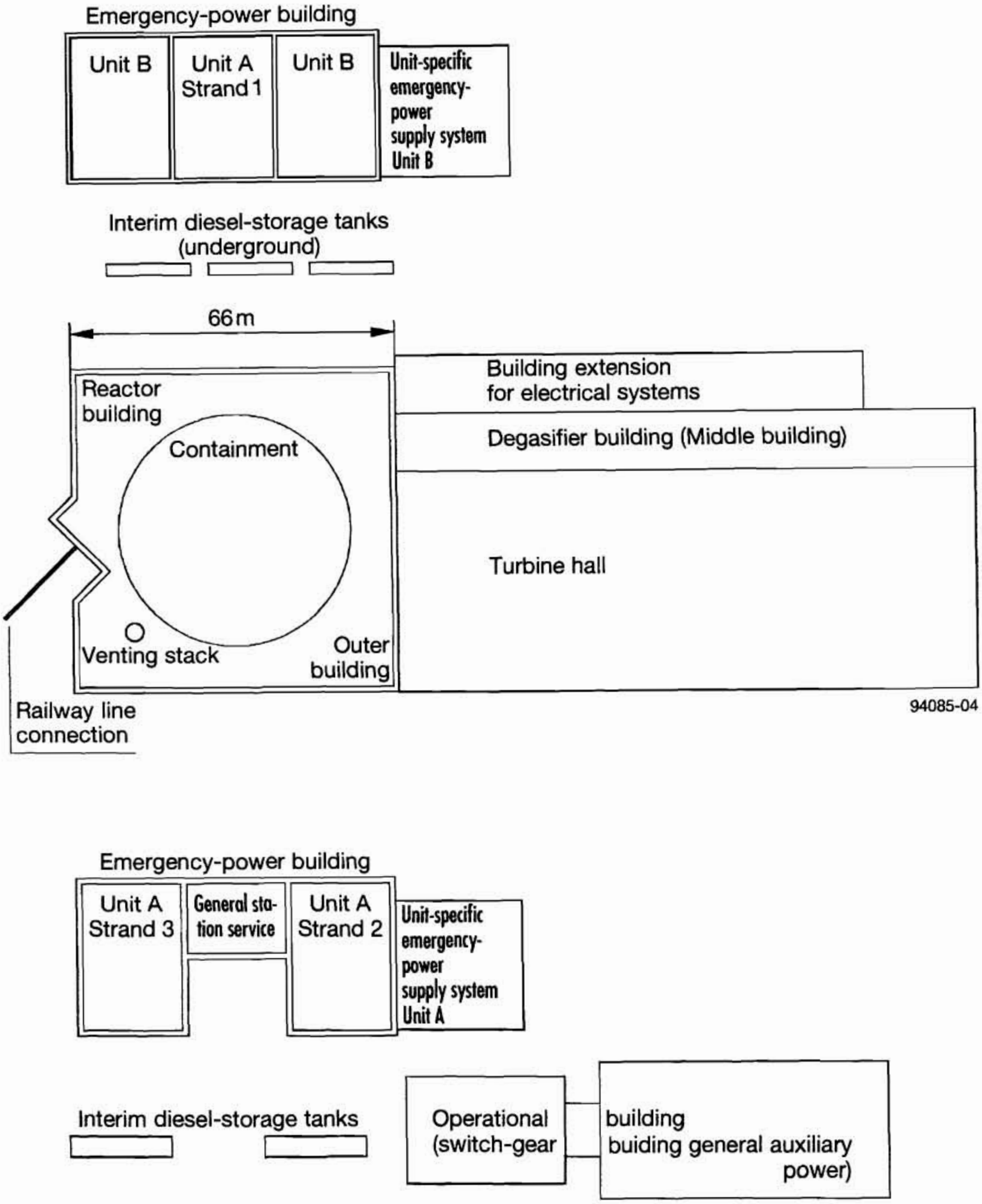
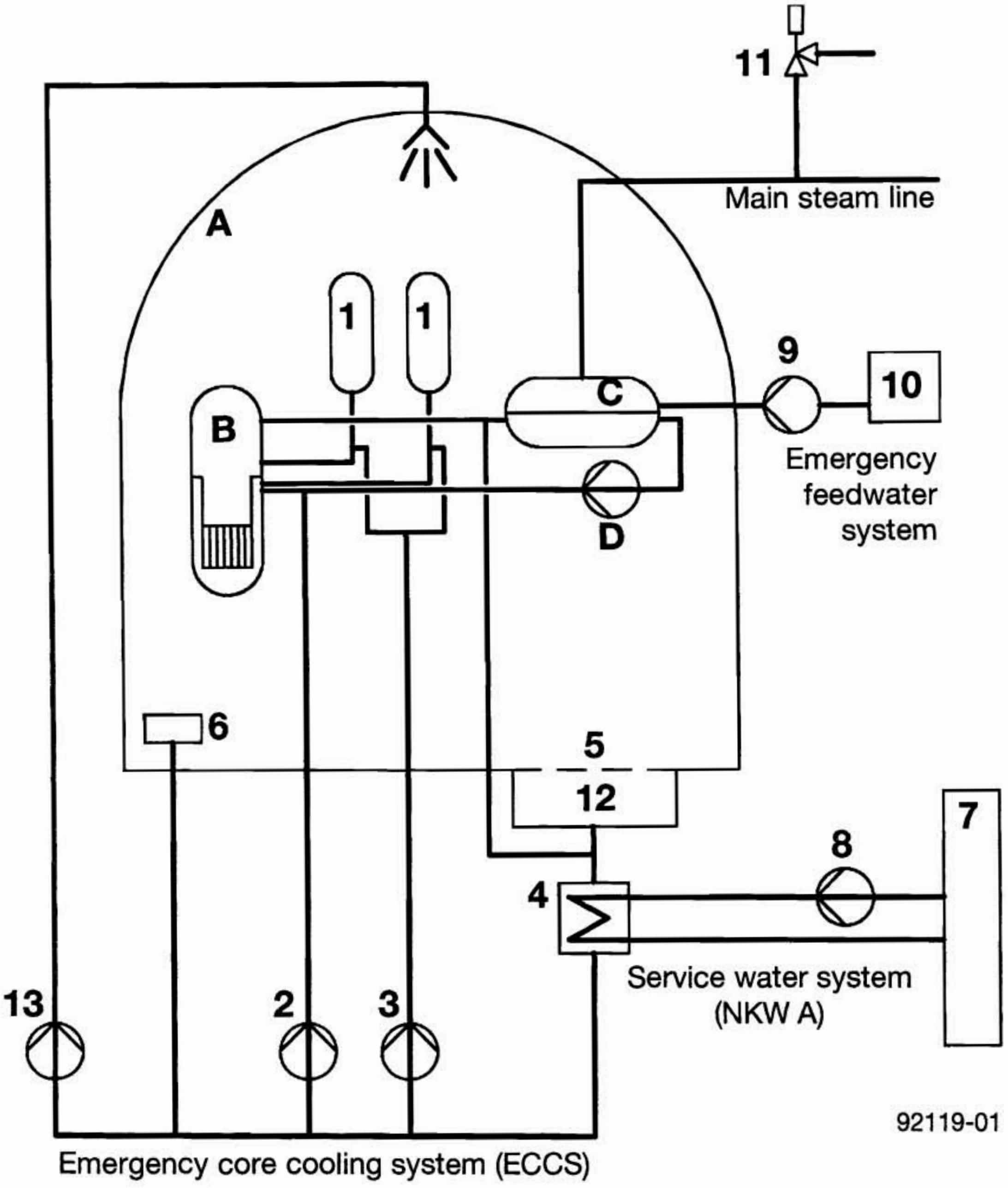



**Fig. 2.1-2** Layout Unit A

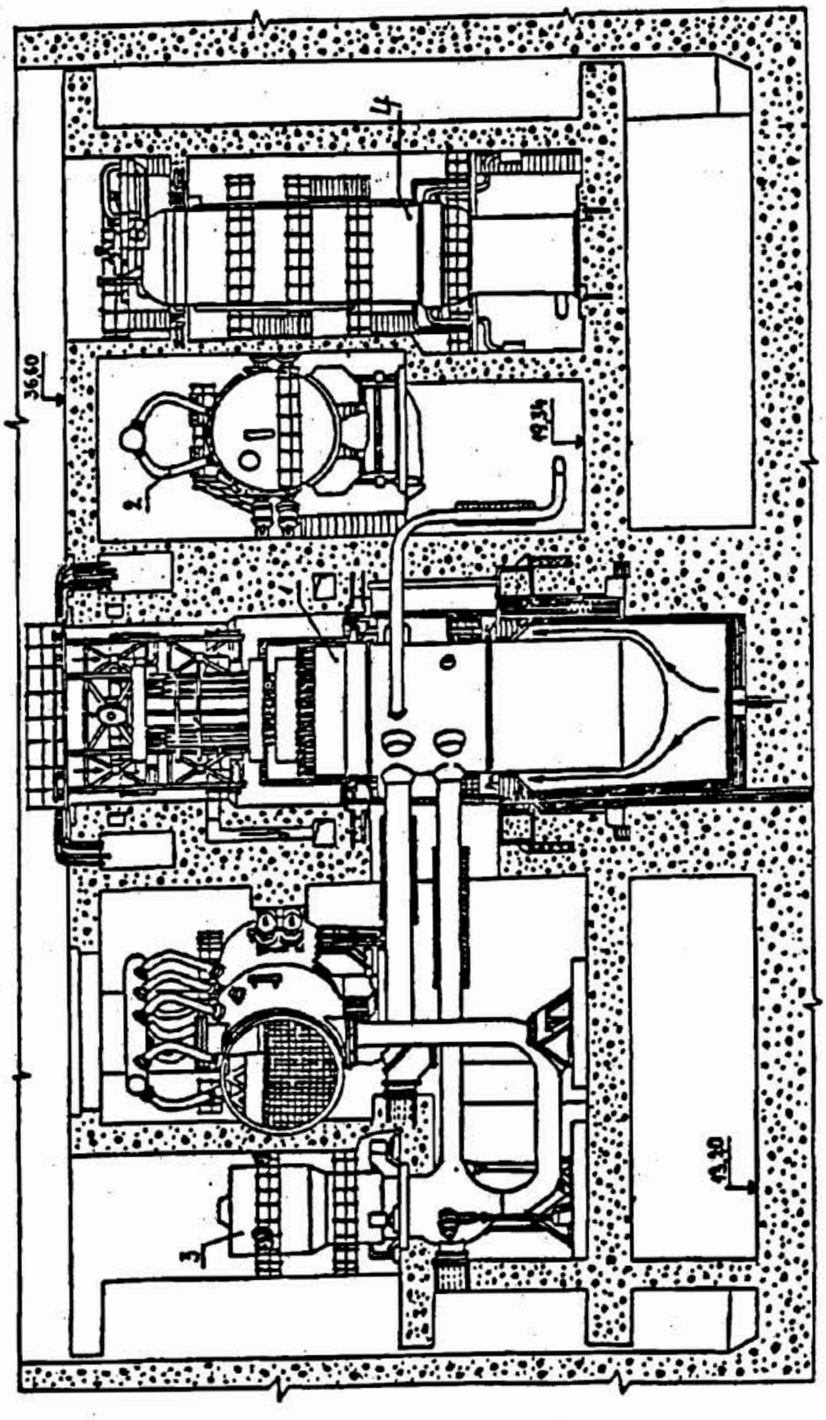



**1** Accumulator
**2** High pressure safety injection pump
**3** Low pressure safety injection pump
**4** Emergency cooler
**5** Containment sump
**6** Storage tank
for concentrated boric acid
**7** Cooling pond
**8** Service water pump
**9** Emergency feedwater pump

**10** Emergency feedwater tank
**11** Steam dump station
(into atmosphere)
**12** Boric acid storage tank
**13** Containment spray pump

**A** Containment
**B** Reactor pressure vessel
**C** Steam generator
**D** Main coolant pump

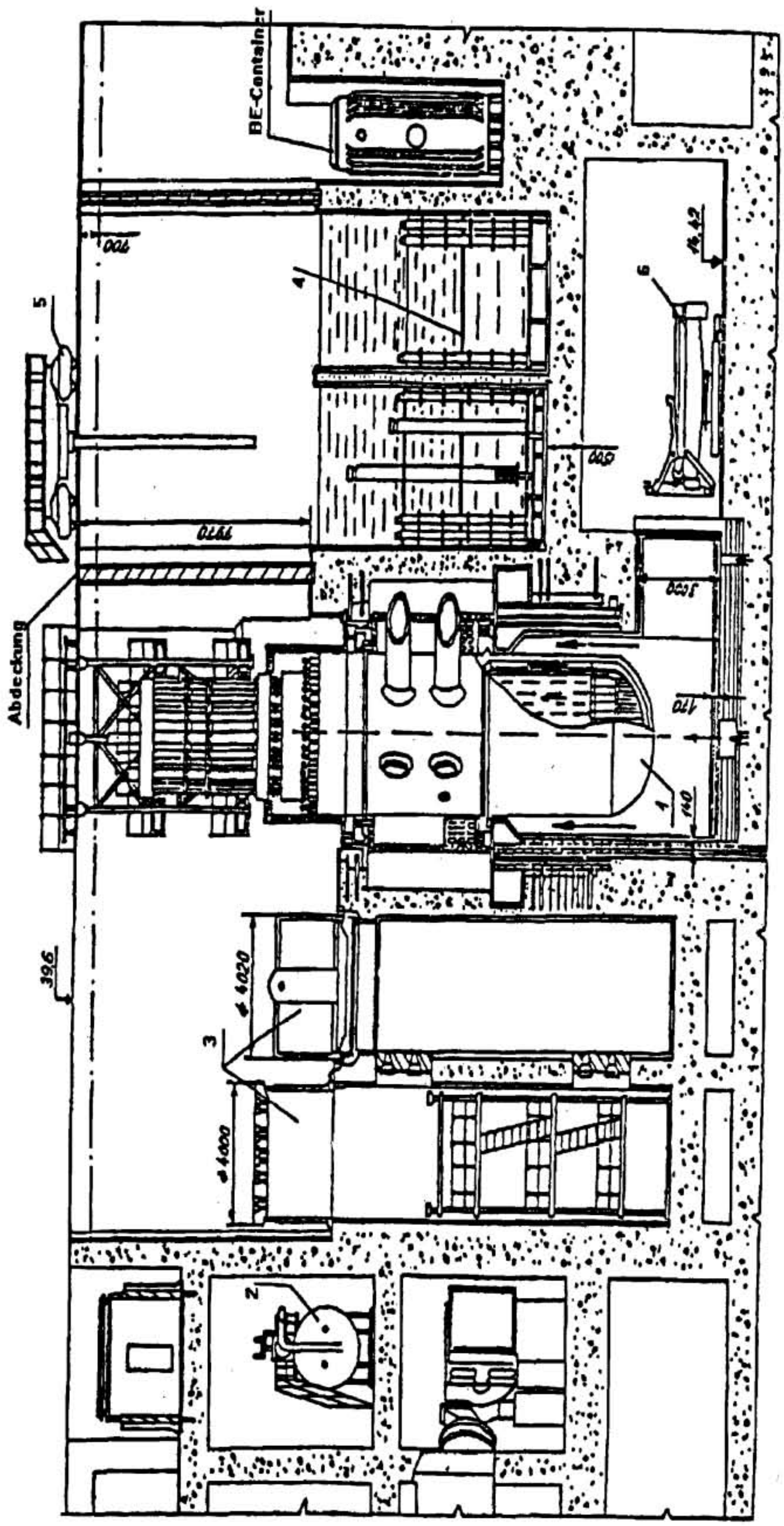
**Fig. 2.1-3** Engineered Safeguards



| Position | Component |
|---|---|
| 1 | Reactor |
| 2 | Steam generator |

| Position | Component |
|---|---|
| 3 | Main coolant pump |
| 4 | Pressuriser |

**Fig. 2.2-1** WWER-1000/W-320 (Temelin NPP), Section of reactor building in the area of the reactor pressure vessel and the steam generator

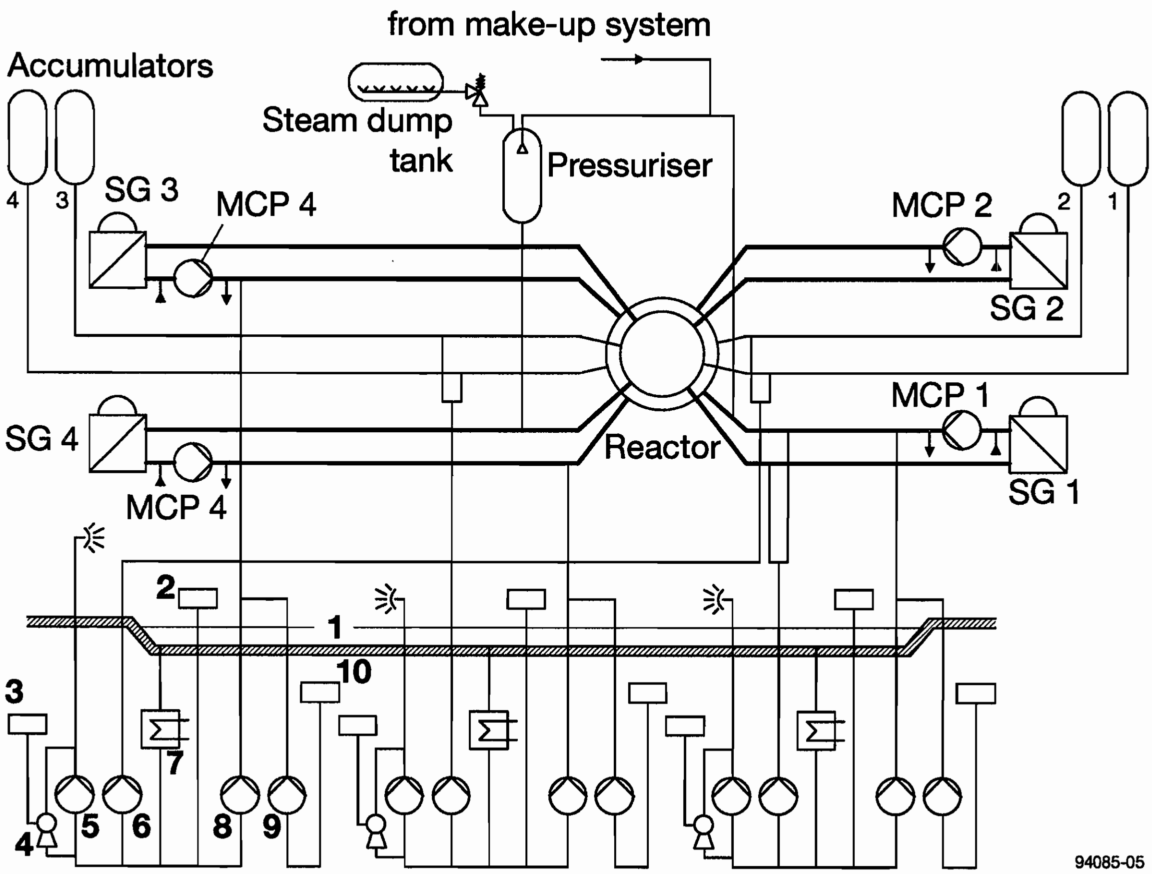



| Position | Component |
|---|---|
| 1 | Reactor |
| 2 | Steam dump tank |

| Position | Component |
|---|---|
| 3 | Inspection shaft |
| 4 | Spend fuel pit |

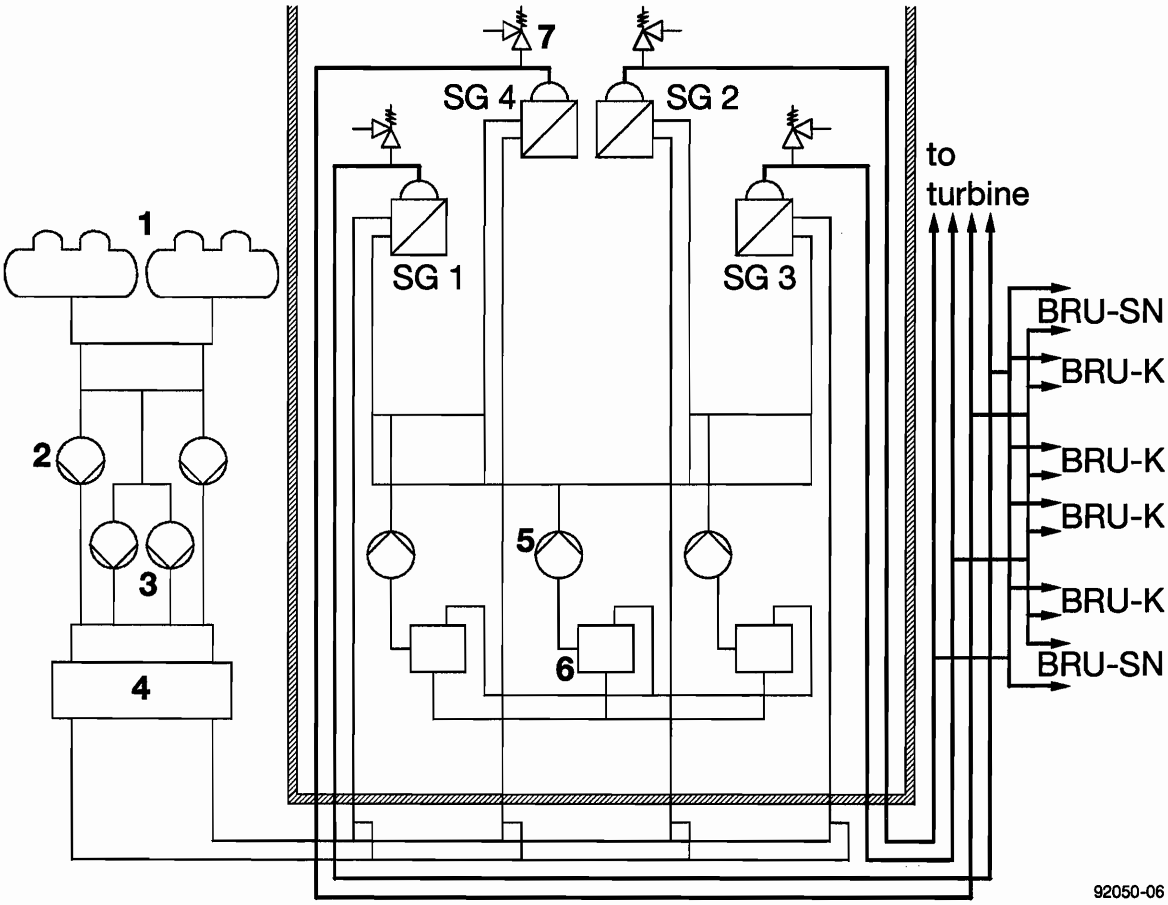
| Position | Component |
|---|---|
| 5 | Refuelling machine |
| 4 | Manipulator, Inspection machine |

**Fig. 2.2-2** WWER-1000/W-320 (Temelin NPP), Section of reactor building in the area of the reactor pressure vessel and reloading systems




**1** Boric-acid storage tank
**2** Storage tank
for concentrated boric acid
**3** Reagent tanks
**4** Ejector
**5** Containment spray pump
**6** LP-safety injection pump
**7** Emergency cooler
**8** HP-safety injection pump
**9** HP-boron injection pump
**10** Emergency storage tank
for concentrated boric acid

**Fig. 2.2-3** Elementary diagram: primary system with engineered safeguards




**1** Feedwater tank with deareator
**2** Feedwater pump (turbo driven)
**3** Auxiliary feedwater pump
**4** HP pre-heater system
**5** Emergency feedwater pump
**6** Emergency feedwater tank
**7** Pressure relief safety valve and steam dump station BRU-A

BRU-K By-pass station to turbine condenser
BRU-SN By-pass station to technological condenser

**Fig. 2.3-1** Elementary diagram secondary system with engineered safeguards

# 3 Licensing and Codes and Standards

## 3.1 Legal Licensing Principles and Assessment Criteria

The legal framework for the peaceful use of nuclear energy is established by The Atomic Energy Act. The Atomic Energy Act was adopted in 1959 and has since been amended repeatedly /ATG 92/.

Sec. 7, Subsec. 2 of the Atomic Energy Act lists the licensing prerequisites . It says that a license may only be issued if

- the provisions necessary in the light of the state of the art have been made against damage arising from the construction and operation of the plant,
- the necessary protection against disturbances or other impacts created by third parties is ensured,
- predominating public interests, especially those pertaining to clean water, air and soil, do not stand in the way of the choice of site.

These safety-related licensing prerequisites are not defined in more detail in the Act, but are spelt out in subsequent legal ordinances, guidelines and technical regulations. The most important codes and regulations are:

- **The Radiation Protection Ordinance /SSV 89/**

The Radiation Protection Ordinance contains the basic principles of radiation protection. The supreme principle is the requirement of minimising radiation. It implies that radiation exposure and contamination be minimized, in accordance with the state of the art and also considering the circumstances prevailing in the specific case, even below defined dose limits. This principle applies both to normal operation and to a possible accident.

Apart from the regulations on monitoring, radiation protection regulations are listed in the Radiation Protection Ordinance, for example:

- principles of radiation protection, especially Sec. 28
- protection of population and environment against the dangers of ionising radiation, especially Sec. 45
- occupational exposure to radiation, especially Sec. 49.

It is necessary to determine whether these regulations are observed by the design and operational planning of the installations.

- **Safety Criteria for Nuclear Power Plants /SKK 77/**

The safety criteria for nuclear power plants contain principles for safety-related requirements on which the design of nuclear power stations is based, especially to ensure the precautions required according to the state of the art against damage caused by the construction and operation of the plant and the necessary protection against disturbances or other impacts of third parties.

According to the principles of the safety precautions, the nuclear power station has to be designed in such a way that the reactor plant can be shut down safely and kept in the shut down state, that the residual heat can be removed and that, in accordance with the state of the art, the exposure of staff and the environment to radiation under normal operating conditions and during accidents can be kept as low as possible, even below the respective dose limits determined by the regulations of the Atomic Energy Act and the ordinances issued on the basis of the Atomic Energy Act.

- **The Accident Guidelines /SFL 83/**

The Accident Guidelines were set up for more recent nuclear power plants equipped with pressurised water reactors. They apply to plants for which the partial construction licences had not been issued before July 1, 1982. Consequently, these Guidelines cannot be referred to directly, but only indirectly in assessing the Stendal NPP.

On the basis of previous experience accumulated in safety analysis, expert assessment and the operation of nuclear plants, the Accident Guidelines define those accidents on which the safety-related design of nuclear power plants with pressurised water reactors must be based and the verification which must be produced by applicants, especially with respect to the observance of accident planning levels as specified in Sec. 28, Subsec. 3 of the Radiation Protection Ordinance.

For plants to which the Accident Guidelines apply, the radiological effects of the following representative accidents have to be examined:

- Double ended break in a main coolant line
- Leakage outside the containment of a measurement pipe carrying primary coolant
- Leakage with sealing capability in a main steam line outside the containment accompanied by simultaneous defects in steam generator tubes.
- long-term failure of the main heat sink due to plant leakages in the steam generator tubes.
- Leakage in a pipe in the offgas system
- Fuel element damage during handling
- Leakage of a vessel filled with radioactively contaminated water
- Leakage of a vessel under seismic impacts.

The list of these accidents can correspondingly be applied to the Stendal NPP. In addition, other WWER-specific accidents may need to be considered.

The necessary provisions required according to the state of the art have to be taken .

- **The RSK Guidelines for Pressurised Water Reactors /RSK 84/**

On the basis of the fundamental safety goals contained in the Safety Criteria, the German Advisory Committee on Reactor Safety (RSK) formulated in more detailed and precise guidelines the safety requirements to be met by the construction and operation of pressurised water reactors.

The requirements listed and specified in more detail in the RSK Guidelines for Pressurised Water Reactors /RSK 84/ are of special importance for the analysis and the safety-related assessment of the plant. Examples of this are:

- To determine the maximum accident pressure acting on the containment, the energy and mass inventories of the secondary side of one steam generator must be considered, in addition to the energy and coolant inventories of the primary system, for pressurised water reactors in West Germany
- Building structures, systems and components important to engineered safeguards must be designed against external events (earthquakes, airplane crashes, etc.)
- The design of and the requirements to be met by the reactor scram system (criteria for activation, dropping times of the shut-down rods, design details).

- **Technical Codes**

The requirements of the ordinances and guidelines are specified in the KTA Codes. They are not described here in detail. Appendix 2 contains a list of the KTA Codes and DIN Standards used for this assessment.

## 3.2 Application of the Federal German Codes to the Stendal Plant

The German technical codes and standards, especially the BMI Criteria, contain design requirements which safety systems must meet in terms of redundancy, diversity, demeshing, and physical separation of the different trains of systems. Above and beyond the criteria contained in the Soviet codes and standards pertaining to safety systems /PBJ 74/, /OPB 73/ and /OPB 82/, not only single failures, but also the absence of one level of redundancy because of repair must be included /GRS 91/. In addition to the single failure, the Soviet codes and standards assume failures having an influence on the accident sequence on components which are not subjected to functional tests during operation. This corresponds to the procedures of the German codes and standards. Here safety provisions to control possible consequences of defects of parts of the plant where recurrent examinations to detect possible defects cannot be carried out, must to be taken.

In the Soviet codes and standards the single-failure concept is restricted to active components. In the German codes and standards, passive components are also taken into account. The single failure, according to the Soviet codes and standards, need not be considered, if the respective (active) components have a high degree of reliability. A comparative restriction of the single-failure concept in the German codes and standards is only permissible with respect to passive single failures. Here, the application of the single-failure concept can be omitted, if special requirements are met in terms of reliable design, manufacture and monitoring.

The ordinances and guidelines mentioned define the requirements and approaches which have proved to work satisfactorily for many years of safety assessment and safety practice of nuclear power plants. The provisions are mainly based on the concepts and designs of light water reactor (especially pressurised water reactor) designs customary in West Germany. Technical alternative solutions to meet safety goals or to guarantee safety functions, respectively, which meet the rules of the codes and standards analogously, are therefore not excluded. This aspect has to be borne in mind when evaluating reactors of different designs, in this case the plant concept of the Stendal Nuclear Power Plant.

It must be examined, therefore, whether the existing design satisfies the protection goals underlying the codes and whether sufficient provisions have been made to avoid and manage accidents.

Where applicable codes and regulations are not met, it must be investigated whether such deviations give rise to a safety deficit and, if so, what measures can be taken to make up for such a safety deficit.

In the case of a Federal German licensing procedure, a safety report according to "Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor" (List of Notes with Subdivision for a Standard Safety Report for Nuclear Power Plants with a Pressurised Water Reactor or a Boiling Water Reactor) /BMI 76/ and further documents according to "Zusammenstellung der im atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen" (Summary of the Information required for Examination in the Legal Licensing and Supervisory Procedure for Nuclear Power Plants) /BMI 82/ are to be submitted.

**References, Sec. 3**


/ATG 92/ Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz), (Act on the Peaceful Use of Nuclear Energy and the Protection against its Hazards), (Atomic Energy Act). As promulgated on July 15, 1985 (Bundesgesetzblatt I, No. 41, of July 31, 1985, taking into account the last amendments of February 28, 1982 as published in Bundesgesetzblatt I, p. 376 of February 28, 1992.

/BMI 76/ Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor (Structured list of criteria for a standard safety report for nuclear power stations with a pressurised water reactor or boiling water reactor) Public notice of the Federal Minister of the Interior of July 26, 1976 as published in Gemeinsames Ministerialblatt No. 15, 1976.

/BMI 82/ Zusammenstellung der im atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen (ZPI) (Summary of the information required for examination in the legal licensing and supervisory procedure for nuclear power plants),passed by the Länderausschuß für Atomenergie (Committee of the Laender on Nuclear Energy) on September 7, 1982, Public notice of the Federal Minister of the Interior of October 20, 1982 as published in Bundesanzeiger No. 6a of January 11, 1983.

/GRS 91/ Gesellschaft für Reaktorsicherheit Vergleich der Sicherheitskriterien für Kernkraftwerke des BMI vom 21. Oktober 1977 mit den Prinzipien der Gewährleistung der Sicherheit von Kernenergieanlagen bei Projektierung, Bau und Betrieb (Comparison of safety criteria for nuclear power plants of the BMI of October 21, 1977 with the principles for ensuring the safety of nuclear power stations during design, construction and operation)

OPB-73 (1973)
Internal Report, 1991

/OPB 82/ State Committee for the Application of Atomic Energy in the USSR
General Safety Regulations of Nuclear Power Plants during Design, Construction, and Operation,
OPB-82

/OPB 73/ Ministry for Energy and Electrification of the USSR
General Safety Principles of Nuclear Power Stations during Planning, Construction and Operation,
OPB-73

/PBJ 74/ State Committee for the Application of Atomic Energy in the USSR
Nuclear Safety Regulations for Nuclear Power Plants,
PBJa-04-74

/RSK 84/ Reaktor-Sicherheitskommission
RSK-Leitlinien für Druckwasserreaktoren
(RSK Guidelines for Pressurised Water Reactors)
3rd edition, October 14, 1981, including amendments as published in Bundesanzeiger No 69 on April 14, 1982 under consideration of the amendments as published in Bundesanzeiger No. 106 on June 10, 1983, and Bundesanzeiger No. 104 on June 5, 1984.

/SFL 83/ Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV, Störfall-Leitlinien
(Guidelines for Assessing the Design against Accidents of Nuclear Power Plants with Pressurised Water Reactors in accordance with Sec. 28, Subsec. 3 of the Radiation Protection Ordinance, Accident Guidelines)
Public notice of the Minister of the Interior of October 18, 1983, Bundesanzeiger No. 245a of December 31, 1983.

/SKK 77/ Sicherheitskriterien für Kernkraftwerke (Safety Criteria for Nuclear Power Plants),
passed by Länderausschuß für Atomenergie (Committee of the

Laender on Atomic Energy) on March 22 and October 12, 1977, Public notice of the Federal Minister of the Interior of October 21, 1977, Bundesanzeiger No. 206 of November 1977.

/SSV 89/ Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung-StrlSchV), (Ordinance on the Protection against Injuries by Ionizing Radiation) (Radiation Protection Ordinance) as promulgated on June 30, 1989, Bundesgesetzblatt I, No. 34 of July 12, 1989 under consideration of the corrections and amendments until the 2nd amendment according to Bundesgesetzblatt II, No. 35 of September 1990.

# 4 Core Design and Pressurised Components

## 4.1 Core Design

### 4.1.1 Core Arrangement and Fuel Elements

- **Description**

The core of the WWER-1000 of the W-320 type at Stendal A consists of 163 hexagonal elements having a width across of 23.4 cm. Each fuel element contains 312 fuel rods, a central tube and 18 guide tubes for the control element. The core is equipped with 61 control elements. The fuel element is, as customary for a pressurised water reactor (PWR), open at the outside and does not have a closed fuel assembly box like the WWER-440. The fuel element differs from the WWER-440 fuel element in its width across (14.3 cm) and the number of fuel rods (126), as well as the use of absorber rods compared with the special construction of the control elements in the WWER-440, consisting of a lower fuel assembly and an upper absorber assembly.

The fuel elements are located in the shaft (core barrel), a thin annular structure with its lower part formed into a perforated elliptical bottom end. The coolant flows out of the four cold legs of the main coolant loops via the annulus between shaft and reactor pressure vessel, through the openings of the elliptical shaft bottom and into the perforated support tubes for the fuel elements (see Fig. 4.1-1). The fuel elements are fastened from the top of the element heads through the protecting tube unit. The heated coolant flows into the upper part of the shaft, through the side openings into an annulus and then into the hot legs of the main coolant loops. This annulus in the hot area is sealed towards the annulus below with a separation ring.

The core is loaded with fuel elements having different fuel enrichments.

It is significant for the nuclear core design that the original reactor project started out from a two-year service life of the fuel elements, while for other WWER-1000 projects only three-year fuel lives are currently planned.

For the two-year fuel life, the initial core loading provided is:

| | | |
|---|---|---|
| 42 | fuel elements with an enrichment of | 3.3 % |
| | 6 of these have an enrichment profile, | |
| 42 | fuel elements with an enrichment of | 3.0 % |
| 79 | fuel elements with an enrichment of | 2.0 % |

For the three-year fuel life the initial fuel loading provided is:

| | | |
|---|---|---|
| 54 | fuel elements with an enrichment of | 4.4 % |
| | 30 of these with have an enrichment profile | (3.6 %), |
| 55 | fuel elements with an enrichment of | 3.0 %, |
| 54 | fuel elements with an enrichment of | 1.6 % |

For reloading, fuel elements with a higher enrichment are used.

The fuel rods in one fuel element normally have the same enrichment. For fuel elements with a high enrichment profiled designs are also provided, the outer row of fuel rods having a lower enrichment. The essential data for the fuel elements and the control rods are summarized in Tables 4.1-1 and 4.1-2. Fuel elements with burnable poisons like boron carbide are available for the WWER-1000; designs with gadolinium or other materials are in the development and test phase.

The fuel pellets have an inner bore hole, the size of which is optimized depending on the experience gathered from in-pile tests.

The core is loaded corresponding to an outer-inner concept, where fuel elements having a high enrichment are inserted near the core periphery in the first year of service life and reloaded to the inner area of the reactor core in the subsequent operational cycles.

- **Assessment Criteria**

The safety requirements can be derived from the general principles referring to design and quality assurance as well as from the requirements of the BMI-safety criteria, criterion 3.1 (Reactor Design), criterion 3.2 (Inherent Safety), the RSK Guidelines pertaining to pressurised water reactors, Section 3 (Reactor Core) and the requirements contained in KTA-Rule 3101, Part 2 (Neutron Physics Requirements to be met by the Design and Operation of the Reactor Core and Adjacent Systems).

- **Assessment**

Because of the numerous changes in the planned core loading, owing, to the transition from the two-year service life to the three-year service life and the associated optimization of the fuel element design, there are only incomplete design calculations for the neutron physics behaviour of the reactor core. Statements by the Soviet side of the technical project are not detailed and are outdated because of further development of the core loading. The core loading, for the two-year service life is problematic with respect to safety, because of the high boron concentrations at the beginning of the cycle and the resulting positive moderator temperature reactivity coefficients. For the three-year service life there are calculations from the German side by K.A.B. /KAB 91a/. These calculations do not, however, consider the intended use of gadolinum as burnable poison in the fuel and, in addition, these calculations are incomplete. For this reason, the assessment of the nuclear design can only be preliminary, in the sense of a conceptual assessment. A complete core-design report is to be presented for the three-year service life of the fuel elements (R 4.1-1).

The differences in the design of the fuel elements compared with other PWR fuel elements do not seem to be essential. The intended optimisation with respect to the use of burnable absorbers corresponds to the respective developments for western nuclear power stations. The loading strategy according to the outer-inner concept results in a high neutron flux at the core edge and thus in a high neutron irradiation for the reactor pressure vessel. In German pressurised water reactors at the present time, only low-leakage loadings according to the inner-outer concept are used. This core-loading additionally allows a better fuel utilization, but requires the use of burnable poisons, like gadolinium for example, in the fuel elements. The arrangement of the fuel elements in the core should therefore be optimised so that a low-leakage loading, to reduce neutron irradiation of the pressure vessel, can be aimed at here (R 4.1-2).

### 4.1.2 Power Control and Shutdown Safety

- **Description**

The control and protection (SUS) system, the make-up system for boric acid and deionized water or the HP-emergency boron injection system are used to control the reactor power or to shut it down, respectively.

The systems are used for the following functions:

- Startup of the reactor after loading to hot zero power and low power
- Automatic power control within the power range including xenon control
- Compensation of reactivity changes by burn-up of fuel
- Reactor scram by drop of all control elements
- Shutdown of the reactor by boron injection.

Shutdown functions have priority over the functions for power control.

The reactor core is equipped with 61 control elements which are subdivided into ten groups. Eight groups contain six control elements, one in each of six azimuthal sectors. One group comprises nine control elements, while Group 5, used for controlling xenon, contains the central control element and three further control elements.

The operational burn-up compensation is effected by group 10 comprising six control elements. The other groups of control elements, are inserted during operation to shutdown the reactor upon request. The operational speed of the control elements is constant and is 2cm/s. Drop of the control elements for shutdown takes 1.5 to 4 s. The operator can insert any combination of control elements.

The degree of reactivity compensation, by either control elements or by changes in boron concentration, does not seem to be limited by technical devices. No insertion limit of the control elements is provided.

To control power, several principles are specified (see also Section 6.4.2.3). According to one control principle, the main steam pressure is kept constant throughout the power range and the average coolant temperature steadily increases with power, to correspond with the heat transfer variation in the steam generator. According to another control principle, up to 80 % of nominal power the main steam pressure is kept constant, but at higher power the average coolant temperature is constant, i.e. the main steam pressure is reduced. The first control principle with constant main steam pressure is the preferred operational mode. The reactor power is determined by the position of the control elements and the boron concentration, adjusted by the operator. This operational concept changes the operational parameters of the reactor core over a wider range, affecting the effective reactivity coefficients and the reactivity balance for shutdown safety.

To demonstrate shutdown safety, a margin of 1 % subcriticality is specified, taking into account the failure of the most effective control element.

The reactor scram occurs after activation of the reactor protection, by insertion of the control elements. The reactor scram is, for example, activated by the "neutron flux high" signal from the ex-core power range detectors. Previous measures to limit the reactor power without actuating reactor scram are to interlock the control elements or to insert the control elements with the normal insertion speed. Activation of reactor scram is explained in detail, together with the emergency protection system and the instrumentation and control for reactor scram in Section 6.4.3.1.

The effectiveness of the reactor scram system is only sufficient for shutdown to the "hot, subcritical" state /KAB 91a/. For this reason additional boron must be injected before the plant can be transferred to the "cold, subcritical, xenon-free" state. Depending on the underlying course of events, the make-up system, the emergency cooling system for accidents involving leaks or the HP-emergency boron injection system are principally available for the boration.

The assessment of process technology related to the shutdown systems is described in Section 6.2.

- **Assessment Criteria**

For assessing power control, the assessment criteria already mentioned in Section 4.1.1 are used. For assessing shutdown, the BMI-criterion 5.3 (Devices for Controlling and Shutting down the Reactor Core), as well as the requirements for shutdown systems of light water reactors in KTA 3103 are used. Two independent and diverse shutdown systems are required in these criteria, to terminate the chain reaction with a sufficient shutdown reliability.

One of the two shutdown systems, the reactor scram system, on its own must be capable of taking the reactor core rapidly from any operating condition and any accident situation to a subcritical state and to hold it there long enough, even with failure of the most effective control element, so that the specified limits of the reactor plant are not exceeded.

The reactor scram system and the reactor core are to be designed in such a way that after shutdown, until sub-criticality has been ensured by the liquid-poison system, the net shutdown margin verified by calculation does not fall below 1 %. The liquid-poison system must be able to keep the reactor in the "cold, xenon-free, sub-critical" state. A calculated net shutdown margin of 1 % is to be verified by proven design calculation procedures.

Liquid-poison systems which are to fulfil the function of a second shutdown system independent of the reactor scram system must be able to render the reactor subcritical, independent of the control rod system, for all operational conditions which do not require fast reactivity changes, and to hold it sub-critical even in the most reactive state which can occur after shutdown. A shutdown margin of 1 % is to be demonstrated by calculation for liquid-poison systems taking the function of a second shutdown system, when neutron flux and absorber concentration are monitored. If these provisions do not exist, the liquid-poison system will have to be designed in such way that a calculated shutdown margin of 5 % is maintained.

The required effectiveness and speed of the two shutdown systems in fulfilling their tasks are to be determined by representative analyses of assumed courses of events.

If components of the shutdown systems are used for operational control purposes, it is to be ensured by their design and by technical safeguards in operation that the effectiveness of these components required for shutdown is maintained under any operational state.

- **Assessment**

The power control concept as intended so far, leaves the operator or the operational regulations too much freedom in the use of control elements or of boration and dilution of the coolant, leading to frequent movement of the control elements. A limitation of the admissible control element insertions is to be provided to ensure the effectiveness of the reactor scram for all operational conditions (R 4.1-3). The priority of emergency protection over operational requirements of the reactor scram system is discussed in Section 6.4 referring to instrumentation and control.

Great changes of the operational parameters are possible because of the intended flexibility in the operational mode of the reactor core. The results of the investigations relating to shutdown safety for a three-year cycle, as currently planned, are not available. In these investigations it is to be demonstrated that shutdown leads to a sub-criticality of at least 1 % until sub-criticality is ensured by the liquid-poison systems (R 4.1-4). The assessment of the speeds of the shutdown systems required has to be carried out within the framework of accident analysis.

The difficulties with xenon oscillations known from operational experience are certainly caused by the power and power density distribution control. The equilibrium between power density distribution and xenon concentration is disturbed by the frequent movement of control elements for power control. The temporal changes of the xenon concentrations and their effects on the power density distribution can necessitate additional movements of the control elements so that xenon oscillations are finally stimulated. An improved control concept for power and power density distribution can reduce power density changes and therefore also avoid the onset of xenon oscillations. Part-length control elements for xenon control have been introduced in the meantime, which according to the latest /MRE92/ shall, however, no longer be used in the future. These part-length control elements should be avoided

(R4.1-5). Control of power density distribution including xenon control is to be automated (R 4.1-6).

To supplement the effectiveness of the reactor scram system in the long-term and as a second shutdown system, boron injection systems are provided. For both functions there are no reactivity balances. It therefore has to be demonstrated for the boron injection system intended as a supplement to the reactor scram system, that it can also render the reactor sufficiently sub-critical in the presence of a single failure, in accordance with the requirements (1 % net shutdown margin) (R 4.1-7). For the second shutdown system it must be demonstrated by calculations that the shutdown margin is 1 % when neutron flux and absorber concentration are monitored; without monitoring measures it must be 5 % (R 4.1-8).

### 4.1.3 Core Instrumentation

- **Description**

The object of the core instrumentation is to provide adequate monitoring of the admissible states of the core.

The core instrumentation consists of an outer instrumentation as part of the SUS system for the startup, transition and power range, from which the signals "neutron flux high" and "reactor period high" for reactor protection are derived, and the in-core instrumentation consisting of neutron flux and temperature measurments.

The in-core instrumentation measures the fuel element outlet temperatures at 95 positions by thermocouples above the fuel elements. The neutron flux distribution is measured in 64 measuring lances having seven rhodium detectors each, i.e. in 448 measuring positions. Calibration is by comparison with the overall thermal power and by comparison with the fuel element outlet temperatures. The values measured by the rhodium detectors are transformed into power density values with the help of coefficients in the computer. In-core instrumentation only provides information on the state of the core, not at present for derivation of active measures for power limitation or shutdown. The values measured by the detectors are compared with the limits for the different axial heights. If the limits are exceeded, there is a warning for the operator, so that the permissible power density is restored by manual measures.

Studies to use the deviations from the limits in an automatic limitation system, for example by interlocking the control rod withdrawal being performed. The measuring heads with the rhodium detectors have been developed further in the meantime so that coolant temperatures can additionally be measured in the intake and outlet of the instrumentation probe. These directly assigned temperature measurements are to improve the calibration of the detector measurements.

- **Assessment Criteria**

The BMI safety criteria, the RSK guidelines for PWR, the KTA Rule 3101, Part 2 as well as KTA Rule 3501 mentioned in Section 4.1.1 are used as assessment criteria.

- **Assessment**

The in-core instrumentation for measuring neutron flux density and coolant temperatures is very extensive with respect to the number of detectors, but, for both measuring systems, open questions remain.

The temperatures measured by the thermocouples at the top of the fuel elements cannot be directly assigned to the power of one fuel element, as the coolant can mix between the fuel element outlet and measuring position, a distance of between 30 and 50 cm. The analysis of the operational experience showed a dependence on the position of the control element.

The indications of the rhodium detectors cannot be checked by an additional system, like, for example, movable fission chambers or an aeroball flux measuring system as in German pressurised water reactors. The present operational experience for the system of measuring power density distribution therefore has to be illustrated and analysed more precisely. Measuring accuracy and its evaluation are to be demonstrated during operation. (R 4.1-9).

The concept of in-core instrumentation should be examined in order to supplement the existing detectors, measuring power density distribution, with an additional system for calibrating and testing (R.4.1-10).

In-core instrumentation should not be used for power density distribution alone, but it should be extended through a link with the control element control system to develop an automatic power density limitation system (R 4.1-11).

### 4.1.4 Thermohydraulic Core Design

- **Description**

The object of the thermohydraulic design is to demonstrate a sufficient cooling of the fuel rods to ensure the integrity of the fuel rod cladding tubes which tightly enclose the radioactive inventory.

For PWR the parameter for a sufficient cooling is the DNB correlation which for every fuel rod section is calculated from the relation of the critical heat flux density to the current heat flux density. To take the most unfavourable cooling conditions into account, a hot channel defined by the hot channel factors is examined.

The following factors were used in the examinations of K.A.B.:

| | |
|---|---|
| radial power factor for the fuel elements | $K_V = 1.30$ |
| local power factor within the fuel elements | $K_\mu = 1.20$ |
| axial power factor | $K_z = 1.50$ |
| technical channel factor for heat flux density | $K_q = 1.16$ |
| $F_{\Delta H}$-factor | $F_{\Delta H} = 2.03$ |
| Maximum rod linear power | 448 W/cm |

The correlation of Besrukov/Astachov is used to calculate the DNB values. Minimum DNB ratios of 1.50 to 1.75 were calculated for selected operational states using this correlation. The permissible DNB ratios for steady reactor operation can directly be derived from the accident analyses for the complete failure of all main coolant pumps, for which evidence is to be provided that the values do not fall below the minimum permissible DNB correlation. The most adverse conditions for reactor power and mass flow in the initial state as well as the effectiveness of reactor scram are to be taken into account here. There are currently no detailed investigation results available for these consitions.

The accuracy of the thermohydraulic correlations is important for the assessment of the results. For VVER reactors the correlation of Besrukov/Astachov is preferred for the design. In connection with the Greifswald NPP, Unit 5, the following statements were made by the Kurchatov Institute: On the basis of 800 experimental points a mean of 1.01 was determined and a root mean square error of = 13.1 % was stated.

- **Assessment Criteria**

The requirements of the thermohydraulic core design are determined in KTA-Rule 3101, Part 1, Principles of the Thermohydraulic Design.

- **Assessment**

The input quantities for the thermohydraulic design with respect to the hot channel factors used can be derived from the nuclear design. The complete nuclear calculations for a three-year cycle are not available to determine the most adverse power distributions. Taking into account the most unfavourable initial conditions referring to reactor power, axial power distribution and core flow rate, the design-determining transients like, for example, the complete failure of the main coolant pumps of one main coolant pump are to be analysed to verify the observance of the minimum permissible DNB ratios (R 4.1-12).

All statements referring to the thermohydraulic correlation available so far which are not unambiguous must be checked. In particular, a description of the experimental background of the DNB correlation including a justification of the accuracy and the tolerance limit must be provided (R 4.1-13).

It should be possible to prove that the permissible DNB ratios are observed. In this context it is to be examined whether a system for power density limitation including a DNB signal for reactor scram, derived from core instrumentation, is necessary for safety-related reasons (R 4.1-14).

### 4.1.5 Reactor Pressure Vessel Internals

#### 4.1.5.1 Construction

- **Description**

Reator pressure vessel internals are the components within the vessel which serve the routing of the flow of coolant and the accommodation of the reactor core. The core internals which are located inside the reactor core (fuel elements, control rod elements) are dealt with in the subsequent Section 4.1.6.

- Shaft

The shaft directs the coolant to the reactor and contains the isolating steel sheet, the fuel elements and the protecting tube unit (see Fig. 4.1-1). The shaft consists of a vertical cylinder formed from eight steel sheet sections and an elliptic bottom surface. At the upper end, its flange leans on the bearing area of the pressure vessel flange. Between inlet and outlet nozzles of the reactor pressure vessel there is a collar, the inner part of which bears on the shaft to reduce leakages between the hot and cold coolant. In the lower section of the shaft surface there are channels. The collar is connected with the reactor vessel and centred by these channels with the help of claws, but it can move axially. In the area of the outlet nozzles of the reactor pressure vessel the shaft is provided with bore holes through which the mixed hot coolant escapes. In addition, there are two penetrations for the admission of emergency cooling water. In the lower part of the cylinder there are six groups of webs in an upright position to centre the isolating steel sheet.The lower end of the shaft consists of a perforated cyclindrical bottom section. Through these perforations the coolant enters the interior of the shaft in a mixed state. In the bottom there are 163 support cylinders installed, bearing the fuel elements. The lateral location in the upper part of these cylinders is ensured by a diagrid which at the same time supports and locates the isolating steel sheet. The upper parts of the 163 support cylinders are perforated so that the coolant can directly flow into the fuel elements.

- Isolating Steel Sheet (Shell of the Reactor Core)

The isolating steel sheet defines the lateral limitation of the reactor core and further absorbs a part of the neutron irradiation penetrating to the outside. It consists of four different forged sheets which are connected to each other with screws and centred with bolts. The inner surface is adjusted to the outer contour of the reactor core. The isolating steel sheet is connected with the diagrid low down in the shaft, mentioned above, with six threaded bolts. On the outside there are six channels into which the six web groups engage.

- Protecting Tube Unit

The protecting tube unit during operation contains the control elements, centres the heads of the fuel elements and bears the core instrumentation.The protecting tube unit consists of the shell and the upper and the lower grid. The two grids are connected by 61 protecting tubes. The lower grid contains devices for centering and holding down the fuel elements. Furthermore, in both the upper and the lower grid there are bore holes for the protecting tubes and the core instrumentation. The protecting tube unit, and thus also the fuel elements, are held down by fitting the RPV head.

- **Assessment Criteria**

The present assessment of the construction restricts itself to a comparison with the corresponding parts in German pressurised water reactors; especially by comparison with the Brokdorf Nuclear Power Station. Referring to the operational experience of WWER-1000 reactors there is no information available on reactor pressure vessel internals.

- **Assessment**

- Shaft

This vessel internal essentially corresponds to the core barrel in German pressurised water reactors with respect to function, design, bearing in the reactor pressure vessel and the loadings. There are minor differences with respect to coolant intake and outlet as well as the bearings of the fuel elements.

- Isolating Steel Sheet

This internal as such does not exist in German pressurised water reactors. The core embracement as the external bordering of the core in these reactors is directly fastened to the core barrel with the help of formed ribs.

- Protecting Tube Unit

This internal in its function essentially corresponds to the upper core housing in German pressurised water reactors.

#### 4.1.5.2 Materials

- **Description**

- Base materials

For the parts of the reactor core internals described above the austenitic material 08Ch18N10T was employed. The requirements for mechanical stress properties and elongation at rupture are defined in /SPE 90/. No information on chemical analyses for this material can be derived from /SPE 90/.

- Welding

In the documents available there also is no information on welding materials.
For the base material and the welding fillers used, identical materials are assumed, from operational experiences on the previous operation of other VVERs.

- **Assessment Criteria**

KTA-Rule 3204 as well the German Standards DIN 17440 and DIN 8556 are used for assessment.

- **Assessment**

- Base material

The basic requirements for stress and elongation at rupture parameters according to /SPE 90/ approximately correspond to the ones of the material X 6 CrNiTi 18 10 (1.4541) in DIN 17440 or KTA 3204, respectively. As no information on the analytical values for this material can be derived from /SPE 90/ and as, on the other hand, this material was also used for other NPP of the Soviet type, reference is made to the respective specification of the Greifwald Nuclear Power Station, Unit 5 /SKO 83/. According to this specification, this material in its chemical analysis also corresponds to the material X 6 CrNiTi 18 10 (1.4541) in DIN 17440 or KTA-Rule 3204, respectively. The further requirements of KTA-Rule 3204 are also fulfilled. Only the somewhat higher carbon content deviates from this rule. Additionally, the minimum value for the limit of elasticity at 325C for 08 Ch18N10T is somewhat higher.

Because of its analytical chemical values and its stress parameters, the material generally meets the requirements of KTA 3204 and therefore from today's viewpoint appears to be suitable for the purpose of this application, although the carbon content, which according to KTA is too high, is still to be assessed (R 4.1-15).

- Welding Fillers

Identical welding fillers as in the corresponding internals of the Greifswald Nuclear Power Station, Unit 5, are assumed, here too the welding fillers stated in /SKO 83/ according to their chemical analyses approximately correspond to the material 19 9Nb permitted in DIN 8556 and in KTA 3204. But the carbon content is higher than permitted in DIN 8556 and in KTA 3204 and must therefore still be examined like the base material (R 4.1-15).The base materials used and the welding fillers must be assessed with respect to their material specifications, particularly with regard to their carbon content, taking operating experience into account (R 4.1-15).

#### 4.1.5.3 Design

- **Description**

- Operation

During operation the internals have to fulfil specific functions. Loads resulting therefrom must be borne and absorbed. According to /OKB 81/ stress calculations of internals for operational conditions were performed.

- Accidents

According to /OKB 81/ calculations of the impact on the reactor vessel internals as well as calculations of the toughness of the internals under accident conditions are available.

- **Assessment Criteria**

According to the RSK Guidelines and the KTA-Rules the internals have to be designed and arranged in such way that they can be shut down safely in all operational states and during accidents and that adequate coolability of the core can be ensured.

The following requirements result therefrom:

- Accommodation of the weight and deformation forces of the fuel elements
- Ensuring the position and alignment of fuel elements
- Accommodation of the shocks produced by the control elements in cases of reactor scram
- Coolant flow configuration in the reactor pressure vessel

- Accommodation of the in-pile irradiation specimens for brittle fracture control of the reactor pressure vessel material.
- Ensuring the stability of the core geometry under accident conditions.

The effects of gamma and neutron irradiations must also be considered here.

- **Assessment**

- Operation

The requirements mentioned are met by the reactor pressure vessel internals from a constructional and functional view. This is also proven by the operational experience of plants of the same type.The verification of the calculations in /OKB 81/, still to be carried out, will have to show whether the requirements mentioned are also met with respect to toughness (R 4.1-16).On the basis of the previous operational experience of plants of the same type and the measurements of the internals as well as the suitability of the materials, there are no indications from today's view that the internals because of their design could not fulfil the requirements to be met by them.

- Accidents

With the present calculations on loss-of-coolant accidents /OKB81/ only a part of the necessary verifications is available. The verification of the calculations in /OKB 81/, still to be performed, will have to demonstrate whether the load due to external events (safe shutdown earthquake, airplane crash, explosion blast wave) is covered (R 4.1-16).
It is considered that the internals of the reactor pressure vessel having about the same wall thicknesses as those of the German pressurised water reactors and a smaller diameter will have to absorb smaller loads arising from accidents.

### 4.1.6 Core Internals

#### 4.1.6.1 Construction

- **Description**

The reactor core essentially consists of 163 fuel elements, 54 of which contain burnable absorbers and 61 control elements. There are 64 guide tubes to accommodate the incore instrumentation /KAB 91a/.

Each hexagonal, laterally open fuel element consists of 312 fuel rods arranged in ten rows around a central guide tube. Fuel elements with burnable neutron poison contain 18 absorber rods each. The fuel rods are fastened by a skeleton consisting of a top, distance pieces (grids) and 18 control rod guide tubes. The top of the skeleton is elastically connected with the head of the fuel element by screwed connections so that thermal expansion and growth induced by irradiation are not obstructed. The lowest distance piece is connected to the fuel element foot and serves as support for the fuel rods which are cottered. The fuel element is fastened to the bottom of the shaft with the cylindrical fitting piece of the foot.

The fuel rod consists of a gas-tight, welded cladding tube in which the $UO_2$-fuel pellets, with central bore hole and dishing (trough-shaped deepening at the pellet ends), are located. A helium filling improves the heat conduction in the gap between fuel and cladding.

The control elements consist of a head part to which 18 absorber rods are fastened, some of which are absorber rods of half length.

- **Assessment Criteria**

The present assessment of the construction of the core internals restricts itself to a comparison with the corresponding parts in German pressurised water reactors.

- **Assessment**

The fuel elements among other things differ from the fuel elements in German pressurised water reactors with respect to their hexagonal instead of square cross-section, the elastic connection between fuel element head and skeleton and the execution of the fuel pellets (with central bore hole). These differences are assessed to be suitable for the design principle chosen.

An essential difference of the control elements compared with German pressurised water reactors is the smaller number of absorber rods, 18 as opposed to 20, some of which are half length, their elastic fixing in the head part as well as the use of $B_4C$ instead of AgInCd as absorber.

It can be derived from reference /KOL 91/ that a design has been chosen for the fuel elements which with respect to the essential constructional features

- fuel rod,
- fuel rod arrangement and
- structure of fuel elements

has been employed successfully in many plants. According to Kolyadin /KOL 91/ the maximum fuel rod failure rate was only 0.02 %. There are no documents relating to the causes of failure; these must be compiled (R 4.1-17).

The concept of the control elements in essential parts is comparable to the one of German pressurised water reactors and meets the functional requirements.

#### 4.1.6.2 Materials

- **Description**

The cladding tube material of the fuel rods consists of zirconium alloy with 1 % niobium. Austenite is used for control rod guide tubes, grids and for the head and foot. The fuel consists of sintered uranium dioxide pellets with central bore hole and dishing.

The absorber of the control elements consists of boron carbide pellets, the cladding tube of austenite. The burnable absorber consists of pellets of $CrB_2$ in an aluminium alloy matrix, located in a cladding tube of zirconium alloy.

- **Assessment Criteria**

KTA-Rule 3103, Shutdown Systems of Light Water Reactors, is used for assessment.

- **Assessment**

References /KOL 91/, /PAZ 91/ show that ZrNb1 has been proven as cladding tube material under the present operational conditions. This can also be regarded as proven for the austentic materials and $B_4C$.

ZrNb1 differs from zircaloy by a smaller toughness and greater embrittlement at higher temperatures. With respect to plasticity, stress corrosion cracking and uniform corrosion there is a comparable or more favourable behaviour, respectively. To what extent the existing differences are significant can only be determined after a complete examination (R 4.1-18). Some analyses of double ended cold leg breaks of a recirculation loop showed that cladding tube damage is not expected.

#### 4.1.6.3 Design

- **Description**

According to the work report /KAB 91b/ of the Kraftwerks- und Anlagenbau AG the limit for fuel rod failures for normal operation according to the Soviet standards is determined by the established level of the coolant activity in the primary system and, in terms of the number of defective fuel rods (1st project limit for fuel rod failures), is

- 1 % fuel rods with gas leaks
- 0.1 % of the fuel rods with direct contact between the coolant and nuclear fuel

In case of loss-of-coolant accidents, the emergency cooling system must ensure the following limits (2nd project limit for fuel rod limits):

- Temperature of fuel rod cladding ≤ 1200 °C
- Local depth of oxidation of cladding tubes ≤ 18 % of initial wall thickness
- Fraction by mass of the reacting zirconium ≤ 1% of the total zirconium mass in the reactor core.

After a loss-of-coolant accident, cooling and shutdown of the reactor must be ensured.

The following points were taken into account to ensure the integrity and functioning of the fuel rods during defined operation within the set limits: Limitation of the respective fuel temperature, observance of the permitted toughness limits, corrosion resistance of materials and the influence of reactor operation, interactions between pellet and cladding tube as well as expansion under the influence of temperature and irradiation.

As stated in work report /KAB91c/, a leak-tightness check of all fuel elements to be unloaded shall be performed during the transfer of fuel elements and damaged ones shall be placed in special positions.

- **Assessment Criteria**

For the design of the core internals it is to be required that they withstand the loads of the defined operation. For accidents it must be demonstrated in accordance with their probability of occurence that depending on the transient the fuel elements can be used further or that the integrity of the cladding tubes is given, respectively. For accidents which are not expected to occur during the entire life-time, e.g. loss-of-coolant accidents with large leakage cross-sections, it must be demonstrated with respect to the core internals that the extent of failure remains so small that residual heat removal and shutdown are ensured and the permitted failure limits are observed.

According to RSK Guideline 22.1 emergency core cooling must ensure during loss-of-coolant accidents that

- the calculated maximum temperature of fuel rod claddings does not exceed 1200 °C,
- the calculated depth of oxidation of the cladding at no point exceeds 17 % of the actual cladding tube wall thickness,
- not more than 1 % of the entire zirconium contained in the cladding tubes reacts during the zirconium-water reaction,
- the releases of fission products owing to cladding tube damage mentioned in Section 2.2 (4), No. 2, are not exceeded,
- no changes occur in the geometry of the reactor core which prevent a sufficient cooling of the reactor core.

KTA 3103 determines the design of the control elements. For fuel elements there is no independent KTA Rule. In accordance with the present state of the art, design criteria for example for temperatures, pressure loads, tensions, extensions, corrosion and hydrogen absorption have been established on the basis of general codes, experimental examinations and operational experiences. These criteria have been proven and are therefore used.

- **Assessment**

A check was made to determine whether the basic constructional features meet the functional requirements and to what extent the design can be compared with the German pressurised water reactors.

From references /PAZ 91/, /PLA 91/ it was concluded that the computer programs currently used for fuel rod design satisfactorily describe the behaviour in VVER reactors up to burnups of 40 MWd/kgU.

The first Soviet project limit for fuel rod damage in normal operation (1 % gas leakage, 0.1 % fuel/coolant contact) does not correspond to the Federal German requirements. According to German codes and standards, core internals, considering the intended mode of operation, must be able to withstand the loads throughout their entire in-pile life. The respective verifications relating to this are to be provided (R 4.1-19).

In accordance with the Soviet criteria mentioned above, startup with defective fuel elements is permitted at the beginning of a cycle. This procedure does not correspond to the Federal German assessment criteria, according to which, following the principles of the minimisation rule of the Radiation Protection Ordinance, each cycle normally is to be started with intact fuel elements. For this purpose fuel elements with suspected fuel rod damage are checked for leakage at the end of a cycle and the defective fuel elements are removed from the core or they are repaired for reuse so that damaged fuel elements are not used. The leak-tightness check should also be performed for fuel elements of the Stendal plant /KAB 91c/ so that defective fuel elements can be identified.

The three requirements of the 2nd Soviet project limit correspond to the first three requirements of RSK Guideline 22.1. In this Guideline it is further required that the release of fission products resulting from cladding tube damage remains closely restricted and that no changes in the geometry (cladding tube expansions) of the reactor core occur which prevent a sufficient cooling of the core. It is not known whether the manufacturer has undertaken investigations with respect to this requirement.

It must be demonstrated that the core internals are designed in such a way that the design limits required by emergency core cooling in accordance with RSK Guideline 22.1 can be observed under accident conditions (R 4.1-20).

To sum up, the basic constructional features meet the functional requirements for fuel elements and control elements. Core internals of this design have so far been used successfully in numerous plants.

A detailed assessment of the core internals can take place after presentation of the documents listed in recommendations R 4.1-17 to R 4.1-20.

For further investigations, for example, within the framework of a Federal German licensing procedure, documents according to the BMI survey of information required for examination in the legal licensing and supervisory procedure for nuclear power plants (ZPI) would have to be presented.

## 4.2 Pressurised Components

### 4.2.1 Object and Aim of the Assessment

The pressurised installations (containment and casing) and pipes of the primary and secondary system are the object of this assessment. The following items were included in the assessment:

- Installations and pipes of the primary system, which are under operational pressure, i.e. reactor pressure vessel, pressuriser, casing of the main coolant pumps, steam generator, main coolant lines, pressure maintaining system and
- installations and pipes which are required for cooling the nuclear fuel, i.e. the emergency cooling system and core flooding tank of the primary system as well as feedwater and main steam system, feedwater tank and preheater of the secondary system.

Pipes of a nominal diameter less than DN 250 were only considered in individual cases, as their replacement or reinforcement is possible without restriction, if required.

The aim of the examination is to analyse the preventive measures to avoid large leaks in the primary and in the secondary system of the reactor plant. It was necessary to examine whether the integrity can be proven for the above scope of plant with the required safety. Loads during normal operation, operational transients and accidents are to be taken into account here. For this evidence, the following items need to be analyzed:

- Suitability of the materials used.
- Mechanical and thermal loads assumed in stress analyses

- Technical design details with respect to stress peaks and non-destructive testing
- Interactions of structural materials with plant coolant
- Quality assurance measures in fabrication, pre-assembly, and assembly.

The documents available for this analysis were insufficient for the assessment of the reactor plant so that some questions could not be answered adequately and had to remain partially unanswered.

### 4.2.2 Description of the Components

#### 4.2.2.1 Arrangement of the Components

Figures 4.2-1 and 4.2-2 provide an overview of the arrangement of the components in the reactor building. The reactor pressure vessel rests on a supporting ring below the lower nozzle ring and is a fixed point of the system. Further fixed points of the system are the points where main steam, feedwater and emergency feedwater pipes penetrate the containment. The four steam generators and the four main coolant pumps rest on non-rigid bearings. The steam generators are additionally restrained by shock absorbers. The main coolant pumps are located in the cold legs. The main coolant lines of all four loops are executed almost identically. The volume control line (surge line) branches off one loop via a fitting on the hot side to the lower pressuriser nozzle. A connecting pipe DN 300 leads from the upper nozzle of the pressuriser to the cold leg. This line contains a control valve with bypass. Nozzles, to which the pipes of the make-up and the emergency cooling system are welded, are positioned on the main coolant lines. Futher nozzles for measurement lines are also welded on. The individual components are separated from each other by concrete walls. Supports and carriers are partially designed as pipe whip restraints.

#### 4.2.2.2 Design of the Components

Low-alloy ferritic steels are used for the pressurised walls of containments, casings and the main coolant line. All inner surfaces having contact with coolant are weld-cladded with stabilized austenitic material, partially in several layers. The pipes of the main coolant lines are welded in the same way. The joint of connecting pipes less than DN 426 is first austenitically buffered and the connecting circumferential weld is performed austenitically. The nozzles for the temperature measuring points are excepted from this procedure.

- **Reactor Pressure Vessel**

The vessel wall (Fig. 4.2-3) consists of three seamless forgings with a wall thickness of 200 mm, two seamless forgings having a wall thickness of 292 mm with neck-shaped nozzles, a seamlessly forged, cone-shaped flange with 54 blind holes for the stud bolts and a curved bottom having a wall thickness of 225 mm. Four nozzles DN 850 each, to which the hot or cold legs of the loops, respectively, are connected, and two nozzles DN 300 each to connect the pipes with the core flooding tanks (Fig. 4.2-4) are arranged on two levels. The upper nozzle ring additionally includes a nozzle DN 100 to lead through control cables. The individual forgings are connected with each other by 2/3 X-circumferential welds. The nozzles DN 300 for emergency core cooling contain thermal shock sleeves (Fig. 4.2-5). Below the level of the main coolant nozzles there are no penetrations in the containment.

The head (Fig. 4.2-6) consists of a head flange and a curved head part having a wall thickness of 292 mm. The head is installed on the containment with a loose flange (pressure ring) with stud bolts. In the head there are 90 nozzles (Fig. 4.2-7) for the control and protection (SUS) system as well as the system for monitoring the fuel element temperatures and a nozzle DN 40 for venting gas from the reactor pressure vessel. The standpipes for the control and protection system are flanged to the head nozzles.

The coating of the inner surfaces by tape coating or manual electro-coating has a thickness of 8 mm at the vessel wall and 7 mm in the head with a tolerance of -2 mm each. In addition, the seal areas of head and vessel wall are coated as well as the base of the welds at the outer surface of the head.

- **Pressuriser**

The pressuriser (Fig. 4.2-8) consists of three seamlessly forged rings having a wall thickness of 165 mm, a seamlessly forged ring having a wall thickness of 250 mm, through which the heating elements are inserted and two curved ends with a wall thickness of 176 mm. The individual sections are connected with each other with 2/3 X-circumferential welds.

Several nozzles DN < 50 are welded on in the cylindrical area for level and temperature measuring points. The root areas of the welds are rebored. At the upper end, besides several nozzles DN < 50, a raised manhole DN 400, a nozzle DN 200 to connect the spray line (Fig. 4.2-8) and a nozzle DN 220 to connect the line to the safety valves are welded on. The root area of the connecting welds is rebored. The spray nozzle contains a thermal shock sleeve. The root of the austenite-ferrite connector is not penetrated (Fig. 4.2-9). At the lower bottom a nozzle DN 400 to connect the volume control (pressurise surge) line is welded on. The root area of the seam is rebored. The nozzle comprises a thermal shock sleeve as well as several small weld-on nozzles for measuring temperature. The areas of the manually cladded shot welds are set off geometrically (Fig. 4.2-9).

- **Steam Generator**

The horizontal steam generator (Fig. 4.2-10) consists of two curved ends (120 mm thick), two cylindrical shells (145 mm thick), into which the hot or the cold collector, respectively, are welded with a nozzle DN 810, and 2 x 2 cylindrical shells having a wall thickness of 105 mm. The connecting weld of the collectors at the steam generator shell (Fig. 4.2-11) is executed austenitically. In both ends there is a manhole DN 400 each and in one end the austenitic emergency feedwater nozzle DN 100, which penetrates a local external cladding. The emergency feedwater nozzle (Fig. 4.2-12) and main feedwater nozzle DN 450 have thermal shock sleeves. "Antler-shaped" tubes from the steam header are welded to nozzles DN 345. The desalination lines are welded to nozzles DN 96 and DN 77, measurement lines to nozzles DN 15.

The collectors with DN 850 have a wall thickness of 160 mm in the area penetrated by the heater tubes (steam generator tubes). At the top they are sealed with a head DN 500 with 20 stud bolts. The heater tubes DN 16 x 1.4 mm wall thickness are disseminated through radial bores and tightly welded on the primary side. On the secondary side there remains an approx. 20 mm deep gap between heater tube and collector bore (Fig. 4.2-13). Severe plastic deformations at the inner surface of the bores and pollutant concentrations in the remaining gap during operation in several steam generators have led to crack formations in ligaments between the heater tube bores in the collector. These crack formations with total lengths of more than 1 m were primarily found in the fringe area between the solid and perforated parts of the cold collector wall after 7000 to 60000 hours of operation.

At the inner wall of the steam generator shell there are numerous carriers for heater tube supports and to fasten steam sieves welded on. At the outer surface of the shell, brackets are welded on for the supports and shock absorbers. The shock absorber carriers are not root penetrated.

- **Core Flooding Tank**

No constructional drawings are available for assessment.

- **Main Coolant Lines**

Plain tubes DN 850 and elbows DN 850 of the main coolant lines are forged seamlessly. The wall thickness of the tubes is 70 mm; the wall thickness of the elbows is 80 mm. The elbows are directly adjusted to the wall thickness of the tube at the weld (Fig. 4.2-14). Nozzles DN 30, DN 50, DN 100, DN 140 and DN 300 are welded onto the main coolant lines. The root area of all nozzle welds is rebored or hollowed, respectively. The cladding is performed as a tightly welded sleeve (Fig. 4.2-15). The austenitic-ferritic connector is part of the nozzle construction, the root is not penetrating. At all nozzles, leakage control of the compartment between cladding sleeve and pressure bearing wall is possible. The nozzles DN 100 have a thermal shock sleeve (Fig. 4.2-15).

### 4.2.3 Assessment Criteria - Comparison of the Essential Requirements of Codes and Technical Regulations

The primary system and the parts of the secondary systems which are located within the safety confinement (steam generator, main steam and feedwater pipes) were designed by Soviet engineering offices. The parts of the secondary system which are located outside the safety confinement (except tanks and turbine with attached auxiliary systems) can be designed by national engineering companies on the basis of the basic Soviet data (operational parameters and pipe dimensions).

The components of the pressurised encapsulation of the primary system were basically constructed, designed and manufactured in accordance with the technical regulations and standards for nuclear technology corresponding to the state of the art in the Soviet Union of the early 80s. The components of the secondary system were constructed and designed according to the applicable rules of the conventional steam and pressure technology.

Comparing the individual technical requirements of the codes and standards, it must be considered that the technical codes and standards reflect the technical experience gathered which has developed for specific constructions, for the use of specific materials and for the use of specific test procedures on the basis of the specifications provided by the manufacturer. The individual technical requirements therefore cannot be transferred unchecked to components of other reactor types.

#### 4.2.3.1 Scope of the Comparison of Regulations

The following comparison of the regulations in the national codes and standards considers criteria and requirements for damage prevention contained in the following main regulations:

- **Soviet Regulations**
  - Basic Principles for Ensuring Safety during Design, Construction and Operation of Nuclear Power Plants (OPB-73, Moscow 1973)

- Nuclear Safety Regulations for Nuclear Electric Power Plants (PBJa-04-74, Moscow 1974)
- Standards for Stress Calculation of Reactor Elements, Steam Generators, Containers and Pipes for Nuclear Power Plants, Experimental and Research Reactors and Nuclear Technology Plants (Moscow 1988)
- Regulations for the Erection and the Safe Operation of Installations of Nuclear Power Stations, Experimental and Research Reactors and Nuclear Technology Plants (Moscow 1982)
- Basic Conditions for Joint and Building-up Welding at Structural Elements and the Construction of Nuclear Power Stations, Experimental and Research Reactors and Nuclear Technology Plants (OP-1513-72, Moscow 1974)
- Control Regulation for Joint and Building-up Welding at Structural Elements and the Construction of Nuclear Power Stations, Experimental and Research Reactors and Nuclear Technology Plants (PK-1514-72, Moscow 1974)
- Temporary Methodology for Calculating Resistance to Brittle Fracture of Reactor Pressure Vessels (Moscow 1981)
- Joint and Building-up Welds of Nuclear Power Plant Installations. Methods of Ultrasound Control (OST-108.004.108-80, Moscow 1981)
- Heat Transfer Media of the Primary System of WWER-440 Nuclear Power Reactors (OST 9510165-85, Moscow 1985)
- Water Chemical Management of the Secondary System of WWER-Type Nuclear Power Stations (OST 34-37-769-85, Moscow 1986)
- Wasserchemische Fahrweise des Sekundärkreislaufes von Kernkraftwerken des Typs VVER, Änderung Nr. 1 (Water Chemical Management of the Secondary System of WWER-Type Nuclear Power Stations, Amendment No. 1) (OST 34-37-769-85, German translation of 1990)

Subordinate detailed regulations like GOST or OST, for example, for NPP products were taken into account only to a limited extent.

- **Federal German Regulations**

Sicherheitskriterien für Kernkraftwerke (Bundesministerium des Inneren, in der Fassung vom 21.10.1977) (Safety Criteria for Nuclear Power Plants) (Federal Ministry of the Interior, October 21, 1977)

- RSK-Leitlinien für Druckwasserreaktoren (GRS, Geschäftsstelle der Reaktorsicherheitskommission, 3. Ausgabe, 14. Oktober 1981)
  (RSK Guidelines for Pressurised Water Reactors), GRS, Office of the Reactor Safety Commission, 3rd edition, October 14, 1981)

- Sicherheitstechnische Regeln des KTA, Komponenten des Primärkreises von Leichtwasserreaktoren (KTA 3201).
  Teil 1: Werkstoffe (6/90)
  Teil 2: Auslegung, Konstruktion und Berechnung (3/84)
  Teil 3: Herstellung (12/87)
  Teil 4: Wiederkehrende Prüfungen und Betriebsüberwachung (6/90)

- (KTA Safety Regulations, Components of the Primary System of Light Water Reactors (KTA 3201).
  Part 1: Materials (6/90)
  Part 2: Design, Construction and Calculation (3/84)
  Part 3: Manufacture (12/87)
  Part 4: In-Service Inspections and Operational Monitoring (6/90)

- Sicherheitstechnische Regeln des KTA, Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises (KTA 3211)
  Teil 1: Werkstoffe (6/91)
  Teil 2: Auslegung, Konstruktion und Berechnung (3/91)
  Teil 3: Herstellung (6/90)
  Teil 4: Wiederkehrende Prüfungen (Entwurf 6/90)

- (KTA Safety Regulations, Pressurised and Active Components of Systems outside the Primary System (KTA 3211).
  Part 1: Materials (6/91)
  Part 2: Design, Construction and Calculation (3/91)
  Part 3: Manufacture (6/90)
  Part 4: In-Service Inspections (Draft 6/90))

- Sicherheitstechnische Regeln des KTA, Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren (KTA 3203, 3/84)
  (KTA Safety Regulations, Monitoring Radiation Embrittlement of Materials used in Reactor Pressure Vessels of Light Water Reactors (KTA 3203, 3/84)).

Subordinate standards like DIN, for example, and other rules and regulations applied to NPP products were only taken into account to a limited extent.

#### 4.2.3.2 Results of the Comparison between the Codes and Standards

In accordance with the object of the comparison, those requirements, the fulfilment of which can exclude a global failure of the pressurised installations and pipes because of manufacture-related deficiencies, were preferentially compared.

- **Requirements to be met by Materials**

According to KTA Rules, materials employed for the manufacture of installations and pipes of nuclear power stations must be licensed for their respective purpose of application. It must be possible to produce and process these materials in a controlled way and they must lead to an increased operational safety of the plant components compared to conventional use.

A consequence of this material concept is that only few, but proven, materials can be used, for which - subdivided into quality levels - special quality characteristics and proofs, and especially analysis and strength requirements apply. The requirements to be met by quality characteristics and proofs are on a higher level than those of the conventional codes and standards. The category and scope of the proofs are determined in the respective codes and standards.

The corrosion resistance is directed at the particular use. In the area close to the core possible damage by neutron irradiation is to be taken into account and limited.

Additional expert opinions on materials are to be prepared or additional requirements (superior qualities) compared to the requirements of the conventional codes and standards apply, respectively. Similar requirements apply to austenitic materials. For welded, austenitic plant components, for example, only stabilized materials are used. The requirements must principally be met by the base material, the weld material and by the heat affected zones.

A detailed comparison of the regulations in the German and Soviet codes has been compiled in Table 4.2-1. The measures provided for damage prevention in principle are similar with respect to the suitability and the selection of the materials. An exception is the lower depth of verification of satisfactory toughness of the base materials and welds. It is not required to reduce the area in the through thickness direction. The heat-affected zone is not incorporated into material testing. In the wall area close to the core of the reactor pressure vessel significantly higher neutron fluence values are permitted.

- **Requirements to Limit Stress**

Increased safety factors for design calculations are determined in the German and Soviet codes and standards for the different stress and load categories which essentially correspond to each other. Stress of operational transients and accidents are more strictly limited in the Soviet standards for calculation than in the German ones.

To protect against brittle fracture, the initiation of crack formation is to be avoided according to the Soviet codes and standards. By contrast, the German codes and standards in principle, permit a limited crack expansion. In both codes and standards specific verifications for the reactor pressure vessel are concurrently required for thermal shock loads as a result of cold water injections.

According to reports provided for informative purposes, the depth of verification for accommodating the loads actually occurring and those postulated, like, for example for temperature stratification of the operational medium in pipes with a temporarily stagnating flow (pressure maintaining system, feedwater) is significantly lower in Soviet licensing procedures.

- **Constructive Requirements**

German codes and standards recommend optimised constructional solutions, which have been proven in practical operation, in avoiding stress peaks and being suitable for complete testability using non-destructive methods.

The constructional recommendations relating to the execution of weld joints in the Soviet codes and standards only insufficiently consider the requirement of unlimited testability. Here weld joints with an unpenetrated root are permitted too.

- **Requirements to be met by Water Chemistry**

The permitted concentrations for oxygen and pollutants can be compared to the values stated in the VGB-Guideline and the EPRI Guidelines /BER 76/.

- **Requirements to be met by Quality Assurance including Non-Destructive Testing**

- General Requirements

It is the object of quality assurance to ensure compliance with the technical requirements contained in the codes and standards. For this purpose it is necessary to plan and determine the quality required, to produce this quality during fabrication and to always keep it at the required level during operation taking the loads into account.

The organisation of quality assurance corresponds to the division of tasks set forth in national legislation and regulations. In the Federal Republic of Germany, apart from quality assurance by manufacturers and operators of a plant, independent examinations are also performed by experts commissioned by the licensing authority. This examination by experts of a technical supervisory organisation independent of manufacturer and operator represents an essential element of the entire quality assurance. In the USSR, supervision was performed by different state organisations.

- Requirements to be met by the Qualification of Manufacturers and the Supervision of Fabrication

According to KTA standards only qualified manufacturers and optimised fabrication technologies are permitted for the fabrication of the product forms, installations and pipes. The manufacturer in particular must have a reliable quality assurance.

For the examination of the manufacturer, staff, fabrication procedures, fabrication and testing devices are to be taken into account. Already prior to the start of fabrication pre-examined fabrication documents and test sequence plans adapted to the fabrication sequence must be presented. Supervision of fabrication, performance of examinations and documentation of test results are carried out by the quality department of the manufacturer. In addition, there are supervisions and examinations by independent technical control organisations commissioned by the licensing authority depending on the quality level.

Special requirements apply to welding technology and supervision of welding. The specified requirements to be met by quality (mechanical-technological parameters) of base and weld material and heat-affected zone are to be demonstrated by procedural, work and batch tests. Principally only quality approved weld fillers are permitted. In accordance with the results of simulation examinations, the welding conditions are to be adjusted in such way that the toughness of the heat-affected zone is limited for ferritic materials and the heat treatment lamination technique is used, if possible. If diameter and fabrication sequence do not permit this, it should be counter-welded and the weld should be ground on the inside and outside. All examinations performed during fabrication are to be documented by the manufacturer, supplier and, if required, by the independent technical supervisory organisation.

In Table 4.2-2 the individual requirements of the German codes and standards are compared with the Soviet codes and standards. For essential requirements the regulations are similar.

- Requirements to be met by Non-Destructive Testing

German as well as Soviet codes and standards require non-destructive examinations during fabrication and assembly. The examination requirements, especially for weld joints (test procedures, verification sensitivities and calibration procedures) are determined in codes and standards. Both national codes and standards restrict permissible failure configurations. The RSK Guideline, however, requires an assessment of the indication together with the state of the material. It is the object of this kind of procedure to avoid impairments of quality by unnecessary repairs.

Contrary to the requirements of the German codes and standards, the influence of existing test restrictions on the safety of a component does not need to be assessed according to the Soviet codes and standards. Such test restrictions in particular exist in the form of permitted excess weld materials and non-penetrated roots for numerous nozzle constructions.

The test sensitivity according to KTA required for US-examination is not met in some wall thickness ranges. The number of acoustic irradiation directions required by the Soviet codes and standards during US-examination (number of directions, from which each volume element must be examined) is lower than according to KTA, and is partially even below the requirements for pressurised or active plant components outside the primary system, respectively. The Soviet codes and standards leave the operator the choice between the penetration method of testing and magnaflux testing. The German codes and standards, however, require magnaflux testing in cases where measurements of components and material properties allow this. For radiographic testing, the German codes and standards require a higher contrast of the radiographies.

### 4.2.4 Results of the Analyses

#### 4.2.4.1 Suitability of the Materials Employed

For the installations and pipes of the primary system of the WWER-1000 reactor plant, only constructional materials are employed which are permitted according to Soviet codes and standards (Table 4.2-3). These standards or subordinate regulations, respectively, specify the chemical composition, heat treatment as well as mechanical-technological parameters, on which the design is to be based. All materials permitted were tested and assessed with respect to their intended use, independent of the manufacturer and processor by the responsible Soviet ZNIITMASCH institute. Here attention is to be paid to the deviations from the German codes and standards (e.g. testing in thickness direction, toughness concept, simultaneous examinations) mentioned in the comparison of the requirements in the codes and standards (Section 4.2.3).

The resistance of the reactor pressure vessel material and its weld joint against neutron irradiation in the area close to the core is of particular importance for safety. For this purpose fracture toughness and the shift of ductile-to-brittle transition temperature as a function of neutron flux density, neutron fluence and copper, phoshorus and nickel contents are to be assessed. It is to be taken into account here that because of the relatively small water gap between the core and RPV wall, the integral neutron fluence at the RPV wall can exceed the limit determined in the German codes and standards in the course of the designed service life-time of the plant. To monitor the state of the material, examinations of suspended samples (notched bar impact bending tests, tensile tests, fatigue tests, mechanical fracturing tests) are intended.

For the base material of the reactor pressure vessel 15Ch2NMFA in the area near the core the special quality 15Ch2NMFA-A is used, for which Cu 0.08 % and P 0.010 % are specified very low and the ductile-to-brittle transition temperature $T_{KO}$ -25°C is specified low. Nickel increases the inclination of the material to neutron embrittlement. The influence of a nickel content of up to 1.5 % on the shift of the ductile-to-brittle transition temperature can, however, not be assessed yet.

For the weld material in the area near the core Cu is specified with 0.08 %, P with 0.012 %, Ni = 1 %. In contrast thereto, Vishkarov et al. /VIS 83/ in 1983 published results on embrittlement behaviour of the weld material, according to which the nickel content of the weld material must be at least 1.6 % to reach the required toughness parameters. With this modified weld material the embrittlement behaviour during irradition with fast neutrons was examined. It remains unclear whether these are laboratory examinations or secured results for industrial use.

With respect to neutron embrittlement of the base material and the weld material of the reactor pressure vessel more detailed analyses are still necessary, which apart from the influence of integral neutron fluence and nickel content also take into account the frequently discussed influence of neutron flux density on the progress of embrittlement (R 4.2-1).

Radko et al. /RAD 85/ in 1985 reported on the crack formation tendency of the interface base material cladding in the root area of the assembly welds of the main coolant line. During in-service examinations it had been found that local hardening and cracks below the cladding had been the starting point of fatigue cracks. More detailed investigations (welding technology, detectability with non-destructive test methods, loads on the respective positions) are still necessary with respect to this problem (R 4.2-2).

There were no original Soviet documents for the assessment of the material 06Ch12N3DL which is used for the spiral housings of the main coolant pumps (R 4.2-3).

#### 4.2.4.2 Load Assumptions and Design of the Components

There are no calculations with respect to load assumptions and to the design of the components including a service-life analysis (R 4.2-4).

#### 4.2.4.3 Construction and Inspectability

Constructive execution and inspectability with non-destructive methods were tested and assessed exclusively on the basis of drawings assigned to the Stendal A NPP project. The applicability of automatic ultra-sound testing, especially for in-service testing was in the centre of interest. Existing test restrictions of the individual components are mentioned.

- **Testability of the Components of the Primary System Using Non-Destructive Methods**

For non-destructive testing of the pressurised encapsulation of the primary system the following essential requirements are contained in the RSK Guidelines or the KTA-Rules, respectively:

All product forms (steel sheets, forged parts, cast parts, tubes, welds) are to be tested completely in a non-destructive way by volume and at the surfaces. All components are to be designed in such a way that testing is possible during fabrication and that in-service testing is possible to a sufficient degree.

As Soviet regulations for NPP prefer radiographic testing to ultra-sound testing, while RSK-Guidelines and KTA Rules prescribe ultra-sound tests, the applicability of ultra-sound tests is to be investigated (R 4.2-5).

The inspectability of the individual primary system components is considered below. Only constructional drawings are available for this purpose. Test restrictions caused by the construction are shown. There is no assessment of the non-destructive tests of product forms at the manufacturers'. In those areas, however, where an examination is not possible in the final state, testing for failure during an earlier fabrication stage has to be incorporated into the assessment.

- Reactor Pressure Vessel

The bottom of the reactor pressure vessel below the nozzle rings can be examined from inside using ultra-sound without significant restrictions. In the area of the nozzle shots, where the circumferential welds are, there are testing restrictions owing to the constructive execution of weld-on parts and changes of the wall thicknesses. The welds from the inside can only partially be tested from one direction. The test result can be improved by the use of additional test angles. Owing to the constructive execution, ultra-sound testing from the outside in the area of the nozzles is strongly restricted. Nozzles and nozzle edges near the surface can be tested with ultra-sound from the inside. There are test restrictions for ultra-sound testing of the outer surface owing to the constructive execution of the nozzle joints. The nozzle welds of the emergency feedwater line cannot be tested from the inside because of the thermal protection tubes welded on. The test restrictions indicated require an analysis of the fabrication tests (R 4.2-6).

The area outside the nozzle field at the head of the reactor pressure vessel can be tested with ultra-sound. The inspectability is restricted at the circumferential weld of the head. Testability of the nozzle field in the head of the reactor pressure vessel is restricted so that fabrication testing becomes highly significant (R 4.2-6). An in-service ultra-sound testing of the volume of the webs between the nozzles is hardly possible from the outside, as lines of the head nozzle flange connections indicating leakages run on the surface of the head. In-service surface crack tests of the inner and outer surface are possible. A sufficiently representative scope of web testing must be ensured which can be achieved by the development of manipulators and possibly by changing the constructive design. The weld joints of the nozzles penetrating the head also require thorough testing. An examination concept for testing nozzles and the perforated area is therefore to be established. Here it is to be taken into consideration how the nozzles are manufactured and assembled (R 4.2-7). Owing to the restricted inspectability of the nozzle field, leak-monitoring systems for localising leakages at the RPV-head penetrations are necessary (E 4.2-8).

Stud bolts, nuts, plain washers and threaded blind holes of the reactor pressure vessel can be tested.

In summary, it can be said that a positive examination result for the reactor pressure vessel including the head can only be arrived at if the restrictions of ultra-sound inspectability indicated can be compensated by an analysis of the manufacturing documentation (R 4.2-6).

- Pressuriser

The pressuriser consists of ferritic steel with an austenitic inner coating. The seamless shots and the curved bottoms are connected with each other by circumferential welds. Such weld joints can principally be tested using ultra-sound. At the changes of wall thicknesses like, for example, from the plain cylindrical part to the bottoms or from the shot with the heater rods, respectively, ultra-sound inspectability is restricted. Excess weld materials also impair inspectability. Owing to the geometric execution and partially to thermal protection sleeves, there are test restrictions for ultra-sound testing at the nozzles. Shots and bottoms appear to be suitable for ultra-sound testing (outside the welds).

- Steam Generators

The steam generators are part of the primary system. There is one circumferential weld in the collectors. The collectors can be examined from the inside, but ultra-sound testing is restricted owing to the geometry of the acoustic irradiation direction. The non-destructive examination of the secondary welds between collectors and collector nozzles of the steam generators seems to be difficult. For these welds possibilities for inspection must be created. The collector heads, screw bolts and nuts are accessible for non-destructive examinations.

According to the documents presented, inspectability of the steam generator tubes in the bent areas is restricted. It is therefore considered to be necessary to develop an examination concept for in-service examinations on the basis of the eddy-current test method, which is also able to detect operationally induced damage in the bend areas in time (R 4.2-9). At some positions nozzles are arranged too close to the circumferential welds of the steam generator shell so that there are locally limited test restrictions for these circumferential welds.

- Main Coolant Lines

The main coolant lines are of seamless tubes and elbows of ferritic steel with austenitic cladding, which are welded with circumferential welds. Such weld joints can principally be examined with ultra-sound. As there are differences in the wall thicknesses between elbows and tubes, here an acoustic irradiation is only possible from the tube side so that there are test restrictions. Furthermore, there are local excess weld materials in the root area which also impair ultra-sound examination. Apart from the weld joints, the base material areas of the main coolant lines are to be examined in a non-destructive way which is possible because of the relatively simple geometry of the tubes und elbows.

- **Testability of Secondary System Components with Non-Destructive Methods**

There are no documents available relating to this point.

#### 4.2.4.4 Interactions of Constructional Materials and Operational Media

- **Primary System**

For the inner surfaces of the primary system having contact with media, the same materials as in the WWER-440 type are used. Therefore the chemistry of the primary system largely corresponds to the mode of operation practised in WWER-440 type plants: The alkalisation is effected with caustic potash solution; for radiolysis suppression ammonia is added. For reactor physics reasons, the boric acid concentration at the beginning of power operation of a cycle, with a maximum of 13.5 g/l, is higher than for plants of the WWER-440 type (8 g/l). All other standard values are almost identical.

According to operational experience gathered during the operation of WWER plants, this type of operation is proven. No significant damage induced by corrosion has been observed at the austenitic inner surfaces of the primary system or on the fuel rod cladding. From the viewpoint of material technology, it is not necessary to alter the management of water chemistry.

As operational experience proves, the admission of filter material and chemicals for decontamination via facilities for water treatment or the make-up system cannot be excluded. The possibilities of pollution are to be analysed and to be eliminated by structural measures (e.g. by installing mechanical resin catchers) (R 4.2-10). To increase operational safety, backfitting with a state of the art, automatic monitoring system is recommended, which renders continuous or almost continuous monitoring of the essential chemical parameters in the primary system and the make-up system possible (R 4.2-11).

- **Secondary System**

The secondary system, as in plant of the WWER-440 type, is designed as a combined construction. Besides nickel chromium and non-alloy or low alloy steels, copper alloy is also used. For the Stendal Nuclear Power Plant condenser, turbine and turbo-injection pump, tubes of CuNi10Fe or CuNi10Fe1Mn are intended.

The different corrosion behaviour of the types of material mentioned renders compromises in the water chemical management necessary. The plurality of operators of VVER plants in recent years followed the Soviet special domain standard, OST 34-37-769-85. According to this standard, the conditioning of the secondary system was effected by hydrazine, whereby pH-values of 7.5 - 8.5 in the feedwater were reached. The standard further provides a 100 % treatment of the turbine condensate with mixed bed exchangers.

This mode has not been proven. The relatively low pH-value in almost oxygen-free water leads to considerable erosion corrosion of non-alloy steels. The high-pressure preheater and separator/intermediate superheater are most affected, resulting in high corrosion product concentrations in the feedwater. Especially in plants of the WWER-1000 type, it is impossible to observe the set limit of 15 µg/kg for iron concentration, according to the current operational experience. The admission of corrosion products into the steam generator leads to a rapid growth of coats which promote corrosion.

In an amended version of the special domain standard, a modified water chemical management of the secondary system is recommended. The pH-value of the feedwater shall be increased to 9.0 +/- 0.2 by an additional dose of ammonium on the suction side of the feedwater pumps. A reduction of the erosion corrosion ratio can be expected from this mode, particularly in the region of one-phase flow. On the other hand, additional problems for the operation of the condensate cleaning system are introduced with this measure (e.g. impairment of the pure condensate quality and increase of regeneration cost) and corresponding consequences for local corrosion in the steam generators owing to increased pollution cannot be excluded.

In areas particularly endangered by erosion corrosion, steel alloys (e.g. 10CrMo910) are employed to prevent locally limited corrosion. The installation of electromagnetic or mechanical high-temperature filters is being investigated. For the given material concept, reliable protection of the secondary system components against damage induced by corrosion with water chemical means only cannot be ensured. The basic preconditions for the application of the high-AVT-mode, which has been proven in German plants and can also be expected to control the corrosion behaviour in VVER plants, can be created by doing without copper-alloy materials and realising technically tight condensers. The material concept of the secondary system is therefore to be revised throughout, with the objective of preventing local corrosion at steam generator tubes and erosion-corrosion in the condensate and feedwater areas by improved water chemistry conditions (R 4.2-12).

For monitoring all relevant chemical parameters an automatic measuring system, appropriate to the water chemical mode, is to be installed (R 4.2-11).

#### 4.2.4.5 Quality Assurance

For the erection of nuclear power stations outside the USSR it was contractually agreed to mutually accept product-related quality assurance. The realisation of the requirements in the codes and standards in component-related quality assurance programs and the fulfilment of these programs outside the USSR is only known in individual cases. Realisation of requirements and fulfilment of quality assurance programs therefore cannot be assessed at present. For this purpose it will be necessary to visit the producers of the components.

So far it has not been possible to inspect and assess test results and evidence for quality during fabrication of the components at the manufacturers' facilities.

The analysis of quality assurance was therefore confined to the quality characteristics of components which can be checked by non-destructive methods even in the assembled state, i.e. failures at the surface and in the volume of the materials. The behaviour of the components then could be assessed on the basis of preset loads, known material properties and the failure status determined.

Applicability and limitations of applicability of ultra-sound testing were described and assessed for the specific components in Section 4.2.4.3.

### 4.2.5 Safety-Related Assessment and Necessary Improvements

The safety-related assessment of the pressurised components and pipes of the primary and secondary systems are carried out in two steps:

1. Assessment of the criteria set forth in codes and standards
2. Assessment of the realisation of the requirements set forth in codes and standards during design and construction of the components.

#### 4.2.5.1 Assessment of the Criteria set forth in Codes and Standards

The comparison of the criteria of the codes and standards shows that the intended measures for damage prevention differ in part. On essential points, the requirements principally correspond to each other. In comparison with the German codes and standards, Soviet codes and standards do, however, require:

- a less rigorous verification of satisfactory toughness of base materials and weld material (lacking evidence in thickness direction) as well as for testing of the heat-affected zone.
- a less rigorous verification of the absorption of occuring and postulated loads (operational transients, accidents, earthquakes) for installations, tubes, pipe whip restraints and supports.
- a less rigorous verification of non-destructive examinations, especially for ultra-sound testing (smaller number of acoustic irradiation directions and test angles)
- no safety-related assessment of restrictions for non-destructive testing,
- no immediate limitation of neutron fluence in the area of the reactor pressure vessel close to the core (neutron fluence is, however, indirectly limited by ductile-to-brittle transition temperature).

It is considered that the codes and standards under consideration permit deviation from the preset values in individual cases, if technical arguments are presented at the same time to demonstrate that the proposed deviation does not adversely affect the quality of the components. The individual components and pipes must therefore be examined to identify whether the less rigorous approach to damage prevention can be removed sufficiently by additional evidence and inspection, supplementary material testing, as well as measures for reducing operational loads.

#### 4.2.5.2 Assessment of the Components

The application of the preset values set forth in the codes and standards to the design of the installations and pipes was examined with the help of the present constructional drawings and material specifications. For a conclusive assessment of the measures to prevent damages, specifications and quality assurance programs of the individual component manufacturers as well as material tests, especially of 06Ch12N3DL (spiral housing of the main coolant pumps), are still to be examined. Toughness calculations are to be presented (R 4.2-3). With respect to life-time analysis of components and pipes, it is still to be clarified whether the stress reversals indicated for the loads mentioned are criteria for calculating the design or their results (R 4.2-4). It could not be derived from the documents (specifications) examined, whether the entire plant or parts of the plant were based on a concept excluding breaks. Thus breaks, for example, in the main steam, feedwater and emergency feedwater system outside the containment and also breaks with consequential damage, because of the unfavourable routing and lack of pipe whip restraints, cannot be excluded (R 4.2-13). This is to be taken into account for the accident analysis (cf. R 5.1-12).

The operational experience of VVER plants has shown that the water chemistry of the primary system is well suited to safe operation. However, experience also shows that, through the special water treatment system, and the make-up system, filter material or decontamination chemicals can be brought into the primary system. Here it is necessary to eliminate these possibilities by constructional measures (R 4.2-10).

- **Reactor Pressure Vessel**

Material selection and constructional form of the reactor pressure vessel largely correspond to the Soviet codes and standards. The knowledge of the influence of the nickel content on the inclination of the reactor pressure vessel to neutron embrittlement is to be broadened. Until presentation of a status report, measures to ensure sufficient long-term safety reserves, e.g. by shielding elements on the edge positions of the reactor core are to be taken (R 4.2-14). Suspended samples are provided to monitor the progress of neutron embrittlement of the pressure vessel wall in the core area. These suspended samples are arranged at the edge of the reactor core within the core baffle. It is still to be investigated whether the higher neutron flux density in the area of the suspended samples and, in particular, whether the higher irradiation temperature allows direct conclusions from the suspended samples to be applied to the wall of the reactor pressure vessel (R 4.2-15). The validity of the material investigations using these samples is therefore to be reconsidered, as the results will possibly be too favourable.

Furthermore, the reactor pressure vessel as a precaution will have to be protected against cold overpressure events (R 4.2-16).

No restrictions are expected from the stress analysis of the reactor pressure vessel still to be analysed.

The constructional design has examination restrictions at some positions (geometry of the nozzles to connect the main coolant line, thermal protection sleeves in the emergency injection nozzles). To ensure sufficient damage prevention, it will be necessary to adapt the examination techniques available for the respective examination tasks and to assess the remaining examination restrictions from the viewpoint of safety (R 4.2-17).

The assembly and welding of the nozzles in the head of the reactor pressure vessel are still to be analysed in more detail. A concept for the examination of the webs between the nozzles (field of perforations) and the welding of the nozzle at the inner surface of the head is to be elaborated (R 4.2-7). These recommendations are further supported by operational experience, according to which there were failures in this area in the plants of other manufacturers.

Although the constructional design of the vessel and head show points with restricted inspectability, a sufficiently representative statement on the quality is possible by adaptation of the examination techniques and returning to the manufacturer's documentation.

- **Pressuriser**

Material selection and form of construction largely correspond to the criteria set forth in the Soviet codes and standards. The operational experience of different types of pressurisers (WWER-440, for example) shows that thermal stress as a result of the non-steady operation was underestimated, particularly with respect to the design of the spray nozzle. Here calculations with refined methods (e.g. FEM) for the spray nozzle and also for the connection of the volume control line are possibly required (R 4.2-4).

The constructional design at some points restrict examination, especially in those areas with changes in the wall thickness, for example at the ends, in the section with the heat elements, at the nozzles because of non-penetrated roots, but also at the excess of weld material between the sections. The restrictions can largely be removed by improving the geometry and adapting the examination techniques available. If the adapted examination techniques do not prove to be sensitive enough for fault detection in the areas of non-penetrated roots, changes in the design of the respective nozzle will have to be considered (R 4.2-5).

Although the constructional design of the pressuriser has some points with restricted inspectability, reworking, adaptation of the examination techniques available and, if necessary, constructional changes render a sufficiently representative non-destructive examination possible.

- **Steam Generator**

The selection of the material and the form of construction of the steam generator shell as well as the collectors and the heater tubes largely correspond to the Soviet codes and standards. The operational experience with different types of steam generators, but with the same materials, show that an intensified control of water chemistry is required (R 4.2-11). The different corrosion behaviour of the materials (the materials of the entire secondary system are to be taken into account here) makes compromises in the water chemical management necessary. Under the given conditions, reliable protection of the heater tubes in the steam generators against hole corrosion and stress corrosion cracking and the simultaneous prevention of erosion-corrosion of non-alloy steels, especially in the separator/intermediate superheater and the high-pressure preheater, is not possible by water chemical means alone. The realisation of technically tight condensers and the change-over to the high-AVT-mode has been proven in German plants doing without copper-alloy materials (R 4.2-12).

Operational experience shows that, until the end of 1991, inner leakages from the primary into the secondary system were detected in 36 steam generators, after a relatively short operational period. The damage exclusively occured at the cold collectors. Starting out from the secondary side, corrosion and crack formation in the webs between the holes, where the heater tubes are installed, led to leakages. The starting point of this damage is the lowest row of holes. Constructional changes, changes in the manufacturing technology and additional heat treatment so far have proven unable to reliably prevent the damage. It is necessary to extent the knowledge of the damage mechanism and to analyse the influence of the cracks on the integrity of the collectors. At the same time, non-destructive test procedures for early detection of the cracks must be worked out and implemented. (R 4.2-18).

In addition, the effects on the steam generator shell of failure of the steam generator collector with fast relief of the primary system pressure are to be investigated. Impacts of jet and reaction forces on neighbouring steam generators as well as the effects on the integrity of the containment are also to be analysed, if necessary (R 4.2-19).

A part of the steam generator shell can only be tested by ultra-sound to a limited extent, becaus of excess weld material, nozzles too close to the circumferential welds and different wall thicknesses. By reworking and using special examination techniques, supplementary examination results can however be achieved. At present it cannot stated conclusively whether the changes from ferritic to austenitic material (joint weld shell/collector) can be examined sufficiently. For some nozzle constructions changes are therefore necessary (R 4.2-5).

An examination concept for in-service inspections which is also able to detect possible operationally induced damage in bent areas with time, must be worked out (R 4.2-9).

- **Core Flooding Tank**

An assessment is not possible, as there are no design documents or specifications available.

- **Main Coolant Lines**

The selection of material and the form of construction of the main coolant and connecting lines largely correspond to the Soviet codes and standards. Documents relating to the qualification of materials could, however, not be examined. The calculations for the static loadings of the pipes were not available for examination so that no statements can be made on the stress level, in particular at the junctions of joining pipes (R 4.2-20). As there frequently are higher loads at the junctions of pipes, caused by temperature stratification and fluctuations, it is also not possible to comment on the assumed life-time. The same applies to the accommodation of loads from earthquakes as well as jet forces in case of large leakage. Corresponding stress calculations are to be presented (R 4.2-4). It also remains unclear whether the supports can assume the functions of pipe whip restraints. Further analyses are necessary with respect to this problem (R 4.2-21).

Owing to non-penetrated roots, restrictions for non-destructive examination exist at almost all nozzles on the main coolant line. It is still to be clarified to what extent the adaptation of different wall thicknesses of plain tubes and elbows influences inspectability. Welded thermal protection sleeves in the emergency injection nozzle make the examination of the inner surface of the bearing wall in the nozzle area impossible. The surfaces of the weld joints are not sufficiently level in different areas to use ultra-sound testing for in-service inspections to the required extent. The surfaces are therefore to be reworked accordingly. For the still remaining areas of examination restrictions, surface crack tests can be carried out from the inside, if necessary (R 4.2-5).

- **Main Coolant Pumps**

An assessment is not possible, as there are no design documents or specifications available.

- **Emergency Cooling System**

The three-train LP-area (LP-emergency cooling system) of the emergency cooling system is fed from a common non-sectioned emergency boron tank (V = 630 $m^3$).The three pipes (DN 600) from the tank sumps to the isolating valves upsteam of the suction nozzles of the pumps are particularly important for safety as they lead through the containment wall.

In case of failure of one or more pipes between tank sump and isolating valve, coolant from the containment would be lost continuously. At the same time the containment would be opened at the break.

As the containment sumps cannot be locked directly, the materials used, as well as design, inspectability and monitoring of these pipes, have to meet special requirements. According to the present knowledge, larger leakages at these pipes between the tank sump and the isolating valve cannot be excluded (cf. Section 6.3.1.3 and R 6.3-3).

- **Feedwater and Main-Steam System**

There were no reliable documents available for a detailed assessment of the feedwater and main steam system. The operational experience of the same type of plants shows that the non-alloy or low-alloy steels, respectively, used for containment and pipes are only suitable for operation to a limited degree. The different corrosion behaviour of the steels and the copper-containing materials used in the condensers makes compromises in the water chemical management necessary. Under the given conditions, neither local corrosion at the heater tubes of the steam generators nor erosion-corrosion in the condensate system can be prevented by water chemical means alone. Here it makes sense to revise the material concept of the secondary system as a whole. In German plants doing without copper-alloy materials, the realisation of technically tight condensers and change to the high-ATV-mode have been proven (R 4.2-12).

According to the present state of knowledge (use of simple steels, lacking results from material testing, unfavourable routing, lacking pipe whip restraints), breaks of pipes and even consequential damages cannot be excluded in the main steam and feedwater area (R 4.2-13).

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be met by quality characteristics and proofs are on a higher level than those of the conventional codes and standards. The category and scope of the proofs are determined in the respective codes and standards.

The corrosion resistance is directed at the particular use. In the area close to the core possible damage by neutron irradiation is to be taken into account and limited.

Additional expert opinions on materials are to be prepared or additional requirements (superior qualities) compared to the requirements of the conventional codes and standards apply, respectively. Similar requirements apply to austenitic materials. For welded, austenitic plant components, for example, only stabilized materials are used. The requirements must principally be met by the base material, the weld material and by the heat affected zones.

A detailed comparison of the regulations in the German and Soviet codes has been compiled in Table 4.2-1. The measures provided for damage prevention in principle are similar with respect to the suitability and the selection of the materials. An exception is the lower depth of verification of satisfactory toughness of the base materials and welds. It is not required to reduce the area in the through thickness direction. The heat-affected zone is not incorporated into material testing. In the wall area close to the core of the reactor pressure vessel significantly higher neutron fluence values are permitted.

- **Requirements to Limit Stress**

Increased safety factors for design calculations are determined in the German and Soviet codes and standards for the different stress and load categories which essentially correspond to each other. Stress of operational transients and accidents are more strictly limited in the Soviet standards for calculation than in the German ones.

To protect against brittle fracture, the initiation of crack formation is to be avoided according to the Soviet codes and standards. By contrast, the German codes and standards in principle, permit a limited crack expansion. In both codes and standards specific verifications for the reactor pressure vessel are concurrently required for thermal shock loads as a result of cold water injections.

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## Tables, Section 4

## Table 4.1-1 Description of the Fuel Element and the Fuel Rod

| | |
|---|---|
| **Description of the fuel element** | |
| Width across | 23.4 cm |
| Number of fuel rods | 312 |
| Number of guide tubes for absorber rods | 18 |
| Central guide tube | 1 |
| Arrangement of fuel rods and guide tubes in triangular grid with a distance of | 12.75 mm |
| Length of the fuel elements with head and foot part | 4.75 m |
| Length of the fuel rod | 3.825 m |
| Length of the fuel zone | 3.53 m |
| Number of spacers | 15 |
| Axial distance of spacers | 0.255 m |
| Spacer material | 08X18H10T |
| (with 69.5 % Fe, 18 % Cr, 11 % Ni, 1.5 % Mn) | |
| **Description of the fuel rod** | |
| Outer diameter of the fuel rod cladding tube | 9.1 mm |
| Thickness of the cladding tube | 0.69 mm |
| Outer diameter of the fuel pellet | 7.57 mm |
| Diameter of the inner bore in the fuel pellet | 1.4-2.3 mm |
| Height of the fuel pellet | 9 - 13 mm |
| Cladding tube material | zirconium-niobium alloy with 1 % niobium |
| Initial internal pressure (cold state) | 1.2 MPa |

**Table 4.1-2 Description of the Guide Tubes for the Absorbers and the Instrumentation as well as the Absorber Rods**

| | |
|---|---|
| **Description of the guide tubes for the absorber rods:** | |
| External diameter of the guide tubes | 12.6 mm |
| Wall thickness of the tubes | 0.85 mm |
| Tube material | 06X18H10T |
| **Absorber rods of the control elements:** | |
| Axial height of absorber material | 3.71 m |
| Diameter of absorber rod | 8.2 mm |
| Absorber material | boron carbide (natural composition of boron) |
| Service-life of absorber rods | 1 to 2 years |
| Dropping time of control elements | 1.5 to 4 s |
| Speed of control elements | 2.0 cm/s |
| **Absorber rods of burnable poisons:** | |
| Axial height of absorber material | 3.55 m |
| External diameter of absorber | 9.1 mm |
| Wall thickness of the cladding tube | 0.65 mm |
| Diameter of absorber pellet | 7.72 mm |
| Burnable absorber material | boron carbide $B_4C$ or chromium-boron compound $CrB_2$ in aluminum alloy matrix or gadolinium |
| Service-life of absorber rods | 1 year |
| **Description of the central guide tube for instrumentation:** | |
| External diameter | 11.2 mm |
| Wall thickness of the tube | 0.8 mm |
| Tube material | zirconium alloy with a niobium content of 1 % |

**Table 4.2-1 Technical Requirements to be met by the materials**

| Technical Requirements to be met by the materials (German regulations) | Annotation or determined in Soviet codes and standards, respectively: |
|---|---|
| fine-grained melting (determination of grain-size) | 1) WM 54 |
| material expertise | 1) WM 54 |
| low inclination to embrittlement upon radiation<br>(areas close to core) | 1)<br>TGL 43 272: 3.1.3 or 4.2.1, resp. |
| welding suitability<br>- hardening suitability<br>- investigation of cross-sectional or tangential polish, resp. | partially 1); TGL 43 272: 1.1.1<br>2)<br>3) |
| Analyses:<br>- restrictions of analyses<br>- special qualities | 1) WM 54 |
| Liquation behaviour | 3) |
| Corrosion resistance | 1) WM 54 |
| Samples:<br>- simulated heat treatment<br>- immediate samples | 1) 2)<br>TGL 43 272: 5.5.13 |
| Hardness test of quenched and subsequently drawn steel (heat treatment control)) | 1) 2 |
| Testing at the centre of wall thickness at s ≥ 150 mm | 3) |
| Testing in thickness direction | 3) |
| Strength behaviour:<br>- $A_V$-T-curves<br>- 68 J-criterion<br>- 100-J upper position | 1) 2)<br>TGL 43 272: 4.2.1 or 4.2.2, resp. |
| Component strength (large samples) | 3) |
| Ductile-to-brittle transition temperature:<br>- determination<br>- evidence | 1) 2) |
| Austenite only:<br>Ik-resistance | 1)<br>TGL 43272: 4.1.6 |
| Austenite only:<br>- testing for non-metallic inclusions<br>- grain size (US-testability) | 1) 2) |

1) in principle met
2) documents contain references to information available
3) information at present not available, secondary testing possibly required

**Annotations**

| | |
|---|---|
| TGL 43 272 | corresponds to the Soviet regulation for the construction and the safe operation of equipment and pipes in NPP, experimental and research reactors and plants (constructors' regulations) |
| WM 54 | part of TGL 43 272 (list of materials) |
| TGL 43 273 | corresponds to the Soviet OP 1513-72 (design regulation) |
| TGL 43 274 | corresponds to the Soviet PK 1514-72 (test specification) |

**Table 4.2-2 Technical Requirements to be met by Manufacture**

| Requirements to manufacture (German regulations) | Annotations or determined in the Soviet codes and standards, resp. |
|---|---|
| Requirements to be met by the manufacturer(qualification, quality assurance, audits) | 1) or 2), resp.TGL 43 272: 2.1, 6.11, 10<br>TGL 43 273: 1.1, 1.4 |
| Production documents<br>Pre-examination documents | 2)<br>TGL 43 272: 5.1.5, 5.4.1 |
| Documentation | 2)<br>TGL 43 272: 12.4, 15.1 |
| Welding engineering and monitoring | 1)<br>TGL 43 272: 5.4, 6.2.5, 6.3<br>TGL 43 273: 1.5, 5, 6.4 |
| Welding examinations | 1)<br>TGL 43 272: 5.4.2<br>TGL 43 273: 5 |
| Constructive design<br>(e.g. length of examinations) | 2)<br>TGL 43 272: 3.4.3, 3.4.4, 3.4.14 |
| Welding design | 2)<br>TGL 43 272: 3.4.2 |
| Procedural, work and batch testing | 1) or 2),<br>resp.TGL 43 272: 6.2, 6.7, 6.10<br>TGL 43 273: 1.4 |
| Welding fillers (qualification test) | 1)<br>TGL 43 272: 4.5<br>TGL 43 273: 3.9 |
| Offset of edges (preset values) | 1)<br>TGL 43 272: 5.3.8 to 5.3.11 |
| Heat treatment | 1)<br>TGL 43 272: 5.5<br>TGL 43 273: 10 |
| Repairs<br>(modes of procedure and categories) | 1) or 2),<br>resp.TGL 43 272: 12, 13<br>TGL 43 273: 11 |
| Strength behaviour of weld joints as well as admissible hardnesses<br>(GW, WEZ and SG) | 1) or 2),<br>resp.TGL 43 272: 6.7<br>TGL 43 273: 6.2 |
| Hardening and tempering technique | 3)<br>TGL 43 273: 8.4 |
| Re-shaping, bending, etc. | 2)<br>TGL 43 272: 3.3.2 |

1) in principle met
2) documents contain references to information available
3) information at present not available, secondary testing possibly required

**Annotations**

| | |
|---|---|
| TGL 43 272 | corresponds to the Soviet regulation for the construction and the safe operation of equipment and pipes in NPP, experimental and research reactors and plants (constructors' regulations) |
| WM 54 | part of TGL 43 272 (list of materials) |
| TGL 43 273 | corresponds to the Soviet OP 1513-72 (design regulation) |
| TGL 43 274 | corresponds to the Soviet PK 1514-72 (test specification) |

**Table 4.2-3 Essential Quality Characteristics of the Materials for the Primary System Components (Manufacturers' Information)**

| Com-ponent | Material | Cemical Composition | | | | | | | | Mechanical Parameters (RT) | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | C | Mn | Cr | Ni | Mo | P | S | Cu | $R_{p0.2}$ | $R_m$ | $T_{KO}$ |
| | | % | % | % | % | % | % | % | % | MPa | MPa | °C |
| Reactor Pressure Vessel (RPV) | 15Ch2NMFA | 0.13 - 0.18 | 0.3 - 0.6 | 1.8 - 2.3 | 1.2 - 1.5 | 0.5 - 0.7 | ≤ 0.020 | ≤ 0.020 | ≤ 0.3<br>V<br>0.18 - 0.12 | ≥ 441 | ≥ 539 | ≤ -10 |
| RPV Core Area | 15Ch2NMFA-A | 0.13 - 0.18 | 0.3 - 0.6 | 1.8 - 2.3 | 1.2 - 1.5 | 0.5 - 0.7 | ≤ 0.010 | ≤ 0.012 | ≤ 0.08<br>V<br>0.08 - 0.12 | ≥ 441 | ≥ 539 | ≤ -25 |
| Pressuriser, steam generator (SG) | 10GN2MFA | 0.08 - 0.12 | 0.8 - 1.1 | ≤ 0.3 | 1.8 - 2.3 | 0.4 - 0.7 | ≤ 0.020 | ≤ 0.020 | V<br>0.03 - 0.07 | ≥ 343 | ≥ 539 | ≤ 15 |
| SG tubes | 08Ch18N10T | ≤ 0.08 | ≤ 1.5 | 17.0 - 19.0 | 10.0 - 11.0 | Ti: 5C- 0.6 | ≤ 0.035 | ≤ 0.020 | ≤ 0.30 | 548 | | |
| Main coolant line | 10GN2MFA | 0.08 - 0.12 | 0.7 - 0.9 | ≤ 0.3 | 1.7 - 2.0 | 0.4 - 0.6 | ≤ 0.020 | ≤ 0.020 | V<br>≤ 0.04 | ≥ 343 | 539 - 673 | ≤ 15 |

**Table 4.2-4** **Essential Quality Characteristics of the Welding Fillers for the Reactor Pressure Vessel (Manufacturers' Information)**

| Com-ponent | Material | Cemical Composition | | | | | | | | Mechanical Parameters (RT) | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | C | Mn | Cr | Ni | Co | P | S | Cu | $R_{p0.2}$ | $R_m$ | $T_{KO}$ |
| | | % | % | % | % | % | % | % | % | MPa | MPa | °C |
| Root (core area) | Sv08AA | < 0.11 | 0.6 - 1.2 | < 0.15 | < 0.25 | | ≤ 0.012 | ≤ 0.015 | ≤ 0.08 | 216 | 352 | ≤ 0 |
| Weld (core area) | Sv09ChGNMTAA-Wi m.NF18M | 0.04 - 0.10 | 0.45 - 1.10 | 1.2 - 2.0 | 1.0 - 1.5 | | ≤ 0.012 | ≤ 0.012 | ≤ 0.08 | 422 | 539 | ≤ 0 |
| Cladding 1st layer | Sv07Ch25N13 | ≤ 0.09 | 0.60 - 1.70 | 21.0 - 26.5 | 11.0 - 14.0 | ≤ 0.05 | ≤ 0.030 | ≤ 0.020 | ≤ 0.05 | | | |
| 2nd + 3rd layer (core area) | Sv04Ch20N10 G2B | ≤ 0.05 | 0.90 - 1.90 | 17.5 - 20.5 | 8.0 - 11.0 | ≤ 0.05 | ≤ 0.030 | ≤ 0.020 | ≤ 0.05 | 265 | 490 | KCV ≥ 29.4 $Jcm^{-2}$ |
| 2nd + 3rd layer | Sv08Ch19N10 G2B | ≤ 0.10 | 1.30 - 2.20 | 17.5 - 20.5 | 8.0 - 11.0 | ≤ 0.05 | ≤ 0.030 | ≤ 0.020 | ≤ 0.05 | 314 | 490 | KCV ≥ 29.4 $Jcm^{-2}$ |

## Figures, Section 4

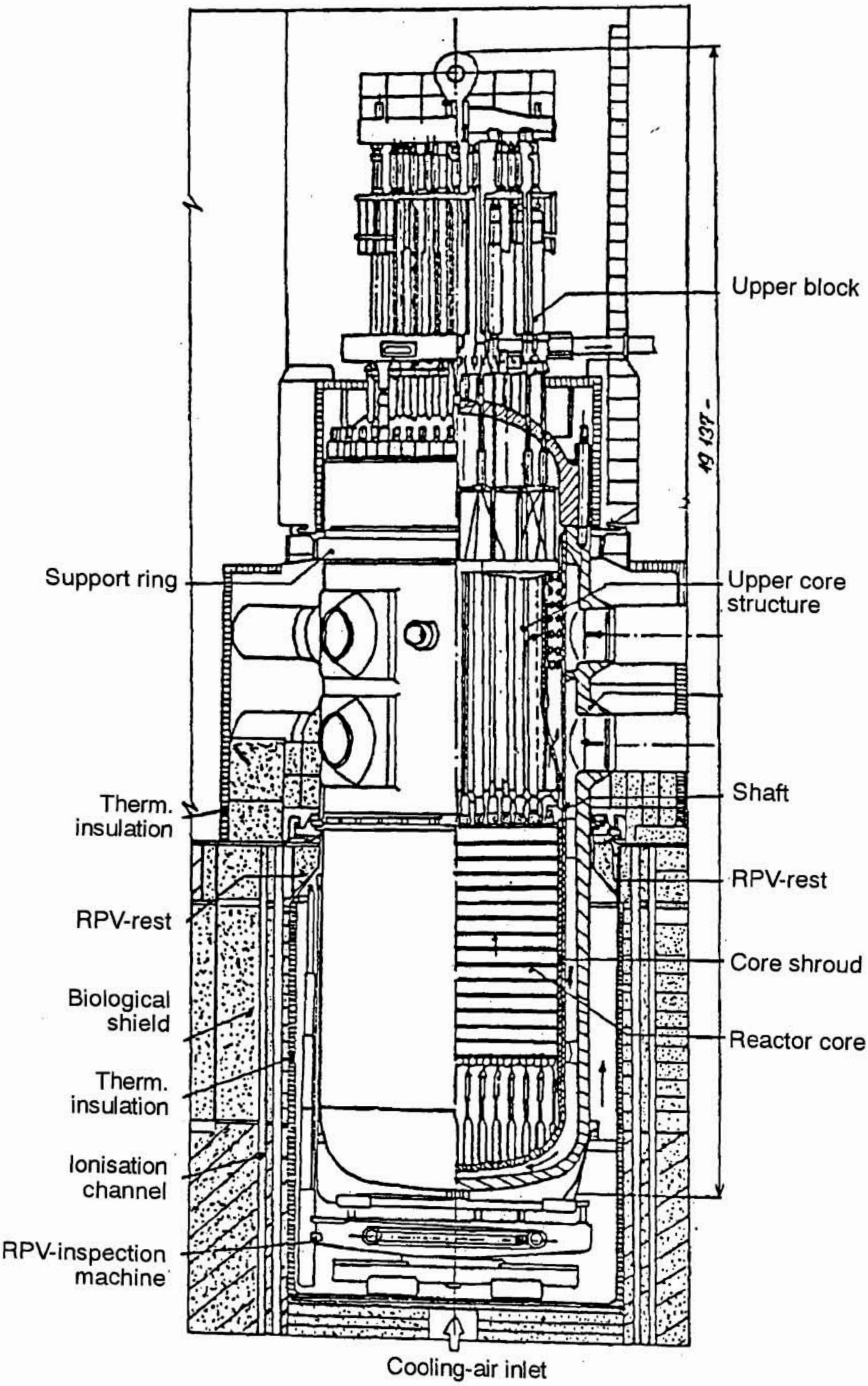
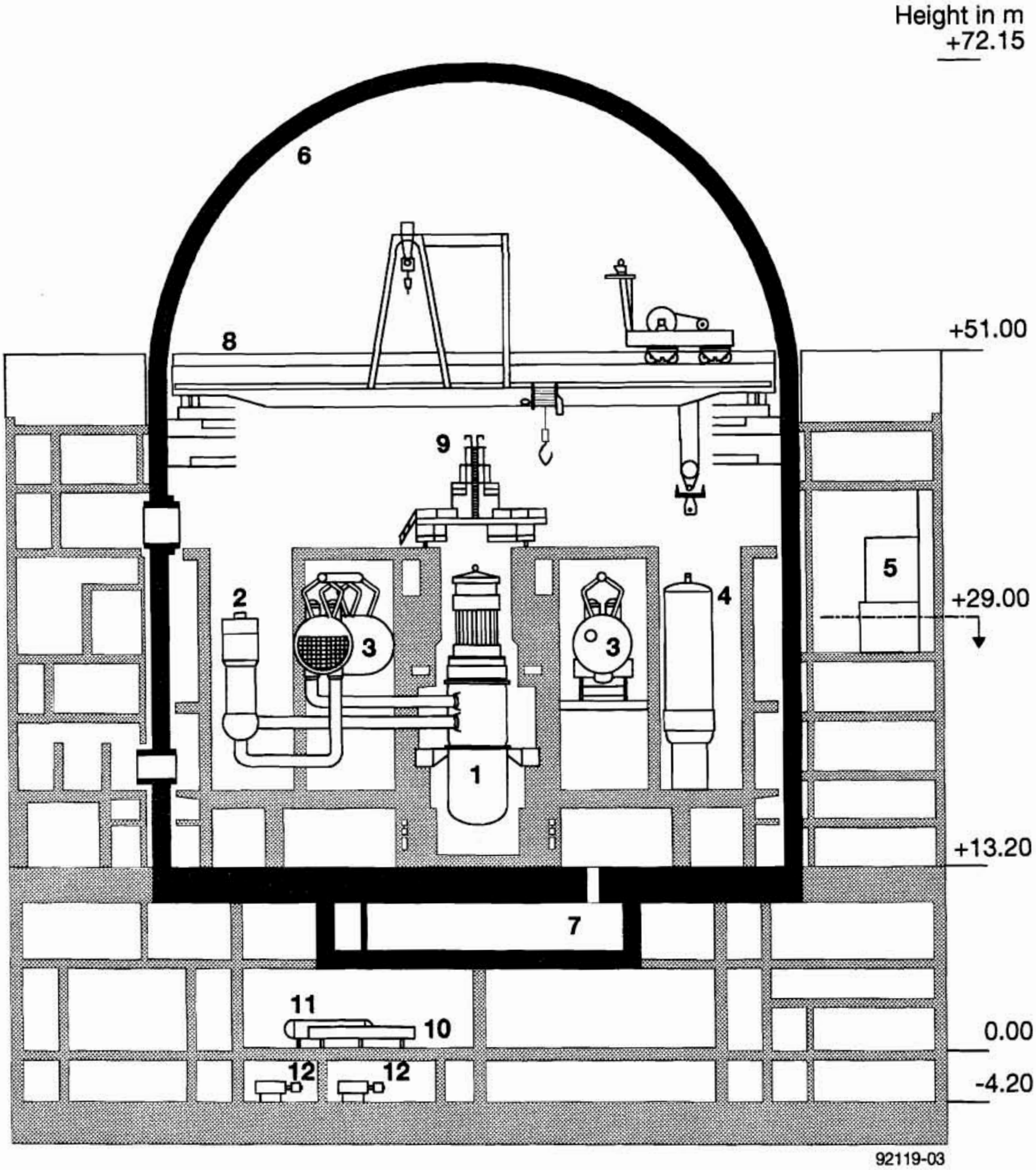
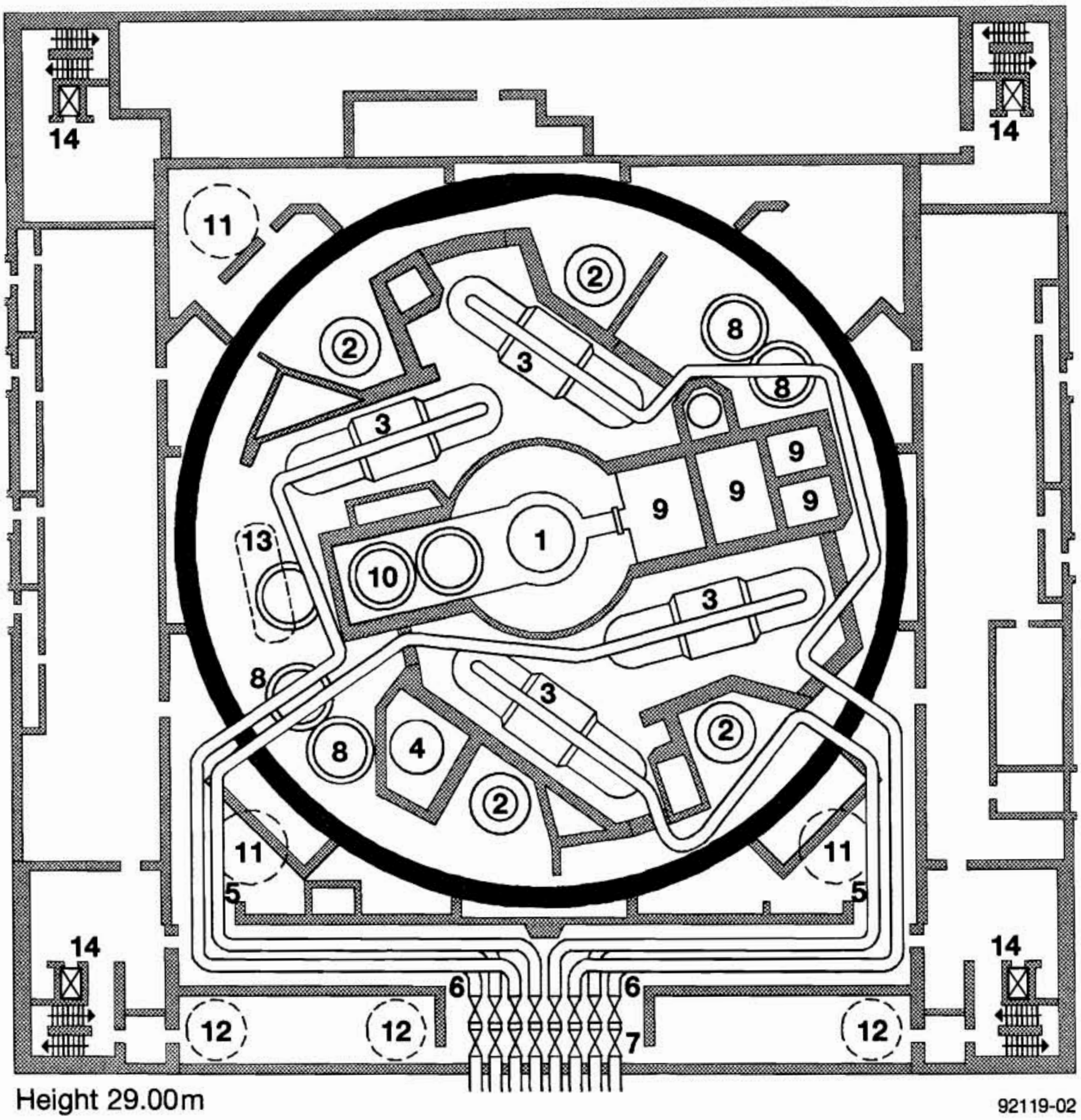



**Fig. 4.1-1** Reactor pressure vessel (RPV) with internals




**1** Reactor
**2** Main coolant pump
**3** Steam generator
**4** Pressurizer
**5** Storage tank for service water
**6** Containment
**7** Boric acid storage tank
**8** Main crane 3.2 MN
**9** Fuel assembly handling machine
**10** Emergency cooler
**11** Spent fuel pit heat exchanger
**12** Emergency core cooling pump

**Fig. 4.2-1** Reactor pressure vessel with containment, layout of the components, vertical projection




**1** Reactor
**2** Main coolant pump
**3** Steam generator
**4** Pressurizer
**5** Main steam pipe
**6** Feedwater pipe
**7** Steam dump station (to atmosphere)
**8** Pressure accumulator
**9** Spent fuel pit
**10** Inspection shaft
**11** Storage tank for service water
**12** Fire fighting water
**13** Pressurizer relief tank
**14** Staircase with elevator

**Fig. 4.2-2** Reactor pressure vessel with containment, layout of the components, horizontal projection

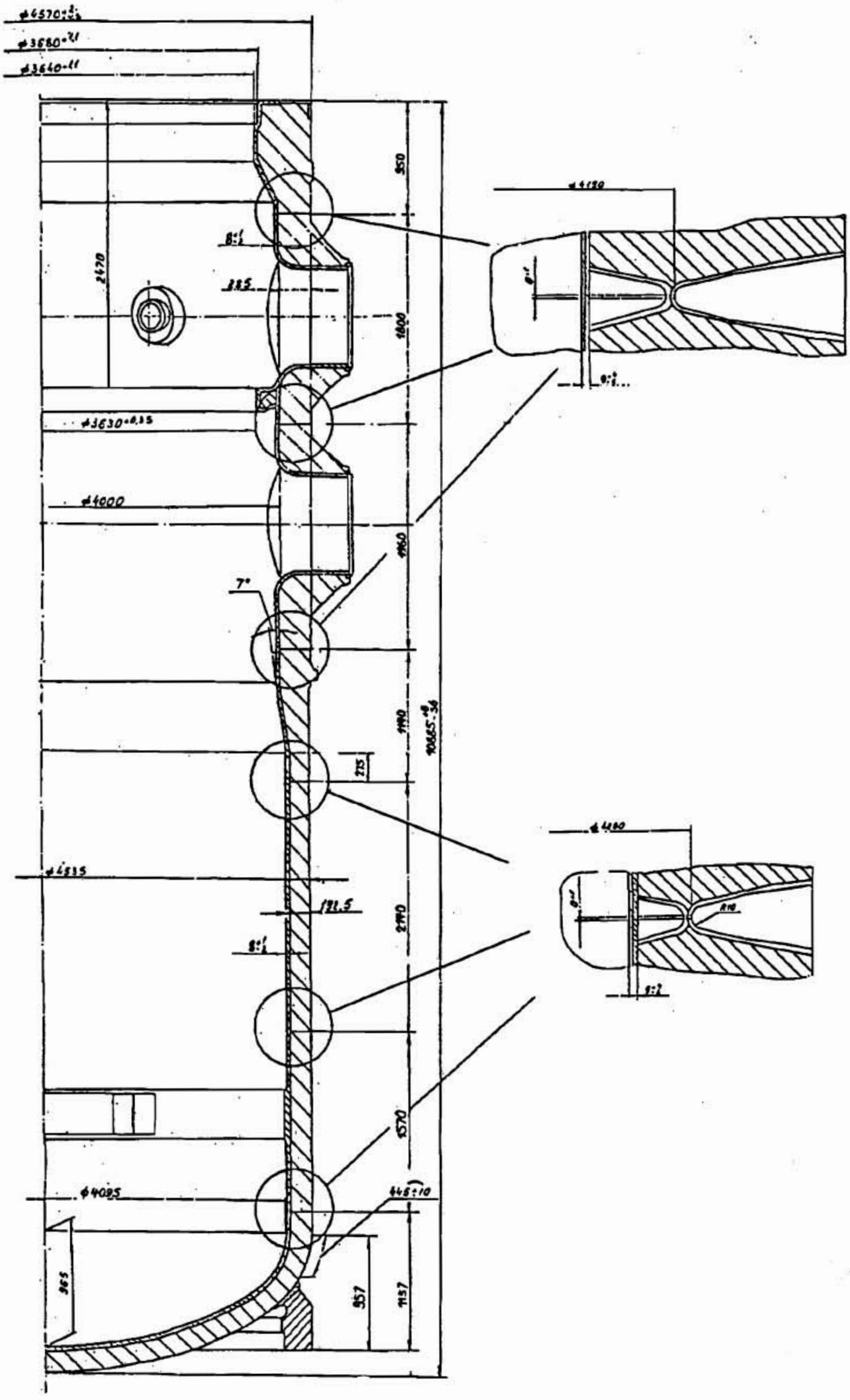


**Fig. 4.2-3** Reactor pressure vessel, overview and execution of main welds

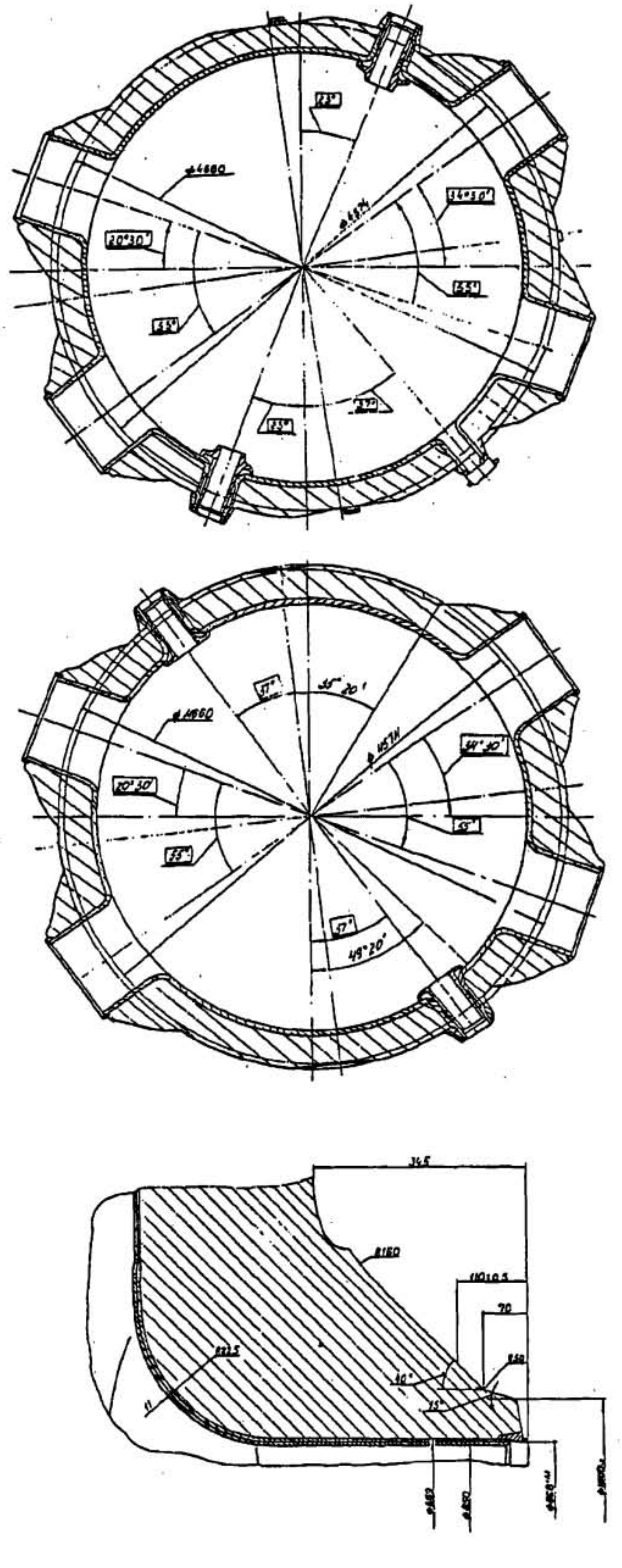


**Fig. 4.2-4** Reactor pressure vessel, hot and cold nozzle ring, execution of nozzles DN 850 for main coolant line

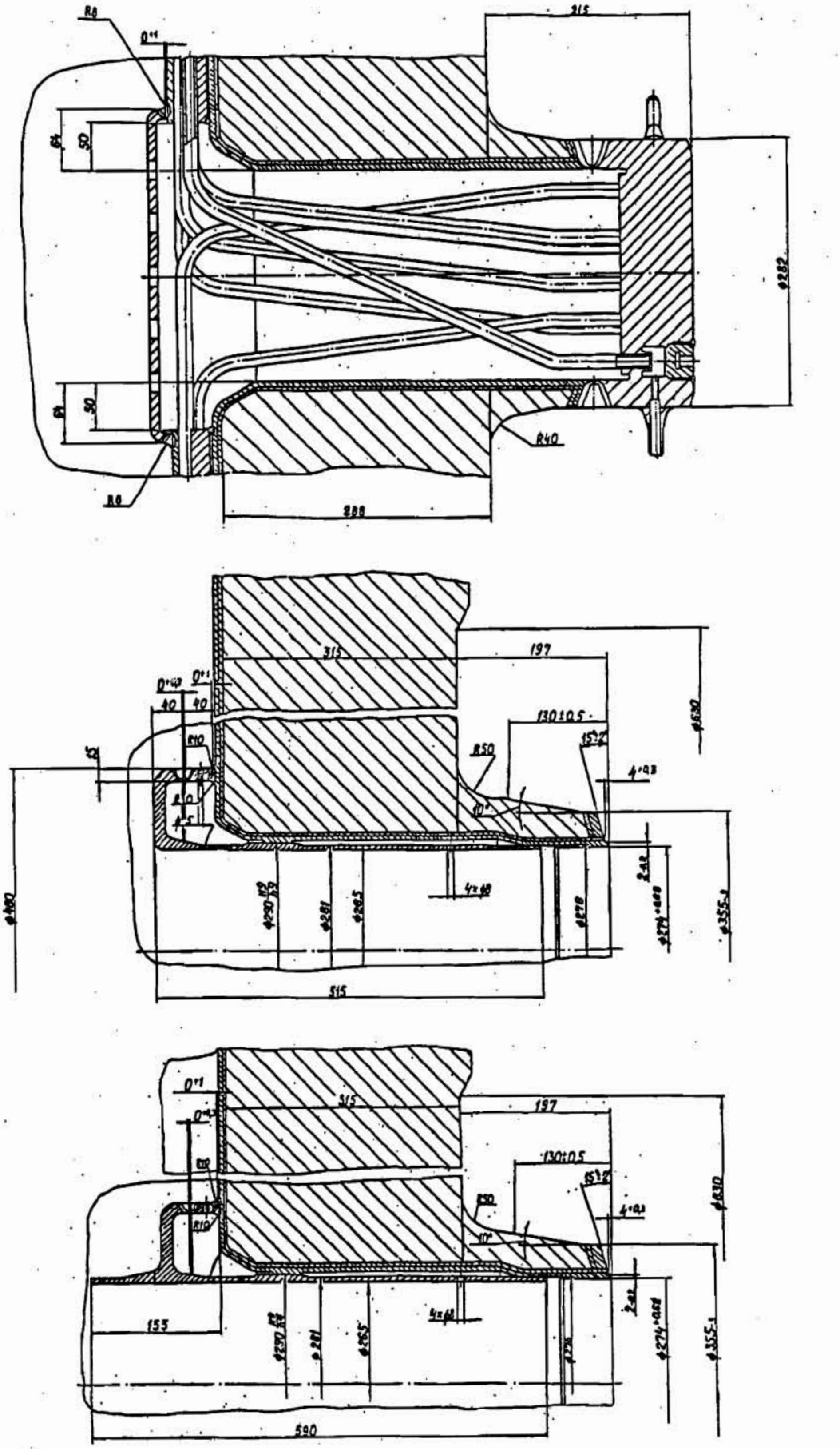



**Fig. 4.2-5** Reactor pressure vessel, nozzle for measurement lines, hot and cold nozzle for emergency cooling tubes

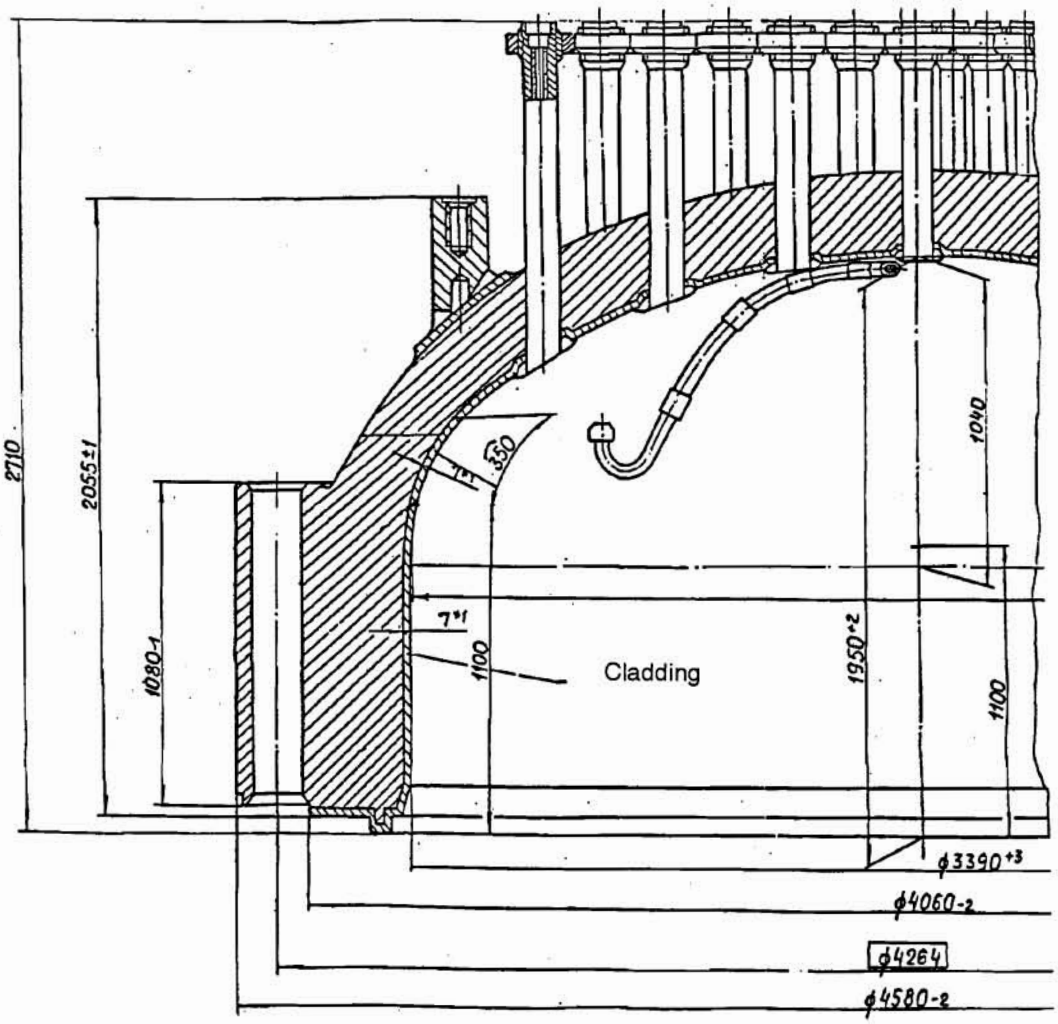



**Fig. 4.2-6** Reactor pressure vessel head, overview

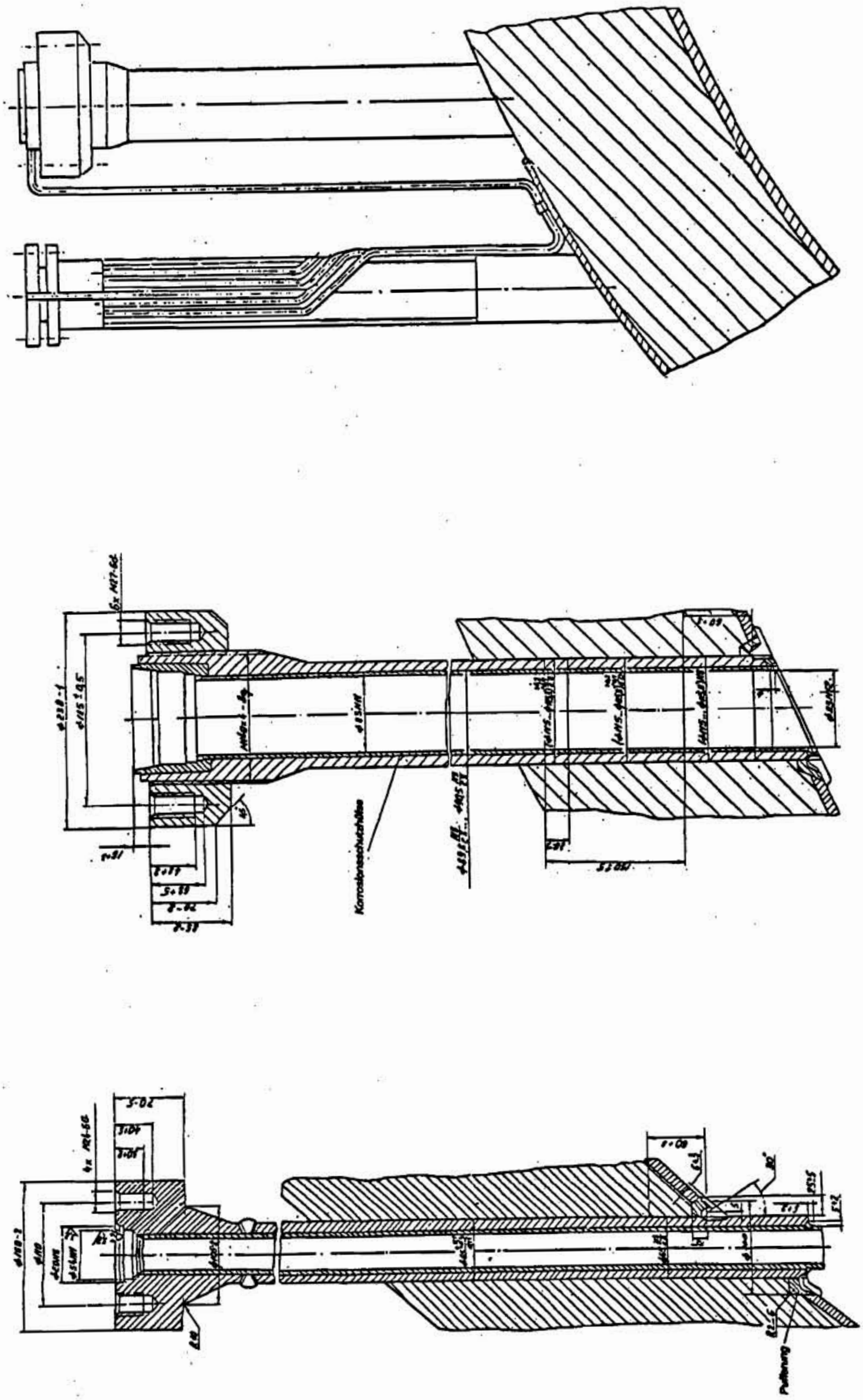


**Fig. 4.2-7** Reactor pressure vessel head, execution of nozzles for KAT, SUS and leakage control pipes

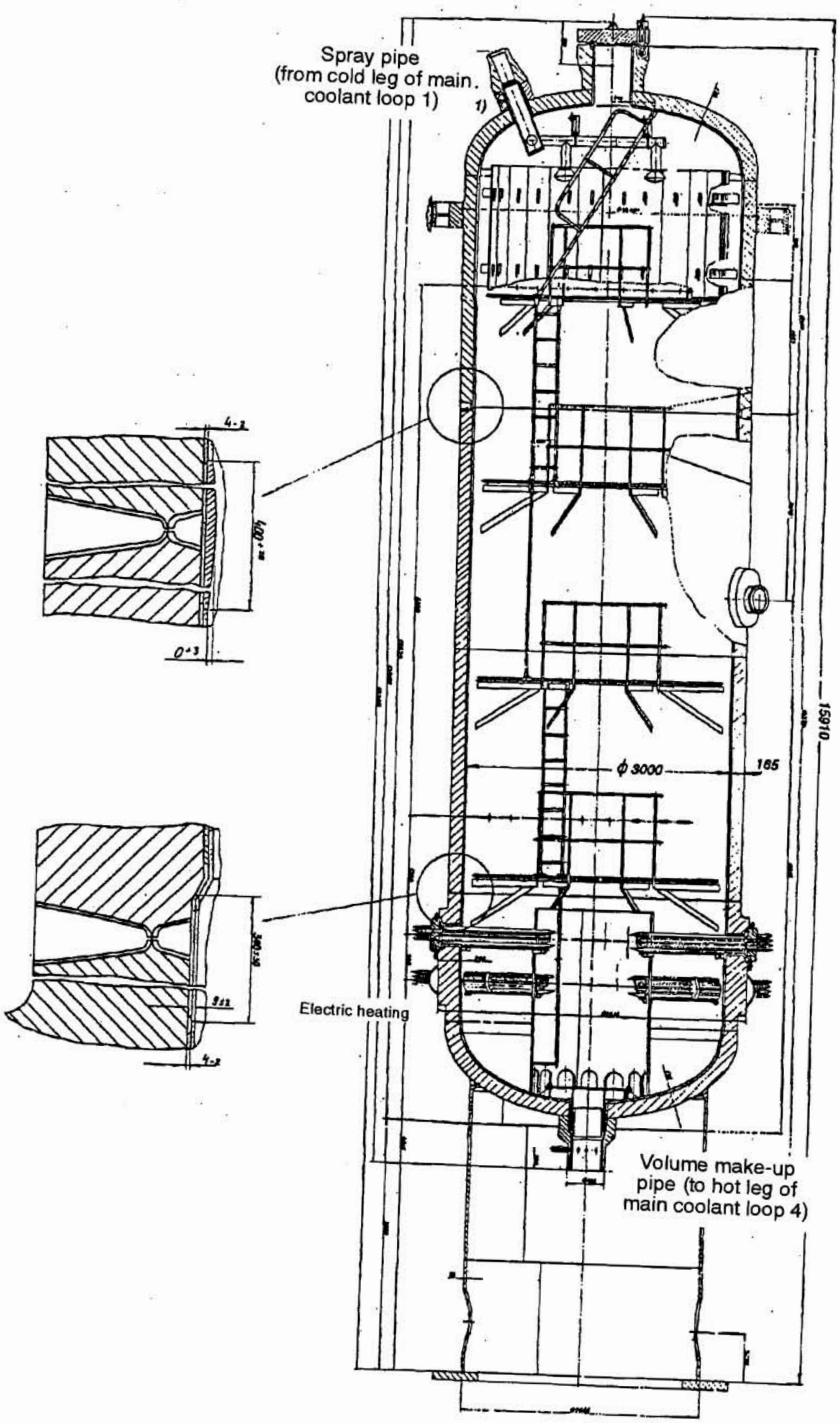



**Fig. 4.2-8** Pressuriser, overview and main welds

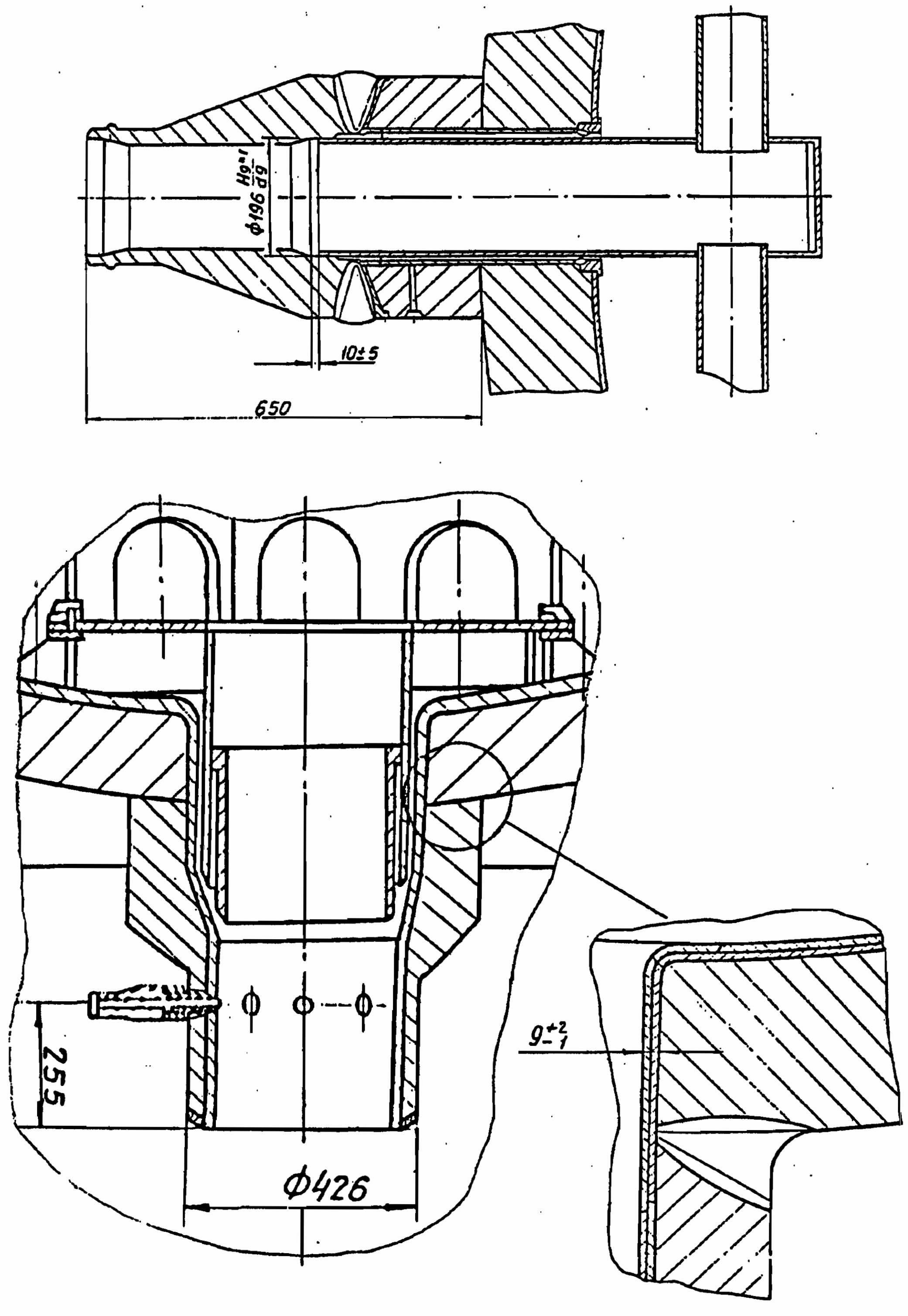



**Fig. 4.2-9** Pressuriser, nozzle for spray line and nozzle for volume control line

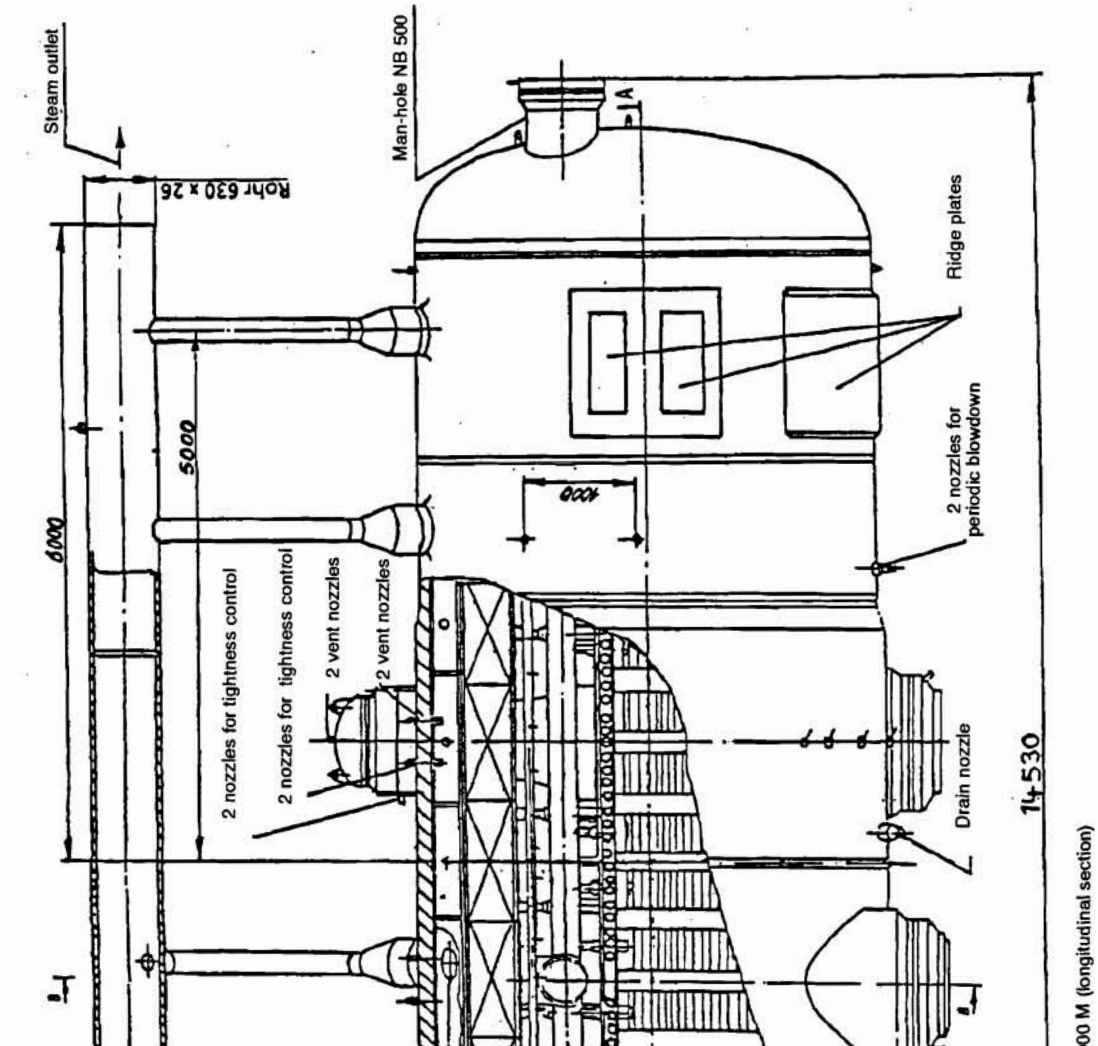



**Fig. 4.2-10** Steam generator, overview

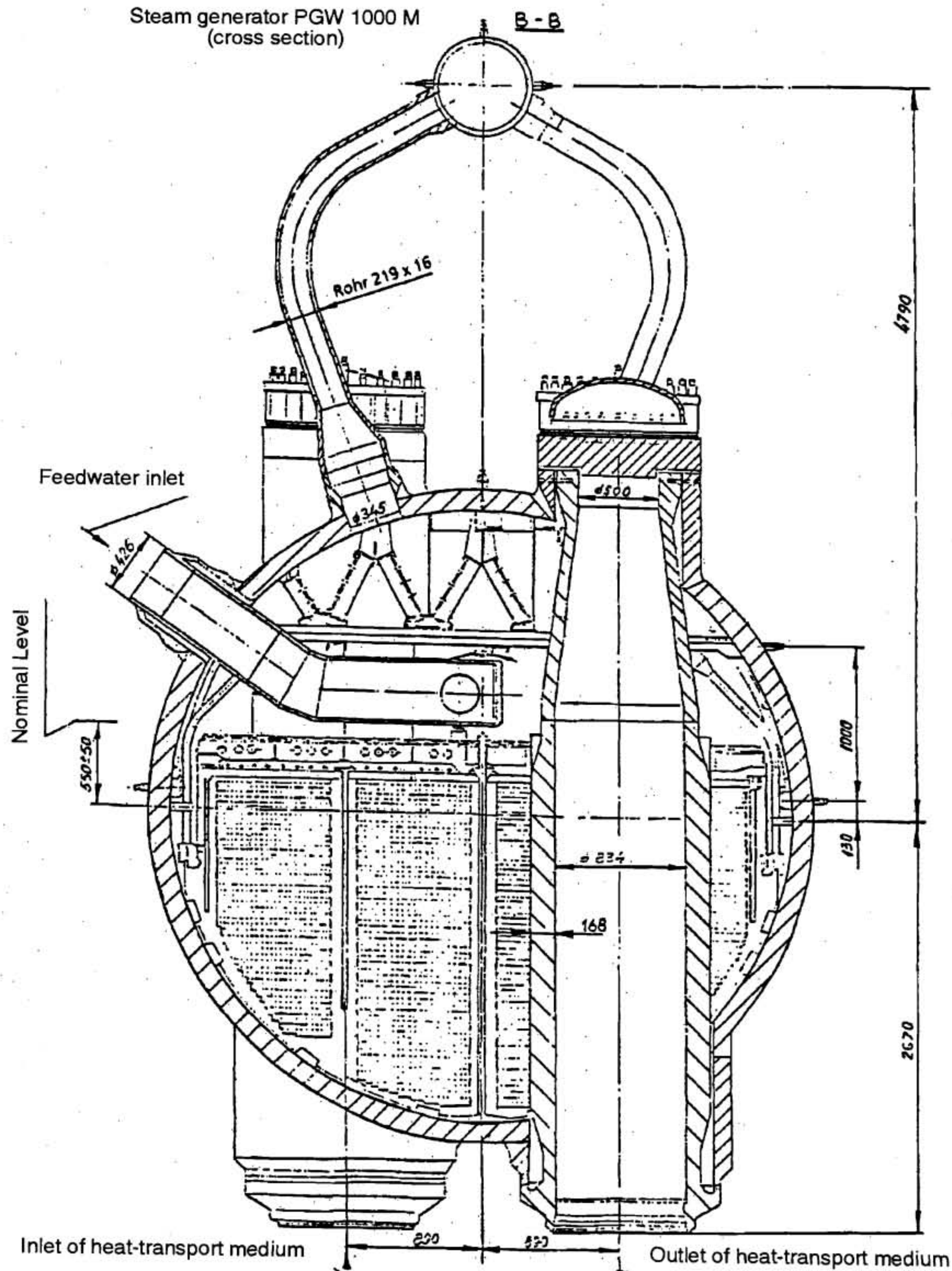
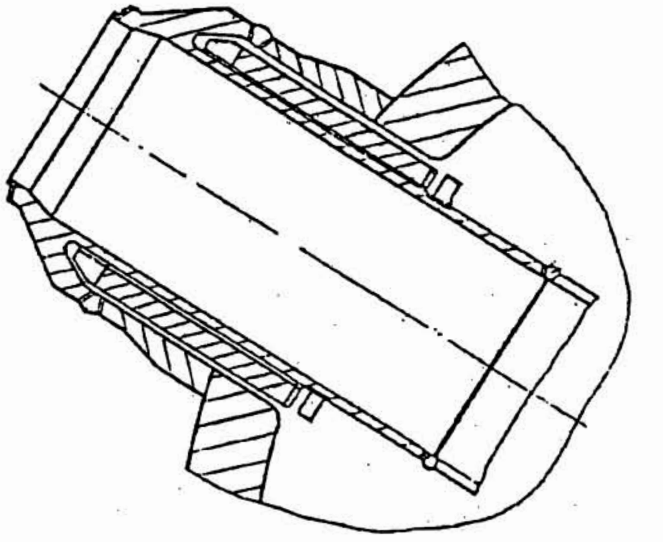



**Fig. 4.2-11** Steam generator, collector



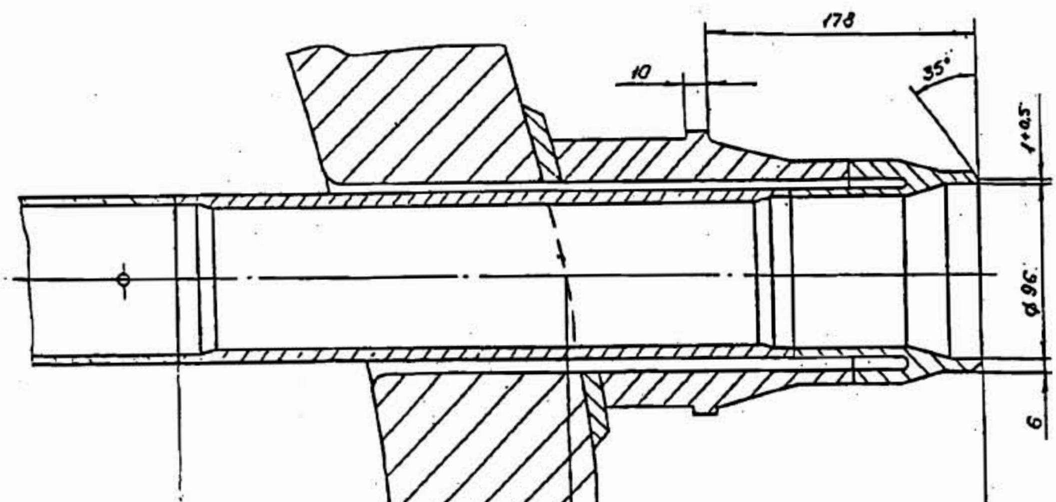



**Fig. 4.2-12** Steam generator, feedwater nozzle and emergency feedwater nozzle

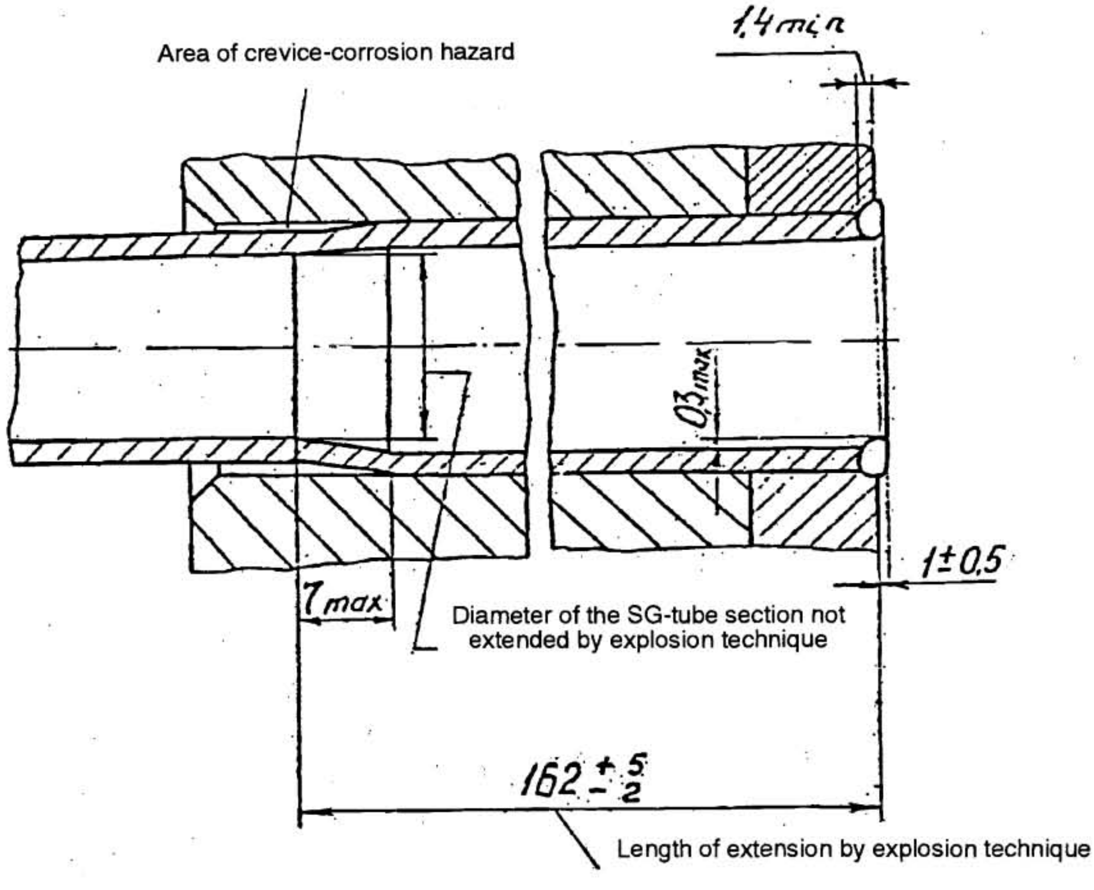



**Fig. 4.2-13** Steam generator, connection of heater tubes to collector

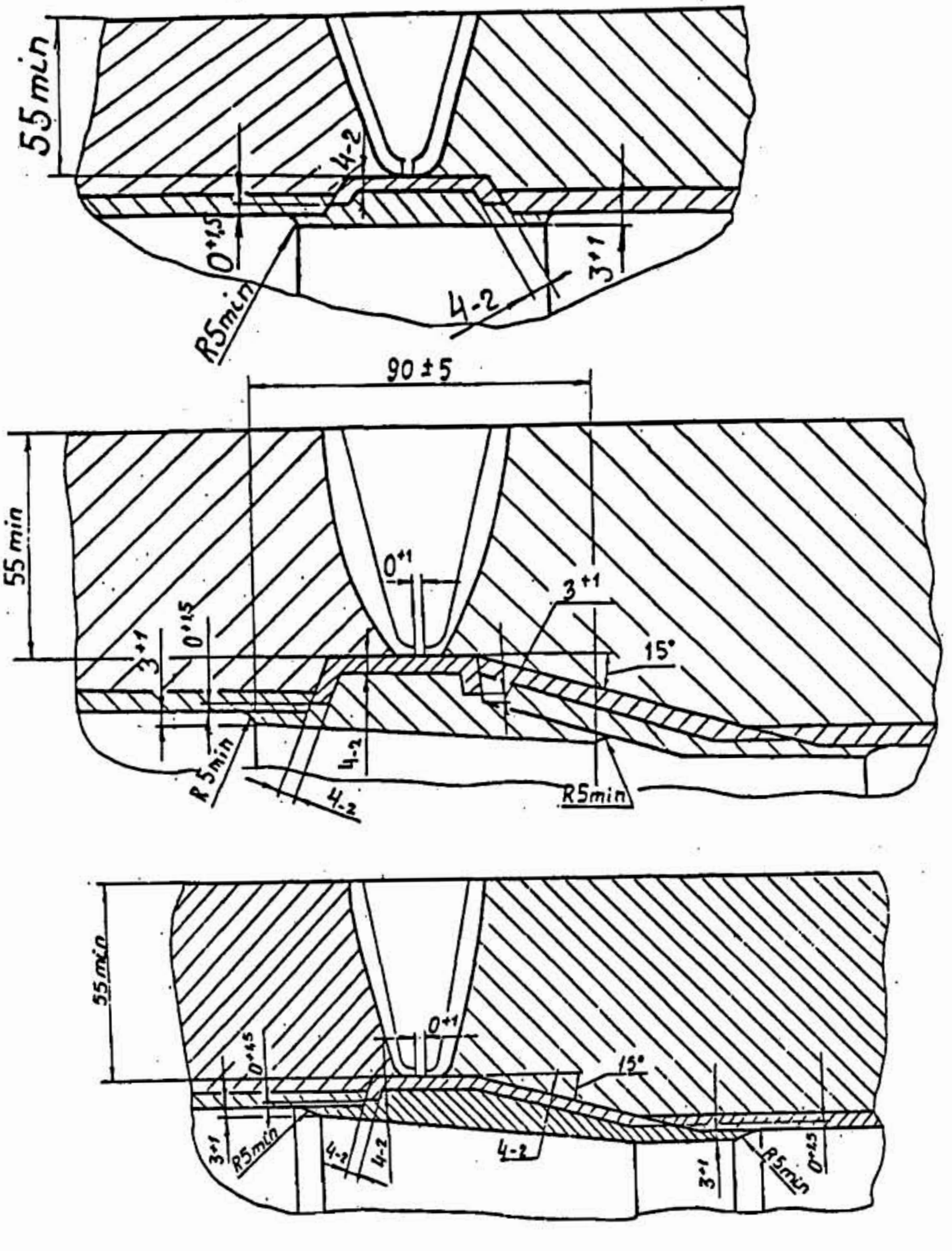



**Fig. 4.2-14** Main coolant line, weld joints plain tube and three variants plain tube-elbow

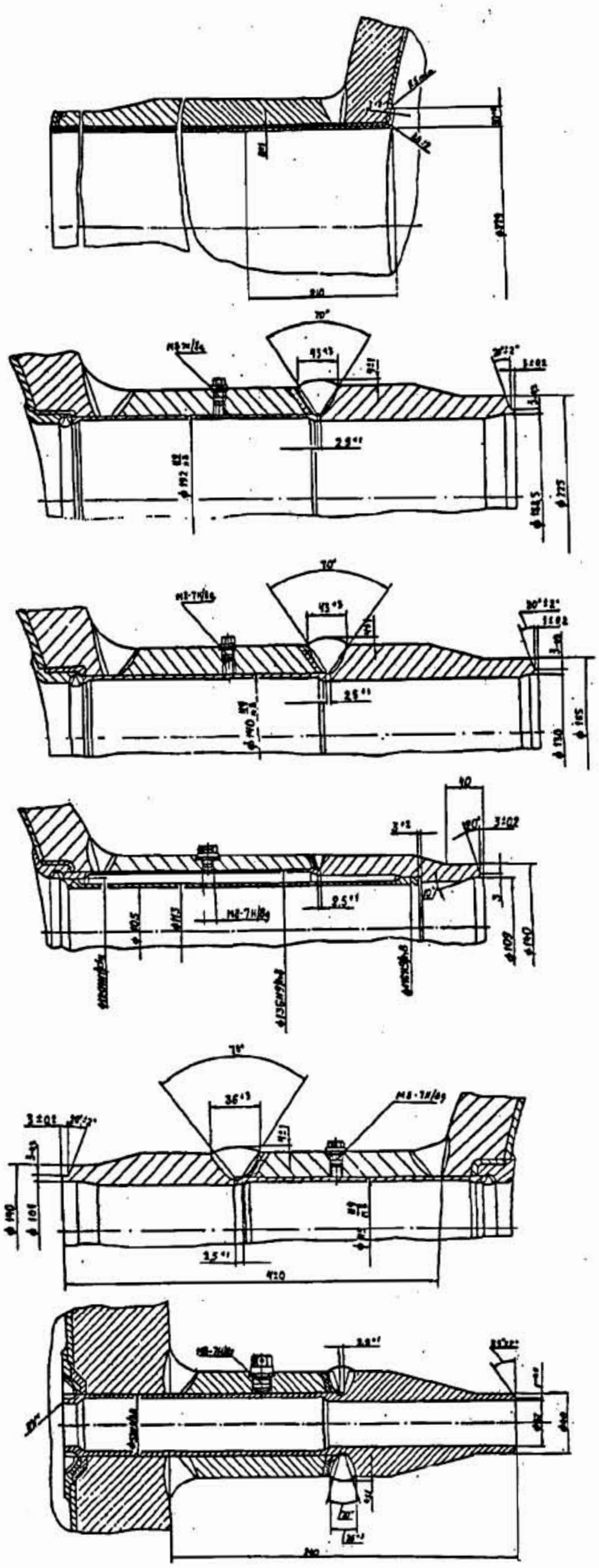


**Fig. 4.2-15** Main coolant line, execution of the nozzles as a function of the nominal diameter

# 5 Accident Analysis

According to the Federal German codes and standards, safety precautions must ensure that if an accident occurs the residual heat of the reactor can be safely dissipated, the reactor can be shutdown, that long-term subcriticality can be maintained and radiation exposure of staff and environment can be kept as low as possible and also below the dose limits determined by the regulations of the Atomic Energy Act and the subordinated ordinances, taking into account the state of the art. Additionally, for many accidents it is required that further protective targets are met. Thus it must be demonstrated for accidents or incidents with a higher probability of occurence, like operational transients, for example, that the heat flux densities at the fuel-rod-cladding tubes are sufficiently remote from the critical heat flux density, that the release of energy in the fuel rods is so low that melting is avoided and the pressure in the primary system is so low that safety valves do not open.

To prove precautions against inadmissible effects of accidents, an accident analysis is to be performed for the plant under consideration, in which sequence and effects of the accidents are investigated. The qualification of the methods of analysis and of the computing programs must be verified with tests in experimental plants or experiments in the reactor plant. The requirements and the boundary conditions for accident analysis are conservatively defined by the Federal German codes and standards.

## 5.1 Analyses relating to Loss-of-Coolant Accidents and Transients

For the safety evaluation of the Stendal Nuclear Power Station, Unit A, accident analyses conducted by the project engineer and the architect engineer were assessed against the background of the Federal German codes and standards for nuclear power stations. Furthermore, accident analyses of further institutions relating to reactor plants of the WWER-1000/W-320 type were assessed. Whenever values of the accident analyses deviated from the codes and standards,the extent of any resulting safety deficit was examined and what replacement measures could be taken, if necessary.

A detailed description of the evaluation of the present accident analyses including bibliography is given in /HOC 91/. The range of accidents dealt with in this section does not comprise all accidents which can be thought of. Thus, accidents with the reactor being shutdown and incidents (accidents) beyond design limits cannot be evaluated here, as no relevant analyses are available. It is recommended to carry out analyses of accidents with the reactor being shutdown, for startup and shutdown processes, as well as analyses of accidents beyond design limits (R 5.1-21). Accidents in handling fuel elements are contained in Section 5.3.2. Accidents during cooling of the fuel elements in the spent-fuel pools are briefly discussed in Section 6.3.4. On the other hand, WWER-specific accidents which are not contained in the accident analyses of western reactors are dealt with in Section 5.1. An overview of the requirements to be met by safety devices, derived from the safety analysis, is provided in Table 6.1-1.

### 5.1.1 Assessment Criteria

The assessment of the accident analyses is based on the Federal German codes and standards. The respective paragraphs of the BMI Safety Criteria, the Accident Guidelines of the BMI, the respective paragraphs of the RSK Guidelines for Pressurised Water Reactors and the respective KTA-Rules must be referred to here. To examine the completeness of the accident range the List of Notes with Sub-division for a Standard Safety Report for Nuclear Power Plants with Pressurised Water Reactor or Boiling Water Reactor /BMI 76/ must further be referred to. Apart from the accidents applying to the WWER-1000 listed there, such accidents, which result from the structural peculiarities of the Stendal plant, Unit A, compared to a Federal German plant are also to be analysed.

RSK Guideline 22.1 of the Federal German codes and standards is to be referred to for assessing the design calculations relating to loss-of-coolant accidents. In this guideline it is required among other things that

- the calculated maximum fuel rod cladding tube temperature does not exceed 1200 °C,
- the calculated depth of oxidation of the cladding at no point exceeds the value of 17 % of the actual wall thickness of the cladding tube,

- during the zirconium-water reaction not more than 1 % of the entire zirconium contained in the cladding tubes reacts with steam,
- only small fractions of the core inventory (10 % of the noble gases, 5 % of the volatile solids, 0.1 % of other solids) may be released into the containment. It has to be assumed that 10 % of all fuel rods fail, unless a lower percentage of failure can be demonstrated in a core damage analysis,
- that no changes in the geometry of the reactor core occur which prevent a sufficient cooling of the reactor core.

In addition, subcriticality of the reactor core must be ensured for long-term cooling after a loss-of-coolant accident.

### 5.1.2 Loss-of-Coolant Accidents

#### 5.1.2.1 Leaks and Breaks from the Primary System to the Containment

Besides a description and evaluation of accident analyses available from other institutions for the Stendal Nuclear Power Plant, Unit A, or other plants of the WWER-1000 type, respectively, the results of two new analyses performed by GRS in the context of this safety evaluation are summarized and assessed.

The evaluation of the analyses performed by other institutions together with a bibliography is described in detail in /HOC 91/. The GRS analyses are documented in /KIM 91a/ and /KIM 91b/.

- **Existing Analyses**

- Completeness of the Accident Spectrum

The present analyses of different institutions describe leaks in the cold leg having cross-sections of 7, 20, 38, 133, 254, 707 and 11349 $cm^2$, in the hot leg of 707 $cm^2$, at the pressuriser of 20 and 254 $cm^2$, as well as of 54 $cm^2$ at the reactor vessel head.

In addition thereto, there is an analysis by K.A.B. referring to the rupture of a measurement pipe as a leak from the primary system into the environment bypassing the containment.

- Initial and Boundary Conditions

The initial and boundary conditions of the analyses are insufficiently documented and can only partially be checked.

In all analyses reactor scram was initiated after activation of the first criterion. Assumptions on the failure of redundancies of the emergency cooling system take into account a single-failure and feeding to the leak. The repair case is not normally supposed. In almost all analyses a loss of off-site power occuring simultaneously with the leak is assumed. Manual measures to control accidents during the first 30 minutes were not assumed in most analyses.

- Calculation programs

The calculation programs TETSCH, SONA, KANAL, and RELAP4/MOD6 were used for the analyses.

- **Assessment of the Existing Analyses**

- Assessment of the Completeness of the Accident Spectrum

The scope of the present analyses referring to the sizes and positions of leaks at the main coolant lines suffices for assessment. However, according to RSK Guideline 21.1 (3), a leak of approx. 20 $cm^2$ below the lower edge of the reactor core is to be assumed additionally for the design of the emergency core cooling system. By this leak, damage to the reactor pressure vessel is included which is not recognized in time by in-service examinations. It is recommended to carry out a corresponding analysis (R 5.1-1).

- Assessment of the Boundary Conditions

The initial conditions of the analyses like reactor power, pressure in the primary and secondary system, water levels, temperatures, nuclear core mass flow, gap heat transfer figures, were, as far as documented, determined according to conservative views. Leak opening time and decay heat were also determined conservatively, as far as documented. The initial and boundary conditions essentially meet the requirements of the RSK Guidelines. Owing to the insufficient documentation, a complete examination of the initial and boundary conditions is not possible.

As far as can be seen, reactor scram in all analyses was initiated after the activation, of the first criterion. In larger loss-of-coolant accidents during nominal power operation, reactor scram by the second criterion only leads to an insignificant change of the accident sequence. In smaller loss-of-coolant accidents during full load operation and leakage accidents during partial load operation, temporal delays until reactor scram can dominate, especially if there are leaks in the upper pressuriser section.

According to the BMI safety criterion 4.3 and RSK-Guideline 22.1.2 (3) relating to assumptions on the failure of redundancies in the emergency cooling system, the single failure and repair case have to be presumed in analyses. Further, it has to be taken into account, by the choice of the leak position, that a redundancy can, wholly or partially, feed to the leak. The requirement of assuming a repair case generally is not met by the present emergency cooling analyses. In the analyses referring to leaks in the upper pressuriser and at the reactor top the requirement of assuming the simultaneous repair case has, however, been met.

If a pressuriser safety valve is opened and remains open erroneously, an additional analysis without manual secondary shutdown during the first 30 minutes is necessary. It is recommended to consider an early shutdown of the secondary side by use of automatic criteria for all loss-of-coolant accidents to be able to use the water reservoirs of the emergency-cooling system more effectively (R 5.1-2).

- Assessment of Computer Programs

There is no proof for verifying the TETSCH, SONA, KANAL computer programs. For the analysis of accidents in WWER plants RELAP4/MOD6 was partially verified by recalculations of experiments at the Hungarian PMK-NVH experimental plant. There are, however, no verifications for the refill and flood phase of a large break. Because of the simplified model of the phenomena occuring here, e.g. the progress of rewetting fronts, RELAP4/MOD6 is only suitable to a limited extent for these phases.

- Assessment of Results

The analyses of large breaks are incomplete as they do not analyse the entire accident range with a program qualified for this purpose. The results of the remaining analyses, taking own analytical and experimental knowledge on accident behaviour of PWR plants in general and WWER plants in particular into account, permit the conclusion that the accidents under consideration are either also covered by the large break or can be coped with by the emergency core cooling system assuming loss of off-site power, single failure, feeding to the leak and, in some cases, additionally assuming the repair case.

As the present accident analyses, as explained above, only partially meet the requirements of the Federal German codes and standards, in case of a licensing procedure it would be recommended to perform the analyses relating to loss-of-coolant accidents anew, taking into account the respective RSK Guidelines and BMI Safety Criteria for the accident spectrum according to the accident guidelines, extended by WWER specific accidents with the respective accident code corresponding to the present state of the art. The finally determined set values of the safety system would have to be used here (R 5.1-3).

- **Own Analyses**

To supplement and ensure the statements made in the previous paragraph, own analyses were carried out. The design accident for the emergency cooling, the double-ended break of the cold leg of a main coolant line was analysed with the GRS ATHLET/FLUT computer programs /KIM 91a/, /KIM 91b/. Conservative assumptions according to the design concept of the reactor plant were made for the initial and boundary conditions. The availability of the emergency cooling system was reduced to one leg by considering the single failure. In addition, it was assumed that one of the two accumulator feeding into the upper plenum, was not available. The case of a ball valve, positioned in the interior of the accumulator, failing in the closed position is thus also taken into account. The repair case according to RSK Guideline 22.1.2 (3) in a further leg was not assumed. The position of the break was chosen in such a way that there are as many feeding positions as possible in the immediate vicinity of the break. It was thus assumed that injection lines directly adjacent to the leak also break, i.e. the line of one redundancy of the HP-emergency cooling system and one of the two lines of a redundancy of the LP-emergency cooling system (cf. Fig. 5.1-1). Owing to the assumption of the simultaneous loss of off-site power, the remaining trains of the active emergency cooling system feed into the primary system with a time delay resulting from the run-up time of the emergency diesel and the connection schedule of the users.

The calculated sequences of primary and secondary pressure, of the steam content in the core and the cladding tube temperatures for the hot rod in the inner core channel are illustrated in Fig. 5.1-2 to 5.1-5.

To round up the accident area "Large Break", and as a study of parameters for WWER-1000, commissioned by the Finnish operator Imatran Voima Oy, performed earlier by GRS and DRUFAN had resulted in higher cladding tube temperatures for the 0.5A break in the cold leg than for the design accident, an analysis of this accident was carried out with ATHLET/FLUT too. The input data as well as the initial and boundary conditions are largely identical with those of the design accident.

Furthermore, own calculations were performed with respect to leaks from the primary system into the environment circumventing the containment. The outflow rates from the break of a measurement pipe were determined as the basis for establishing the radiological impacts (see Section 5.3).

- **Assessment of Own Analytical Results**

For the hot rod with the maximum rod linear power of 448 W/cm prior to the occurence of the accident, the analysis showed a maximum value for the cladding tube temperature of 775 °C for the 2A-break, while a maximum temperature below 600 °C was calculated for medium load rods. Temperatures above 600 °C for the hot rods only occured for a period of less than 6 s, at pressures of more than 5 MPa in the primary system. With such a short dwell period and the external pressure still being relatively high, the fuel rod claddings of zircalloy would not break, so that a separate analysis on the extent of damage would be unnecessary. For the fuel-rod-cladding tubes in the WWER-1000 of a zirconium alloy of about 1 % niobium it was demonstrated in a recent investigation /ADA 89/ that the expansion and rupture behaviour does not differ substantially from the cladding tubes of zircalloy-4. Consequently a separate analysis on the extent of damages for the cladding tubes of the WWER-1000 of ZrNb-1 is not compulsory.

The results for the 0.5A-break illustrate that the reactor core can be completely rewetted through the accumulator during the injection phase, which lasts up to about 80 s. Although the maximum cladding tube temperature of 720 °C calculated for the hot rod is lower compared to the 775 °C for the design accident, temperatures exceeding 600 °C have been determined for a somewhat longer period. While such temperatures for the design accident have only been calculated for up to 6 s, this phase in the case of an 0.5A-break lasts about 15 s. This period too is so short that using zircalloy, cladding tube damage is hardly to be expected. With respect to an analysis on the extent of damage, the same applies as for the 2A-break (see above).

The calculations for the 2A-break and the 0.5A-break show that the reactor core can be sufficiently cooled under the chosen conditions. It thus has been verified that the emergency cooling system to control the design accident has been designed sufficiently according to the Soviet criteria.

To fulfil the safety criteria of the BMI and the RSK Guidelines, the repair case has to be taken into account. It is expected that the design accident cannot be controlled with an additionally assumed repair case in one redundancy. It is recommended to design the emergency cooling system in such a way that the requirements of taking the single failure and a simultaneous repair of one redundancy into account are observed. As a substitutional measure, narrowly-defined and justified repair-time limits are to be provided (R 5.1-4). According to recent sources of information /MRE 92/, an overall repair-time limit of 72 h exists in all WWER-1000 plants. After this period the plant is to be shutdown and cooled-down. When, owing to a repair, one redundancy is unavailable, the remaining redundancies are put into operation. According to more recent operational regulations (approx. two years old), the pumps are shutdown again after a successful test of the valves remaining open. If the repair takes more than two shifts, the pumps will be started and tested again. According to older operational regulations the pumps in a repair case ran permanently.

### 5.1.2.2 Leaks and Breaks from the Primary to the Secondary System

Own analyses have not been performed for this group of accidents. Accident analyses for the Stendal NPP or other plants of the WWER-1000 type conducted by other institutions are summarised and evaluated.

- **Existing Analyses**

- Completeness of the Accident Spectrum

Analyses relate to the break of a steam generator tube in different variants, to the rupture of the steam generator collector top and to the rupture of the entire collector.

The safety report of the Technical Project does not contain any analyses relating to the rupture of the steam generator collector top and to the rupture of the entire collector. This case is not a design accident. But the analysis of this case is of a special importance for safety, as such a case already occured in one WWER-440 plant. Stendal-specific analyses of such cases have already been started by the architect-engineer K.A.B. Most of these analyses have, however, not been completed. For this reason analyses for the Kosloduj-5 and Rovno-3 plants were additionally included into the assessment. More recent (1989, 1991) Russian analyses /MRE 92/ referring to the break of the collector top were not available to GRS for assessment.

The accident spectrum thus formally has been taken into account completely.

- Initial and Boundary Conditions

Best-estimate conditions, i.e. nominal conditions for power, pressures, temperatures and mass flow rates, were used as initial conditions.

Different cases with and without simultaneous loss of off-site power were analysed.

- Computing Programs

The analyses were carried out with the computing programs DINAMIKA and RELAP4/MOD6.

- **Assessment of the Existing Analyses**

- Assessment of the Completeness of the Accident Spectrum

Besides the break in the steam generator tube, the rupture of the steam generator collector top and the rupture of the entire collector, as beyond design basis accidents, are particularly important because the containment is circumvented with a possibly significant release of activity into the atmosphere. The break of the head already occured in one WWER plant (Rovno-1 plant, type WWER-440/W-213 in the former USSR in 1982). According to a Soviet comment on the safety evaluation of the Greifswald Nuclear Power Plant, Unit 5, conducted by GRS, these accidents must be analysed in every WWER plant in connection with the establishment of special technical and organisational measures to reduce risks (accident procedures). We agree to this recommendation (R 5.1-5).

For the analyses in the first accident phase, it is important to determine the release of activity into the environment and to determine the maximum pressures in the secondary system. The proof of sufficient cooling of the fuel rods plays a role in the later accident phase only. The use of best-estimate initial conditions compared to more conservatively chosen initial conditions therefore are of a minor relevance.

- Assessment of Initial and Boundary Conditions

The boundary conditions have not always been chosen conservatively. Thus, in the Bulgarian analysis of a rupture of the steam generator collector top it is, for example, assumed that only one HP-emergency cooling pump is available. This may be considered as a conservative assumption from the viewpoint of core cooling during the initial phase. But in the long-term, the water reserves of the primary side are consumed earlier with all HP-emergency cooling pumps being available. Furthermore, this results in a higher release to the environment via the BRU-A, if all HP-emergency cooling pumps inject.

Leak rates set conservatively high from the viewpoint of emergency cooling analysis result in actuating reactor scram upon break of a steam generator tube. Using the best-estimate calculation of the leak rate considering pipe friction, the criterion "primary system pressure < 14.7 MPa" will possibly not be reached, if the high pressure injection pumps of the make-up system overfeed the leak. There are no statements on reactor shutdown in the description of the analysis relating to the postulated steam generator tube in the Rovno-3 plant, only available in text form.

In a K.A.B. analysis on the break of a steam generator tube it is conservatively assumed that when opening the BRU-A, a safety valve in the defective steam generator opens simultaneously and remains open due to a wrong reference input. These assumptions go beyond the single failure assumption.

- Assessment of the Computer Programs

There are no proofs verifying the DINAMIKA computer program used by K.A.B. The RELAP4/MOD6 computer program used for the Kosloduj analyses, has been partially verified for the accident analyses in WWER plants by recalculations of experiments at the Hungarian test facility PMK-NVH. The simulation of the secondary side of the steam generators with only one control volume is insufficient for the reproduction of the phase separation processes during heater tube leakage.

- Assessment of Results

The K.A.B. analyses relating to heater tube leaks have not been completed. The provisional conclusion drawn by K.A.B. that it is not to be expected that the admissible limits of activity release to the environment are exceeded, appears to be premature, as the problems arising from a possible admission of water to the BRU-A and to the main steam lines of the defective steam generator were not dealt with. The requirement of isolating valves to the steam generator safety valves put forward by K.A.B is not supported. In contrast to that, it is suggested that the BRU-A be equipped with isolating valves, so that the steam generator concerned can be shut off when the BRU-A fails in the open position (R 5.1-6).

It can be derived from the present analyses relating to the rupture of the collector top in the steam generator that the early shutdown with an intact secondary side is highly significant. In this context, feeding with the startup and shutdown system is also very important. It can further be derived that the pressure loads of the secondary side can be accommodated. The HP-emergency cooling pumps are insignificant for core cooling, when the primary system pressure can be lowered to a level below the actuation pressure of the BRU-A at an early stage. It is questionable whether the BRU-A can be closed again after a longer admission of water (in the analyses at least 10 minutes) and whether the main steam lines withstand the admission of a mixture of steam and water. This, above all, is also important with respect to safeguarding long-term core cooling, as with a BRU-A remaininig open, emergency cooling water is lost into the atmosphere.

Some details of the results of the Kosloduj-5 analysis relating to the break of the collector top are not plausible. Qualitative criteria are mentioned for the automatic closure of the main steam isolating valve in the steam generator assumed for the analysis, which do not correspond to the present interlock lists (e.g. Stendal NPP, K.A.B.). K.A.B. in its evaluation arrives at the conclusion that inadmissible releases cannot be avoided. This is confirmed by Russian analyses of 1989 and 1991 /MRE 92/ which are, however, not available to GRS. Structural changes of the collector top entirely excluding the accident or at least limiting its effects to a considerable degree, are therefore required. We principally agree with this recommendation (R 5.1-7). In addition thereto, the subcritical rupture in the collector itself should be analysed. This is also to be seen against the background of the damage which occured in the collectors of many steam generators of the WWER-1000 type (cf. Sections 4.2.2.2 and 4.2.5.2).

It is recommended that reliable analyses of the entire spectrum of possible leaks between the primary system and the secondary system should be performed.

The evaluation of the present analyses for the Stendal NPP shows that no accident procedures to control these accidents in the Stendal NPP are available yet. After the development of accident procedures, new accident analyses to prove their effectiveness are required for a final safety assessment (R 5.1-8). The requested development of a suitable accident procedure should consider the following points:

1. the introduction of a time criterion for manual measures (e.g. 30 minutes)
2. ensuring automatic reactor scram
3. cooling down the primary system by shutdown initiated automatically via the intact steam generators with justified shutdown gradients, preferably via the BRU-K; ensuring sufficient capacities for auxiliary and emergency feeding of the steam generators
4. automatic primary-side depressurisation at sufficient subcooling until pressure equalisation with the secondary side of the defective steam generator; ensuring sufficient capacities for spraying in the pressuriser; preventing the

actuation of the HP-emergency cooling pumps for small leaks by the appropriate selection of the shutdown gradient (not too large), or as a substitutional measure, the prevention of long-term pressure effect by HP-emergency cooling pumps, possibly by lowering the zero-lift of these pumps below the actuation pressure of the secondary-side safety valves

5. isolation of the main steam line of the defective steam generator by criteria like, for example, "activity in the main steam line high"(in reactor protection quality and sufficiently spaced apart from the other main steam lines), possibly logically linked to "water level in the defective steam generator high".

6. additional borating of the primary system to prevent recriticality, e.g. upon backflow from the defective secondary side

7. safeguarding the lockability of the BRU-A of the defective steam generator after a previous outflow of water and mixture, for example by an isolation valve, qualifying the pipes belonging to the BRU-A for the admission of two-phase mixture; possibly increasing the actuation pressure of the safety valves with a sufficient distance to the actuation pressure of the BRU-A.

8. ensuring sufficient quantities of borated water to supplement the primary coolant

According to recent information /MRE 92/, similar suggestions with a far-reaching automation of the above measures are currently being discussed in Russia.

### 5.1.3 Transients

#### 5.1.3.1 Reactivity Accidents

The assessment is exclusively based on analyses conducted by other institutions. Own analyses have not been performed within the course of this safety assessment.

- **Existing Analyses**

- Completeness of the Accident Spectrum

In the Technical Project only one single reactivity accident, the uncontrolled withdrawal of a group of control elements, was described in writing. An analysis relating to the ejection of a control element in combination with a leak at the reactor pressure vessel head was conducted by the VUJE research institute (CSFR) with the RELAP4/Mod6 computer program. The thermohydraulic result of this analysis was also used for the assessment in Section 5.1.2.1. There are written descriptions of more recent analyses with the DYBERCORE computer program performed by K.A.B. relating to the withdrawal of control elements, with failure of the reactor scram, from hot zero power and from power operation, to the ejection of a control element from full load, as well as to a loading accident (incorrect loading).

There are no analyses for further reactivity accidents which are to be examined according to the BMI List of Notes for a standard safety report and according to the Accident Guidelines, for example:

- Uncontrolled withdrawal of the most effective control element from the cold and the hot, subcritical state
- Ejection of a control element
- Erroneous drop or erroneous insertion of control elements
- Cold water injection into the reactor cooling system
- Inadvertent reduction of the boron content in the reactor core area
- Inadvertent change of the boron concentration in the coolant (injection of clean condensate)
- Coolant temperature transients (e.g. break of main steam line)

- Initial and Boundary Conditions

The initial and boundary conditions apply to the 2-year cycle and are no longer up-to-date. They are insufficiently documented.

- Computer Programs

The programs TESCH-M with SONA 2 (thermohydraulics of the core) and the TWEL module used by the Soviet project engineer are three equation equilibrium models with point kinetics, two core channels and a bypass channel. There is no information on verification. The nuclear computation system PHYBER-WWER-1000 used by the architect engineer comprises enlarged versions of the NESSEL, KASTALIA and PYTHIA programs, the new PREPAR, TRAPEZ and POLEX programs for a more exact calculation of the core and the DERAB program for a fine mesh calculation. The computation system DYBER-CORE for analysing the core behaviour during transients branches out into the zero- and one-dimensional branch with the PYTHIA, DERAB, INCO and FLOPOIN programs (isolated cooling channel) and the three-dimensional branch with the RAUDY or DYN3D, FLEX or DERAB and INVER (cooling channels with cross-exchange). Some results on the previous verification of the computation systems are available.

- **Assessment of the Analyses**

- Assessment of the Completeness of the Accident Spectrum

The statements in the Technical Project relating to reactivity accidents are completely inadequate for an assessment with respect to the accident spectrum and the results of the analyses.

The further documents available referring to reactivity accidents are insufficient for safety assessment. Analyses on some important accidents are missing completely. With the exception of the Technical Project, the existing analyses relating to reactivity accidents had not been intended for presentation within the framework of a licensing procedure. They were frequently classified as "Provisional Assessment ..." or "Provisional Study to Prepare an Accident Analysis" by the authors.

- Assessment of Initial and Boundary Conditions

The reliability of the present accident analyses relating to reactivity accidents is strongly undermined weakened by the fact that initial and boundary conditions for the outdated 2-year cycle have been used, which are furthermore insufficiently documented.

- Assessment of the Computer Programs

There is no information verifying the TESCH-M computer programs, including the SONA 2 and TWEL modules, used by the Soviet project engineer. The existing verification material for the program system used by the architect engineer K.A.B. is incomplete.

Because of the unsuitability of the RELAP4/MOD6 computing program for this purpose, the analysis conducted by the VUJE research institute (CSFR) relating to the ejection of a control rod combined with a leak from the reactor pressure vessel head can only to a restricted extent be regarded as a reactivity accident analysis.

- Assessment of the Results of the Analysis

The results of the present K.A.B. analyses are plausible, for the given initial and boundary conditions. The reactivity feedback coefficients used have to be regarded frequently as provisional estimates. The resulting uncertainties still would have to be quantified. It can, however, be seen from the estimates:

- A safety assessment of selected reactivity accidents, like for example, uncontrolled movements of control elements or breaks in main steam lines, is required for the actual core loading.
- Restrictive measures for the control and start-up concept, i.e. the determination of inadmissible combinations of control element positions, are urgently required.
- If all six control elements belonging to the control group are ejected, the critical surface heat flux will only be exceeded in the hot channel. Investigations relating to the fuel rod behaviour after exceeding the critical surface heat flux are still to be performed.

- With the present core instrumentation an incorrect loading of the fuel elements cannot clearly be identified before starting power operation.

The entire spectrum of reactivity accidents is to be analysed anew under conservative boundary and initial conditions with verified computer programs and up-to-date nuclear data. The results are to be evaluated according to the requirements set forth in the codes and standards (R 5.1-9).

According to recent information /MRE 92/, there are additional Russian analyses of reactivity accidents. These were, however, not available for safety assessment, as they were not included in the scope of the contract for the Stendal NPP when the project was terminated.

#### 5.1.3.2 Leaks and Breaks in the Secondary System

No new analyses relating to this group of accidents were performed. Existing accident analyses by other institutions for the Stendal NPP, Unit A, or other plants of the WWER-1000 type are summarised and assessed.

- **Existing Analyses**

- Completeness of the Accident Spectrum

The Technical Project contains the analysis of the double-ended break in the main steam line as the only case of this group of accidents.

K.A.B. performed analyses of the double-ended main steam line break (DN 500 and DN 600) downstream of the main steam isolating valve and the check valve with the ANDY-1000 computer program. In the analysis relating to DN 500, two cases were considered: one with closure of the main steam isolating valve in accordance with the intended purpose and the other with failure in the open position. The analysis of DN 600 was carried out with an assumed failure of the main steam isolating valve to close.

In addition, the following accidents were analysed by K.A.B. with DINAMIKA, although not to the extent required for an assessment of the radiological releases:

- Break of the main steam line downstream of the main steam isolating valves with the control and protection devices, including closure of the main steam isolating valves, functioning in accordance with their intended purpose, coincident with the a break of one steam generator tube and loss of off-site power .

- A steam generator safety valve remaining open and the simultaneous 2A-break of a heater tube and loss of off-site power

- several cases of an inadvertent opening of valves in the main steam system (BRU-A, BRU-K, DE-SIV).

A Soviet analysis of a break of the main steam header for a prototype of the WWER-1000 is quoted in the report /DOE 88/.

No analyses referring to leaks and breaks in the feedwater system have been performed so far. According to the Soviet codes and standards these cases are not design basis accidents and have not been considered in the Technical Project.

- Initial and Boundary Conditions, Availabilities of the System

The ANDY analyses of K.A.B. have been documented sufficiently with respect to the assumed initial and boundary conditions.

Concerning the initial conditions of the other analyses, there are either no statements or incomplete statements. When available, the parameters of the primary and the secondary system for nominal power operation were chosen. Different cases were analysed, some with and some without a simultaneous loss of off-site power. The criteria for actuating automatic measures in case of accidents of this class, e.g.:

- reactor scram and turbine tripping

- closure of the main steam isolating valves in the main steam line

- closure of the feedwater control valves

- shutdown of the main coolant pumps

in the accident descriptions of K.A.B. have been chosen largely corresponding to the interlocks intended for the Stendal NPP /K.A.B. 91a/. Different values for the pressure decrease in the main steam header as a reactor protection signal were, however, used in the analyses. According to recent Russian statements /MRE 92/, the shutdown signal "pressure decrease in the main steam collector high" has not been used in reactor plants for about four years, because of the unreliable measuring technique. It was replaced by the signal "Difference of the saturation temperatures between primary and secondary system more than 75 °C with main steam pressure less than 49 bar".

- Computer Programs and Models

The analysis in the Technical Project was performed with an unknown program with a homogeneous representation of the primary-side fluid dynamics. The remaining analyses were conducted with the DINAMIKA and ANDY-1000 programs.

For DINAMIKA there is no information relating to modelling and there are no proofs of verification.

The ANDY-1000 code is a K.A.B. development for transient analysis with a one-phase coolant in the primary system. All necessary main components, safety devices, control units and interlocks are programmed in the code as modules. The kinetics are represented by a point model. The thermohydraulic behaviour of the reactor core can be described by a normal and a hot channel having six axial zones. A subdivision of the primary system is possible with variable nodalisation into two loops.

- **Assessment of the Existing Analyses**

- Assessment of the Completeness of the Accident Spectrum

The present accident analyses do not comprise the entire spectrum of leaks and breaks in the secondary system.

The statements in the safety report of the Technical Project relating to secondary-side leaks are completely insufficient.

In addition to the K.A.B. analyses of breaks in the main steam system, new analyses relating to this group of accidents are to be performed. The most unfavourable combination for sub-cooling the primary coolant in the core must be found by a systematic variation of leak position and leak size. The necessity to apply 3D-core models may arise here. For these analyses, plant data for the current core loading are to be used. Cases, which have not been analysed so far, like main steam line breaks from hot zero power condition, are to be performed, to determine whether recriticality occurs. It is also recommended that 3D-core models are used for this additional analysis (R 5.1-10).

There are recent Russian analyses /MRE 92/ of main steam line breaks including 3D-computations, but these are not available to GRS.

One peculiarity of the Stendal plant is that the main steam isolating valves are not positioned directly at the penetration of the main steam lines through the containment, but after a distance of several metres. It is therefore recommended that analyses be performed to demonstrate accident control for the accident category "main steam line break between the containment penetration and the isolating valve, with simultaneous break in a steam generator tube or leaks or breaks in the steam generator collector". To prove the basic safety of the collector, it is only still to be demonstrated analytically that the rupture of the collector head can be tolerated, possibly considering structural measures to reduce the consequences of a break of the head. An alternative exist in structural measures to preclude main steam line leaks between the containment penetration and the fast-acting isolating valve (R 5.1-11).

It is further recommended to analyse consequential breaks of main steam and feedwater lines in the region where these lines are close together near the penetrations through the containment (cf. also R 4.2-13). These analyses serve the purpose of proving accident management requirements. They will not need to be performed if the pipes are sufficiently protected from each other by spatial separation (partition walls) (R 5.1-12).

As a summary, it is recommended to carry out analyses of leaks and breaks in the main steam system corresponding to the entire accident spectrum of the accident guidelines.

In accordance with the accident guidelines analyses of leaks and breaks in the feedwater line are also to be requested (R 5.1-13). Analyses of leaks in the purging line are also to be requested, unless sufficient preventive measures (e.g. double-walled pipes) can be demonstrated. It is, however, to be expected that the consequences of the rupture of a feedwater line at the steam generator, with respect to sub-cooling of the primary system and the reactivity increase in the core, are milder than the rupture of a main steam line.

- Assessment of the Initial and Boundary Conditions Applied

The initial and boundary conditions selected for the case of a double-ended break of the main steam line described in the Technical Project are very inadequately documented and can therefore not be assessed. The assumed position of the break cannot clearly be recognised. Presumably the break is located between the steam generator and the main steam isolating valve, but inside the containment.

The assumed initial and boundary conditions of the K.A.B. analyses with the ANDY-1000 computing program largely correspond to the requirements set forth in the German codes and standards. The interlocks used largely correspond to the K.A.B interlock lists for the Stendal NPP, but different values for the reactor protection signal "Pressure decrease in the main steam collector high" are, however, used in the analyses. Because of the introduction of a new reactor protection signal "Difference of the saturation temperatures between primary and secondary system high at main steam pressure low" the assessment of the present analyses is only partially appropriate. The recommended future analyses have to be carried out using the current reactor protection criteria (R 5.1-14).

The statements referring to initial and boundary conditions of K.A.B. analyses using DINAMIKA are incomplete.

The statements referring to initial and boundary conditions in the Soviet analysis /DOE 88/ relating to the break of the main steam header are incomplete; in particular it cannot be derived from this analysis how many main steam isolating valves are assumed to fail in the open position.

- Assessment of the Computing Programs and Model Assumptions

As the computing programs used by the Soviet side are not known, an assessment cannot be provided.

The ANDY-1000 computing program employed by K.A.B. seems to be principally suitable for the simulation of secondary-side leakage accidents, on the basis of the information available. Statements on code verification are, however, not available.

The statements relating to the model assumptions for the cases investigated with ANDY are generally plausible, but a complete mixing of the primary coolant of the defective loop with the coolant of the intact loops in the annulus and in the lower plenum was assumed without verification. In future analyses, experimentally supported assumptions for coolant mixing are to be made, to be able to evaluate the effects of this model assumption, for example, on the increase of reactivity in the core. 3D-core models must also be used in these analyses.

The DINAMIKA computer program also used by K.A.B., on the basis of the information available seems to be principally suitable for the simulation of secondary-side leakage accidents. Statements on code verification are, however, not available. Because of the restricted modeling capabilities of the DINAMIKA computer program, the function of the turbine controller cannot be modelled to the required extent. Future analyses should consider the turbine controller behaviour in the appropriate form.

- Assessment of the Results of the Existing Analysis

The results of the analysis of the double-ended break in the main steam line described in the Technical Project are incompletely documented. The analysis in the Technical Project is useless for a safety assessment.

The assessment of the present analyses basically restricts itself to main steam line breaks downsream of the isolating valves with or without a presumed failure of the valves in the open position, analysed using ANDY-1000. The sufficiently described results are plausible and can be reconstructed.

As an essential result of the analyses with single failures, positive reactivity increments due to cooling down the primary system, of the order of the shutdown reactivity of the control rods, were identified. It is not certain whether recriticality as a result of sub-cooling can be avoided. The very precise result of the analyses in this respect is based on the assumption of complete mixing of the primary coolant in the annulus outside the shaft and the lower plenum. As long as there is no experimental verification for this assumption, it must be expected that in some parts of the core recriticality can temporarily occur.

The results of recent Russian analyses relating to a break in the main steam line which cannot be isolated /MRE 92/, which are not available to GRS in detail, confirm that recriticality occurs in the respective quadrant of the reactor core after reactor scram and that about 30 % of the nominal power is reached. DDN can also occur for a short period. For these analyses, mixing in the downcomer and in the lower plenum was not assumed conservatively.

The cases of breaks in the main steam line which can be isolated and of incorrect opening or defective non-closure after actuation of main steam line system valves, partially overlapped by the double-ended break of a steam generator tube calculated by K.A.B. for the early accident phase only, do not permit any statements on the sub-cooling of the primary coolant or on the radiological effects. They do, however, provide a first impression of the accident progression with the relevant automatic actuations. It is recommended to continue these analyses or to perform them again.

The Soviet analysis quoted in the report of the American Department of Energy /DOE 88/ relating to the break of the main steam collector for a prototype of the WWER-1000 cannot be reconstructed. Here there is a very strong primary-side cooldown with subsequent recriticality even after emergency injection of 540 t/h with a boron content of 30 g/l. The analysis was performed for the prototype of the WWER-1000, Unit 5, of the Novo-Voronesh NPP so that the transfer to the Stendal NPP is not appropriate. Presumably the failure of all four main steam isolating valves was assumed. If a failure in the open position is assumed for not more than one main steam isolating valve and the reactor is shutdown by an appropriate reactor protection signal, the break of the collector will be covered by the analysis of the main steam line break.

#### 5.1.3.3 Operational Transients

With respect to operational transients there are analyses by the plant manufacturer of the Stendal Technical Project, analyses by the architect engineer (Kraftwerksanlagenbau, K.A.B.) and additional analyses by the Technical Project Rovno, Unit 3.

- **Present Analyses**

- Completeness of the Accident Spectrum

From the Stendal Technical Project the transients

- failure of all main coolant pumps,
- turbine tripping (with and without failure of the first BRU-K and first BRU-A),
- load rejection of 100 % to 30 % with a re-increase of the load to 100 %,
- load change from 80 % to 100 %,
- loss of off-site power (total failure of power),
- failure of the main feedwater supply,
- change of the supply frequency

were evaluated.

In addition, in the K.A.B. analyses

- blockage of one of four main coolant pumps,
- blockage of one of two main coolant pumps,
- break of the shaft of a main coolant pump with four pumps running,
- load rejection to auxiliary power supply
- turbine tripping
- main feedwater failure

are evaluated.

Analyses of the Technical Project of the Ukrainian nuclear power plant Rovno, Unit 3, available as an English translation by the U.S. Department of Energy were additionally evaluated:

- blockage of one of four main coolant pumps
- blockage of all main coolant pumps

- decrease of the supply frequency
- turbine tripping with and without opening the 1st BRU-K and the 1st BRU-A
- load alteration including load rejection to zero
- failure of the main feedwater supply
- failure of the high pressure preheaters
- inadvertent closure of a main steam isolating valve
- loss of off-site power

- Initial and Boundary Conditions

In the analyses of the Technical Project, according to the documents, the respective most adverse combination of the parameters reactor power (± 7 %), pressure (± 3 bar), flow rate (± 800 $m^3/h$) and temperature (± 2 °C) are used. The nuclear data correspond to the fresh first core of the two-year load. The maximum hot-spot factor in the core is 2.72.

The K.A.B. analyses were generally started from the nominal conditions of the primary system and the secondary system. In case of pump failure, a thermal reactor power of 107 % is assumed. The nuclear data correspond to the first cycle of the 2-year load. The control and protection devices are assumed to function as intended.

In the Technical Project as well as in the K.A.B. analyses the activation of reactor scram on the first reactor protection criterion is presumed.

- Computing and Models

An unnamed computer program was used for the analyses in the Stendal as well as Rovno Technical Projects. The K.A.B analyses were performed with the DINAMIKA, DYBLO and ANDY computer programs.

- **Assessment Criteria**

For the operational transients, it must be demonstrated according to RSK Guideline 3.1.3 (2) that the heat flux densities have a sufficient margin to the critical heat flux density, that the pressure in the primary system generally remains below the actuation pressure of the safety valves and that the release of energy in the fuel rods is so low that melting is avoided.

- **Assessment**

- Assessment of the Completeness of the Accident Spectrum

With the exception of the main heat sink all essential operational transients are taken into account.

- Assessment of the Initial and Boundary Conditions

The boundary conditions selected in the Technical Project can generally be assessed as being conservative. This does not apply to most of the K.A.B. analyses. The use of the respective first reactor protection criterion cannot be assessed as being conservative.

- Assessment of Computing Programs and Models

The DINAMIKA, DYBLO and ANDY computer programs appear in principle to be suitable for the analysis of operational transients. There were, however, no documents verifying these computing programs. Futhermore, the degree of specification of the modelling was not indicated or there was insufficient detail.

- Assessment of the Results

The results from the Stendal Technical Project are generally plausible. With the working limitations and protective devices, inadmissible states of the plant do not occur. Except for the loss of off-site power, there is no actuation of the steam generator safety valves. In the latter case, according to the present analyses, opening of the pressuriser safety valves cannot be excluded. The documentation in the Stendal Technical Project is, however, incomplete. Sometimes the computations end before steady conditions are reached and the sequences of relevant parameters, for example of the DNB correlation, are frequently missing. Apart from that, there are at times contradictions between the description in the text and the diagrams.

The K.A.B. analyses partially have a provisional character and most cases require completion. Compared to the Technical Project, the computations distinguish themselves by a greater degree of specification with respect to the modelling of the affected systems. The present results are generally plausible. They prove that, with the reactor protection and the safety devices operating as intended, no endangering states are to be expected in case of operational transients.

The analyses of the Technical Project referring to Rovno, Unit 3 are partially identical with those of the Stendal Technical Project. In the Rovno Technical Project it is also stated that, during load alternations, axial xenon oscillations with subsequent inadmissible power density distributions are possible, which have to be controlled manually by the operators. This is probably due to the core loading strategy with the 2-year cycle on which the analysis is based and the use of part-length absorber rods. These results are possibly outdated because of the planned change to the three-year cycle and the renouncement of part-length absorber rods. The final data are, however, not yet available. It is recommended to examine the stability behaviour of the reactor core with the final core data (R 5.1-15). The measures for avoiding xenon oscillations should be automated. (cf. R 4.1-6).

Because of the deficiencies of the analyses in the Technical Project and because of the provisional character of the K.A.B. analyses it is recommended to analyse anew the entire spectrum of operating transients in accordance with the BMI List of Notes for a standard safety analysis report, using the finally determined set values of the reactor protection system for controlling the safety system (R 5.1-16).

It is also recommended to evaluate systematically the operational transients that have occured in WWER-1000-type plants, with the aim of re-calculating those cases that are well documented and suitable for code verification with an advanced accident code (R.5.1-17).

#### 5.1.3.4 ATWS Accidents

- **Present Analyses**

- Completeness of the Accident Spectrum

Two new K.A.B. analyses were evaluated:

- failure of the main feedwater supply
- failure of the main heat sink during failure of the auxiliary-power supply

In addition, a provisional assessment of K.A.B. relating to the withdrawal of control elements or groups of control elements from zero load and full load with a failure of reactor scram as well as an ATWS study of OKB Gidropress for the reactor concept WWER-1000/88 were used for assessment.

- Initial and Boundary Conditions

In the K.A.B analyses nominal conditions at full load, beginning of cycle, of the primary core (BOL) and failure of the absorber rods to insert on request were assumed. All other systems function in accordance with the design, if their ability to function is not impaired by the initiating event. As there were no reactor physics data available for the reactivity feedback during the three-year cycle at the time of the analysis, provisional estimates were used. No single failure assumptions were made. For the loss of off-site power it was assumed that the make-up pumps are not available. The loss of off-site power was calculated in two variants.

- Variant A: without additional boration of the coolant
- Variant B: with injection of the HP-emergency boron injection system

- Computer Programs and Models

The Soviet DINAMIKA program was used for the analyses. Nothing is known about the state of verification of the program, especially for high pressures.

- **Assessment of the Analyses**

- Assessment of the Completeness of the Accident Spectrum

The present documents on ATWS are insufficient with respect to accident spectrum and quality of the analyses. It is recommended to analyse operating transients with a presumed failure of reactor scram (ATWS) according to RSK Guideline 20 (R 5.1-18).

- Assessment of Initial and Boundary Conditions

The reactivity coefficients used are to be assessed as a provisional estimate. The uncertainties resulting therefrom jeopardise the quantitative statements made in the analyses. With this restriction, the results with the given initial and boundary conditions are plausible.

- Assessment of Computing Programs and Models

It is doubted that the modelling of the reactor kinetics (point kinetics), of the two phase leakage and the weighing of the cololant density effects in the DINAMIKA program is adequate for ATWS cases. The same applies to the verification of the program in the range of higher pressures.

- Assessment of the Results of the Analyses

The pressure in the primary system in both cases analysed is limited by opening one (ATWS-main feedwater failure: $P_{max}$ = 18.56 MPa) or more (ATWS-loss of off-site power: $P_{max}$ = 18.81 MPa) pressuriser safety valves. It is to be expected in both cases that there will be an outflow of mixture through the safety valves in the course of the transient. The respective pipes, the pressuriser safety valves and the relief tank of the plant therefore are to be designed for this purpose (R 5.1-19). Individual contributions of the reactivities, outflow rates via the pressuriser valves, the efficiency of the steam generator and the DNB-ratio are missing in the description.

Referring to an ATWS with loss of off-site power, the analysis period is too short as the pressure at the end of the analysis is still higher than 18 MPa and as the reactor power is still significantly higher than the steam generator power. In variant B of this case there is a boron injection via the HP-emergency boron injection pumps. Because of an apparent fault in the program, the results of the analysis can only conditionally be assessed. It furthermore has not been illustrated, on the basis of which signals the pumps of the HP-emergency boron systems are initiated.

The present analyses have not proved long-term heat removal and long-term sub-criticality.

The provisional K.A.B. analyses relating to the incorrect withdrawal of groups of control elements with subsequent failure of the reactor scram demonstrate that from zero load as well as from full load the actual DNB value can fall below the permissible DNB limits. The pressuriser safety valves do, however, limit pressure so that it is not expected that the integrity of the primary system will be endangered by a failure of pipes or components. In the analysis, measurements in the Bulgarian nuclear power station Kosloduj 5 are also quoted, according to which - analogous to the Technical Project for Stendal - the boron injection even at a pressure of about 12.8 MPa only becomes effective about 170 s after opening the valve in the core.

In the Gidropress study of ATWS for the WWER-1000/88 reactor concept, in which a series of ATWS accidents were analysed, the conclusion is drawn that without a fast boration system the second project limit according to OPB-82 is exceeded and partial melting of the core cannot be excluded for these cases. Actuation criteria, pump head, injection rates and boron content of the suggested fast boron injection system are, however, not indicated.

Therefore it is recommended to provide an efficient additional borating system for shutting down the reactor and ensuring long-term sub-criticality during ATWS-accidents (R 5.1-20). For accident control, it must be able to inject effectively with a sufficient boron content, even with the pressures ; 18 MPa to be expected. This system is to be designed as a second shutdown system in the sense of the BMI criterion 5.3 and RSK-Guideline 3.1.2. The dimensions are to be justified by analyses.

### 5.1.4 Summary of the Recommendations relating to Accident Analysis

From the viewpoint of the work group "Accident Analysis" the recommendations given here and listed in Section 10 represent a precondition for licensability of the Stendal plant in the Federal Republic of Germany. A final safety assessment of the plant could be carried out after the recommendations have been observed.

Because of the number and the significance of the recommendations, it is considered to be necessary to perform the entire safety analysis anew as a part of an actualised safety report with progressive, verified computing programs and using up-to-date data for the reactor plant. WWER-specific accidents like, for example, the rupture of the steam generator collector top, as well as accidents which have not been analysed so far, like, for example, ATWS, consequential breaks in the main steam and feedwater system and accidents in the shutdown state of the plant will have to be investigated for this analysis.

## 5.2 Accident Analysis for the Containment

### 5.2.1 Pressure and Temperature Sequence of a 2A-Break of a Main Coolant Line

#### 5.2.1.1 Procedure

The time sequence of the release of mass and energy from the primary and/or secondary system (leak function) essentially determines the pressure and temperature sequences in the containment, which again have an influence on the leak function. This made it necessary to link the analysis program for determining the leak function (ATHLET/FLUT) with the one intended for the determination of pressure and temperature sequences in the containment (CONDRU). With the coupled program system ATHLET/FLUT-CONDRU the leak function was determined until up to about 800 s. The further release of energy, especially from the secondary side of the steam generators was extrapolated /KIM 92, RIS 92/. The computation of the pressure and temperature sequences.of a 2A-break of the main coolant line is performed with the one-zone program CONDRU. The multizone-program RALOC was used for verifying pressure maxima. Both programs were verified in numerous experiments (Batelle, HDR). In an examination of the design of the containment RSK Guideline 5.1 (Design Basis of the Containment) and KTA-Rule 3413 are to be considered.

#### 5.2.1.2 Sets of Data for the CONDRU and RALOC Computing Programs

The containment encloses a confinement system consisting of 63 compartments /WFF 91/, in which the pressurised components of the primary system are located. Its net volume is about 61000 $m^3$. Fig. 5.2-1 shows a cross-section through the reactor building. To occlude radioactive substances, a negative pressure of a maximum of 200 Pa is maintained in the confinement system during normal operation.

CONDRU determines the time sequences of pressure and temperature in the containment during leak accidents (outflow of primary and/or secondary coolant) in a one or two-compartment system. RALOC is a multizone-model which among other things additionally determines local gas concentration, convection currents, temperature stratifications, etc.

The sets of input data /WFF 91/ were prepared from the design documents available in /TEP 81/ and /HER 91/. The volumes of the individual compartments in the containment were added to a total volume for CONDRU computations and to eight zones with 20 connections for RALOC computations. Heat absorbing structures of concrete and steel were considered.

The containment-spray system (sprinkler system) essentially contributes to pressure reduction. Contrary to Konvoi plants, the sprinkler system is necessary in WWER-1000, as the horizontal steam generators are located at about the same height as the in- and outlet nozzles of the reactor pressure vessel. The energy content of the steam generators of the three intact loops on the primary side is thus introduced into the containment by injection, in particular via the LP-emergency cooling system . It is assumed that one of the three injection lines is being repaired and one has failed due to single failure. The preset flow rate of one line according to /TEP 81/ was determined to be 700 t/h, the temperature after re-cooling 30 °C. It is assumed that the containment-spray system from 60 s onwards feeds into the dome-shaped roof of the containment with full capacity and that the sprinkler heads are 100 % effective.

The following computations were performed:

- pressure maximum
    - CONDRU-best-estimate, without increases necessary for licensing
    - CONDRU with increases necessary for licensing in accordance with RSK Guideline 5.1 (2), like, for example:

- 2 % smaller volume of the containment
- in addition to blowdown, the content of a secondary-side steam generator up to the first isolating valves discharges into the containment

- RALOC-best-estimate to verify CONDRU computations.

- Long-term Pressure Sequence

  - CONDRU-best-estimate, initial temperature in the containment 30 °C, temperature of the sprinkler system 30 °C, from 1400 s on the decay heat power is bounded by the emergency cooling system, heating the water without steam generation

  - CONDRU-best-estimate, initial temperature in the containment parameterised up to 60 °C, otherwise as above

  - CONDRU-best-estimate, from 1400 s on the decay heat power is not bounded by the emergency cooling system (steam generation).

#### 5.2.1.3 Results

- Maximum Pressure

With 382 or 386 kPa the pressure maxima calculated with CONDRU and RALOC on the basis of best-estimate assumptions are practically the same (Fig. 5.2-2). The temporal delay of about 6 s is insignificant on the load on the containment.

In accordance with the requirements of the German codes and standards, i.e. especially under additional consideration of mass and energy of the secondary-side inventory of a steam generator, a pressure of 432 kPa in the containment was calculated. The requested 15 % safety increase on the overpressure for adverse operational states and the calculation uncertainties lead to a pressure of 482 kPa which thus remains below the design pressure of 500 kPa.

- Maximum Temperature

The calculated maximum transient temperature in the atmosphere of the containment, according to RSK Guideline 5.1 (3), is 134 °C (see Fig. 5.2-3). In individual compartments somewhat higher temperatures can occur according to RALOC computations. The maximum transient temperature does not reach the design temperature of 150 °C given in /TEP 81/. But there is a temperature of 120 °C to 130 °C in the containment for about 1500 s, which is important for the design of cables, seals, etc.

- Sequence of Long-Term Pressure

After the first maximum the pressure drops to approx. 340 kPa and from 400 s to 1070 s steadily rises to 354 kPa (Fig. 5.2-3). The second pressure maximum thus is clearly below the first. In this sequence the conservative design assumptions according to RSK Guidelines only have little influence. The release of steam from the leak, the sprinkler system and the heat removal into the concrete structures have an essential effect on pressure.

With the temporal decrease of the leak outflow the heat sinks, like the sprinkler water and the concrete structures, dominate so that at 5000 s pressure will have dropped below 120 kPa. In the longer term, pressure asymptotically approaches values somewhat above the operational initial pressure, depending on normal operation temperature and sprinkler water temperature during cooling of the structures heated before. A defined negative pressure according to /TEP 81/ can only be reached, if the sprinkler water temperature is clearly below the operational initial temperature of the containment, or the operational initial pressure in the containment is aleady correspondingly low. The maximum temperature in the containment mentioned in /TEP 81/ of 60 °C at a relative humidity of 90 % cannot be used here, as it is not conservative, neither with respect to maximum pressure nor to the negative pressure which can be reached in the long run, as it does not correspond to the operational circumstances and is additionally technically undesirable (e.g. corrosion). 25 to 45 °C with a relative humidity of 20 to 50 % are realistic. Sprinkler water temperature in summer realistically ranges from 30 to 40 °C. The pressure in the containment which can be reached asymptotically then is up to 10 kPa above the operational initial pressure. Only if the sprinker temperature, for example in winter, is at least 10 °C below the operational temperature of the containment or if the starting temperatures in the containment are unrealistically high, can negative pressure in the containment be expected in the long run after the start of the accident. The design concept for the containment according to /TEP 81/, which after a 2A-break of a main coolant pipe is to provide a negative pressure after a few hours, is thus not observed here. Further investigations for long-term accident management are therefore deemed necessary (R.5.2-1).

A computation to estimate the influence of the efficiency of the emergency cooling system showed that from about 1400 s onwards the spray system alone is capable of discharging the decay heat power out of the containment and to decrease pressure (120 kPa after 1d)

- Minimum Pressure

The design minimum pressure for the equipment in the containment is stated to be 85 kPa /TEP 81/. This is the value which can be reached theoretically at 24 °C in the containment, if the adverse initial conditions according to /TEP 81/ before closure of the isolating valves to seal the containment are 60 °C and 90 % relative humidity. The admissible negative pressure for the containment of 50kPa mentioned in Section 7.1.1.4 is not reached by far.

### 5.2.2 Leak in the Secondary System

The maximum pressure upon break of a secondary system line within the containment was estimated. Two sets of fast-acting valves in the steam and feedwater lines of the individual secondary system loops prevent water or steam from other steam generators flowing to the break, even if the single failure criterion is applied. After isolation of the broken loop, it is essentially the energy of the steam generator concerned which flows into the containment. If, for non-fulfilment of RSK Guideline 21.2, a larger damage of the steam generator tubes or the steam generator collector must be assumed, the primary system in this way can also discharge into the containment. The energies resulting therefrom have already been considered for the case of a 2A-break of a main coolant line with the simultaneous discharge of the secondary side of a steam generator (Fig. 5.2-1), but the outflow periods get longer because of the smaller cross-section of the break. Therefore, in this case it is not to be expected that higher pressures occur in the containment than for the 2A-break of a main coolant line with a discharge of a secondary-side steam generator. For a more exact determination of the pressures to be expected, detailed analyses of the locking mechanisms and control of the secondary system's isolating valves, of the expected break dimensions in the secondary system pipes and within the steam generators, of the heat removal from the primary system via the remaining steam generators, etc. are recommended (R 5.2-2).

### 5.2.3 Pressure Differences within the Containment

There are no exploitable documents, design guidelines or analyses relating to pressure differences between the compartments of the containment during outflow processes of loss-of-coolant accidents. But because of the large connecting cross-sections between the compartments /HER 91/, no unusually high pressure differences and, because of the composite steel cell construction technique, no far-reaching damage to walls and ceilings are to be expected. To verify this evaluation, the plant is to be examined in detail with respect to the loads resulting from pressure differences and their absorption according to RSK Guideline 5.1(4) (R 5.2-3).

## 5.3 Radiological Impacts

Three of the eight accidents discussed in Section 3.1 were analysed with respect to the radiological impacts on the environment. A further group of accidents, which is not provided in the Technical Project, i.e. the break of a steam generator collector or the break of a collector top, is not dealt with here, as there is no relevant documentation.

Potential radiation exposures during different design accidents are determined using the computation procedures set forth in the accident computation principles /SBG 83/, considering modifications resulting from the general administrative regulation of Section 45 of the Radiation Protection Ordinance /AVV 90/. The following exposure pathways are examined:

- external exposure by $\beta$-irradiation within the exhaust plume ($\beta$-submersion, organ concerned: skin)
- external exposure by $\gamma$-irradiation from the exhaust plume ($\gamma$-submersion)

- external exposure by $\gamma$-irradiation via the contaminated soil (radiation of the soil)
- internal exposure by radionuclides which are inhaled with the air (inhalation)
- internal exposure by consumption of food (ingestion).

If the ingestion pathway is concerned with the consumption of food or feeding stuff located within a radius of 2000 m of the release location and contaminated via epigeous plants, it is assumed for the computation of the potential radiation exposure in accordance with computation procedures set forth in the accident computation principles /SBG 83/ that their harvest or use, respectively, is terminated one day after the first accidental release of activity.

### 5.3.1 Break of a Primary Coolant Measurement Pipe Outside the Containment

- **Description**

There is a radiological assessment of this accident by K.A.B. (KAB 91b/. The estimates show that it must be expected here that the accident design values of Sec. 28, Subsec. 3 of the Radiation Protection Ordinance are clearly exceeded. The individual computation assumptions to determine the source term therefore have been checked by GRS. For a pipe of 60 metres, in accordance with /KAB 91b/ having a diameter of 0.01 m and a pipe friction with Lambda = 0,02 (smooth pipe) simplified ATHLET simulations /KIM 91c/ resulted in outflow rates with an upper limit of 0.8 kg/s which is, however, one decades below the maximum value of the K.A.B. document. The outflow rates stated in the K.A.B. document are apparently based on an insufficient consideration of the friction pressure losses in the measurement pipe. For this reason the outflow rates as a function of pressure and temperature were determined anew for the two shutdown variants with a cooldown rate of 30 or 60 K/h and the source terms were calculated again similarly to the parameters of /SBG 83/. The release period is 15 h in the first case and 11 h in the second case. The released activity of every nuclide of the first source term is above the respective value of the second source term so that the first case is bounding. For this reason, only the first source term is considered further. For I 131 an activity discharge of $2.6 \times 10^{11}$ Bq, of $9.3 \times 10^{10}$ for CS 134 and $9.3 \times 10^{10}$ Bq for CS 137 is obtained integrally. The activitiy is released via the stack having a height of 100 m. Because of the dimensions of the reactor building, 66 m high and 66 m wide, the influence of the building on the atmospheric spreading was assumed, in accordance with the accident computation principles, which resulted in an effective release height of 84 m. In the time interval up to 8 h 64.5 % of the total release of the noble gases and 96.5 % of the total release of the remaining radionuclides are released, 10 % of the iodine being elemental iodine and 90 % aerosol iodine. The minimum distance to the fence of the site, in the South-Easterly direction, is 420 m.

- **Results**

The most adverse conditions are represented by the atmospheric spreading conditions according to the Pasquill spreading class E, for which the accident computation principles establish a rain intensity of 5 mm/h. The maximum values of radiation exposure through ingestion are found within a distance of 2000 m, for inhalation and external irradiation from the cloud and the ground at the fence of the plant, in a distance of 420 m. The most adverse irradiation exposure in relation to the limit is the effective dose of an infant with 18 mSv compared to the limit of 50 mSv. The thyroid dose of an infant is 49 mSv compared to the limit of 150 mSv. For adults the effective dose is 15 mSv and the thyroid dose 23 mSv with limits of also 50 mSv or 150 mSv, respectively.

The main proportion of the total for the effective dose, with 92.2 %, results from external irradiation from the ground by the nuclides Cs 137 and Cs 134. The ingestion dose via milk, predominantly caused by I 131, contributes 5.6 %.

- **Assessment**

On the whole, the potential radiation exposures calculated for the accident under consideration remain below the accident design values. The results for the second source term are qualitatively the same, resulting in dose values of about 70 % of the values of the first source term.

This also applies to supplementary computations when a termination of the release by manual measures is assumed after 30 minutes. In this case the dose values are 1.5 to 2 % of the above values.

### 5.3.2 Fuel Element Damage during Handling

- **Description of the Procedure**

The accident analysis for the Greifswald NPP, Unit 5, generated certain experience on the radiological impact of these accidents /GRS 92/. It could be perceived that the effects on the environment are low. Without having to perform a complete radiological computation, a rough estimate of the source term for the Stendal NPP can be made here. As the main proportion of the radiation exposure is caused by the nuclide I 131, the comparison is exclusively performed for this nuclide.

In the safety assessment relating to the Greifswald NPP, Unit 5, the radiological impacts of a fuel element handling accident were computed. Radiation exposures associated with various exposure pathways were determined and the total dose was calculated. The maximum values for an infant, as the critical person, and the thyroid, as the critical organ, are encountered at a distance of 2000 m (from the stack) for ingestion, at 500 m for inhalation and external irradiation from the cloud, and at 360 m for external irradiation from the ground. The total thyroid dose is 33 mSv, as compared to the limit of 150 mSv. The calculated potential radiation exposures for the accident considered are thus clearly below the accident planning levels.

To estimate the impacts of the same accident at Stendal NPP, the assumptions on which the calculation for the Greifswald plant were based were compared with the given conditions at Stendal NPP.

The following points were found to be different:

- At Stendal there is a ventilation cutoff after 30 minutes so that there is no release of activity after this period. In Greifswald, however, the release continues over a period of seven days, also via the stack.

- The thermal power per fuel rod is higher at Stendal: 59.0 kW (Greifswald 31.3 kW). Thus, a higher iodine inventory must be assumed.

- The number of damaged fuel rods with 21 fuel rods per fuel element is higher at Stendal (Greifswald 13).

- At Stendal a decay time of 24 h only after shutdown of the reactor is assumed as compared to seven days at Stendal.

The release of I 131 into the environment for Stendal was calculated with the following data:

| | |
|---|---|
| - Total iodine inventory upon reactor shutdown: | $1.90 \times 10^{19}$ Bq, 15.4 % thereof I 131 = $2.93 \times 10^{18}$ Bq |
| - number of fuel elements: | 163 |
| - number of fuel rods per fuel element: | 312, 21 of these are damaged /TÜV 92/ |
| - iodine release into water: | 5 % |
| - distribution coefficient water/gas: | $10^5$ |
| - volume water tank: | 1900 $m^3$ |
| - effective volume gas compartment: | 20000 $m^3$ |
| - airflow above stack: | 40000 $m^3$/h |

The release into the environment for 30 minutes after a decay time of 24 hours at Stendal is calculated to be $5.8 \times 10^9$ Bq. The respective source term at Greifswald in the first eight hours (without long-term phase) was already $3.5 \times 10^{10}$ Bq.

- **Assessment**

The comparison with Greifswald shows that the release of I 131 into the environment upon damage of a fuel element will be lower at the Stendal NPP. Thus for the radiological impacts too, a lower value is to be estimated than at Greifswald, i.e. for Stendal too the values during this accident clearly remain below the accident planning levels.

Furthermore, this calculation is based on the conservative assumption that the unfiltered exhaust air is led via the stack. In reality, however, the automatic switch-over of the exhaust air routing via the aerosol and iodine filters is provided after a period of 10 s after actuation of activity monitoring.

This would lead to a further reduction of the activity released.

### 5.3.3 2A-break of the Main Coolant Line

- **Description**

The analyses of the fuel rod loads from a double ended break of the main coolant line illustrate that cladding tube damage is not expected here (cf. Section 5.1.2.1). This case therefore is considered to be of minor importance from the radiological point of view. As the activity release in this case is only composed of the coolant activity, the results relating to the break of a primary coolant measurement pipe can be used for a rough estimate of the radiological consequences.

Upon break of a measurement pipe an activity release via the stack into the environment of a total of $2.6 \times 10^{11}$ Bq as a maximum for the radiologically most important isotope I 131 was determined. The coolant inventory of I 131 at the beginning of the accident according to the accident calculation principles /SBG 83/ was determined to be $3.1 \times 10^{12}$ Bq. The spiking effect further increases this activity during the accident and it was accordingly considered in the calculations of the source term. The release of I 131 resulting from a break of the measurement pipe can therefore be compared with the release of:

- 8 % of the coolant inventory at the beginning of the accident (without spiking).

Upon rupture of the main coolant line, it is assumed that the entire coolant flows into the containment within approx. 20 s, with about 42 % thereof evaporating. In this case there is no increase of the coolant activity by spiking. With the exception of the noble gases, the activity release is carried in the entrained droplets of water in the steam. According to /SBG 83/ a proportion of 10 % of the steam discharge is to be assumed in consideration of a coolant activity additionally concentrated by the evaporation. For iodine this means an activity release of:

- 10 % x 42 % / (100 % - 42 %) = 7.2 % of the coolant inventory at the beginning of the accident (without spiking).

In contrast to the break of a measurement pipe, the radionuclides released do not reach the environment directly, but they first get into the containment. Until the ventilation valves are closed, a small part can escape from the containment. From then on the activity discharge only takes place via containment leakage. After about two hours pressure equalisation with the external atmosphere is reached (also see Fig. 5.2-2) and the release can be considered to be terminated. From the coolant activity released into the containment therefore only a small fraction reaches the environment.

- **Assessment**

This means that the radiological impact of the accident "break of the main coolant line" will remain clearly below the radiation exposure of the accident "break of the measurement pipe ".

### 5.3.4 Steam Generator Collector Damage

Possible damages of the steam generator collector with radiological impacts to be analysed according to Section 5.1.2.2 are the break of a steam generator collector or the rupture of the steam generator head. There are no detailed documents referring to this group of accidents.

According to /MRE 92/, upon steam generator collector damage, large quantities of the primary system inventory are released directly into the atmosphere within minutes via the blow-off control valve (BRU-A). On the basis of the examinations conducted, especially with respect to the break of a primary coolant measurement pipe, it can be estimated that the radiological impact on the environment in these cases will exceed the accident planning levels according to Sec. 28, Subsec. 3 of the Radiation Protection Ordinance. These accident groups therefore have to be investigated with respect to the radiological impact (R 5.3-1).

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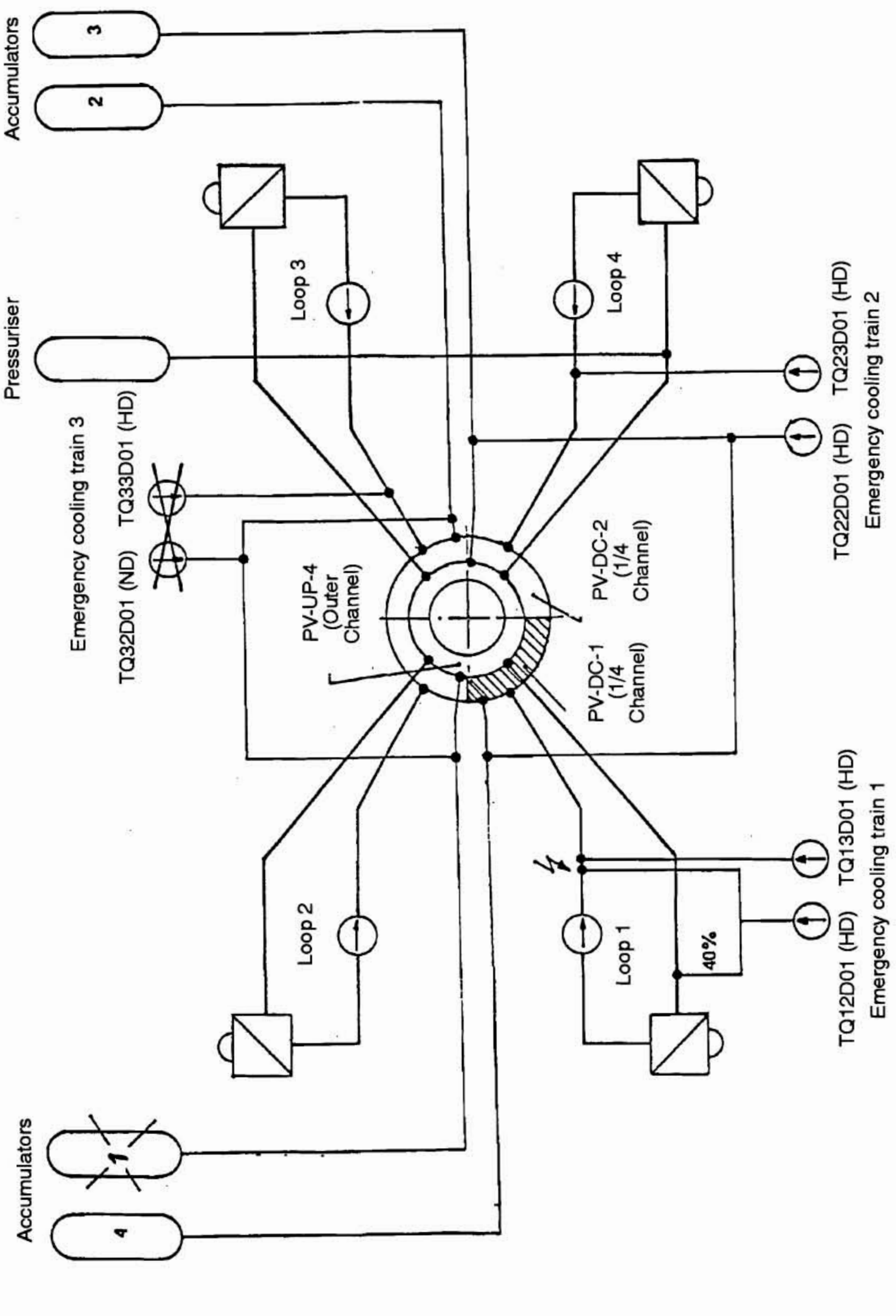



**Fig.5.1-1** Availability of the emergency cooling system

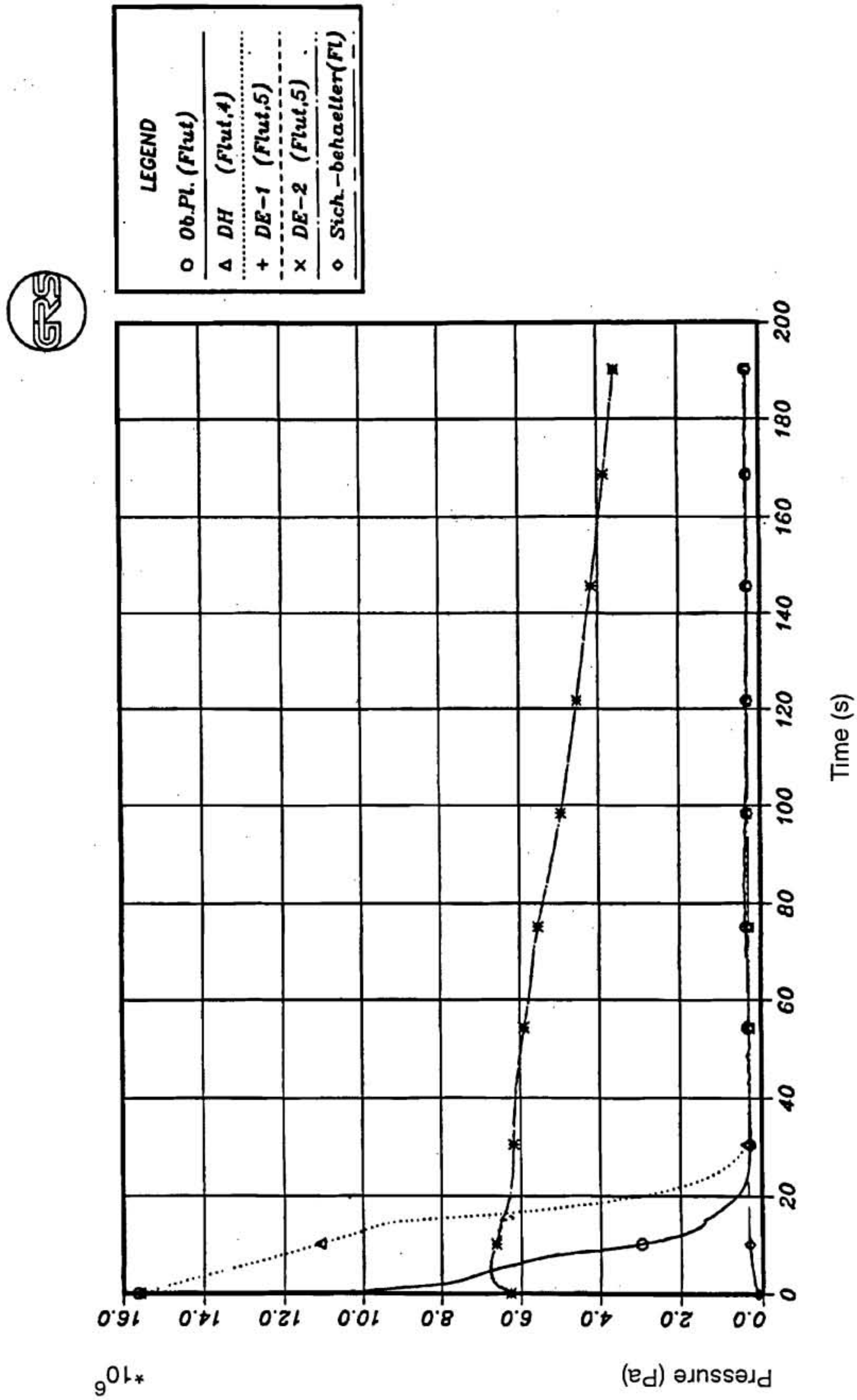



**Fig 5.1-2** Pressure in the primary and secondary system

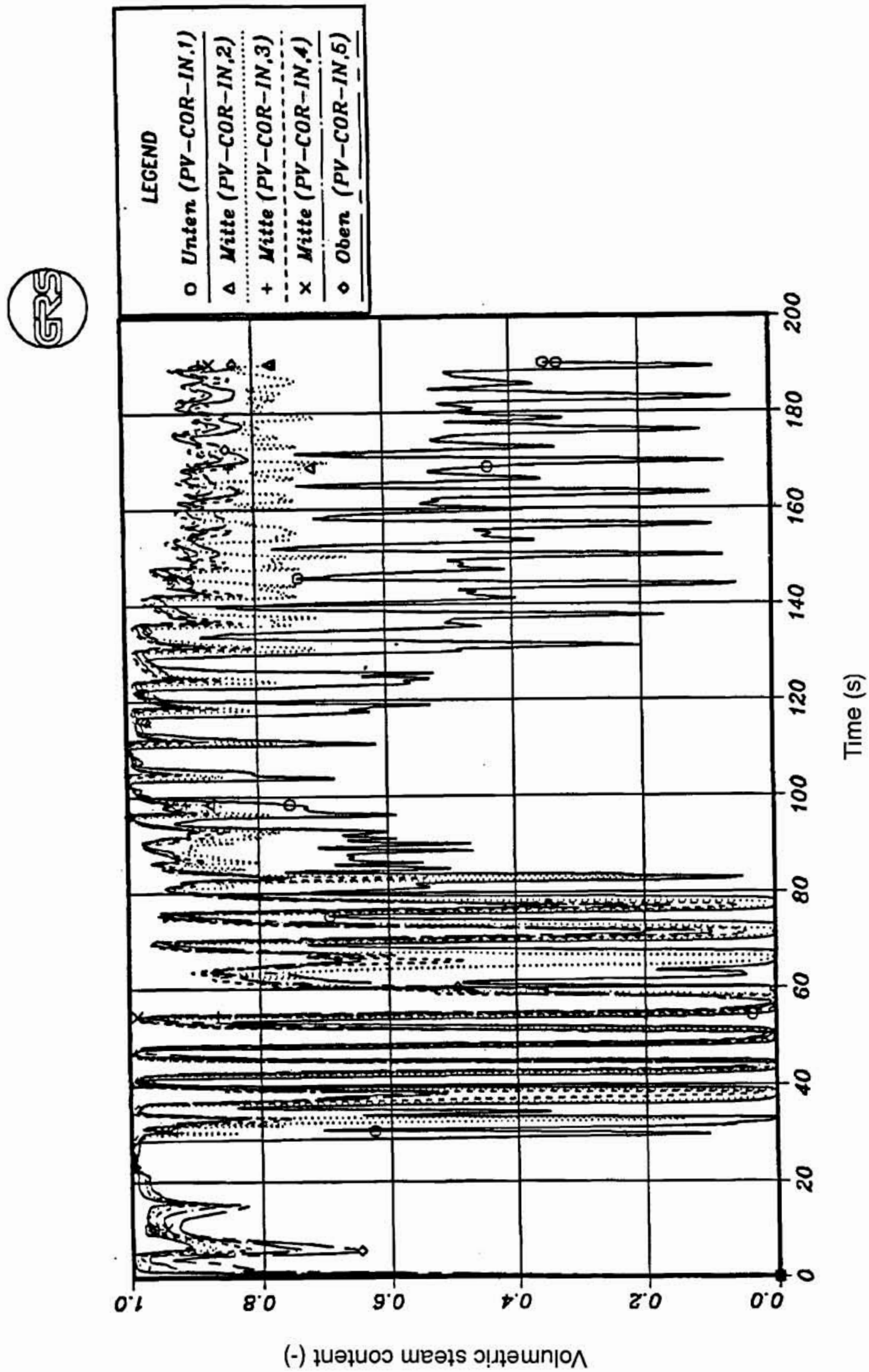



**Fig. 5.1-3** Steam content in the inner core channel

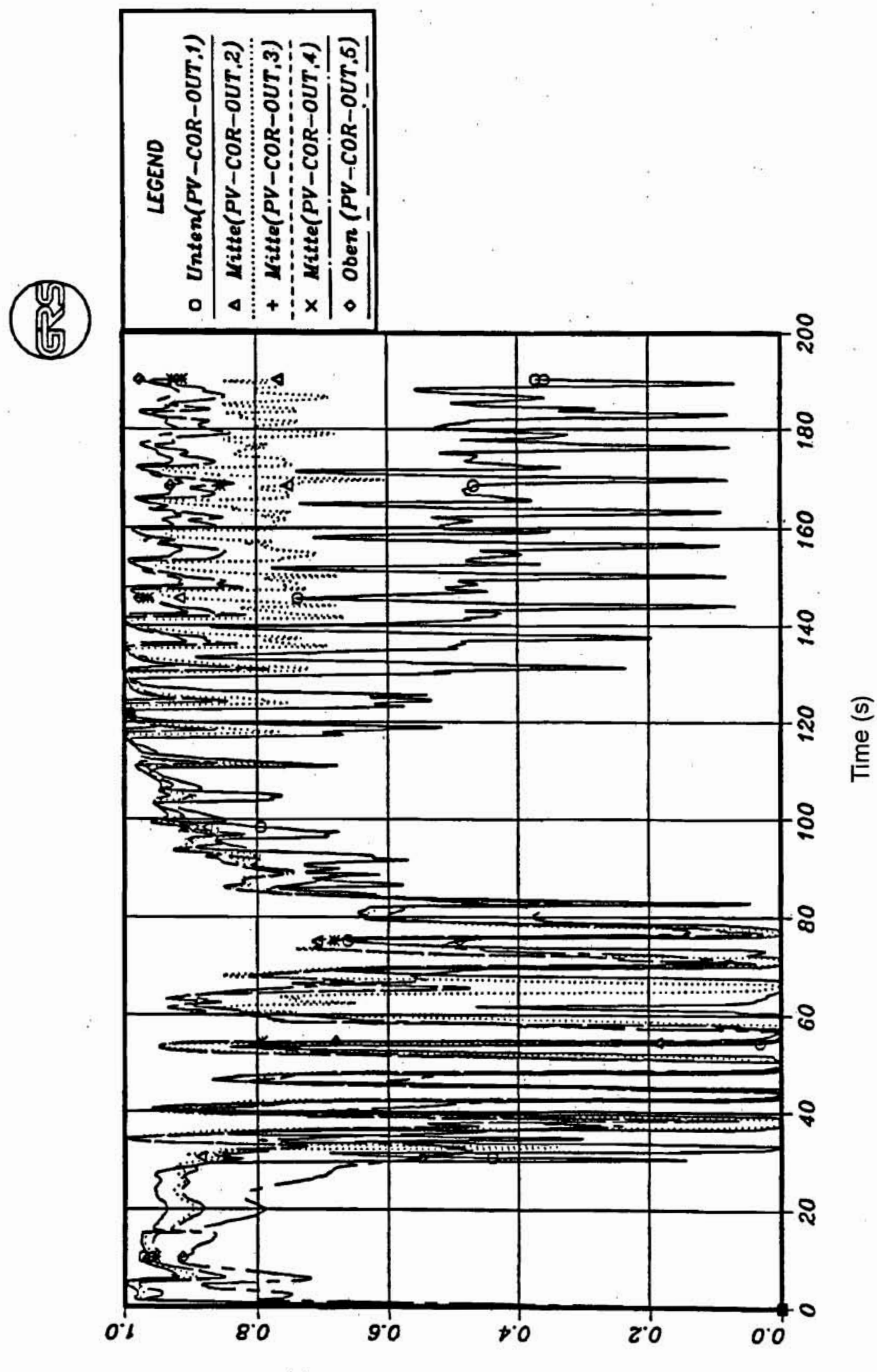



**Fig. 5.1-4** Steam content in the outer core channel

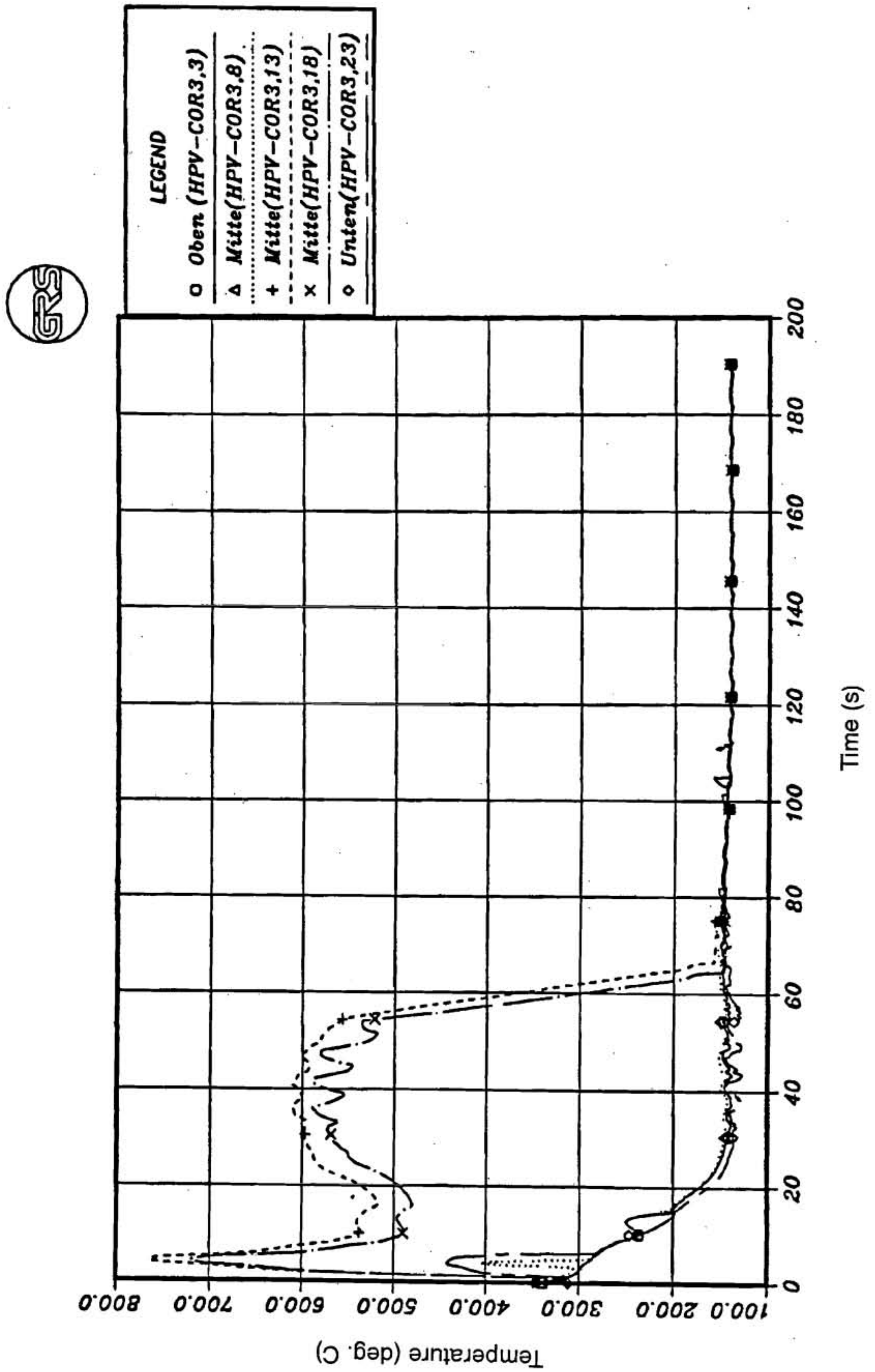
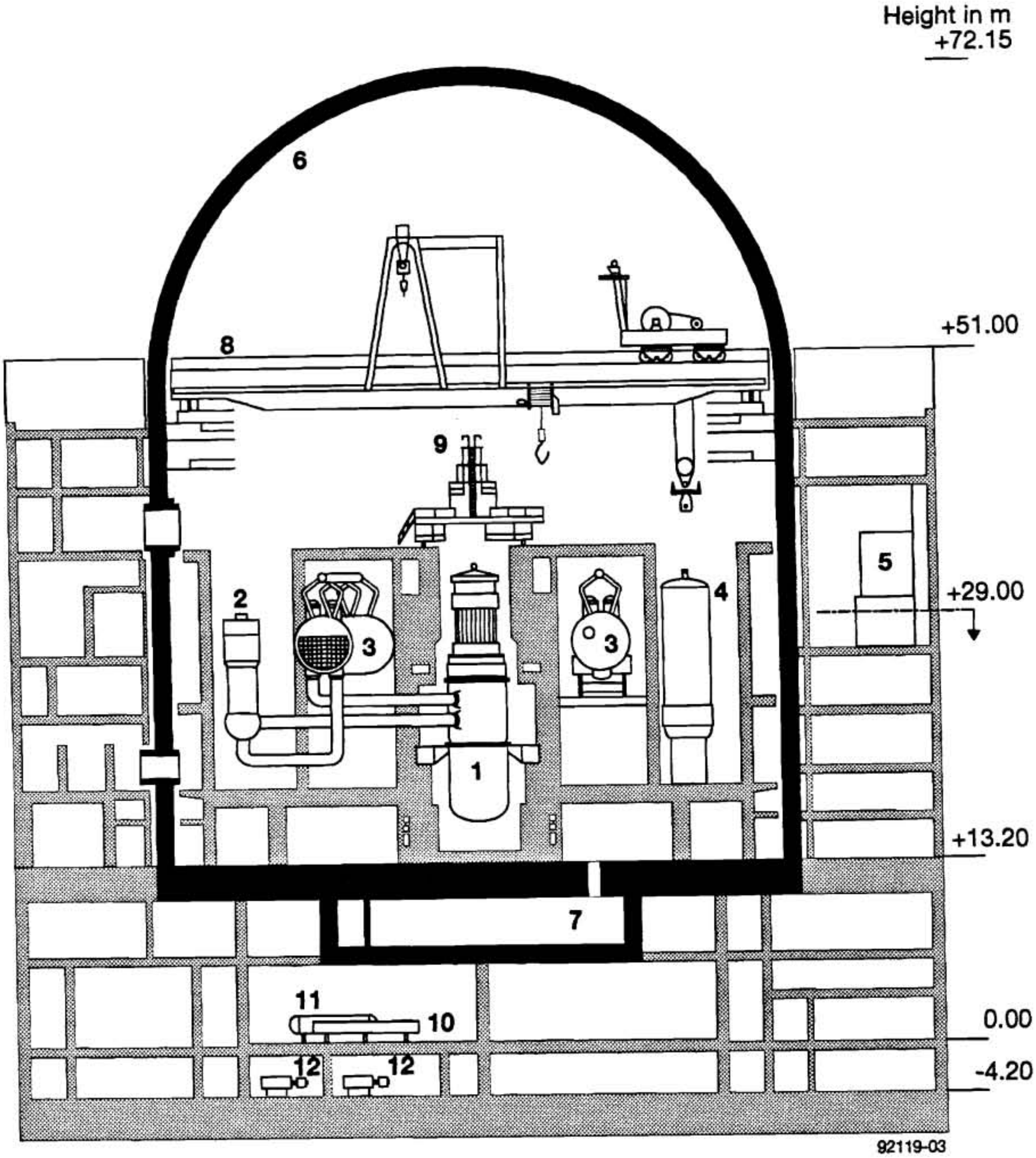



**Fig. 5.1.5** Cladding tube temperatures of the hot rod in the inner core channel




**1** Reactor
**2** Main coolant pump
**3** Steam generator
**4** Pressurizer
**5** Storage tank for service water
**6** Containment
**7** Boric acid storage tank
**8** Main crane 3.2 MN
**9** Fuel assembly handling machine
**10** Emergency cooler
**11** Spent fuel pit heat exchanger
**12** Emergency core cooling pump

**Fig. 5.2-1** Reactor building with containment, arrangement of the components, elevation

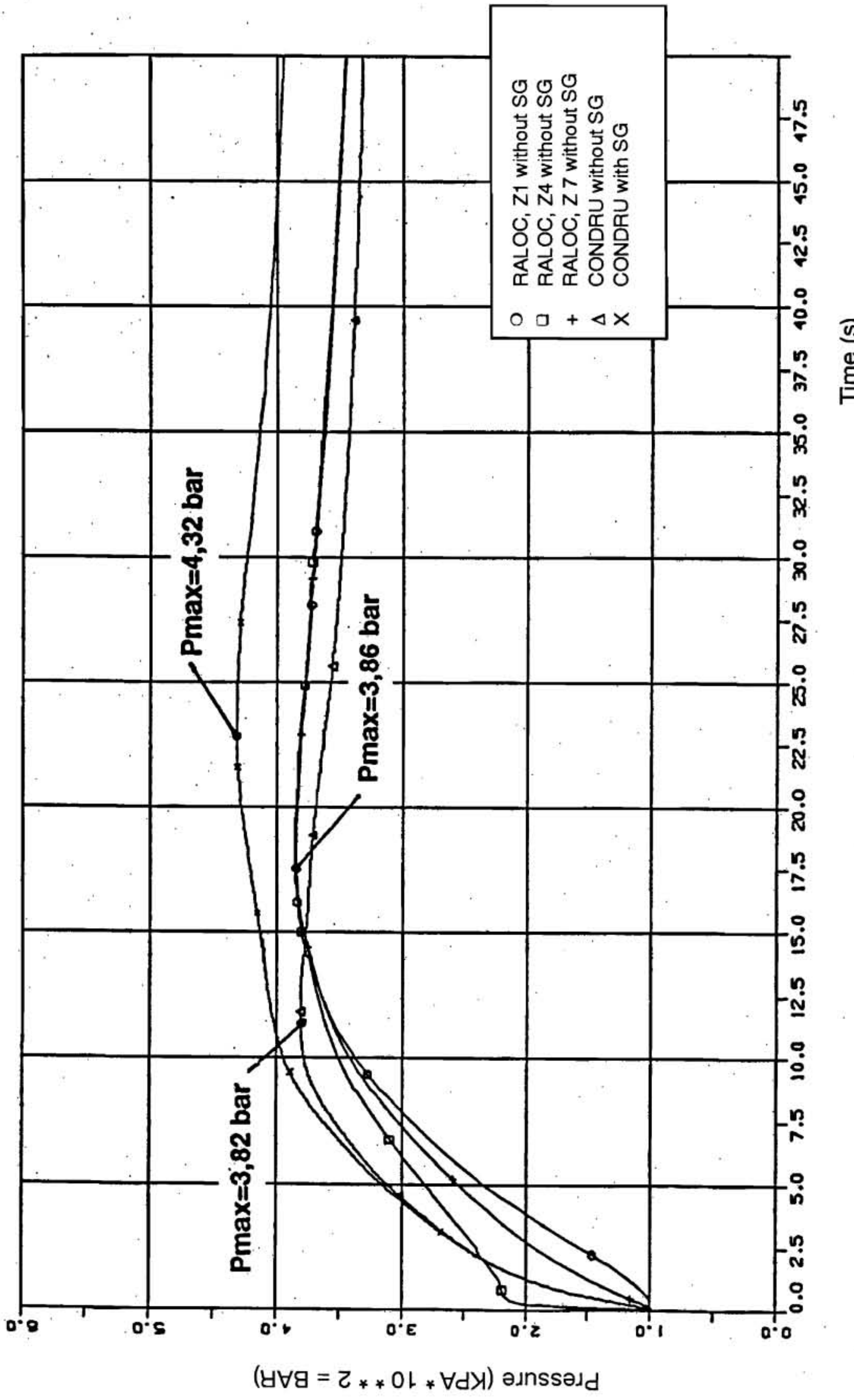
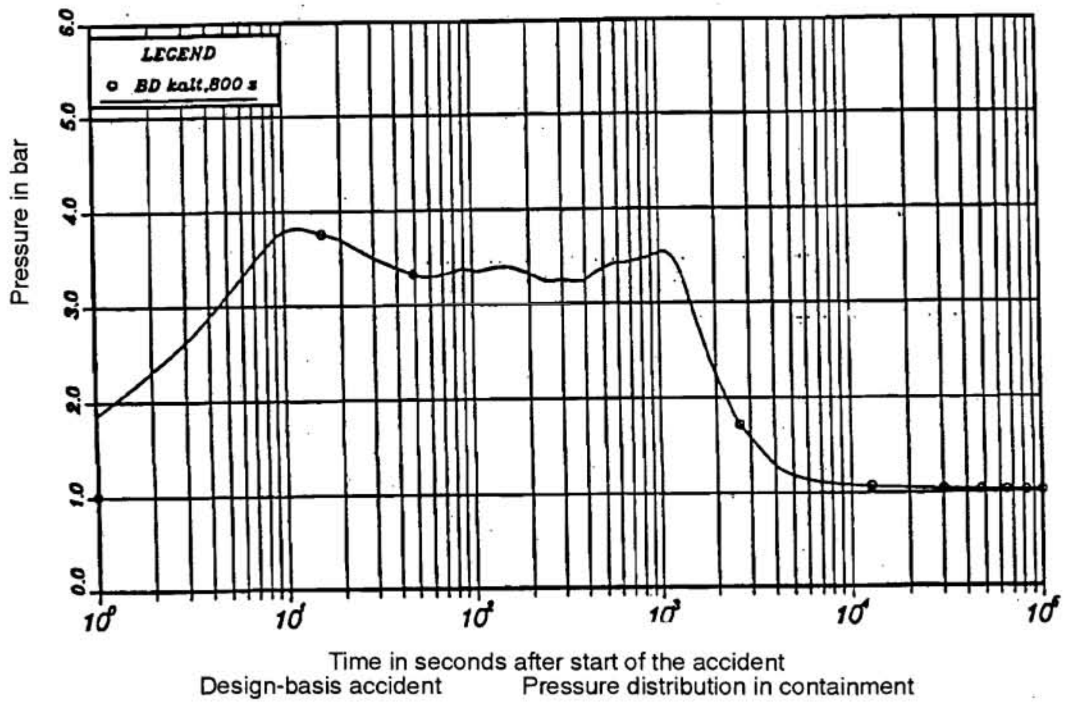



**Fig. 5.2-2** Pressure sequences (maxima) in the containment with and without steam generator, comparison CONDRU/RALOC

CONDRU, BD cold extended, no SG, 1 sprinkler , tw=30




Blowdown cold extended, no SG, 1 sprinkler, tw=30, Tagami

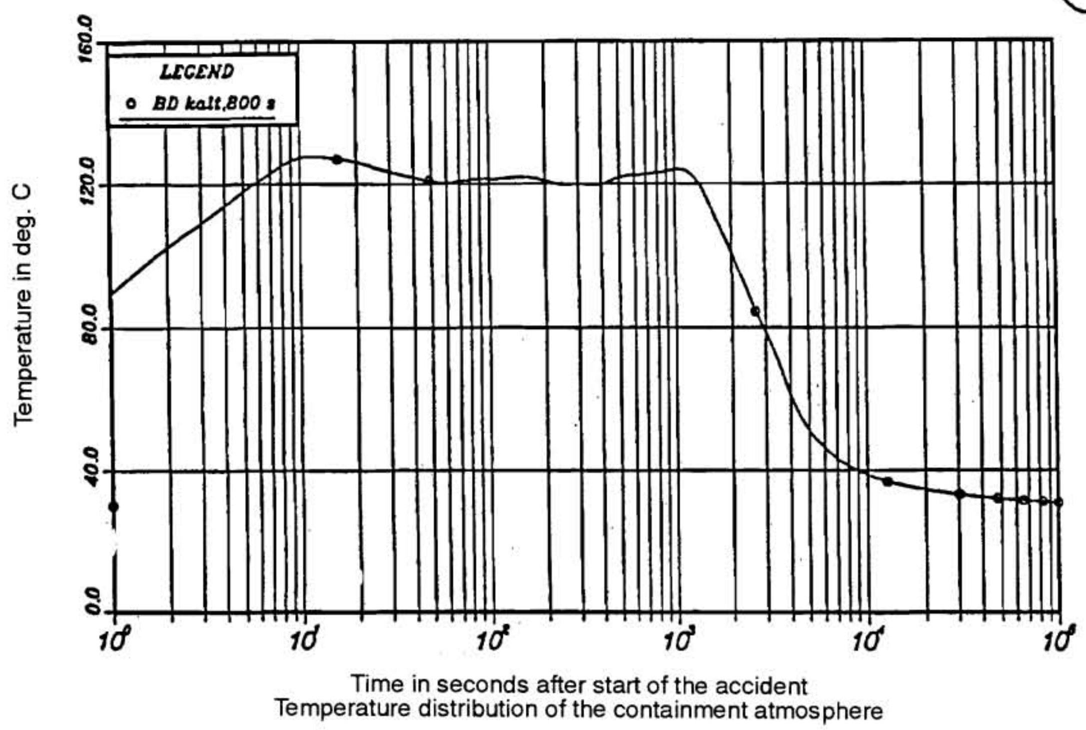



**Fig. 5.2-3** Long-term pressure and temperature sequence in the containment

# 6 Analysis of the Safety System

## 6.1 Introduction

### 6.1.1 Requirements to be met by the Safety System

In the KTA codes and regulations the safety system is defined as the totality of facilities in a reactor plant having the function of protecting the plant against inadmissible strains and, should accidents occur, to keep their impacts on staff, plant and environment within the prenset limits. The safety system, including the safety-relevant operational facilities, must ensure that the following protective aims can be achieved in case of disturbances and accidents:

- interruption of the nuclear chain reaction and maintenance of sub-criticality
- maintenance or re-establishment of the coolant inventory
- removal of accumulated heat and decay heat
- encapsulation of radioactive substances and protection against radioactive irradiation

For the assessment of the effectiveness of the safety system, disturbances and accidents are to be assumed which are activated by

- internal events within the plant (failure of active and/or passive components, fire, flooding)

or by

- external impacts (caused by nature, e.g. earthquakes, floods; caused by civilisation, e.g. air plane crash, explosion blast wave).

Section 5 deals with the spectrum of the different accidents to be assumed.

A high reliability of the safety systems shall above all be achieved by the following design principles:

- Redundancy (sufficient effectiveness even during failure of up to two redundant trains of an engineered safeguard)

- Demeshing (functional separation) and physical separation of the redundant trains
- Diversity of working mechanisms and/or components (as far as possible and appropriate).

The requirements to be met by the assumed accident spectrum and the design of the safety system with a diverse degree of specification are summarised in:

- the BMI Safety Criteria
- the Accident Guidelines
- the RSK Guidelines
- the KTA-Rules and
- the List of Notes for a Standard Safety Report.

In the following paragraphs, to what extent the facilities of the safety system, together with the safety-relevant operational facilities, including the respective auxiliary and supply systems required, fulfil the above design principles will be analysed; considering the state of the art. Where there is sufficient information on the design of the systems, whether the specific requirements contained in the indicated guidelines and rules are fulfilled is also examined. The subsequent safety assessment largely restricts itself to an analysis of the technical design of the plant. Detailed investigations, like an examination of whether safety precautions according to BMI Criterion 1.1 are observed, for example with respect to the consideration of sufficient safety increases during system design or the realisation of maintenance friendliness, are not carried out.

The effectiveness and, above all, the reliability of the facilities of the safety system can be influenced strongly by the operational mode of the plant. An assessment of these influences on the Stendal plant is possible to a limited extent only.

### 6.1.2 Design Principles for the Safety System of the Stendal NPP

During the design and planning phase of the Stendal NPP the "Allgemeine Richtlinie zur Gewährleistung der Sicherheit von KKW ..." (General Guideline for Ensuring the Safety of NPP ...) /OPB 73/ was binding. Drafts of /OPB 82/ have, however, already largely been taken into account /MRE 92/.

The sudden break of the main coolant line under loss of off-site power conditions was assumed as the design accident for the Stendal NPP. Further, the break of a line with a smaller diameter in the primary system, the break of a main steam line, the loss of off-site power as well as different reactivity accidents, etc. were considered as internal initiating events (also see Section 5).

In addition to the internal initiating events, the loss of off-site power and the coincidental failure in a redundancy of an engineered safeguard or the entire safety system, respectively, were assumed in the design phase. The consideration of consequential failures and the repair case are not directly required and are not set out consistently. Earthquakes, blast waves and airplane crash were considered as external impacts.

According to /OPB 82/ the engineered safeguards shall

- serve to fulfil only one aim (para 2.1.7)
- use passive facilities (para 2.1.8)
- render technical and functional examinations possible without reducing the safety level (para 2.1.9 to 2.1.11) and
- provide means which exclude incorrect actions of the operational staff which could lead to an aggravation of the consequences of the failure (para 2.1.12)

Section 3 describes the differences in the application of the single-failure criterion between the German codes and standards and OPB 82. An essential difference is the assessment of the passive components.

### 6.1.3 Preconditions and Boundary Conditions for the Systems Analysis of the Stendal NPP

For the systems analysis of the Stendal NPP, the Technical Project of the Stendal NPP /TEP 81/, supplementary system descriptions of the power plant facilities and the results of a meeting with Russian experts on the Stendal NPP /MRE 92/ are available as binding documents. The state of information corresponds to that of a German nuclear power plant at the concept appraisal stage; in some paragraphs it goes beyond it. To the extent to which the information necessary for safety assessment is not available, it must be presumed provisionally that the nuclear power plant is licensed, built and operated according to the Soviet codes and standards, in particular according to the /OPB 82/ outlined in Section 6.1.2. But the present experiences referring to the construction and operation of nuclear power plants in the former COMECON area prove numerous deviations from the preset codes and standards (see Section 8).

A newly erected plant corresponding to the present project and according to the regulations governing quality assurance is presupposed for this study.

Federal German codes and standards are used as the yardstick for assessing the engineered safeguards of the Stendal NPP. In case of differences between the planned execution and the requirements of the codes and standards, an examination for correspondence of the general meaning or the applicability of the codes and standards is carried out and incorporated into the assessment.

The main emphasis of the safety assessment is on the shutdown systems, the emergency cooling system (emergency core cooling and residual heat removal system), the engineered safeguards of the main steam system, the emergency feedwater supply system, the service water system A (nuclear service water system), the emergency supply system and the I & C systems important to safety, as well as their auxiliary and supply systems.

The engineered safeguards of the Stendal NPP required for accident management are listed in Table 6.1-1. The selection of the accidents in Table 6.1-1 is based on the Federal German Accident Guidelines (Tables I and II) as well as on the List of Notes with Subdivision for a Standard Safety Report for Nuclear Power Plants. In addition, WWER-specific accidents not contained in the Federal German codes and standards were also considered (Table 6.1-1, No. 5, 15). Owing to its more conceptual character, a detailed examination of the minimum requirements to be met by the design of the engineered safeguards is not carried out in the present safety assessment.

Essential preconditions for the compilation of Table 6.1-1 are documented in Section 6.1-1. Accidents are generally considered up to the reactor state "sub-critical cold".

Apart from the accidents listed in Table 6.1-1, in particular

- earthquake,
- airplane crash,
- explosion blast wave and
- fire

are to be taken into account as external impacts for plant design.

The design of the building is dealt with in Section 2.1. The protection of the plant against fire is discussed in Section 7.2.

In case of earthquake, airplane crash and explosion blast wave, the failure of the turbine hall with a coincidence loss of off-site power is presumed so that the accident sequence principally corresponds to accident No. 16, break of the main steam line outside the containment vessel downstream of the fast-acting gate valves with simultaneous loss of off-site power. The engineered safeguards required for controlling accident No. 16 in principle are also to be designed earthquake-safe and to withstand the loads of an airplane crash and an explosion blast wave.

The areas of the service water system A which are located outside the reactor building can be excluded from the design against the loads resulting from an airplane crash, if these are sufficiently remote from the reactor building.

As the simultaneous occurence of an accident induced by external events and a loss-of-coolant accident is to be excluded according to the Federal German codes and standards, then particularly the primary system has to be designed in such a way as to remain leak-tight in case of an accident due to external events.

The engineered safeguards listed in Table 6.1-1 are subsequently described and assessed, together with the auxiliary and supply facilities required for their operation as well as their I & C systems important to safety and their emergency power supply.

In addition to the safety systems, there are safety-relevant operational facilities which fulfil safety functions under certain boundary conditions. To these belong the make-up system, the startup and shutdown system, the main steam bypass stations BRU-K and the pool cooling system of the spent-fuel pool. They are incorporated into the following investigations.

Further safety-relevant operational facilities are listed below, which, predominantly because of inadequate information, could not be subjected to the current safety assessment:

- service water system B
- hydrogen-retarded-combustion (off-gas system)
- spent-fuel pool
- ventilation in the rooms of the engineered safeguards

## 6.2 Shutdown Systems

### 6.2.1 General Safety Requirements to be met by Shutdown Systems of Lightwater Reactors

The following facilities are included in the plant concept which can be employed for shutting down the reactor.
Containing reactivity by dropping control elements into the core:

- reactor scram system as a part of the control and protection system

Containing reactivity by injection of soluble boron compounds in the coolant (poisoning systems):

- HP-emergency boron injection system
- make-up system.

- **Assessment Criteria**

The functions and requirements to be met by shutdown systems of lightwater reactors are summarised in KTA-Rules 3101.2 and 3103. According to these rules it is the function of the shutdown systems to reduce the reactor to the sub-critical, zero power state and to keep it permanently subcritical in the most adverse condition.

For the transfer to the sub-critical state, two different systems which are independent of each other are required. The reactor scram system must be able to render the reactor subcritical, sufficiently fast and on its own, starting out from all states of the plant to be assumed and to keep it sub-critical for a sufficiently long period, while the further shutdown system serves the purpose of rendering the reactor permanently subcritical, from all states of normal operation, through the most adverse states.

The reactor scram system is part of the safety system. The independent further shutdown system (boron injection) is then only a part of the safety system, if the reactor scram system alone does not fulfil the function of maintaining permanent sub-criticality.

- **Assessment**

The reactor scram system in Section 4.1.2 was assessed with respect to its effectiveness in transfering the core into the cold, xenon-free, sub-critical state. The examination resulted in the necessity of providing a poisoning system (boron injection system) as a supplement for this function. The three-train HP-boron injection system is best suited for this purpose.

According to the German codes and standards this leads to the following shutdown concept:

- safety system: reactor scram system and HP-emergency injection system
- second independent shutdown system: HP-boron injection system and/or make-up system

Which of the two systems is to be considered the second, independent shutdown system can only be determined after further analyses of their effectiveness have been presented. In the following safety assessment, it is provisionally assumed that the make-up system is at least part of the second shutdown system.

### 6.2.2 Reactor Scram System (Control Elements)

- **Description of the System**

The control elements with their instrumentation and control, the emergency protection system, serve as the reactor scram system. They are part of the control and protection system. Their effectiveness is described and assessed in the section dealing with the core design (Section 4.1). The realisation of emergency protection is discussed in Sections 6.4 and 6.5, the mechanical design of the control elements in Section 4.1.6.

The control elements of the control and protection system belong to the reactor construction. There are 61 control elements (rod cluster control assemblies), each consisting of 18 rods containing absorber material. The control elements are also used for operational reactor control. For this purpose they are moved by an electromagnetic step drive. In response to the emergency protection actuation or failure of the power supply, the electromagnetic step drive mechanisms are de-energised and the control elements drop into the reactor core. Their dropping time from the upper end layer into the lower end layer according to /TEP 81/ is 1.5 to 4 s. The main elements of the drives are the drive cover, the outer part of the drive (magnet arrangement), the inner part of the drive, the spacer (connection between drive and control element) and the position indicator.

- **Assessment Criteria**

According to Section 6.1.3 (Table 6.1-1) the reactor scram system is required for almost all accidents to be assumed for the plant. Accordingly, it has to be designed accident-resistant, emergency-power supplied and resistant to external impacts.

It is part of the safety system and therefore has to met the requirements listed in Section 6.1.1.

Further essential technical requirements of the system result from KTA Rule 3103:

- The reactor scram system must fulfil its safety-related technical function, even when a single failure occurs.
- Operational controls must not impair the safety-related technical function of the reactor scram system.
- A position indication and a final position sensor is to be provided for every control element.
- An uninterrupted emergency power supply is to be provided for the power supply of the indicators.
- Functional safety of the reactor scram system is to be demonstrated.

- **Assessment**

The conceptual design of the reactor scram system was planned in accordance with the requirements of item 2.3.1 of /OPB 82/. These requirements largely correspond to the Federal German codes and standards. For this reason there are no objections against the concept of the reactor scram system described in the Technical Project /TEP 81/.

There are several years' operating experience for the WWER-1000/W-320, which will be dealt with in Section 8.3.1, the assessment of which will lead to the recommendation of up-grading measures.

In the Technical Project /TEP 81/, item 3.2.3.2, testing of the system and its elements by means of test patterns was proposed. The degree of detail in the present documents is not sufficient for conducting a reliable evaluation of the system by means of the above yardsticks. A detailed description of the system is to be presented, as well as an assessment of the test results, in combination with the operating experience, with regard to the wide range of the reported dropping times (R 6.2-1).

### 6.2.3 HP-Emergency Boron Injection System

- **Description of the System**

The HP-emergency boron injection system (see Fig. 6.3-3) is part of the safety system of the reactor and supplements the reactor scram system for ensuring sub-criticality of the reactor in the cold, xenon-free state. Its effectiveness is discussed in Section 4.2.

The HP-emergency boron injection system is designed as a 3 x 100 %-system. The three trains are arranged physically separated, emergency power supplied and resistant to external impacts. Each train is equipped with a storage tank (15 $m^3$ useful volume) for concentrated boric acid solution (40 g/l) as well as a HP-emergency boron injection pump which is cooled by service water A. The nominal delivery rate is 6 $m^3$/h with a delivery head of 16 MPa; the maximum delivery head is 19.1 MPa with a delivery rate of 1.6 $m^3$/h. The injection line of every HP-emergency boron injection train feeds into the unisolatable part of the injection line of the respective HP-emergency cooling pump via check valves. On the suction side there is no connection to other systems. The pumps and tanks are arranged in the reactor building below the containment. At the storage tank there are nozzles for temperature, concentration and level measurements. Upon detection of an accident criterion (see Section 6.4, Table 6.4-1), the system changes to recirculation mode (start of pumps, feeding into storage tank). Switch-over to injection is initiated manually by opening the injection valves and closing the recirculation valve, in accordance with the special regulations relating to the slow control of reactivity.

- **Assessment Criteria**

According to Section 6.3.1 (Table 6.1-2) the HP-emergency boron injection system is required either early or in the long-term for controlling almost any accident to be assumed for the plant. Accordingly it has to be designed accident-resistant, emergency-power supplied and resistant to external impacts. It is part of the safety system and therefore has to meet the requirements mentioned in Section 6.1.1.

Further essential technical requirements result from KTA Rule 3103:

- For all plant-internal accidents and for earthquakes, single failure with simultaneous loss of off-site power and simultaneous non-availability of a redundancy owing to maintenance measures is to be assumed.
- For airplane crash and explosion blast wave, apart from the initiating event only loss of off-site power is assumed additionally.
- The time of the application of the system, as well as the decision whether the application may be initiated manually or whether it must be started automatically, are to be determined by analysing sequences of events.
- To ensure readiness for operation, only few active switching operations shall be necessary for operational activation.
- The boron concentration and the level in the boric acid tanks, as well as the position of the valves, are to be monitored.
- Functional safety of the system is to be demonstrated.

- **Assessment**

The HP-emergency boron injection system fulfils the requirements to be met by an engineered safeguard with respect to single failure, layout, plant construction, emergency power supply, instrumentation and design against earthquake and explosion blast wave.

It is located outside the containment in a sector of the reactor building, which ensures protection against loads resulting from airplane crash by physical separation only. It therefore is to be demonstrated that the HP-emergency boron injection system will not be unacceptably damaged by an airplane crash, as a consequence of induced vibrations (R 2.7-1).

The functional safety of the system cannot be assessed, as the respective documents, in particular those referring to previous operating experience, are not available (R. 6.2-2)

To control certain accidents (see Table 6.1-1), pressuriser spray operation is necessary. At present this function can only be fulfilled by the make-up system, which does not correspond to the requirements to be met by engineered safeguards. It is recommended to establish a connection from the HP-emergency boron injection system to the spray line of the pressuriser (R 6.2-3).

To meet the requirements of the HP-emergency boron injection system in accordance with Table 6.1-1, an integrated concept for using the system for accident control is to be developed (R 6.2-4) which also comprises the necessary automation of the system (see also Section 5.1).

### 6.2.4 Make-Up System

- **Description of the System**

The make-up system is a safety-relevant operational facility with various functions. Within the framework of the shutdown system according to Section 6.2.1, it is assigned the function of the second independent shutdown system. Otherwise, the most important functions of this system during power operation of the reactor are compensating operational leakages, compensating reactivity changes, ensuring the water quality of the coolant by injecting reagents and supplying seal water for the main coolant pumps, as well as compensating changes in volume during startup and shutdown of the unit. Furthermore, the pressuriser with the main coolant pumps out of operation, can be cooled down with the help of the make-up pumps, the core flooding tanks can be filled and re-used and the leaktightness of the primary system can be controlled. The prevention of an uncontrolled injection of clean condensate into the reactor, the functioning of the system in all cases of normal operation and the possibility of an additional injection of boric acid solution into the primary system in transient accidents are part of the design requirements (Fig. 6.3-6).

Under certain boundary conditions during the "steam generator tube rupture" accident the make-up system can take the safety function of "coolant supplementation" and "effecting sub-criticality of the reactor". The pressure compensation between primary and secondary system necessary for controlling this accident is achieved by pressuriser spray via the make-up lines.

The system fulfils the functions of degasing/boric acid control, charging and letdown of primary coolant. The main components for the function "degasing and boric acid control" are two degasers with collecting vessel having a content of 19 $m^3$ each (boric acid solution or clean condensate respectively) and a maximum flow of 65 t/h. The main components for the "charging" function are the three make-up units. Each make-up unit comprises a booster pump (delivery rate 110 $m^3/h$ at 0.48 MPa) and a subsequent injection pump (delivery rate 10 to 60 $m^3/h$ at 17.7 MPa), as well as the thrust bearing coolers of the pumps. The units are physically separated. Two of the three units in case of loss of off-site power, are supplied by the independent 4th and 5th diesel units, which are not resistant to external impacts. The main component for the "letdown" function is a cooler located in the containment downstream of the regenerative heat exchanger, which is cooled by the component cooling system of the reactor building.

Upon detection of the criteria for a leak in the primary system (see Section 6.4, Table 6.4-1) boundary valves in the letdown as well as in the charging lines close. Thus, the system can no longer be used without manual action in such cases.

- **Assessment Criteria**

As a shutdown system the make-up system is subject to the requirements of KTA Rule 3103.

- **Assessment**

The make-up system is an operational facility and can be considered as a part of the second shutdown system according to Section 6.2.1. This double function according to KTA Rule 3103, Item 4.1 (4) is explicitly permitted. There it reads: "Die Abschaltsysteme können ganz oder teilweise zur betrieblichen Steuerung herangezogen werden". (The shutdown systems can entirely or partially be used for operational control). The assessment of the effectiveness of the make-up system as a shutdown system is dealt with in Section 4.1.

With respect to its system-related design, the make-up system basically meets the requirements for an operational facility, an additional shutdown system and a facility for compensating operational or small leakages and changes of volume in the primary system. For better control of the "steam generator tube rupture" accident the make-up system is to be upgraded, as a short-term measure, for example, by automating the pressuriser spray function via the make-up system (R 6.2-5) and as a long-term measure, fulfilling this function via the HP-emergency boron injection system (see Section 6.2.3, R 6.2-3).

## 6.3 Engineered Safeguards and Safety-Relevant Operational Facilities of the Primary System, of the Secondary System and of the Containment

### 6.3.1 Residual Heat Removal Systems

#### 6.3.1.1 Safety-Related Requirements to be met by the Residual Heat Removal Systems

The following engineered safeguards are provided for residual heat removal:

- emergency cooling system
- containment-spray system
- emergency feedwater supply system
- relief valves (BRU-A) and steam generator safety valves in the main steam system
- service water system A.

The residual heat removal systems are part of the safety system and therefore have to meet the requirements listed in Section 6.1.1.

For the systems analysis, Section 6.1.3 (Table 6.1-1) states which of the above engineered safeguards are required for the Stendal NPP to control the individual accidents assumed. This defines the requirements to be met by its accident-related design. These are in particular an accident-resistant design, emergency power supply and design against loads resulting from external impacts. Special accident assumptions are to be taken into account for the design of the pumps, accumulators, heat exchangers and the safety and relief valves for the primary and the secondary system and for dimensioning the coolant reserves. Further detailed requirements to be met by the design of residual heat removal systems are contained in KTA Rule 3301. In the analysis of the functions of the system the possible interactions between operational and safety-related functions as well as between different safety-related functions are to be considered.

#### 6.3.1.2 Combined Efforts of Engineered Safeguards and Safety-Relevant Operational Facilities to Remove Residual Heat in the Stendal NPP upon Request

During power operation, small losses of coolant, eg leakages, sampling, etc., are compensated by the make-up system (volume control system). For losses of coolant which can no longer be compensated by the make-up system, the emergency cooling system (emergency core cooling and residual heat removal system) automatically start operation. For larger losses of coolant the containment-spray system (sprinkler system) also responds, serving to decrease the pressure in the containment and support residual heat removal. The emergency core cooling system and the containment-spray system have a common sump and must be considered together in dimensioning the coolant reserves.

The residual heat from the reactor in the first cooldown phase is dissipated via the secondary system. The auxiliary or emergency feedwater supply system on the feedwater side and the main steam bypass station (BRU-K), the relief valves (BRU-A), or for a short period also the steam generator safety valves, on the main steam side serve this purpose. In the long-term cooldown operation of the unit the residual heat of the reactor in the subcritical hot state is dissipated via the cool down station (BRU-SN) located on the secondary side and in the sub-critical cold state via the primary-side LP-emergency core cooling system. The service water system A is required for residual heat removal from the primary system via the emergency cooler and for cooling safety-relevant components.

The emergency cooling system of the Stendal NPP and the emergency cooling and residual heat removal system of a Federal German PWR-plant are illustrated schematically in Fig. 6.3-2 and 6.3-1, respectively. The essential difference between the two concepts is that at Stendal NPP a containment-spray system is required for accident control and the component cooling system is not required in the residual-heat-removal chain.

### 6.3.1.3 Emergency cooling System (Emergency Core Cooling and Residual Heat Removal System)

- **Description of the System**

The HP-emergency core cooling, the core flooding and the LP-emergency core cooling system (Fig. 6.3-3) belong to the emergency cooling system. The common emergency boron tank with the three sump intakes is assigned to the LP-emergency core cooling system. The predominant functions of the emergency cooling system are to ensure cooling of the reactor and make-up of the primary coolant in loss-of-coolant accidents as well as long-term residual heat removal. It furthermore supports the shutdown systems in ensuring sub-criticality.

- HP-Emergency Core Cooling System

The HP-emergency core cooling system is designed as a three-train system which is resistant to external impacts. Each train is physically separated and emergency power supplied. Each train has a storage tank (useful volume 15 $m^3$) for concentrated boric acid solution (40 g/l) which is located in the containment. Electrical heating is used to maintain the temperature at 55 °C to 60 °C. The HP-emergency core cooling pumps (delivery head 10.8 MPa, nominal delivery rate per pump 160 $m^3/h$) are located below the storage tanks for concentrated boric acid, outside the containment in the reactor building. The pumps are cooled by the service water system A. The injection lines of the HP-emergency core cooling system lead to the pressure-side of the main coolant line (loops 1, 3, 4). During normal operation of the reactor plant the injection valves in the primary system and the valves in the recirculation lines leading to the storage tank are closed. During a loss-of-coolant accident all three HP-emergency injection pumps start operation, the injection valves open immediately and the recirculation valves open after a time delay. Once the delivery rate of the HP-emergency core cooling pumps exceeds 80 $m^3/h$, the recirculation valves close. When the storage tanks of the HP-emergency core cooling train are empty, the HP-emergency core cooling pumps take suction from the emergency boron tank. In the case of the main steam line break, as the main steam valve (SSA) closes, the HP-emergency core cooling system is activated automatically to compensate for the volume contraction and to ensure sub-criticality of the reactor.

- Core Flooding System

The core flooding system is the passive part of the emergency cooling system. The main components are the four core flooding tanks (accumulators), two of which inject on the hot side and two on the cold side. The four trains are arranged in pairs and physically separated in the containment. Each tank has a nominal volume of 60 $m^3$ and is filled with 50 $m^3$ boric acid solution (concentration 16 g/l) which is kept at a temperature of 55 °C by electrical heating. The nitrogen blanket in each core flooding tank has a maximum pressure of 5.89 MPa. In the injection lines DN 300 to the reactor there are two fast-acting valves arranged in series which, by an interlock at a low level in the core flooding tank, provide a gas-tight isolation of the tanks from the primary system, as well as two check valves arranged in series. In each core flooding tank, pressure is safeguarded by two safety valves. The injection valves leading into the reactor are open during normal operation.

If, during an accident, the pressure in the primary system drops below 5.89 MPa, boric acid solution will be injected into the reactor.

- LP-Emergency Core Cooling System

There is only one emergency boron tank (630 $m^3$ boric acid solution, boric acid concentration 16 g/l) for the entire emergency cooling system. It is arranged as the deepest room of the containment and has three sump intakes.The emergency boron tank is double-walled and has a leakage control system. Three outlet lines lead from the tank to the emergency core coolers (residual heat cooler). They are executed as single tubes and in a distance of about 12 m from the emergency boron tank have one isolating valve each.

The LP-emergency core cooling system is a three-train system which is resistant to external impacts. The trains are arranged physically separated and are emergency power supplied. Each train has an emergency core cooler (effective heating surface 790 $m^3$, coolant intake/outlet temperature 150 °C/60 °C) and a LP-emergency core cooling pump (delivery head 2.25 MPa, nominal delivery rate 750 $m^3$/h). The emergency core coolers are positioned below the emergency boron tank outside the containment. To prevent an inadmissible cooldown of the emergency cooling medium by the emergency cooler (danger of brittle fracture within the reactor vessel upon cold water injection), a part of the emergency cooling medium can be led around the emergency cooler via a bypass route. The LP-emergency cooling pumps, at a primary coolant pressure < 2.2 MPa, injects into the primary system. At a primary coolant pressure > 2.2 MPa in the primary system, the boric acid solution is returned to the pump suction side via a recirculation line. The emergency cooler and the LP-emergency cooling pump are cooled by the service water system A. The injection lines of the LP-emergency cooling trains dicharge into the lines from the core flooding tank (trains 2 and 3 each into the lines injecting above and below the reactor core) or into the hot and cold legs of loop 1 (train 1).

The emergency cooler and LP-emergency cooling pumps, at low primary system parameters (shutdown operation to the cold shut down condition), are used for long-term residual heat removal. For this purpose there is a connecting line (cooldown line) between the cold leg of loop 4 and the suction line of every LP-emergency cooling train.

During reactor operation the injection valves of the LP-emergency cooling system are closed and, following appropriate accidents, are opened via interlocks. In case of a leak in the primary system, the LP-emergency cooling system has the function of ensuring residual heat removal and sub-criticality of the reactor at low primary system parameters.

- **Assessment Criteria**

The essential assessment criteria are described in Section 6.3.1.1. Further, it is required by the BMI Safety Criteria, criteria 4.2 and 4.3, Residual Heat Removal after Loss-of-Coolant Accidents, that the injection capacity of the emergency cooling system suffices for accident control during maintenance work in one redundancy and simultaneous single failure in a further redundancy. The RSK Guidelines also, in Section 22, Systems for Heat Removal after Accidents, require ensurance of sub-criticality during long-term operation, redundant, unmeshed trains (common components possible under preset conditions), investigation of the effects of leakages in the emergency coolers and steam generators (sub-criticality and water reserves), protection against the consequences of accidents, HP-emergency coolant injection during sump operation, etc. For emergency cooling calculations the water carried to the break point may not be considered for core cooling.

- **Assessment**

An examination was made of the extent to which the concept of the emergency cooling system meets the requirements of the Federal German codes and standards. The following positive characteristics were found:

- The emergency cooling system during normal operation of the plant does not have a functional connection to the pool cooling system.
- The HP-emergency cooling system is designed in such a way that the HP-emergency pumps can draw from the common emergency boron tank, after the tanks assigned to the train have been emptied.
- The boric acid solution in all storage tanks of the emergency cooling system is continually preheated to 55 °C to 60 °C, which reduces the threat of brittle fracture of the reactor pressure vessel.

The following essential weaknesses and differences compared to the Federal German codes and standards were detected in the course of the analysis:

- The degree of redundancy of the emergency cooling system is not sufficient for all system and accident conditions. During a single failure with an assumed additional repair case, sufficient core cooling cannot be ensured for an unfavourable position of the break (feeding to the leak) (see Section 5.1.2.1, recommendation R 5.1-3).
- In KTA Rule 3301, Item 4.4, Dimensioning of Coolant Reserves, the dimensioning of the coolant reserves is required under consideration of failure assumptions and redundancy requirements. After the train-wise assigned 15 $m^3$ tanks of the HP-emergency core cooling system have been emptied, only the common emergency boron tank with its three sump outlet to all three trains of the emergency cooling system is available for the emergency cooling system and the containment-spray system. A part of the water which reaches the containment during a leakage accident remains in the containment and is no longer available for circulation. The quantity lost is dependent on the size and the location of the leak. According to /MRE 92/

there are measurements on the quantities lost during startup tests of WWER-1000 units. They amount to about 80 $m^3$. This value is to be verified and in addition it has to be demonstrated that there are sufficient quantities of water in the emergency boron tank during all phases of accidents (R 6.3-1).

- According to KTA Rule 3301, Item 6.2.2, special constructional requirements are to be met by the containment sump (intake, retention of insulating material, intake line). Owing to the lack of documents, the operativeness and efficiency of the sump cover and the respective filter devices are to be demonstrated according to KTA-Rule 3301, Item 6.2.2 (R 6.3-2).

- The single failure can be excluded for the emergency boron tank as a passive component, if it meets Item 5.2.2.2 of KTA-Rule 3301 (special requirements to be met by the material, for example). The same applies to the connecting lines to the tank, i.e. the suction lines of the trains of the emergency cooling system 3 x DN 600, the feed lines for special water treatment SWA IV, 3 x DN 100, as well as the flow pipe DN 150 for tank heating. Leaktightness of the tank as well as of the connecting lines are monitored by measuring the levels in the sump outlet lines to the emergency boron tank. The pressure is not checked. Basic safety in accordance with KTA Rule 3301, Item 5.2.2.2, has to be demonstrated for the connecting lines to the emergency boron tank and the tank itself. According to KTA Rule 3301, Item 6.2.2.3, a loss of water with a simultaneous loss of the containment function with respect to the retention of activity under accident conditions could thus be excluded. Even if the basic safety of the connecting lines to the emergency boron tank were given, it would be recommended in accordance with the state of the art and RSK Guideline 22.1.2 (7) that double tubes with leakage detection should be provided between the tank and the isolating valve. It is further recommended to position the isolating valve as close as possible to the emergency boron tank (R 6.3-3). Because of the safety-related importance of the isolation for maintaining the emergency cooling water reserves, separate evidence on the reliability of the isolating valve is recommended.

- The emergency coolers are cooled by the service water system A which dissipates its heat via the spray ponds. The pressure conditions between the two media in the emergency cooler during an accident are not clearly stated. The contamination danger for the emergency cooling system, as well as a dilution of the boric acid solution by overflow from the service water system A into the suction line of the emergency cooling pumps, with a release of activity into the environment cannot be excluded. It is therefore recommended to install a component cooling system in the residual heat removal chain (R 6.3-4).

- According to KTA Rule 3301, Item 5.4.1, it is requested that the injection lines of the emergency cooling system have automatic isolating devices, which are connected in series and the tightness of which can be examined. This means that the locked injection valves of the HP- and LP-emergency core cooling system are to be kept in the "open" position during normal operation (R 6.3-5) and that leaktightness of the check valves in the injection lines of the emergency cooling systems must be monitored (R 6.3-6).

- In case of smaller leaks in the primary system, a longer recirculation operation is to be expected. This leads to a heat-up of the circulating water. It is to be examined whether the installation of a cooler in the recirculation system is required, to keep within the design temperature of the HP-emergency cooling pump (R 6.3-23).

- Parts of the emergency cooling system are located in the reactor building outside the containment where a protection against loads from airplane crash is only ensured by physical separation. It therefore has to be demonstrated that the respective parts of the emergency cooling system are not unacceptably damaged by the vibrations induced by an airplane crash (R 2.7-2).

- Section 8.3.3 reports on breakdowns or malfunctions in the emergency cooling system and corresponding upgrading measures are formulated. According to BMI Criterion 1.1, that in principle only reliable components shall be employed, it is recommended to perform a systematic examination of the operational reliability of all pumps of the emergency cooling system and the containment spray system (R 6.3-7).

#### 6.3.1.4 Containment-Spray System (Sprinkler System)

- **Description of the System**

The containment-spray system (Fig. 6.3-3) is needed to control leakage accidents of the primary and secondary system within the containment. During the accident it has the function of decreasing pressure in the containment as fast as possible, to largely bind the fission products in an aqueous solution upon condensation of the steam atmosphere and to discharge a part of the residual heat as well as ensuring emergency filling of the spent-fuel pool via a connecting line to the pool cooling system during accidents.

The containment-spray system is a three-train system resistant to external impacts. The three trains are physically separated and emergency power supplied. They use the common emergency boron tank (sump) of the emergency cooling system as a water source. Each train has a containment-spray pump (delivery head 1.5 to 0.75 MPa, delivery rate 210 to 975 $m^3/h$) cooled by service water system A, a water jet pump (ejector), a chemical tank (volume 6 $m^3$, diamide hydrate) and a containment distribution ring with sprinkler nozzles.

The containment-spray system is a stand-by system. In case of accident, the containment-spray pumps are activated automatically and the valves in the pressurised line of the containment spray pump and the chemical solution intake line also open automatically.

- **Assessment Criteria**

The installation of a containment-spray system is not required by the Federal German codes and standards. Therefore there are increased requirements to be met by the emergency cooling system and the secondary-side cooldown (100 K/h shutdown). In the Stendal NPP the containment-spray system is, however, required for observing the critical containment pressure in the long-term follwoing a design accident, to discharge the energy content of the steam generator of the systems still intact during a loss-of-coolant accident. Contrary to the Konvoi plants, in the WWER-1000 the energy content is brought into the containment by the LP-emergency core cooling system, as the horizontal steam generators are positioned at almost the same height as the inlet and outlet nozzles of the reactor pressure vessel and the emergency cooling water flows through them.. From its plant concept, the containment-spray system at the Stendal NPP therefore belongs to the safety system. It thus has to meet the respective requirements according to Section 6.3.1.1. In accordance with its functions (also cf. Section 6.1.3, Table 6.1-1) it must be designed to be accident-resistant, emergency power supplied and furthermore, to ensure long-term residual heat removal and cooling of the spent-fuel pool, it must be resistant to external impacts.

- **Assessment**

The system largely corresponds to the requirements to be met by an engineered safeguard. The problems relating to the function of the emergency boron tank illustrated in Section 6.3.1.3 do, however, also concern the effectiveness of the containment-spray system. Furthermore, verifications of the effectiveness of the sprinkler nozzles for all accident conditions, including design pressure of the containment, are missing (R. 6.3-8). A technical solution must be provided for periodic function tests of the containment-spray system up to the last check valve during power operation of the unit and test cycles for the sprinkler nozzles must be determined (R 6.3-9).

Parts of the containment-spray system are located in the reactor building outside the containment where protection against the loads induced by airplane crash is only ensured by physical separation. It therefore must be demonstrated that the respective parts of the containment-spray system are not excessively damaged by vibrations induced by an airplane crash (R 2.7-2).

#### 6.3.1.5 Auxiliary Feedwater System

- **Description of the System**

The auxiliary feedwater system (Fig. 6.3-4) is a safety-relevant operational facility. It consists of two electrically driven injection pumps (delivery rate 200 $m^3$/h, delivery pressure 9.37 MPa) which are each connected to one feedwater tank (volume 210 $m^3$, water content 185 $m^3$, pressure 0.588 MPa, temperature 164 °C) of the main feedwater system and the four steam generators. They are activated automatically upon protective shutdown of the turbine feedwater pumps and reduction of the water level of any steam generator to 220 mm below the normal filling level. The motors of the auxiliary feedwater pumps are supplied from the fourth and fifth unit diesels on loss of off-site power.

To shutdown the unit after reactor scram the two auxiliary feedwater pumps are needed for about ten minutes. Following loss of off-site power, the steam generators are fed with warm water by the auxiliary feedwater pumps, which avoids a thermal shock effect on the steam generator.

- **Assessment Criteria**

There are no separate safety-related requirements to be met by safety-relevant operational facilities.

- **Assessment**

The auxiliary feedwater system serves the residual heat function removal after reactor scram. It's design is neither redundant nor resistant to external impacts, but it is emergency power supplied by the fourth and fifth unit diesels. Except during accidents with external impacts, the auxiliary feedwater system thus is an installation which comes before the emergency feedwater supply system. By feeding the steam generators with warm water from the auxiliary feedwater system after reactor scram, consequential failures owing to thermal shock are prevented.

#### 6.3.1.6 Emergency Feedwater Supply System

- **Description of the System**

The emergency feedwater supply system (Fig. 6.3-4) is an engineered safeguard with a pump capacity of 3 x 100 %. It is resistant to external impacts and emergency power supplied. It is located in the reactor building and each train consists of an emergency feedwater tank (500 $m^3$), an emergency feedwater pump (nominal delivery rate 150 $m^3/h$; delivery head 9.56 MPa) with a minimum flow line, feedwater control units upstream of the steam generators and connecting pipes. The water reserves of an emergency feedwater tank last for about five hours for residual heat removal /MRE 92/. Train 1 of the emergency feedwater supply system can feed to steam generators 2 and 4, or after switch-over to steam generators 1 and 3. Train 2 feeds to steam generators 1 and 4 and train 3 to steam generators 2 and 3. The emergency feedwater tanks of all three trains are arranged in one room in the containment and they are interconnected.

The emergency feedwater supply system is initiated automatically on detection of the leakage criteria for the primary and secondary system, as well as for loss of off-site power. When the filling level in the individual steam generators falls below a fixed limit, the minimum flow lines of the respective emergency feedwater pumps are closed and the steam generators are fed with emergency feedwater which has not been preheated separately. The service water system A (pump and motor cooling), the deionised water system (re-filling the emergency feedwater tank after ten hours at the earliest, depending on the course of the accident and considering a single failure), the emergency power supply and instrumentation and control are required to operate the emergency feedwater supply system.

- **Assessment Criteria**

The essential requirements to be met by the emergency feedwater supply system are included in Section 6.3.1.1. In KTA Rule 330,1 feeding of the steam generators must be ensured during failure of the operational feedwater supply and during accidents with and without losses of coolant corresponding to the respective operational, plant and system states.The requirement to be met by the emergency feedwater supply system as a part of the emergency standby system is summarised in RSK Guideline 22.2. In this guideline an accident-resistant execution, emergency power supply and design against loads resulting from external impacts are required.

- **Assessment**

The analysis showed that the emergency feedwater supply system largely corresponds to the requirements of the Federal German codes and standards with respect to system design. There are differences in the physical arrangement of the emergency feedwater tanks (all three tanks are located in one room) and the partial meshing of the injection lines. Considering a single failure, the water reserves of the emergency feedwater tanks, without re-filling, last for about ten hours. Feeding to the leak and the repair case are not considered here. The requirements of the emergency feedwater supply system as an emergency standby system thus are not met (also see Section 6.3.1.9). If evidence cannot be provided that the operability of the remaining system is not impaired by a leak in one emergency feedwater tank, these tanks will have to be physically separated by train (R 6.3-10). Parts of the emergency feedwater supply system are arranged in the reactor building outside the containment, where protection against loads resulting from airplane crash can only be ensured by physical separation. It is to be demonstrated that the respective parts of the emergency feedwater supply system are not excessively damaged by vibrations induced by airplane crash (R 2.7-2).

### 6.3.1.7 Main-Steam System with Relief Valves (BRU-A), Steam Generator Safety Valves and Main-Steam Isolating Valves (SSA)

- **Description of the System**

The main steam system (Fig. 6.3-4) is an operational facility (main steam pressure 5.9 MPa, main steam temperature 274 °C) which also has to fulfil safety-related functions. The main steam system comprises four steam generators, the main steam lines, the steam bypass station BRU-K and the cool-down system BRU-SN. As engineered safeguards it contains the relief valves (BRU-A), the steam generator safety valves and the main steam isolating valves (SSA).

During normal operation of the unit the steam generators supply main steam to the turbine. During startup and shutdown of the unit excess pressure in the steam generators is led to the turbine condenser via the bypass station BRU-K. The bypass station BRU-K is a safety-relevant operational facility. With the exception of external impacts it is the normal facility for residual heat removal. It is a necessary component during load rejection of the turbine to the auxiliary power supply level. It is not emergency power supplied. A short time after reactor scram, the auxiliary power reducing station BRU-SN can dissipate the steam of the steam generators via the technological condensers. Upon failure of the operational facilities for residual heat removal the engineered safeguards come into operation.

The relief valves BRU-A and steam generator safety valves belong to the engineered safeguards of the main steam system for ensuring residual heat removal.

- Relief Valves BRU-A

There are four relief valves BRU-A (opening pressure 7.26 MPa, closing pressure 6.28 Mpa, flow rate 4 x 900 t/h). They are emergency power supplied and resistant to external impacts. After failure of the bypass station BRU-K and the auxiliary power reducing station BRU-SN they relieve excess steam to the atmosphere and so reduce pressure. They can be controlled automatically or manually. In the automatic position, the cool down process is controlled by cool down rate. The BRU-A do not have any isolating valves positioned upstream of them which can be closed if a BRU-A valve is stuck open.

- Safety Valves of the Steam Generators

Each steam generator has two safety valves in the outlet main steam line (opening pressure 8.34 MPa, closing pressure 6.97 MPa, flow rate 900 t/h each). They are designed to be resistant to external impacts. They also can be operated manually from the main control room.One safety valve of each steam generator is assigned to a given train of the emergency power supply. The third emergency power train, not used for the safety valves of a steam generator is used for the relief valve BRU-A of that steam generator.

- Main Steam Isolating Valves (SSA)

The fast-acting main steam isolating valves (SSA) are located in the main steam lines downstream of the steam generators. They are emergency power supplied and resistant to external impacts. They are activated by the criteria "negative pressure change rate in the steam generator high" and "difference in the saturation temperatures between primary and secondary system high" to isolate the steam generators in case of a leak in the main steam system.

- **Assessment Criteria**

The essential assessment criteria are described in Section 6.3.1.1. A summary of KTA Rule 3301 states that relief of the steam being generated in the steam generator must be ensured for the respective operational, plant and system states upon failure of the main heat sink during accidents with or without losses of coolant and during external impacts. A detailed list of the requirements to be met by the relief valves BRU-A or the steam generator safety valves for accident control is contained in Section 6.1.3 (Table 6.1-1).

- **Assessment**

The concept analysis showed that the safety-related facilities of the main steam system in principle correspond to the Federal German codes and standards relating to system design. After a reactor scram the residual heat is normally removed by the normal operational facilities. During external impacts and upon loss of off-site power, only the relief valves BRU-A and the safety valves of the steam generators are available for residual heat removal. By the separate use of all three emergency power trains for the two steam generator safety valves and the relief valve BRU-A of a steam generator, an independent 3 x 100 % design for the relief of steam from every steam generator was achieved for accidents.

The engineered safeguards of the main steam system together with the main steam and feedwater lines are located in room A 820 of the surrounding outer building (see Fig. 2.1-2) of the containment (height 29 m) without physical separation. In case of pipe failures, consequential failures cannot be excluded. The basic safety of pipes and components must therefore be demonstrated (R 6.3-11). Room A 820 is insufficiently protected against external impacts. If the basic safety of the main steam and feedwater lines cannot be demonstrated, Room A 820 will have to be backfitted accordingly and the systems will have to be separated physically (R 6.3-12). Erroneous opening of the relief valves BRU-A, or their failure to close after opening normally can become an initiating event, as they cannot be isolated. An isolating valve upstream of every relief valve BRU-A is missing (R 6.3-13).

#### 6.3.1.8 Service Water System A and Component Cooling System of the Reactor Building

- **Description of the System**

It is the function of the system to safely dissipate the heat occuring in the safety-relevant cooling positions under all operational conditions of the reactor. To fulfil this function, especially during accidents, the service water system NKW-A, in common with the engineered safeguards to be supplied, is designed in three trains. The trains are largely physically separated from each other and resistant to external impacts. An exception is the cross-over point of the flow lines of two trains in the outer section of multi-unit plants (Fig. 6.3-7).

Each train of the system, in flowpath order, consists of:

- A spray pond with a surface of about 5000 $m^2$ (dimensions 71 m x 75 m) and a water volume of at least 2810 $m^3$ and 8776 $m^3$ at the most,

- two siphon pipes, which are designed for a performance of 100 % each, with the corresponding water jet pumps,
- a drawing-off structure, with primary cleaning systems and adjacent underground gradient line to the
- emergency power building with
  - a pump suction chamber with secondary cleaning systems,
  - two service water pumps (one in operation, one stand-by pump) with a nominal delivery rate of 3600 $m^3$/h at a delivery head of 0.5 MPa
  - emergency power diesels, compressor system and air-conditioning as consumers,
- consumers in the reactor building (emergency cooler, cooler of the component cooling system for systems withing the reactor building, pool cooler, cooler for the make-up system, ventilation systems, emergency cooling pumps, containment-spray pumps, emergency feedwater pumps),
- a valve structure with valves for the automatic switch-over from pool to spray mode, depending on the cold water temperature,
- 30 single nozzles with a flow rate of about 100 $m^3$/h each which spray into the spray pond.

Between the drawing-off structure and the emergency power building, there are the underground suction and pressure lines of the service water system A. Train 2 is routed north of the turbine hall and trains 1 and 3 are routed a distance of 40 m south of the reactor building. These lines are designed as manifolds for several units. For this reason the connecting lines from the manifolds of train 1 to the emergency power building unit A cross the manifolds of train 3 to the other units.

The three trains, even during power operation of the reactor, are constantly in operation to cool the consumers required for this mode of operation. The evaporation and spray losses of the tanks are compensated by an additional water supply with a maximum of 225 $m^3$/h treated water from the River Havel. Upon failure of this supply, an emergency supply from the River Elbe is possible. To prevent the cold water temperature falling below 5°C in winter, heating was installed between the drawing-off structure and the pump station which is fed by the return flow from the heating network with $t_R$ = 70 °C. The electrical requirements of the system are emergency power supplied.

The component cooling system in the reactor building only serves the cooling of consumers of non-safety-related systems carrying radioactive media. It is cooled by trains 1 and 2 of the service water system A.

- **Assessment Criteria**

The service water system A is part of the residual heat removal system and has to meet the requirements stated in Section 6.3.1.1. Accordingly, the trains are to be arranged physically separated, they are to be supplied with emergency power and they are to be designed resistant to external impacts.

For the cooling points of the engineered safeguards according to KTA Rule 3301, two acitivity barriers are required. As the first barrier a passive component (heat exchanger), as the second barrier a second passive component or a corresponding pressure differential can be provided. According to KTA Rule 1504, monitoring of the cooling trains with respect to leakage and activity is to be provided.

- **Assessment**

The service water system A has operational and safety-related functions. Most of the cooling positions are fed directly without intermediate cooling circuits. The requirements according to KTA-Rule 3301, Item 5.4.2 (activation barriers to heat sink) are not met. For this reason a nuclear component cooling system is to be installed for all safety-relevant cooling positions (R 6.3-4). It must be demonstrated that there are sufficient water reserves in the spray ponds of the service water system A under design accident conditions. If this is not possible, the additional water supply will have to be designed in accordance with the KTA Rules for safety-related supply systems (R 6.3-14).

In the outer area the cross-over points of pipes from the service water system A of the three trains must be made safe, to withstand external impacts when multi-unit plants are used (R 6.3-15).

Evidence must be provided of a resistance to external impacts throughout the servive water system A (R 6.3-16).

#### 6.3.1.9 Emergency Standby System

- **Description of the System**

An emergency standby system is not provided in the Stendal NPP project. It must be assessed to what extent other systems can assume the function of an emergency standby system.

- **Assessment Criteria**

The emergency standby system is a further system for residual heat removal. It thus principally is subject to the requirements mentioned in Section 6.3.1.1. The additional requirements to be met by an emergency standby system are determined in RSK Guideline 22.2 (Emergency Standby System). The emergency standby system has the function of transfering the plant into a safe state without any manual measures and keeping it in that state for at least 10 hours in case of inoperability of the main control room. In addition, it must be possible to bring the plant into a state which permits subsequent residual heat removal via the LP-emergency cooling system, with the help of the emergency standby system by relief on the secondary side. The emergency standby system above all shall meet the following safety-related requirements:

1. The emergency standby system must be protected against external impacts and impacts of third parties.

2. There must be a consistent separation between the emergency standby system and other systems, i.e. there must be an independent energy supply for the shutdown of the plant and a system-independent cooling chain.

- **Assessment**

The Stendal NPP contains elements of an emergency standby system. In case of main control room failure there is an emergency control room which can shut down the unit, transfer it to a safe state and hold it there for at least 10 hours /MRE 92/. There are an emergency feedwater supply system, the relief valves BRU-A and the LP emergency cooling system of the primary system for use in long-term residual heat removal. The engineered safeguards do not meet the requirements of an emergency standby system with respect to the following points:

1. The above isystems are not self-sufficient.

2. The emergency feedwater tanks are positioned in one room.

3. The emergency feedwater pumps are not cooled via a system-independent cooling chain.

4. The main steam lines and installations in essential areas are not basically safe and resistant to external impacts.

5. There is no separate supply of electrical energy after failure of the emergency power supply, to fulfil the above mentioned requirements with respect to shut-down and maintenance in the safe state.

It therefore can be said that there is no standby emergency system which meets the conceptual requirements of the Federal German codes and standards. Therefore an emergency standby system must be backfitted (R 6.3-17).

### 6.3.2 Pressure Protection of the Primary System

- **Description of the System**

The pressure maintaining system of the primary system has the function of generating, maintaining and limiting the pressure in the primary system during different operational states. It is located in the containment. The most important component of the system is the pressuriser with its auxiliary systems. It is a vessel with a volume of 79 $m^3$ (55 $m^3$ thereof are water and 24 $m^3$ steam) which is connected with the hot leg of a loop by a line (surge line), without valves. Upon increase of coolant pressure with the main coolant pumps in operation, there is an automatic pressuriser spray with coolant from the cold leg of the loop. With the main coolant pumps switched off, "cold" coolant in this case is led by the make-up pumps via the cooldown line into the pressuriser and sprayed. Upon pressure decrease in the primary system the water content of the pressuriser is heated electrically (max. power 2520 kW).

Upon failure of pressuriser spray, the pressure in the primary system can increase up to the opening pressure of the safety valves. At the pressuriser three pulse safety valves DN 50 with a nominal flow rate of 50 kg/s steam each are installed. The safety valves are emergency power supplied. The response pressures of the safety valves are graded. The first safety valve responds at 17.9 MPa, the two remaining safety valves at 18.3 MPa. The steam from the safety valves is discharged through a pipe DN 200 into the water seal of the relief tank (tank volume 30 $m^3$, of which 20 $m^3$ is water). The water in the relief tank is cooled via a component cooling system by the service water system A. The relief tank is equipped a rupture membrane which is designed for a pressure of 0.5 MPa and ruptures after eight seconds at 100 % feeding of all safety valves. The relief tank then blows off into the containment.

- **Assessment Criteria**

In KTA Rule 3301, Item 4.3.4 it is requested for safety and relief valves of primary and secondary coolant systems that opening and closing pressures, opening and closing behaviour, relief capacity and the aggregate of the medium discharged, as well as the physical conditions on the relief-side are to be derived from accident analyses. RSK Guidelines, Section 3.1.4 demand that the plant is to be designed in such a way that opening of the pressuriser relief valves is only to be expected in case of infrequent transients with a high pressure increase and that the relief valves are to be equipped with an isolating mechanism, upstream, which will close automatically if the relief valves fail to close.

- **Assessment**

The pressure of the primary system is protected by three safety valves with 100 % steam relief capacity each. The safety valve which responds first is not indicated as a relief valve and has no isolating mechanism. The accident-related requirements to be met by safety valves (see Section 5.1) require their operation, even for discharging two-phase mixture. A pressuriser relief valve which can be isolated is therefore to be installed, which can also discharge two-phase mixture and water (R 6.3-18).

Further, the operating reliability of the pressuriser safety valves during discharge of two-phase mixture and water is to be demonstrated. In case of a new installation, the principle of diversity is to be applied (R 6.3-19).

### 6.3.3 Engineered Safeguards and Safety-Relevant Operational Facilities of the Containment

#### 6.3.3.1 General Remarks and Design Principles for Systems with respect to the Containment of the Stendal NPP

Function, design, calculation and the peculiarities of the containment are detailedly discussed in detail in Section 7.1.

The containment of the Stendal NPP is designed as a single-shell, full pressure containment. Because of the composite steel cell construction method it represents a prototype.

The containment is designed in accordance with the criteria contained in the Technical Project /TEP 81/ and the the General Principles of Ensuring Safety of NPP /OPB 73/. In these criteria, penetration isolation devices, i.e. devices for the retention and deposition of radioactive substances limiting the release of activity to permissible values are required.

### 6.3.3.2 Isolation of the Building, Locks, Racking Components

- **Description of the System**

Hermetic compartments (in the containment) and non-hermitic compartments (outside the containment) are connected by pressure-resistant hatches and locks. The containment is equipped with two locks for the staff (main and emergency lock) and a material lock consisting of a gate and a hatch. Apart from that there are several racking components.

The subsequent compilation of /SIE 90c/ provides an overview of the type and number of locks and racking components in the containment wall:

- Containment - cylinder area

| | |
|---|---|
| main lock | 1 |
| emergency lock | 1 |
| tube racking components | 71 |
| ventilation racking components<br>(TL 22/42 exhaust air/supply air and TL 21/41<br>exhaust air/supply, air for repair purposes) | |
| cable racking components | 864 |

- Containment-ceiling + 13.20 m

| | |
|---|---|
| transport hatch | 1 |
| tube racking components | 82 |
| cable racking components | 21 |

Pipe and cable connections through the containment are executed as hermetic racking components. All pipes penetrating the containment wall have at least two, predominantly three active penetration isolation valves (shut-off valves). This also applies to the valves in the drain system. The lines between the emergency boron tank and the valves TQ 10 (20, 30) S01 in the suction lines of the emergency cooling system, in each of which only one active penetration isolation valve is installed, represents an exception (cf. Section 6.3.1.3).

- **Assessment Criteria**

The assessment criteria are determined in BMI Criteria 8.1 and 8.2, RSK Guideline 5.6 and KTA Rules 3402, 3403 and 3409. These regulations require that pipes penetrating the containment and connecting to the pressurised enclosure must have at least two isolating valves. The isolating valves on the inside and outside must be located close to the containment, be remote-controlled and be sufficiently tight. The sudden complete break of a pipe must be controlled and the control and energy supply of the isolating valve must remain operable. In-service inspections must be possible at any lock and any racking component. Locks and ventilation flaps are to be connected to a leak-suction system.

- **Assessment**

The analysis of the confinement isolation, the locks and racking components showed that the design of the Stendal NPP corresponds in principle to the requirements of the Federal German body of rules. The peculiarities resulting from the different execution of the containment of the Stendal plant, unit A, or the safety containment with leak suction, required according to the Federal German body of rules, can only be assessed after an examination of the details.

A leak-suction system at the racking components and locks is not specified. To increase the efficiency of the containment, a leak-suction system is therefore to be installed at all penetrations, for a controlled and filtered discharge of leakages (R 6.3-20).

#### 6.3.3.3 Ventilation System of the Containment

- **Description of the System**

The functions of the ventilation systems for the rooms within the containment are

- establishment and maintenance of an underpressure of 200 Pa during power operation of the unit, to prevent the uncontrolled expansion of air from these rooms to avoid a release of radiaoactive substances,

- the removal of excess heat and humidity,
- the provision of optimum room conditions for normal plant operation,
- the creation of a room atmosphere permitting repair or reloading works during shutdown of the unit or after accidents.

The supply air and exhaust systems TL 42 and TL 22 primarily serve the purpose of fulfilling the two first functions above. Vents and filters are designed as three trains, lines are designed as two trains. The exhaust system comprises three emergency-power supplied vents (one in operation and two standby vents) and filters, as well as three quick-acting flaps in each of the supply air and exhaust lines at the containment boundary, connected in series. These flaps close on breakdown of the underpressure. The flaps, connected in series, are supplied by different trains of the emergency power system.

The supply air and exhaust systems TL 41 and TL 21, like the supply air system 48, are to fulfil the last mentioned function abvoe. As isolating valves at the containment boundary, there are two ventilation flaps per line, connected in series which close when the primary system temperatures exceed 150 °C. The vents of all the above-mentioned systems are located outside the containment.

Within the containment there are the recirculation system TL 49 for the creation of an air curtain above the spent-fuel pool, the TL 01 and TL 04 systems for heat and humidity removal out of the steam generator box and the reactor hall, the TL 02 system for cleaning the air via filters, the TL 03 system for cooling the control and protection drives and the TL 05 system for cooling the reactor vessel compartment.

The TL 01, TL 03, TL 04 and TL 05 systems are designed in triplicate and emergency-power supplied. The TL 02 system consists of an operational and a standby system. Heat removal is performed by the service water system A (TL 01, TL 04, TL 05) or the service water system B (TL 03).

- **Assessment Criteria**

The essential requirements to be met by the ventilation systems are contained in RSK-Guideline 9, BMI Criterion 9.1 and KTA-Rule 3601. Among other requirements there are an automatic isolation of ventilation upon high activity in the containment and the design of the filter system for accidents to ensure a certain filtration efficiency.

- **Assessment**

The ventilation systems can only be assessed conceptually. The safety function "Prevention of the release of radioactive substances into the environment" can principally be achieved with the ventilation system introduced. But the problems of accident-resistance, especially of the resistance towards external impacts, operating reliability etc. could not be investigated.

The available documents referring to the ventilation systems are in parts not sufficiently detailed, partially contradictory and information needed to assess the fulfilment of the requirements set forth in the German body of rules is missing (R 6.3-21).

The "Abschlußbericht zur verfahrenstechnischen Bearbeitung der Systeme, Lüftungssysteme AH (Kontrollbereich)" (Final report on the process treatment of the system, ventilation system AH (control area) of K.A.B. AG of February 1, 1991 /KAB 91/ shows that changes and additions to the components are necessary in the current project.

#### 6.3.3.4 Hydrogen Monitoring and Delimiting System in the Containment

Neither a hydrogen monitoring system nor a hydrogen delimiting system are planned in the Technical Project.

- **Assessment Criteria**

To assess the formation and distribution of hydrogen after design accidents and a local limitation of $H_2$-concentrations to values < 4 %, the requirements contained in the Federal German rules and guidelines, especially RSK-Guideline 24, are to be applied.

- **Assessment**

There is no evidence that the $H_2$-flammability limit is not exceeded during normal operation, as well as during an accident . Measures are therefore to be taken to prevent the formation of flammable hydrogen concentrations. Independent of such measures, a monitoring system must be installed (R 6.3-22).

### 6.3.4 Pool Cooling Systems

- **Description**

It is the function of the pool cooling system to remove the decay heat of the discharged fuel elements in the three spent-fuel pools (fuel element pools) (Fig. 6.3-1). It is designed as a three-train system (3 x 100 %) and independent of the emergency cooling system. Each train is assigned to one spent-fuel pool. The pool cooling trains are physically separated, but, because of the suction and pressure side interconnections, each train can also cool each pool. For cooling, the pool water is taken from about 3 m above the top of the fuel element and again returned to a point near the bottom of the pool via nozzles. Condensation losses are compensated with the help of the filling (pool make-up) system. The pool cooler is positioned on the suction side of the pool cooling pump. Cooling is performed by the service water system A.

During a loss-of-coolant accident the confinement isolating valves in the suction and pressure lines close. Cooling is then possible by the so-called "emergency flooding" with the help of the containment-spray pumps, where the spent-fuel pools are fed by a connection line from the containment-spray system to the pool cooling system. By this means, cooled water is led through the emergency coolers, from the emergency boron tank into the spent-fuel pools.

The pool cooling pumps are emergency power supplied so that cooling is ensured even during loss of off-site power.

The permissible pool water temperature is 70 °C during normal operation and 90 °C during accidents.

- **Assessment Criteria**

The essential requirements to be met by the pool cooling system are contained in RSK-Guideline 22.1.2 (12) and in the KTA-Rules 3303 as well as 3404, Item 3.13.

These state, among other things, the pool water temperatures to be observed for different operational cases:

- $T_1$ = 45 °C at maximum heating rate
- $T_2$ = 60 °C in abnormal system states (e.g. failure of redundancies, active components), or design accidents of the plant and simultaneous failures of active components, or a train not available for operation
- $T_3$ = 80 °C upon design accidents of the plant, when one or more trains are not available.

Furthermore, two acitivity barriers to the heat sink are required.

- **Assessment**

The design principles for the pool cooling system of a WWER-1000 apparently basically differ from those set forth in the KTA-Rules. One severe difference for example is the pool water temperature of 70 °C during normal operation of the plant permitted according to its design, compared to the 45 °C required by KTA Rule 3303. To prevent the escape of water vapour (as well as gases and aerosols) from the surface of the pool in WWER-1000 plants an air curtain is formed above the spent-fuel pool with the help of the recirculating ventilation system TL 49. In addition, the supply and exhaust air systems TL 41 and TL 21 have the function of achieving exchanges of air (even during reloading).

An assessment is thus only possible after knowing the bases and the results of calculations referring to the design. The number of redundancies and the independence of other systems are sufficient according to the requirements of the

KTA Rules. The pool coolers are cooled by one of the three trains of the service water system A each, which removes its heat via the spray ponds. The pressure conditions between the two media are unknown. Pollution of the pool water by service cooling water A, as well as a loss of water from the pool, are undesirable. The installation of an intermediate component-cooling system is therefore recommended (R 6.3-4).

The documents on the pool cooling system available do not suffice for a comprehensive assessment with respect to the fulfilment of the requirements of the German body of rules. Therefore a recommendation can only be made for the pool cooler.

## 6.4 Instrumentation and Control

Instrumentation and control (I & C) comprise the facilities for operational monitoring, adjustment and control of the nuclear power unit as well as I & C systems important to safety. In addition, there are the main control room and the emergency control room serving operational as well as safety-relevant functions.

### 6.4.1 Control Rooms and Control Desks

- **Description**

In the main control room of the WWER-1000 there are the control desks of the reactor operators, of the turbine engineers and the unit supervisor.

In the main control room, situated in the reactor building at an elevation of + 6.6 m, 28 boards are arranged in a U-shaped array. To illustrate the functional connections of the main systems, mimic flow diagrams arranged according to process criteria are positioned on the boards. Directly in front of the workplaces of the operators, eleven black and white or coloured screens of the control computer are located on the desks. In addition, one screen showing the in-core measurement system is located near the reactor operator.

The most important means for illustrating information in the main control room is the control computer system. The computer system carries out the acquisition, processing and illustration display of measured data on screens in the main control room. For their decisions, the control room staff are largely dependent on information from the control computer. Without this control computer system the power plant unit cannot be power operated. If the control computer system fails, the power plant unit must to be shut down within a certain period of time.

The unit in power operation, as well as during startup and shutdown, is controlled from the main control room, among others with the help of function group controls.

To control the safety system including the reactor scram system additional hardware is installed in the main control room. Upon failure of the control computer system and the function group control, the unit can thus be shut down in a controlled way. Power operation is, however, not possible with this hardware. In case of a disturbance, the must important process parameters can be indicated and recorded and, if the automatic mechanisms fail, the engineered safeguards can be actuated manually.

In cases when the main control room cannot be used any more, there is an emergency control room, with the help of which the shutdown of the unit can be initiated and supervised and the work of the safety systems can be monitored and controlled.

In the emergency control room, situated in the reactor building at the elevation of -4.2 m, no systems belonging to the control computer are installed. The indication and operation functions are executed conventionally without computer technology. The positioning at - 4.2 m is to secure the function of the emergency control room during external impacts, like earthquakes or airplane crashes.

During normal operation the emergency control room is not attended. When there is a disturbance in the area of the main control room, so that the emergency control room is entered by the operating staff, the signalling devices are at first out of peration. In the Technical Project, it is stated that the signalling devices of the safety system are to be switched in one after the other by the operating staff. This approach is supposed to be necessary to prevent operating staff losing the overview on entry to the emergency control room because of signals that are no longer up-to-date.

During an accident all functions of the safety systems are started automatically, so that the operating staff in the emergency control room primarily has to perform supervisory functions and will only have to interfere if an automatic mechanism fails.

Circuits shall be provided which do not permit the use of erroneous signals for component control from a defective control room. This is, for example, realised by sending a code upon pressing a key or actuating a switch. This code cannot be generated upon short circuits caused by fires or flooding. It shall thus be ensured that a malactuation caused by a defect in a switch or cable of the main control room or the emergency control room is not possible. More detailed documents on the decoupling of the emergency control room and the main control room are not available.

Besides main control room and emergency control room there are:

- control room for stationary radiation protection monitoring
  (for controlling dosimetric parameters)
- the auxiliary-power supply control room
  (for controlling the station-wide electrical requirements)
- the load dispatching centre
  (corresponds to a station-wide power system control centre)

In addition thereto there are a number of local control desks, especially for local and auxiliary systems.

Documents on the design and function of these control rooms and desks do not exist.

- **Assessment Criterion**

Assessment criterion is the Federal German KTA-Rule 3904.

It must be possible to monitor and control the normal operation of the power plant unit from the control room. The indications and actuation devices required for this purpose are to be positioned in the main control room in such a way that the indications can be seen from the working positions and the actuation devices can be seen and operated. It must be possible to recognise disturbances and to take measures to keep the plant in a safe condition. The control room must be designed against flooding, lightning, storm and against effects resulting from radioactive irradiation during accidents. It should be designed in such a way that it can neither fail owing to fire nor to airplane crash or explosion of a gas cloud. If there is an emergency control room the design is to be chosen in such a way that, as a result of the above events, only the main control room or the emergency control room can fail, not both.

If necessary, it must be possible to transfer the reactor from power operation to the safe shutdown state, and to keep and monitor it in that state.

- **Assessment**

An assessment can only be provided on the basis of the existing documents. Main control room and emergency control are physically separated. Simultaneous destruction of both systems seems to be unlikely.

As the unit cannot be operated in the power operation mode without the control computer in the long term, it must be thoroughly examined if and for how long power operation is possible (R 6.4-1).

The solution concerning the decoupling and preferential switching between the main control room and the emergency control room must be analysed in detail and evaluated as to its admissibility (R 6.4-2).

The transmission of infomation to the emergency control room upon entry of the operating staff must be examined as to its correctness, by use of further documentation (R 6.4-3).

### 6.4.2 Operational Instrumentation and Control

All measurement, adjustment and control systems required for operation, the systems for controlling special process and plant parameters and the control computer belong to the operational instrumentation and control.

It is necessary to investigate the operational instrumentation and control in connection with this assessment, as there are parts of the operational instrumentation and control performing safety relevant functions. The control computer system and the in-core monitoring system SWRK as an important part of the man/machine interface are of special significance in this context. Reliable control circuits adjusted to the unit dynamics and the function group controls are the basis for controlling complicated transition processes to prevent disturbances which could be starting points of accidents.

#### 6.4.2.1 Control Computer System

- **Description**

The control computer system (reference Titan-2) is the most important means for illustrating information in the main control room. Based on four computers of the SM-2 type, it represents a measurement data acquisition, processing and illustration system. Two computers, redundant to each other are each employed for the primary system and for the secondary system including local systems. Both computers of the primary system or of the secondary system process the same tasks and monitor each other's operational reliability.

Analog data acquisition is performed in basic computer units with a time cycle of 100 ms. In the supervisory computers the measured values are, however, only available in 2 s or 4 s cycles. Measuring is performed by a 0 to 5-mA standardised signal at the transmitters and transducers. Binary values are normally acquired in a 1 s cycle and, for about 200 values, in a 100 ms cycle /KKA 90/. For important analog and binary values the input channels of the control computer system are doubled. There are no figures on the exact extent of the measured signals processed.

The control computer system Titan-2 contains information from the in-core measurement system, from the emergency cooling system, as well as from the safety control system on the state of the reactor and the control state of the safety system. It further contains information on the computers of the function group controls.

The measured values acquired are stored in the control computer system, they are processed according to the respective algorithms and output via printers or displayed on colour graphics screens.

The control computer system is supplied by a battery-aided non-interruptable power supply, which during normal operation is fed by a 0.4-kV unit distribution. If necessary, it is possible to manually switch in a supply from the 6-kV distribution of the emergency power system, where only short interruptions occur, via the standby transformer, so that a power supply to the control computer system is also possible under difficult conditions.

- **Assessment Criterion**

The function of the control computer is to acquire and output the process data in such a way that

- the operator of the control desk is informed immediately and comprehensively,
- disturbances are detected early, cleared quickly and counter-measures can be taken so that accidents can occur less frequently,
- decision aids are provided for the management of the plant,
- the processes are documented.

Essential characteristics of the control computer system must be

- high degree of reliability
- good illustration of information for process control
- easy operation and
- sufficiently fast acquisition of analog and binary measurement values (at KWU computer systems 1 s cycle for analog and 10 ms cycle for binary values).

- **Assessment**

The control computer system is designed redundantly. The cycles of the analog and binary signal acquisition are too slow to provide control room staff with information on the screens with a sufficiently short time resolution. The computers employed do not conform to international standards. Statements on reliability and on of the software employed, do not exist. It is therefore recommended to install modern computer technology from the start, should construction of the power plant be resumed (R 6.4-4).

#### 6.4.2.2 In-Core Measurement System (SWRK System)

- **Description**

To control the parameters within the reactor the SWRK system is employed. It fulfils the following functions.

- Collection and processing of meaured values

- Illustration of information on
  - the distribution of the neutron flux in the reactor core,
  - the distribution of the energy release, the temperature of the fuel elements and the coolant,
  - the burnup of the fuel elements
  - the value of the reactivity reserve.

The following data are acquired in the SWRK:

- 95 fuel element outlet temperatures measured by thermocouples of chromel-alumel material, type TXA-2076

- 64 n,β-measuring lances, each equipped with seven Rh-SPN detectors, having a time constant of 1 min, as well as three thermocouples

- signals from the ex-core neutron flux measurement system and from the system measuring the process parameters.

The measuring positions for fuel element outlet temperatures are about 40 cm above the core location in the reactor.

There is no system similar to an aeroball flux measuring system for calibrating the SPN detectors.

In the SWRK, binary values are available on a 2 s cycle and the analog values are available on a 12 s or 60 s cycle /KKA 90/.

For reasons of redundancy the SWRK system consists of two complexes of equipment. Each complex consists of a measured-value acquisition system of the "Hindukusch-1" type with a computer SM-2M connected to it. All measurement signals except those of the measuring lances are connected to each Hindukusch system. The signals of the measuring lances for technical reasons are evenly distributed between both systems. The signals processed are always transmitted to the other system. Both systems are coupled and work in parallel so that further steady operation of the unit is possible after failure of one of the systems.

During normal operation conditions, pre-processing of the measured values as well as a series of operative calculations are performed in the SWRK system. The calculation results are transmitted to the control computer which performs further calculations.

An autonomous operation mode is also possible. Here simplified calculations of the most important parameters of the reactor are performed by the SWRK system and the results are shown on screens or printed out by printers.

The SWRK system does not automatically affect reactor control for limiting power density.

- **Assessment Criterion**

According to KTA-Rule 3101.2 power density of pressurised water reactors is to be limited in such a way that the limits required are kept under normal operating conditions and that, upon events of an abnormal operational condition or accidents, fuel element and cladding tube conditions proven to be permissible are observed. To fulfil these requirements a continuously indicating instrumentation of the reactor core and the cooling systems is to be provided for monitoring the local power density, if required. If necessary, equipment and measures for limiting power density are to be provided. The number and position of the sensors, their calibration and the kind of signal formation are to be selected in such a way that inadmissible increases in the local power density in individual zones of the reactor core to be monitored can be detected.

- **Assessment**

The in-core measurement system SWRK only represents a pure information system.

Because of the relatively large number of measuring positions and measurement data acquisition by two computers, an appropriate reliability and availability of the system can be assumed. The requirement relating to the existence of devices for power density limitation are, however, not fulfilled by this system. A local reduction of the power density is only possible by manual measures of the operating staff. As the distance to the measuring positions of the fuel element outlet temperatures from the fuel elements is too great, the accuracy of their measured values is reduced (cf. Section 4.1.5).The concept of core instrumentation as introduced in the Technical Project of 1981 should be thoroughly revised. In this context it should be extended by a power-limitation system as well as a reliable calibration system (R 6.4-5).

#### 6.4.2.3 Function Group Controls and Control Engineering

- **Description**

- Function Group Control

Apart from the primary system the main process groups are equipped with function group controls.

The actuators are controlled by a single drive control, which allows manual control, control by I & C safety systems with priority or by function group controls. Protection, thyristor controllers as well as control modules are technical equipment for every drive.

In the function group controls, the control logic to automatically control the program of the actuators according to preset algorithms is realised on the basis of the microprocessors MPKA-135-1.

There are connections from the function group controls to the control computer for transmitting additional information on the state of the actuator drives and the measured values.

- Control Engineering

  Control engineering was planned on the basis of the equipment family "Kaskade-2". Different control loop structures can be configured with different electronic modules. Thus it is technically possible to correct specified values by the control computer system and to change the control structure of the function group control on command.

  The most important control loops are:

  - Reactor power controller (ARM)
  - Turbine power controller
  - Pressure controller of the primary system
  - Level control in the pressuriser
  - Level control in the steam generators
  - Controller for the maximum steam pressure and for relief of the secondary system during turbine tripping (BRU-A, BRU-K)

  The reactor power controller (ARM) consists of a controller for stabilising neutron flux density and a controller for stabilising technical process parameters. Each controller consists of three channels and operates according to a 2 of 3 selection principle. ARM can work in the following operational modes:

  - *Operational mode N:* Neutron flux control
  - *Operational mode T:* Control of main steam pressure
  - *Operational mode S:* Monitoring of main steam pressure. When it rises above a certain value, the reactor power is reduced.
  - *Operational mode K:* In the upper range of the reactor power the temperature of the primary system is kept constant, in the lower range of the reactor power main steam pressure is kept constant.

The operational mode N can be used in the range of 3 % to 120 % of the nominal reactor power. For the operational modes T, K and S, which can be used in the restricted range of 20 % to 110 % of the reactor power, neutron flux control serves as an auxiliary control parameter. There is no automatic control for compensating local power fluctuations, but the in-core measurement system provides information to the operating staff, with the help of which a manual compensation of the power density is possible.

There is not sufficient information on the other controllers.

- **Assessment Criterion**

It can be seen from BMI Criterion 1.1 (Grundsätze zur Sicherheitsvorsorge (Principles of Safety Precautions)) that high requirements relating to design and quality are also to be met by the operational instrumentation and control. Operational instrumentation and control even without recourse to engineered safeguards, must ensure operation which is as free from disturbance and environmentally compatible as possible.

- **Assessment**

The instrumentation and control employed for control engineering and function group control is used in several Soviet power plants. It can, however, not be derived from the documents whether they meet the requirements for use in a nuclear power plant. It can be concluded from the analyses of the operating experience of other WWER-1000 units (see Section 8) that the reliability of the operational instrumentation and control is inadequate. This concerns actuations, position indicators and limit-position switches of all isolating valves and control valves (R 6.4-6). The gauges for pressure and differential pressure should be qualified (R 6.4-7). Following negative operating experience in other WWER-1000 units, the I&C concept for the control of dynamic transition processes should be revised (R 6.4-8).

### 6.4.3 I & C Systems Important to Safety

The I & C systems important to safety in the WWER-1000 unit are subdivided into the emergency protection system and the protection system for controlling the safety system.

The designer assigned the emergency protection system to the control and protection system (SUS). To the control and protection system further belong the operational control system of the reactor as well as the reactor power controller ARM. The emergency protection system actuates reactor scram if the respective criteria are fulfilled. The alarm system (reactor power limiting facility) is also part of this system, as a back-up protection.

The protection system for control of the safety system (safety control system) serves to initiate of protective actions of the active engineered safeguards, with the exception of the reactor scram system.

#### 6.4.3.1 Emergency Protection System of the Reactor

- **Description**

The instrumentation and control of this partial system comprises all facilities for monitoring and limiting the reactor power and for actuating and activating reactor scram.

As there was only insufficient information on instrumentation and control of the Stendal NPP, the documents for the same type of nuclear power station at Saporoshje were used /KKS 90a/, /KKS 90b/, /KKS 90c/.

Instrumentation and control of the emergency protection system is subdivided into two independent, physically separated trains for reactor scram as well as one train for actuating the alarm system. The actuation criteria for reactor scram are listed in Table 6.4-1. Exact information on these actuation criteria and on the actuation criteria of the alarm systems can be found in /GID 90/. Upon response of one train of the emergency protection system, all 61 control rods drop into the reactor.

The alarm system according to its function represents a back-up protection. It is actuated by criteria which normally respond before the criteria requiring reactor scram occur.

According to its actions the alarm system is subdivided into:

- accelerated alarm system:
  effects the drop of the control rods of the control group into the reactor

- alarm system I:
  leads to the insertion of control rods according to the normal sequence
- alarm system II:
  generates a signal to prohibit withdrawal

The emergency protection system can be subdivided into an actuation level, logic level and control level. It consists of two trains, each train designed as a 2 of 3 selection circuit.

The actuation level of the system is divided into

- the system for creating signals of the neutron flux
- the system for creating signals from process parameters

The system for creating the signals of neutron flux (AKNP) consists of two independent trains. Each train consists of three measuring channels each for the startup range, the transition (intermediate) range and the power range, as well as the corresponding evaluation modules and limit transducers (see Fig. 6.4-1) /KKS 90b/. Each AKNP train is assigned to a train of the logic level. Information is transmitted via closed-circuit contacts. It could not be determined whether there is an automatic monitoring of the limit adjustment of the neutron flux parameters. Self-monitoring of the measurement channels exists.

The system for creating process actuation signals generally has three transducers for every train of the logic level, having one subsequent limit transducer each. For each actuation signal for pressure transmitters and differential pressure transmitters, only three pulse lines are planned. Two pressure transmitters each from the different trains are connected to one pulse line (see Fig. 6.4-2). In addition, one transmitter each for the train of the alarm system is also connected with each pulse line /TPS 81/. Also, for the process actuation criteria, information is transmitted by limit transmitters to the logic level via closed circuit contacts.

Each train of the logic level consists of three channels (see Fig. 6.4-3) /KKS 90c/. Each channel is installed in a control cabinet.

If one actuation responds in one of the evaluation modules, the square wave of the signal generator is not transmitted from this module onwards, so that the output amplifier module is triggered off. In the following relay module there is a 2 out of 3 weighting so that at least two of three channels of a train must have responded. In the following module there is a relay contact multiplication and a contact provision for the control level.

The emergency protection system is not designed to be self-monitoring in important partial sectors. It is not indicated how the signals are transmitted.

Reactor scram is actuated at the control level by two signal pathways independent of each other. On the one hand a signal for shutting down the operating electronics and for terminating the corresponding power infeed is transmitted to every one of the 61 control element drives. On the other hand the power supply for all control element drives is switched off on the respective boards.

Logic level and control level are connected in such a way that the priority of reactor scram is ensured and that a functional examination of a train is possible up to the last actuation member without drop of the control elements.

The train of the alarm system receives its signals from process actuation criteria, neutron flux actuation criteria and also from the facility for reactor power limitation ROM. If main coolant pumps or feedwater turbo-injection pumps fail or the fast-acting turbine isolation valves close, a reduction of the reactor power will be effected by the ROM system via the alarm system I as long as the neutron flux of the reactor does not exceed the limit admissible for the actual state of operation.

A conclusive quality assurance for the entire instrumentation and control of the emergency protection system cannot be derived from the documents available.

- **Assessment Criterion**

KTA Rule 3501 represents the basis for the assessment of instrumentation and control of the emergency protection system. In this rule is defined under which basic assumptions of failure combinations the emergency protection system shall remain workable. The following failure combinations are to be considered during an analysis of the intended operation of the reactor plant:

- random fault and systematic failure or
- systematic failure and maintenance case or
- random fault and maintenance case with the respective additional consequential failures.

These failure combinations even coincident with a disturbance may not lead to the failure of the emergency protection system.

The effects of systematic failures in the emergency protection system are to be analysed. Depending on the result of the analyses, additional measures for reducing the probability of occurence of systematic failures or their effects are to be taken. In these analyses, it is assumed that as a consequence of a systematic failure all similar equipment of a product fails simultaneously and in the same way in the signal channels. These analyses may be renounced, if using diverse measuring facilities, when a systematic failure of these measuring devices does not have to be assumed.

- **Assessment**

The requirements of the KTA Rules with respect to redundancies and to the separation of the trains of the emergency protection system are largely met. The following deficiencies have been perceived:

There is no diversity in the equipment in the two trains of the emergency-protection system for reactor scram. No evidence is available that this is compensated by special technical and/or organisational measures. Such evidence should be given (R 6.4-9).

- Except in the neutron flux measurement system, there appears to be no self-monitoring available in the emergency protection system. Self-monitoring should be backfitted (R 6.4-10).

- It is possible that the limit values of the neutron flux measurement system as well as those of the gauges of the actuation criteria related to process-engineering may readjust themselves without being noticed. It is recommended to eliminate this deficiency by technical measures (R 6.4-11).

- A case where there is maintenance work going on in one train of the emergency-protection system and a failure occurs simultaneously which renders the entire second train ineffective (e.g. through external or internal impacts) cannot be controlled. It must be examined if and for how long one train may be taken out of operation for maintenance purposes (R 6.4-12).

- A control-element-insertion limitation must be backfitted for ensuring shutdown reactivity (R 6.4-13).

- No information on the reporting and inspection concept could be derived from the documents. As a conclusion from the operating experience in other WWER-1000 units in operation (cf. Section 8) it is recommended to revise the reporting and inspection concept (R 6.4-14).

#### 6.4.3.2 Protection System for the Control of the Safety System (Safety Control System)

- **Description**

The protection system for the control of the safety system (safety control system) is responsible for the initiation of protective actions of the active engineered safeguards.

A train of the safety control system is assigned to each of the three process trains of the engineered safeguards. Each train of this safety control system fulfils its functions independent of the two other trains. The three trains are located in separate rooms with separate cable channels and separate pulse lines and transmitters. Each train of the safety control system fulfils the following functions:

- Measurement and assessment of process variables required for the actuation of protective actions,

- Switching on and off of the process units of the engineered safeguards belonging to the train and opening or closing of the respective valves corresponding to predetermined algorithms.

According to the designer the entire I&C equipment of the safety control system is designed earthquake-proof. Performance tests of the modules as well as accident resistance proofs are not available. There is no equipment diversity.

Below only one of the three trains of the safety control system is described. Each train can be subdivided into actuation level, logic level and control level.

The actuation signals from the process variables are formed at the actuation level. The actuation signals for initiating the safety system are contained in /ATP 87a/, /ATP 87b/. The most important actuation signals are listed in Table 6.4-2.

The actuation signals are acquired in quadruplicate per train of the safety system. The four limit transmitters of an actuation criterion inter-compare the analog signals of the gauge and its transform and report any inadmissible deviation of the measured values. There is no information on the further processing of these signals and their indication in the main control room.

On the logic level the actuation criteria are built from signals of the actuation levels. An actuation signal is built in a 1 out of 4 selection circuit of the respective output signals of four evaluation modules. Each of these four modules first performs a 2 out of 4 evaluation of the actuation signals (see Fig. 6.4-4).

In contrast thereto the respective actuation signal upon loss of off-site power is derived from in a 1 out of 2 selection.

The actuation criteria are further processed on the logic level in two redundant logic circuits and are then fed into the drive controls in a 1 out of 2 selection. On the logic level the signals for starting the units are formed after fitting the actuation criteria. The signals for actuating the priority circuit, including blocking of manual interference by unit staff, are also fromed and the signals for closing the valves of the containment isolation are initiated. During loss of off-site power the logic circuits form signals for activating the diesel generators and the staggered connection of the units.

The logic level is realised by relay connections with working current principle and electronic logic modules.

On the control level the actuation signals for the individual units are formed. For each process unit an independent drive control is planned.

A priority circuit is included in the control level. Upon response of the unit protection criteria in the 2 out of 2 selection circuit the switch-off prohibitions of the safety control system are cancelled. The switch-off itself to protect the unit is not effected automatically, but has to be performed manually.

All modules, beginning with the limit transmitters of the actuation level to the control modules of the control level provide signals on the state of their outputs to the control computer. In addition, there is the possibility of partially performing automatic function tests.

No exact data on securing the power supply of the safety control system can be derived from the documents.

The design of the system was not based on a time criterion, like, for example, the Federal German 30-minute criterion. It therefore cannot be excluded that in individual cases manual measures will be required before 30 minutes have elapsed. Safety-hazard reporting assigned to the manual actuations, required according to KTA Rule 3501, cannot be derived from the documents.

- **Assessment Criterion**

The assessment criteria for the safety control system are the same basic requirements of KTA Rule 3501, as expressed in connection with the emergency protection system of the reactor (Section 6.4.3.1). However since only the actuation of active protective actions is concerned here, the following requirements have also to be considered: The safety control system shall actuate protective actions automatically. The safety system is to be designed in such a way that necessary protective actions to be actuated manually are not necessary for controlling accidents before 30 minutes have elapsed.

Even during maintenance work on the safety system, no accidents with resulting damage may be generated by an initial failure in the safety control system.

- **Assessment**

The basic technical concept of the safety control system meets only the basic requirements of the KTA-Rules with respect to redundancy as well as to a functionally and physically separated design.

The following deficiencies were detected:

- There is no diversity in the equipment within the safety control system. No evidence is available that this is compensated by special technical and/or organisational measures.
- Self-monitoring with fault indication only exists for the limit transmitters and the evaluation modules BFK. These modules transmit a message to the control computer system. No permanent automatic self-monitoring can be recognised for the redundant logic circuits of the logic level as well as for the control level. Backfitting of complete self-monitoring is recommended (R 6.4-15).
- An unnoticed readjustment of the limit values in the signals is possible. It is recommended to eliminate this deficiency with technical measures (R 6.4-16).

- There is no evidence that manual protective measures for accident control do not become necessary before 30 minutes have elapsed. For such manual protective measures, safety-hazard reporting according to KTA 3501 should be backfitted (R 6.4-17).
- It is recommended to provide evidence that the protection system does not initiate safety-significant transients during power cuts (R 6.4-18).
- It is recommended to provide evidence of type inspections conforming to international standards for all equipment used. Whereever this is not possible, the technical equipment should be replaced (R 6.4-19).
- A reconstructable quality assurance does not exist.
- Although in-service inspections of the safety control system was discussed in the documents, no inspection concept could be perceived.

### 6.4.4 Accident Instrumentation

- **Description**

In addition to the acquisition of measured values, their processing and display by the control computer system, all measured values which are essential for the transfer of the reactor into the safe state, for monitoring the work of the safety system and for monitoring the sub-critical state of the reactor, are illustrated in the main control room and also in the emergency control room with the help of conventional technology. This system performs functions which partially correspond to those of an accident instrumentation as required by KTA Rule 3502. However, nothing was found in the documents on the scope of the measured values, on the measurement ranges and on the accident resistance of the technology employed.

According to the Project there shall be two redundant sets of equipment for indicating the measured values and recording each in the main control room and in the emergency control room, which are each supplied by two independent, reliable power supply systems. Recordings are made by indicating recorders. By these measures it is achieved that, should the control computer system fail or the main control room be destroyed, indication and recording of the most important reactor parameters is possible.

It can be seen from the documents of the former Kombinat Kraftwerksanlagenbau /KAB 90/ that this organisation worked on its own concept of accident instrumentation for the Stendal NPP.

- **Assessment Criterion**

BMI Criterion 5.2 and KTA Rule 3502 represent the basis for an assessment criterion relating to accident instrumentation.

In the nuclear power plant there must be facilities for measuring and recording to be able to

- provide sufficient information on the state of the systems to take the appropriate protective measures for staff and plant,
- provide hints with respect to the sequence and to render its documentation possible,
- estimate the effects on the environment

during and after accidents and unforeseeable sequences of events.

The range of measurements of the accident overview indication are to be chosen in such a way that they render possible an assessment of the state of the plant after occurence of an accident with respect to the following criteria:

- effectiveness of reactor scram
- effectiveness of residual heat removal
- effectiveness of primary-side and secondary-side pressure limitation and pressure reduction measures
- effectiveness of the activity enclosure.

The equipment of the accident overview indication, during accidents and their consequences at their respective place of installation, must withstand the environmental conditions which may occur and remain operable.

- **Assessment**

It can be seen from the present documents that facilities are planned in principle for the Stendal NPP which can partially fulfil functions of an accident instrumentation. Accident resistance, scope of measurement parameters as well as accident recordings are, however, not explained or verified. An assessment of these points can therefore not be provided. It is recommended to provide evidence that the requirements of KTA-Rule 3502 concerning accident instrumentation are met by the existing equipment. Backfitting must be carried out where no such evidence exists (R 6.4-20).

### 6.4.5 Summarising Assessment of Instrumentation and Control

The physical separation of main control room and emergency control room is assessed to be good. The physical separation of the three trains of the safety control system are to be evaluated in the same way.

There are no statements on quality assurance, accident resistance and on the reliability of the equipment used for I & C systems important to safety. The individual recommendations are listed in Section 10. Owing to the plurality of recommendations it is suggested that detailed investigations be performed using the operating experience of other units, which will have to clarify whether the intended instrumentation and control can be used in the Stendal NPP or whether it would be better to replace the entire I&C system.

## 6.5 Electrical Energy Supply

### 6.5.1 Grid Connection and Generator

- **Description**

A generator having the following main parameters was designed for the Stendal plant:

| | | |
|---|---|---|
| Capacity: | 1000 | MW |
| Speed: | 3000 | RPM |
| Frequency: | 50 | Hz |
| Efficiency: | 98.75 | % |

The electrical power is led from the generator to two unit transformers having a capacity of 750 MVA each via a power switch. Both transformers, during the first construction phase with one power plant unit, feed into a 220-kV switchyard of the Schwarzholz substation. The Schwarzholz substation further feeds into the grid via three 220-kV double-circuit lines. The generator is followed by a power switch, so that feeding both auxiliary supply transformers via both or one of the two unit transformers is thus possible during failure of the generator. The 110-kV switchyard is fed from the 220-kV switchyard. It supplies the standby auxiliary power supply transformers as well as the auxiliary supply transformers of the general auxiliary power supply.

In the second construction stage of the power plant the two unit transformers are fed in such a way that one feeds into the 220-kV switchyard and the other into the 380-kV switchyard still to be built. The unit shall be designed for load rejection to auxiliary power supply during supply failure.

- **Assessment Criterion**

Basic requirements relating to the assessment of the grid connection are contained in KTA Rule 3701. The following infeeds must at least be available for the electrical energy supply of the safety system of a reactor plant:

- possibility of auxiliary power supply in the unit by the unit generator of the power plant

- two grid-side possibilities of auxiliary power supply
- emergency power supply system.

Following connection of the main grid, the generator switch and the standby grid connection, the standby grid connection must permanently be operable and automatically connectable. Its capacity must be sufficient for shutdown of the power plant, maintaining the main heat sink. There must be a generator switch between unit generator and auxiliary supply branch which renders startup and shutdown of the power plant possible via the main grid connection.

The emergency power supply system and its sources in the nuclear power station are to be designed in such a way that it must at least be possible to obtain the electrical power required for residual heat removal to supply one cooling train.

- **Assessment**

The circuit technology of the grid connection corresponds to the general requirements of KTA Rule 3701.1. In the first construction phase the grid connection is only carried out via a 220-kV switchyard which also feeds the 110-kV switchyard. In case of a defect in the 220-kV switchyard all other grid connections may possibly fail. It is therefore recommended to build a second switchyard, e.g. a 380-kV switchyard, in order to provide a redundancy (R 6.5-1). It is recommended to backfit an emergency grid connection, which so far is not available, by way of an underground cable (R 6.5-2).

### 6.5.2 Classification of Electrical Energy Consumers

The consumers of electrical energy were divided by the designer as follows:

- **Category I**
  - Consumers, which do not permit a power supply interruption and require a reliable constant supply for response of the safety system of the reactor

- Consumers which do not permit a power supply interruption which do, however, not require a constant supply for response of the safety system of the reactor
- Consumers which during normal operation and transitional operational states require a guaranteed supply within 2 s to prevent an incorrect response of the safety system, which can, however, manage without power supply upon voltage failure for response of the safety system.

Consumers of category I are supplied by batteries or by units of the emergency supply system respectively for interruption-free power supply.

- **Category II**
  - Consumers which can have a short-term voltage interruption and which must in any case be supplied for response of the safety system

Consumers of category II are fed by 6-kV or 380-V busbars of the emergency power supply system.

- **Category III**
  - Consumers which do not have high requirements to be met for reliablitiy of supply

Consumers of category III are supplied by the auxiliary supply system.

### 6.5.3 Auxiliary Supply System

- **Description**

6-kV and 380/220-V buses to feed the consumers of category III are provided in the auxiliary power supply grid, eg. cooling water pumps and main coolant pumps.

The auxiliary power supply in the power plant is subdivided into the auxiliary power supply of the turbine hall and the reactor building, as well as the general auxiliary power supply.

The elementary diagram of the auxiliary power supply for the turbine hall and the reactor building (Fig. 6.5-1) shows that four 6-kV unit distributions independent of each other are fed by a total of two auxiliary supply transformers. The two auxiliary supply transformers, having a capacity of 63 MVA each, can be fed by the generator and, upon generator shutdown, also from the grid. In addition, there are auxiliary supply standby transformers with a capacity of 63 MVA each, which make the full auxiliary power possible supply via a 110-kV line from the switchyard.

The 6-kV level is divided into four independent supplies. One main coolant pump is connected to each supply.

The general auxiliary power supply serves the supply of the local systems and the auxiliary systems. To increase supply safety the general auxiliary power supply is equipped with an independent diesel unit with the capacity 6.3 kV/6.3 MW.

The systems fed by the general auxiliary power supply are

- radiation protection monitoring systems
- diesel pumps for the additional injection of diesel fuel from the central storage tank into the intermediate storage tanks of the three emergency power systems and
- rectifier for battery recharging of the emergency lighting.

The general auxiliary power supply consists of four 6-kV busbars fed by two independent transformers with 40 MVA each from the 110-kVswitchyard. The 0.4-kV busbars are supplied from the 6-kV busbars via transformers.

- **Assessment Criterion**

The BMI Criterion 1.1 (Principles of Safety Precautions) as well as KTA Rule 3701 are the basis for the assessment criterion of the auxiliary power sypply system.

An accident-free and environment-friendly operation of the plant without recourse to the engineered safeguards must be ensured by high requirements of design and quality of the plant. Sufficient safety margins, approved materials, maintenance-friendliness of the component parts and a comprehensive quality assurance are in particular to be achieved .

The physical arrangement of the auxiliary power supply system must be organised in such a way that not all supply possibilities can fail due to a single failure initiating event.

Apart from the BMI Criterion the respective VDE and DIN regulations are also to be taken into account.

- **Assessment**

The auxiliary power supply system corresponds to the general requirements of KTA Rule 3701.1 with respect to the physical arrangement. There is a division of the 6-kV level in four independent distributions having one main coolant pump each, as well as the support of the general auxiliary supply system by a diesel unit. It can be derived from operating experience in other operational units of the same type, that the quality assurance particularly of the cables and switches is poor. Cables and switches should be replaced by approved ones (R 6.5-3). In the auxiliary power system, sufficient selectivity to prevent short circuits and protection against consequential spreading impacts between the individual 0.4-kV and 6-kV busbars must be backfitted (R 6.5-4).

### 6.5.4 Emergency Power System

- **Description**

The emergency power system supplies consumers of category I and II (also see Section 6.5.2).

It can be seen from the Stendal NPP Project that consumers are supplied by emergency power which according to the Federal German body of rules would not have to be supplied with emergency power. To these belong the control elements and the instrumentation and control, for example. These consumers are supplied with emergency power to ensure their function during the short-term or also during longer power failures.

Potential disturbances occuring in the operation of the power station shall thus be prevented.

- Emergency Power Supply of the Safety System

To secure the emergency power supply of the safety system, three identical emergency power systems, completely independent of each other, were also built corresponding to the three process trains of the safety system (Fig. 6.5-2). Each of the three emergency power systems is designed in such a way that it can feed into the train of the safety system assigned to it upon full load. The 6-kV busbar of the respective emergency power system during normal conditions is fed via two switches connected in series from one of the 6-kV distributions of the emergency power supply. If this feed fails, an independent diesel generator is initiated automatically.

The consumers of category II are supplied directly or via an intermediate transformer from the 6-kV emergency power busbar.

The consumers of category I are fed from a unit of the interruption-free power supply. This unit is connected with the 6-kV emergency power busbar via an isolating transformer. It consists of two rectifiers, a battery and two inverters. The two rectifiers feed the d.c. busbar, among other things being responsible for recharging the battery. To exclude the influence of short circuits in the 220-V outgoing d.c. circuits onto the inverter operation, a separating diode is installed between the outgoing circuits mentioned and the incoming line to the inverters. The two inverters generate a sinusoidal 380/220-V a.c. voltage to supply the a.c. consumers of category I. It cannot be derived from the documents how these consumers are connected to the inverters. In later WWER-1000 plants four inverters are provided in every emergency power system. The supply of the I & C systems important to safety are thus ensured more reliably.

The switching and distribution plants of the three emergency systems are located in different rooms of the reactor building. The respective cables are routed on different cable routes separated from each other. The diesel generators are in three different, physically separated buildings. Apart from them the following equipment is located in these buildings:

- the compressed-air system with compressor for securing the starter air of the diesel as well as for securing the function of the compressed-air drives of the penetration isolation valves of the respective trains of the safety system,
- the control voltage supply of the diesel generator from a 24-V battery with a capacitiy of 200 Ah,
- the pump station for water supply of the most important consumers of the reactor building of the respective train of the safety system and
- the lubrication system with an oil reserve for 20 days stored in tapping tanks of 5 $m^3$ and 1 $m^3$.

A number of tanks are provided for diesel fuel storage:

- In the emergency power system there is a tapping tank for every diesel generator. This tank has a volume of 15 $m^3$ and ensures the diesel motor operation for seven hours.
- For each emergency power plant there is an underground intermediate tank having a volume of 100 $m^3$ securing diesel motor operation for two days.
- For all three emergency power systems together there is a basic standby storage tank, consisting of two containers having a volume of 500 $m^3$ each, with a common storage capacity of seven days. The necessary pumps are supplied by the general auxiliary power supply system supported by a diesel unit.

The compressed air system per diesel generator for starting the diesel motor consists of two compressed-air bottles, with the content of the bottles capable of six accelerated starts of the diesel motor. Compressed air used is automatically replaced by the compressor unit.

During the selection of the diesel unit the designer determined a maximum peak power demand of 5958 kW and a maximum continuous power demand of 5021 kW. A diesel unit with a continuous power of 6200 kW was chosen.

For the two emergency power transformers of the emergency power plant for feeding the two 0.4-kV busbars of category II, the designer selected two transformers of 1000 kVA each, to meet a total load of 1590 kVA.

For each energy power system, on changeover to emergency power, there is an independent automatic sequence for starting the diesel generator and for the step-wise addition of load. A Diesel generator start is only initiated by undervoltage actuation. Each automatic sequence from the transmitter to signal actuation is at least designed as a 1 out of 2 selection circuit. It ensures:

- that overloads are avoided when the diesel is started by first disconnecting all consumers of category II from the 6-kV and 0.4-kV busbars,
- that the switches to the auxiliary power system are opened,
- that the consumers are connected with a time delay according to a rated program considering boundary process conditions,
- that the operating staff upon failure of the automatic sequence can actuate units when the capacity of the diesel generator has been reached and
- that the operating staff can only shut off units when the actuation criterion for activating the safety system no longer exists.

The emergency power supply of the control element drives is via two separate transformers, through rectifiers and a battery. The emergency power supply of the emergency protection system is realised by a connection to the emergency power systems of the three trains of the safety system.

Each train of the safety control system is fed by the corresponding emergency power system. It can, however, not be concluded from the documents, how this supply is realised in detail.

- Emergency Power Supply of Operational Instrumentation and Control

Two further interruption-free power supply units each with one swithed battery are provided for emergency power supply of the control computer system and function group control (see Fig. 6.5-3). The interruption-free power supply (USV) for the control computer system on the one hand is fed via a separate transformer of the 6-kV unit distribution "BA", but it can, on the other hand, also be manually connected to the 6-kV distribution of the emergency power system I via a standby transformer. It is thus ensured that the amount of information available in the main control room in case of accident is not only illustrated by the conventional secondary equipment, but that the information for large sectors can be provided by the control computer system.

- **Assessment Criterion**

KTA Rules 3701.1, 3702.1, 3703 and 3704 form the basis for the assessment of the emergency power system. The consumers important for the safety of a power station are to be connected to emergency power systems. The emergency power switchyards are always to be kept under such a voltage that the emergency power consumers can obtain the emergency power supply from the emergency power system and, upon failure of this energy supply, from emergency power generating systems. The failure of the auxiliary power supply must be detected by voltage monitoring at every diesel generator busbar as well as by frequency monitoring (as the second actuation criterion). The redundancy of the emergency power generator and distributor systems must correspond to the redundancy of the process systems. The emergency power systems are to be designed in such a way that, after demand for an emergency power system, no manual interferences are required for the operation of the emergency power system for at least 30 minutes. The emergency power operation shall be terminated when the supply from the auxiliary power system is again safely available.

The redundant trains of the emergency power system are to be arranged physically separated from each other or they are to protect each other in such a way that failure-actuating events in one train cannot spread to other trains.

A balance of the effective output for each train-wise arrangement of the diesel units is to be established to determine the efficiency of the diesel motor. A safety margin of at least 10 % must be added to the maximum power determined by the output balances. The compressed-air reserves per diesel unit are to be calculated in such a way that six subsequent automatic starting processes are possible.

For each train, an independent power consumption balance is to be determined for the batteries for interruption-free power supply of emergency power consumers. A safety margin of at least 10 % is to be added. According to a RSK recommendation the discharge time per battery may not be below 2 h.

- **Assessment**

The emergency power supply of the safety system meets the basic requirements of the rules mentioned with respect to redundancy, physical separation and functionality.

There is a consequent physical separation of the switchyard and the three emergency power diesel buildings.

The step-wise loading of the emergency power system with consumers upon voltage failure at the 6-kV emergency power busbar follows one and the same program, independent of the further sequence of the accident. The requirement of KTA Rule 3701 to design the emergency power system assuming the simultaneous failure of the auxiliary power supply with one of the design accidents is thus met.

Since there is no below-frequency actuation of the diesel generator, it should be backfitted (R 6.5-5). It is not possible to switch the electricity supply of the safety system from emergency power back to normal power supply as long as there are still any process-based actuation criteria in effect. Therefore a synchronising device for each diesel generator should be backfitted to make a switch back possible (R 6.5-6).

The concept of the common basic standby storage tank for diesel fuel for all three emergency power system is to be considered. In particular, the failure of the respective fuel pumps upon failure of the respective power supply is negative value.

There is a power consumption balance for the selection of the batteries. But the selection of the batteries as well as the evidence for observing the discharge time of at least 2 h cannot be reconstructed. Evidence should be provided that the discharge time of the batteries is kept > 2 h (R 6.5-7). The designer planned a series of indication, notification and alarm systems as well as a number of protective devices for the electrotechnical installations of the auxiliary power system as well as the emergency power systems. Owing to the insufficient information contained in the documents it can, however, not be assessed whether these installations correspond to the requirements of the KTA Rules. From operating experience in other operational units of the same type (cf. Section 8), it can be derived that the cable and switch concept must be revised in connection with the ensurance of selectivity in case of short circuits (R 6.5-8). The components used in the emergency-power systems must be of approved types (R 6.5-9). As it can be assumed that, as a result of upgrading measures of the safety system, the number of the consumers to be supplied with emergency power will increase, more powerful emergency diesels should be used (R 6.5-10).

An assessment of the earthing and lightning protection is not possible on the basis of the documents available.

### 6.5.5 Summarising Assessment of Electro-Technics

The concept of the physical separation of the three emergency power systems, as well as the separation of the 6-kV auxiliary supply level into four independent busbars, are satisfied. There were no verifications relating to the accident resistance, quality assurance, short circuit resistance and selectivity upon short circuit of he electrotechnical equipment. Inspectability of the emergency power system must be ensured. The individual recommendations are listed in Section 10.

Because of the plurality of recommendations it is proposed to perform investigations to clarify whether the intended electrical technology can be employed in the Stendal NPP or whether it would be better to replace it.

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## Tables, Section 6

## Table 6.1-1 Engineered Safeguards Neccesary for Accident Control

| No. | Accident accord. to | Body of Rules/ Engineered Safeguard | Reactor Scram | HP-boron injection (short-term) | Emergency Cooling System | | | | Sprinkler | Contain-ment Isolation | Pressu-riser Safety Valve | Emer-gency Feed-water | BRU-A | Steam Gene-rator-Safety Valve | Emer-gency Power | Service Water System A | Steam Gene-rator Isolation (SSA) | Venti-lation 20) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | HP-Emer-gency Cooling System | Core Flood-ing Tank | LP-Service Water System | LT 19) | | | | | | | | | | |
| 1 | large leak in primary system | ST-LL I.1.1<br>MA 5.2.16.1-3 | | | + | + | + | | + | + | | | | | + | + | | + |
| 2 | small leak in primary system | ST-LL I.1.2<br>MA 5.2.6.11 | + | | + | | + | + | + 1) | + | | + | + | | + | + | | + |
| 3 | steam generator heater tube leak without loss of off-site power | ST-LL I.1.2<br>MA 5.2.14 | + | | 2) | | (+) | + | | | | | | | | + | + 3) | + |
| 4 | steam generator heater tube leak with loss of off-site powe | ST-LL I.2<br>MA 5.2.14 | + | + 4)<br>+ 5) | 2) | | (+) | + | | | | + 6) | + | | + | + | + 3) | + |
| 5 | steam generator collector break without loss of off-site power | WWER-specific | + | | + | + 1) | + | + | | | | | + 7) | + | | + | + 3) | + |
| 6 | withdrawal of control element or groups, resp. | MA 5.2.1 | + | + 8) | | | | + | | | | | | | | + | | + |
| 7 | erroneous withdrawal or insertion of control element, resp. | MA 5.2.3 | + 9) | | | | | + | | | | | | | | + | | + |
| 8 | ejection of the most effective control element | ST-LL II.1.1<br>MA 5.2.2 | + | | | | | + | | | | | | | | + | | + |
| 9 | reduction of temperature in primary system (e.g. main steam leak) | ST-LL II.1.2<br>MA 5.2.4, 5.2.5 | + | A | A | | | + | | | | | | | | + | A | + |
| 10 | reduction of boron content in primary system | MA 5.2.7 | 11) | + 11) | | | | A | | | | | | | | + | | + |
| 11 | primary system leak outside containment (power operation) | ST-LL I.4.1<br>MA 5.2.16.4, | 12) | | + 13) | | | + | | | | | | | | + | | + |
| 12 | primary system leak outside containment (shut down) | ST-LL II.3 | | | | | + | + | | | | | | | | + | | + |

**Table 6.1-1 Engineered Safeguards Neccesary for Accident Control**

| No. | Accident accord. to Body of Rules/ | Engineered Safeguard | Reactor Scram | HP-boron injection (short-term) | Emergency Cooling System: HP-Emergency Cooling System | Emergency Cooling System: Core Flooding Tank | Emergency Cooling System: LP-Service Water System | Emergency Cooling System: LT 19) | Sprinkler | Containment Isolation | Pressuriser Safety Valve | Emergency Feedwater | BRU-A | Steam Generator-Safety Valve | Emergency Power | Service Water System A | Steam Generator Isolation (SSA) | Ventilation 20) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 13 | primary system leak inside containment (shut down) 14) | WWER-specific | | | | | + | + | A | A | | | | | | + | | + |
| 14 | main steam leak in containment | ST-LL I.3.1<br>MA 5.2.15. | + | A | A | | | + | + | + | | | | | | + | + 3) | + |
| 15 | main steam leak outside containment before SSA | ST-LL I.3.1.II.4<br>MA 5.2.15.2 | + | A | A | | | + | | | | + 6) | + | | + 15) | + | + 3) | + |
| 16 | main steam leak outside containment behind SSA | ST-LL I.3.1<br>MA 5.2.15.1 | + | | A | | | + | | | | + 6) | + | | + 15) | + | + | + |
| 17 | No. 16 and steam generator tube rupture | ST-LL I.3.2<br>MA 5.2.15 | + | + 4)<br>+ 5) | + 2) | | | + | | | | + 6) | + | | + 15) | + | + | + |
| 18 | No. 14 (or steam generator safety valve open) and steam generator tube rupture | ST-LL I.3.2<br>MA5.2.13.3,5.2.15 | + | + 4)<br>+ 5) | + 2) | | | + | | | | + 6) | | | + 15) | + | + 3) | + |
| 19 | break of feedwater line | MA 5.2.12.4 | + | + | | | | + | A | A | + | A | | | | + | + 3) | + |
| 20 | break of emergency feedwater line | MA 5.2.12.5 | + | + | | | | + | ? | ? | + | A 17) | | | | + | + 3) | + |
| 21 | loss of off-site power | ST-LL II.2<br>MA 5.2.11 | + | + | | | | + | | | | + | + | + | + | + | | + |
| 22 | failure of several main coolant pumps | MA.5.2.10.2 | + | + 8) | | | | + | | | | | | | | + | | + |
| 23 | seized shaft of a main coolant pump | MA 5.2.10.4 | + | + 8) | | | | + | | | | | | | | + | | + |
| 24 | failure of feedwater pump | MA 5.2.12.1 | + | | | | | + | | | | + 6) | | | | + | | + |
| 25 | turbine trip without main steam bypass station (BRU-K) | MA 5.2.8.2 | + | | | | | A | | | | | + | + 18) | | A | | A |
| 26 | failure of main heat sink | MA 5.2.9 | + | | | | | A | | | | | + | + | | A | | A |

## Table 6.1-1 Engineered Safeguards Neccesary for Accident Control

| No. | Engineered Safeguard / Accident accord. to Body of Rules/ | | Reactor Scram | HP-boron injection (short-term) | Emergency Cooling System: HP-Emer-gency Cooling System | Emergency Cooling System: Core Flood-ing Tank | Emergency Cooling System: LP-Service Water System | Emergency Cooling System: LT 19) | Sprinkler | Contain-ment Isolation | Pressu-riser Safety Valve | Emer-gency Feed-water | BRU-A | Steam Gene-rator-Safety Valve | Emer-gency Power | Service Water System A | Steam Gene-rator Isolation (SSA) | Venti-lation 20) |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 27 | primary system pressure increase (e.g. pressuriser heating) | MA 5.2.6.2 | A | A | | | | A | | | A | | | | | A | | A |
| 28 | ATWS (anticipated transients without scram) | MA 5.2.17 | | + 8) | | | | + | A | A | + | | | | | + | | + |

**Remarks**

1) dependent on leak size
2) acutuated automatically, must be switched off manually later
to render presssure reduction by spraying possible.
3) also on feedwater side
4) spraying function necessary, at present not possible, impossible owing to lack of line
5) spraying function with the make-up system necessary and possible, but emergency
power supply by diesels 4 and 5 not resistant to external impacts
6) auxiliary feedwater function possible, but pumps and emergency power supply by
diesels 4 and 5 not resistant to external impacts
7) BRU-A concerned not designed for discharge of water, cannot be isolated
8) automatic actuation required
9) reactor scram necessary because of power density is locally exceeded, no actuation criterion
10) on the long-term required for any accident, if no other system can be used for boron injection
11) automatic or manual reactor scram, resp., required, in the shutdown state
HP-emergeney boron injection is required
12) depending on the size of the leak automatic or manual actuation , resp.
13) make-up system suffices for a very small leak, but emergency power
supply by diesels 4 and 5 not resistant to external impacts
14) break of suction line of long-term cooling
15) simultaneous loss of off-site power
16) break between steam generator and gate valve in containment
17) isolation of the emergency feedwater train necessary
18) as reactor scram is not actuated by turbine trip and BRU-A opens slowly
19) LT - long-term heat removal via cooldown line of LP-emergency water system
20) cooling of the components of the engineered safeguards

ST-LL Accident Guidelines/SLL 83/
? unknown
A accident-dependent
MA Lists of Notes
HP high-pressure
\+ required
NP low-pressure

**Table 6.4-1 Actuation Criteria for Reactor Scram**

| No. | Actuation Criterion | Parameter | |
|---|---|---|---|
| 1 | Period in source range | $T_{SR}$ | ≤ 10 s |
| 2 | Period in energy range | $T_{ER}$ | ≤ 10 s |
| 3 | Neutron flux density in source range | $N_{\Phi SR}$ | ≥ $N_{\Phi\ presetSR}$ |
| 4 | Neutron flux density in energy range | $N_{\Phi ER}$ | ≥ 107 % $N_{\Phi presetER}$ |
| 5 | Neutron flux density in energy range | $N_{\Phi ER}$ | ≥ $N_{\Phi preset\ value}$ |
| 6 | Neutron flux density in energy range upon shutdown of 1 out of 4 main coolant pumps | $N_{\Phi ER}$ | ≥ 75 % $N_{nom}$ after 50 s after shutdown of main coolant pump |
| 7 | Neutron Flux Density in energy range upon shutdown of 1 out of 3 main coolant pumps, operation of the 2 main coolant pumps located opposite each other | $N_{\Phi ER}$ | ≥ 60 % $N_{nom}$ after 50 s after shutdown of main coolant pump |
| 8 | Like No. 7, but operation of the 2 adjacent main coolant pumps | $N_{\Phi ER}$ | ≥ 50 % $N_{nom}$ after 50 s after shutdown of main coolant pump |
| 9 | Pressuriser level | $H_P$ | ≤ 4600 mm |
| 10 | Pressure reduction in steam line, difference between saturation temperature of the primary and secondary system | $P_{MS}$<br>and<br>$\Delta t$ | ≤ 4.9 MPa<br><br>≥ 75 K |
| 11 | Pressure difference at main coolant pump | $\Delta P_{MCP}$ | from 0.39 MPa to 0.25 MPa within 5 s |
| 12 | Voltage failure main coolant pump 1 of 2, or<br>2 of 3 at N > 5 % $N_{nom}$ with Tv = 1.4 s<br>2 of 4 at N > 75 % $N_{nom}$ with Tv = 6.0 s | | |
| 13 | Pressure above reactor core | P<br><br>$N_R$<br>P<br>$t_{PS}$ | ≤ 14.7 MPa<br>at<br>≥ 75 % $N_{nom}$<br>≤ 13.72 MPa at<br>≥ 260 °C |
| 14 | Main steam pressure | $P_{MS}$ | ≥ 7.84 MPa |
| 15 | Earthquake | | ≥ Size 6 (MSK) |
| 16 | Level in a steam generator with main coolant pump in operation | $h_{SG}$ | ≤ H-650 mm nominal level in one of the four steam generators |

| No. | Actuation Criterion | Parameter | | |
|---|---|---|---|---|
| 17 | Drop of frequency in 3 of 4 unit distributions of the main coolant pump supply | f | ≤ | 46 Hz |
| 18 | Excess pressure in containment | P | ≥ | 0.029 MPa (excess pressure) |
| 19 | Pressure in primary system | $P_{PS}$ | ≥ | $P_{PS}$ 17.64 MPa |
| 20 | Difference between saturation temperature in the primary system and temperature in the hot train | Δt | ≤ | 10 K |
| 21 | Temperature in one of the hot trains | t | ≥ | $t_{nom}$ + 8 °C |
| 22 | Failure of the control and protection injection 2 of 3 | | | |
| 23 | Actuation switch HS in main control room or emergency control room | | | |
| 24 | Failure high-voltage injection control and protection system, 2 entries with Tv = 3 s | | | |
| 25 | Failure of d.c. injection on field "PAK", control and protection system | | | |

**Table 6.4-2 Automatic Actuation Criteria of the Safety System**

| Component | Automatic Actuation Criteria |
|---|---|
| HP-emergency boron injection pump TQ 14 (24, 34) | - Loss of off-site power: Initiation after start of the diesel generator with Tv = 5s, but only if $t_{PS}$ > 70 °C (recirculation operation)<br>- no process criteria |
| HP-emergency cooling pump TQ 13 (23, 33) | - Loss of off-site power: Initiation after start of the diesel generator with Tv = 5s, but only if $t_{PS}$ > 70 °C<br>- One of the 6 process actuation criteria |
| LP-emergency cooling pump TQ 12 (22,32) | - Loss of off-site power: Initiation after start of the diesel generator with Tv = 5s, but only if $t_{PS}$ > 70 °C<br>- Loss of off-site power: Initiation when diesel generator has been switched on, when the 6-kV emergency power busbar is energised and $t_{PS}$ < 70 °C<br>- One of the 6 process actuation criteria |
| Service water cooling pump VF 10 (20, 30), [QF 11 (21, 31)] | - like TQ 12, but Tv = 10 s |
| Containment-spray pump TQ 11 (21, 31) | - like TQ 13, but Tv = 30 s |
| Emergency feedwater pump | - like TQ 13, but Tv = 40 s |
| Penetration isolation valves | - one of the first two process actuation criteria |

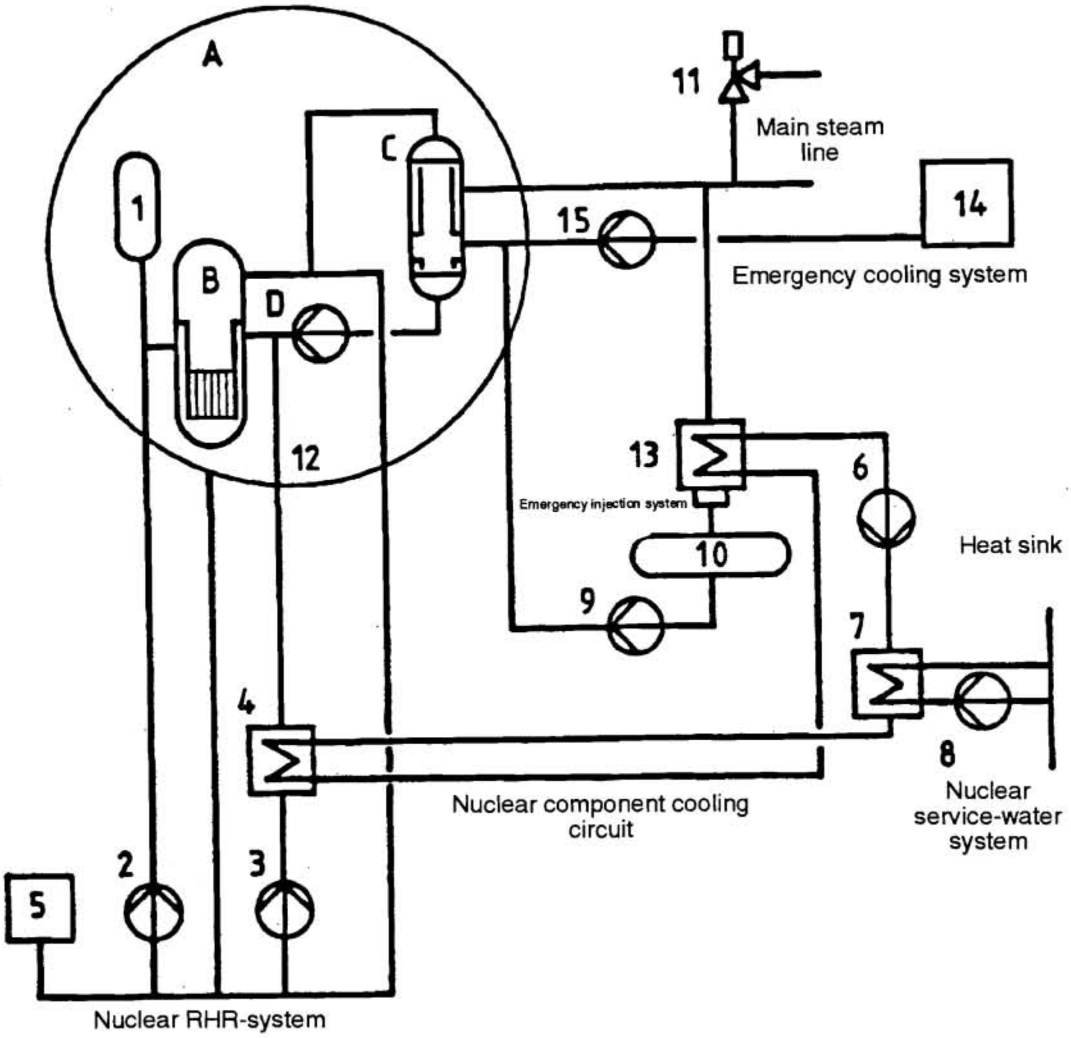
**The six process actuation criteria:**

- Pressure in the containment Pc > 0.129 MPa
- Difference between saturation temperature of the coolant in the primary system (PS) and the maximum temperature of the coolant in one of the four loops of the PS $\Delta t$ <10 K
- Difference between saturation temperature of the primary system (PS) and the saturation temperature of the water in the steam generator (SG) 1 or SG 2 $\Delta t$ > 75 K at a main steam pressure of $P_{MS}$ < 4.9 MPa
- Difference between saturation temperature of the PS and the saturation temperature of the water in the SG 3 or SG 4 $\Delta t$ > 75 K at $P_{MS}$ < 4.9 MPa
- Pressure change rate in SG 1 or SG 2 > 0.149 MPa/s at $P_{MS}$ < 5.1 MPa
- Pressure change rate in SG 3 or SG 4 > 0.149 MPa/s at $P_{MS}$ < 5.1 MPa

The two latter actuation criteria in more recent projects have been replaced by the following actuation criterion:

- Low level in the pressuriser and low main steam pressure

## Figures, Section 6



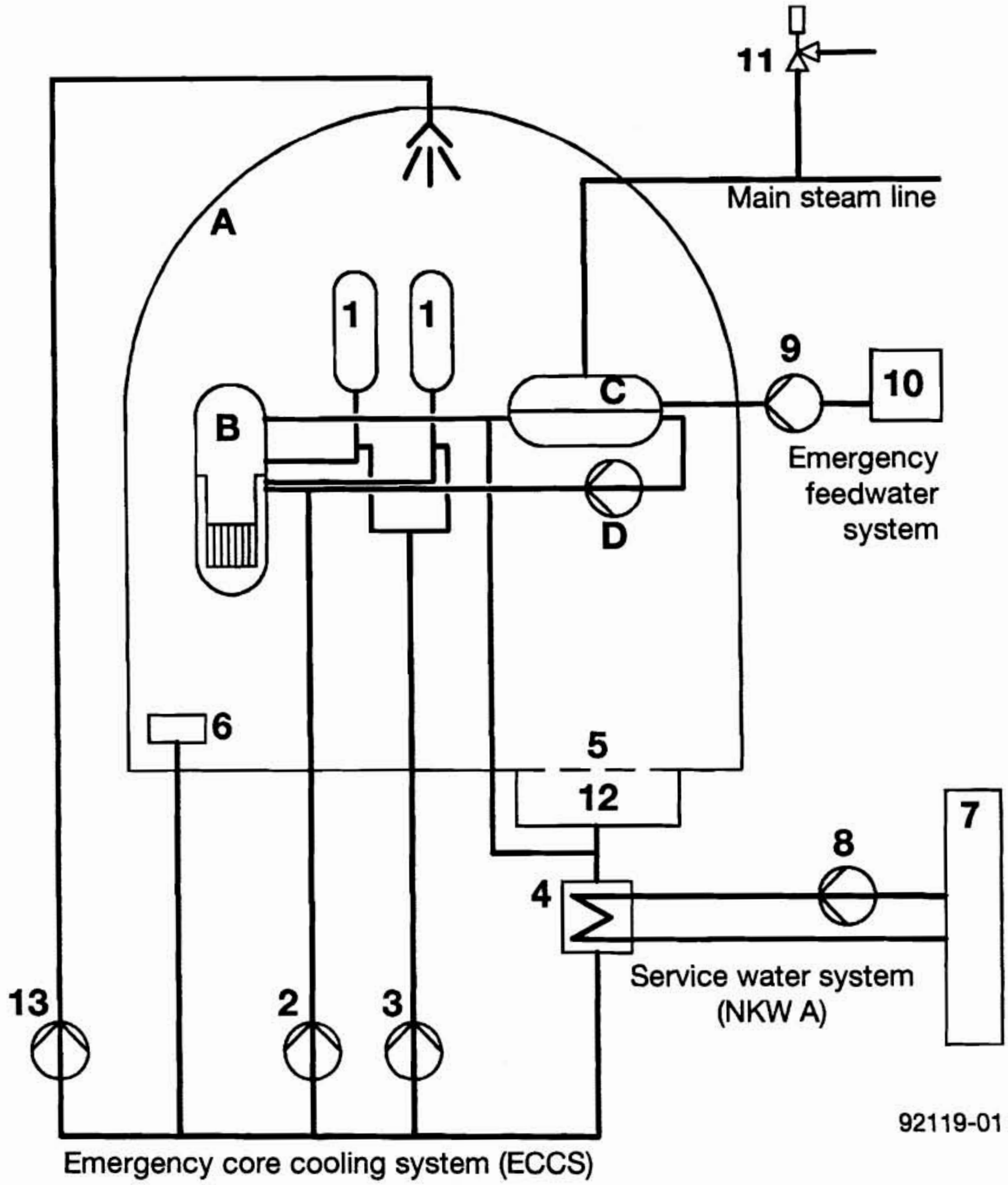

1 Accumulator
2 Safety injection pump
3 RHR-pump
4 RHR-cooler
5 Borated-water tank
6 Closed cooling water pump
7 Component cooler
8 Service-water pump
9 Emergency feedwater pump
10 Emergency feedwater tank

11 Steam dump station
12 Containment sump
13 Emergency condensator
14 Condensate tank
15 Condensate pump

---

A Containment
B Reactor pressure vessel
C Steam generator
D Main coolant pump

**Fig. 6.3-1** Simplified illustration of the residual heat removal systems of the Fed. German nuclear power plant with PWR according to KTA 3301




**1** Accumulator
**2** High pressure safety injection pump
**3** Low pressure safety injection pump
**4** Emergency cooler
**5** Containment sump
**6** Storage tank for concentrated boric acid
**7** Cooling pond
**8** Service water pump
**9** Emergency feedwater pump
**10** Emergency feedwater tank
**11** Steam dump station (into atmosphere)
**12** Boric acid storage tank
**13** Containment spray pump

**A** Containment
**B** Reactor pressure vessel
**C** Steam generator
**D** Main coolant pump

**Fig. 6.3-2** Engineered safeguards

**Hier muß von der Druckerei das ausklappbare Bild 6.3-3**

**(Deutsche Version) eingefügt werden!!!**

**Fig. 6.3-3** Stendal NPP, Primary system with adjacent engineered safeguards

**Rückseite des ausklappbaren Bildes 6.3-3**

**(Deutsche Version)**

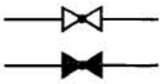
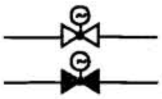
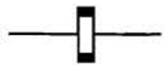
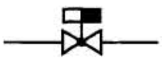
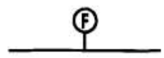
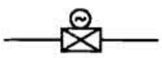
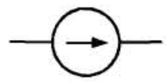
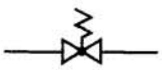
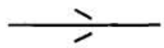
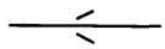
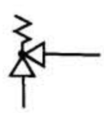
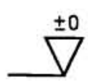
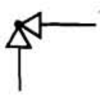
**Hier muß von der Druckerei das ausklappbare Bild 6.3-4**

**(Deutsche Version) eingefügt werden!!!**

**Fig. 6.3-4** Stendal NPP, Main steam and feedwater system with adjacent engineered safeguards

**Rückseite des ausklappbaren Bildes 6.3-3**

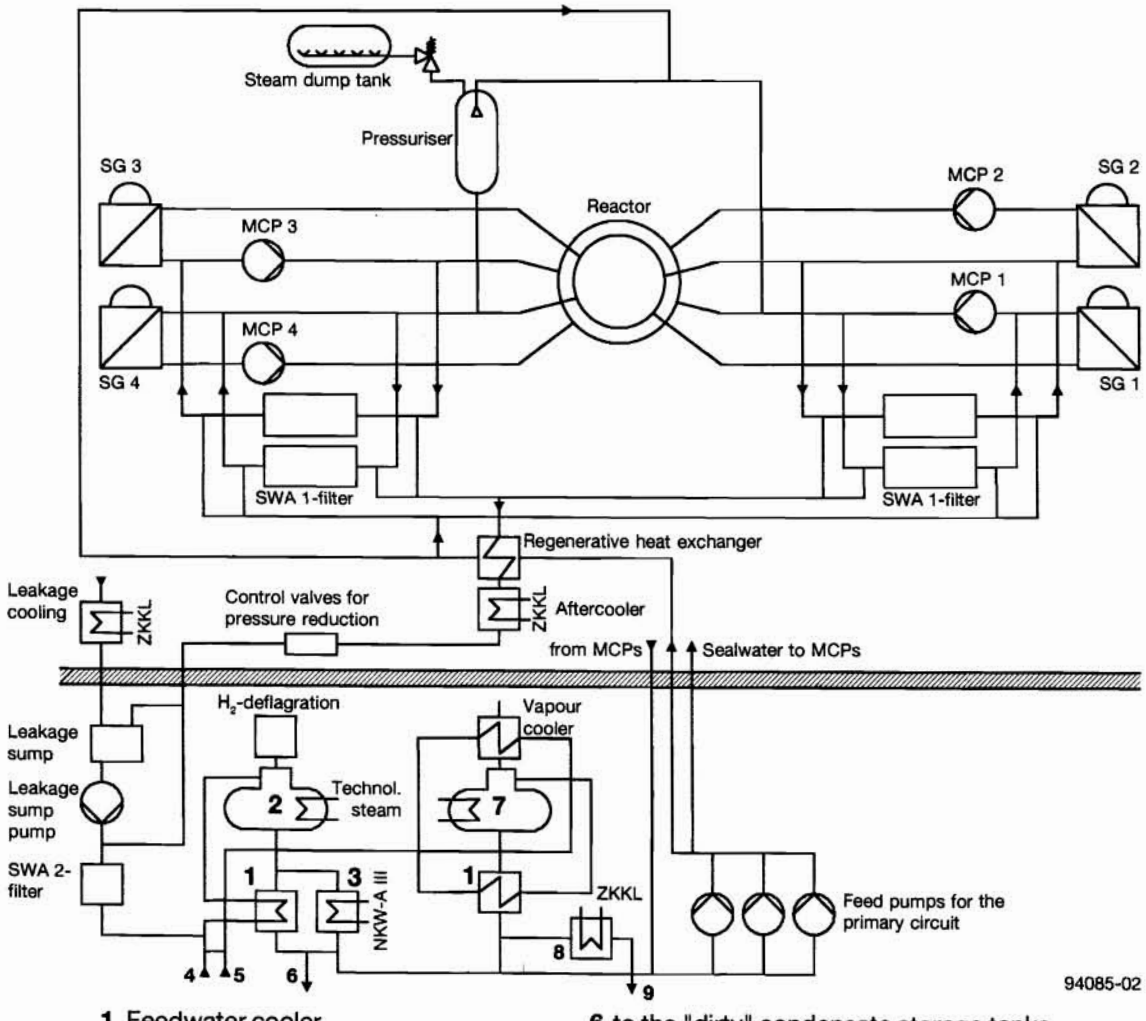
**(Deutsche Version)**

| Symbol | Explanation | Symbol | Explanation |
|---|---|---|---|
|  | Open/closed hand-operated valve |  | Ventilation on unpressurised tanks |
|  | Open/closed valve with electric drive |  | Throttling orifice |
|  | Closed, lock-secured hand-operated valve |  | Discharge limiter |
|  | Fast-closing pneumatically controlled valve |   | Flow meter |
|  | Control valve with electric drive |  | Pump |
|  | Control valve in the secondary circuit |  | Pipe narrowing |
|  | Check valve |  | Pipe widening |
|  | Pressure reducer |   | Room number |
|  | Spring-loaded safety valve |   | Height |
|  | Pressuriser safety valve | | |

NB:
Folded diagrams show valve positions during power operation

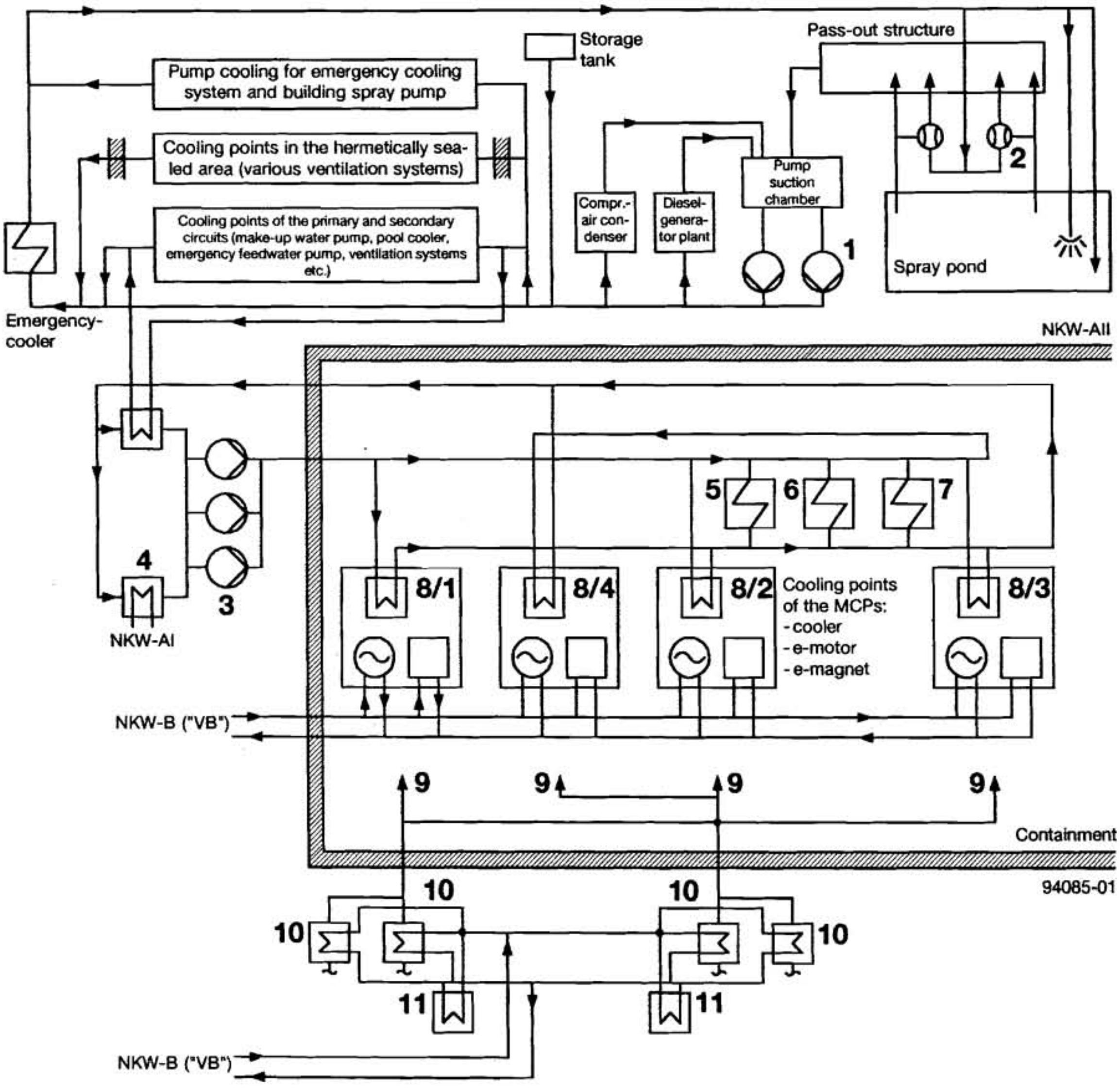
94085-08

**Fig. 6.3-5** Legend of the symbols used in Fig. 6.3-3 and 6.3-4



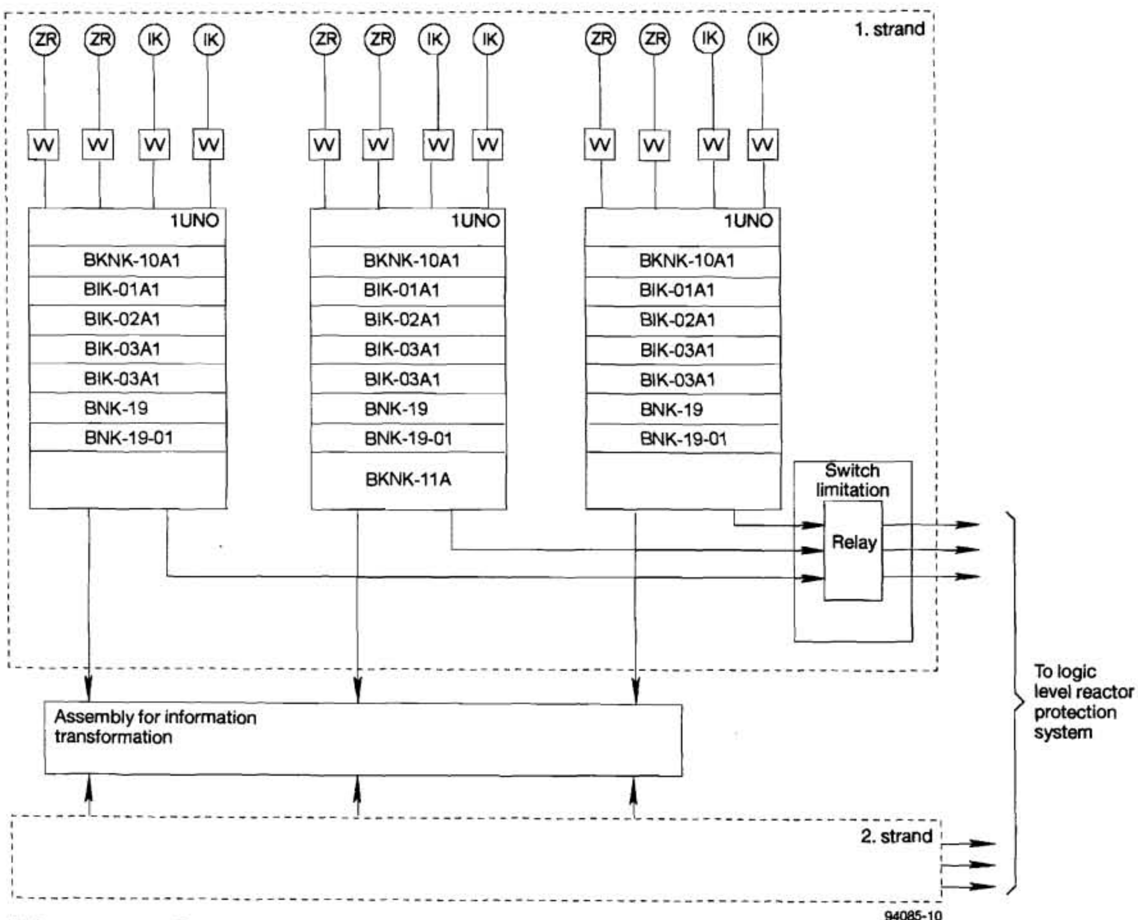

1 Feedwater cooler
2 Make-up water deareator
3 Feedwater aftercooler
4 from the fill-up pumps
5 "clean" condensate
6 to the "dirty" condensate storage tanks
7 Boron control deareator
8 "clean" condensate aftercooler
9 to the "clean" condensate storage tanks

**Fig. 6.3-6** Make-up system




**1** Service-water pumps
**2** Water jets for siphon pipes
**3** Pumps of the component cooling circuit
**4** Coolers of the component cooling circuit
**5** Leakage cooler
**6** Steam dump tank cooler
**7** SWA 1 aftercooler
**8** /1-4 cooling points of the MCPs
**9** Cooling points of the MCPs
**10** Oil cooler
**11** Oil-vapour cooler

**Fig. 6.3-7** Service water system A, consumers of the service water system B and the component cooling system of the reactor building (ZKKL)




ZR = Counting tube
IK = Ionisation chamber
W = Preamplifier
1UNO = NFMS-cabinet
BKNK-10A1 = Calibrating assembly
BIK-01A1 = QB-channel (start-up range)
BIK-02A1 = ZB-channel (transition range)
BIK-03A1 = EB-channel (power range)
BNK-19 = Power supply assembly
BNK-19-01 = Power supply assembly
BKNK-11A = Comparator assembly of the EB-instrument channels

**Fig. 6.4-1** Structure of the neutron flux measurement system

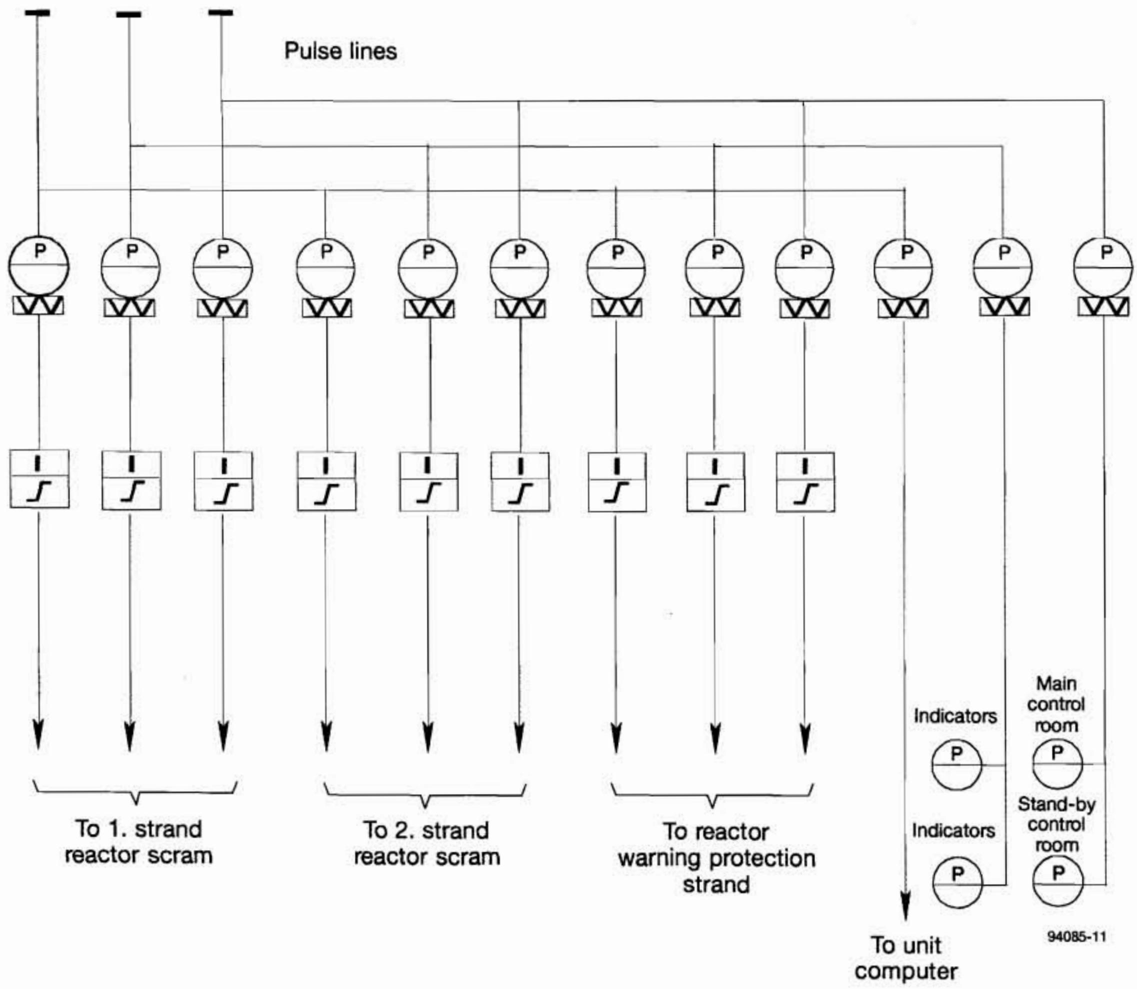
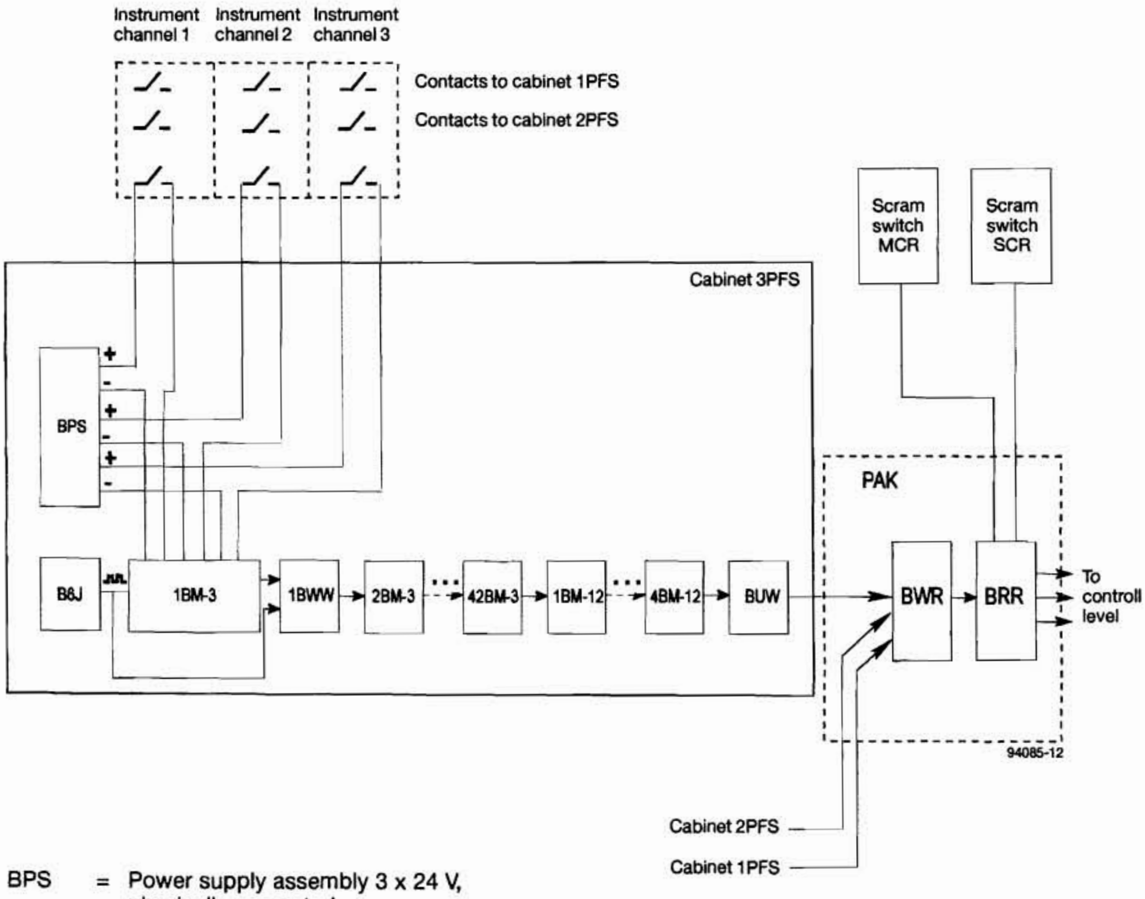



**Fig. 6.4-2** Emergency protection system: Connection principle of the pulse lines- transmitter




BPS = Power supply assembly 3 x 24 V, physically separated
BGJ = Impulse generator
BM-3 = 2-out-of-3 selection assembly
BWW = Time delay assembly
BM-12 = 2-out-of-4 or 3-out-of-4 selection assembly
BUW = Output amplifier assembly
BWR = Relay assembly
BRR = Contact multiplier assembly

**Fig. 6.4-3** Block diagram of a cable of a train of the logic level for reactor scram

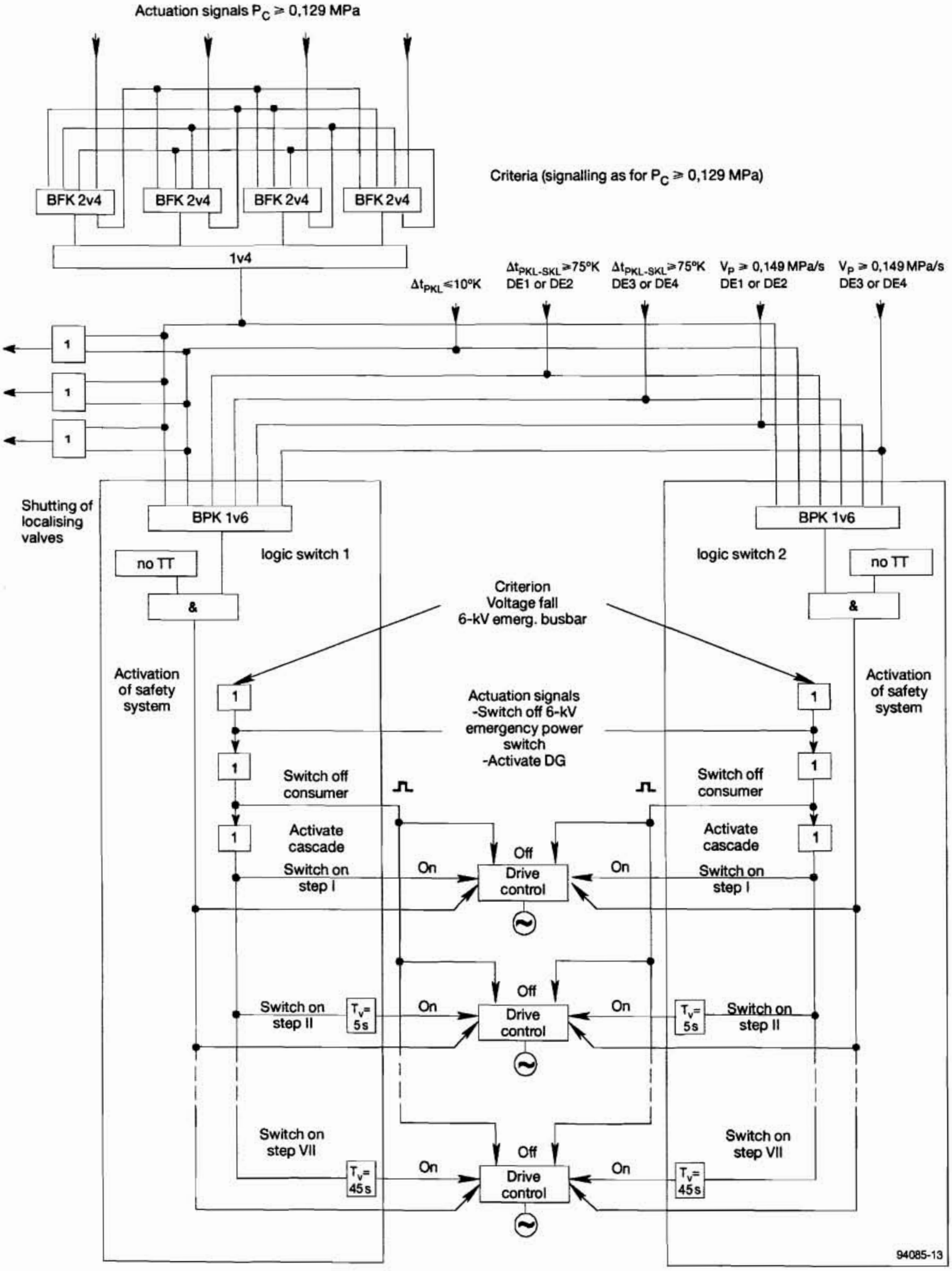



**Fig. 6.4-4** Block diagram of a train of the safety system

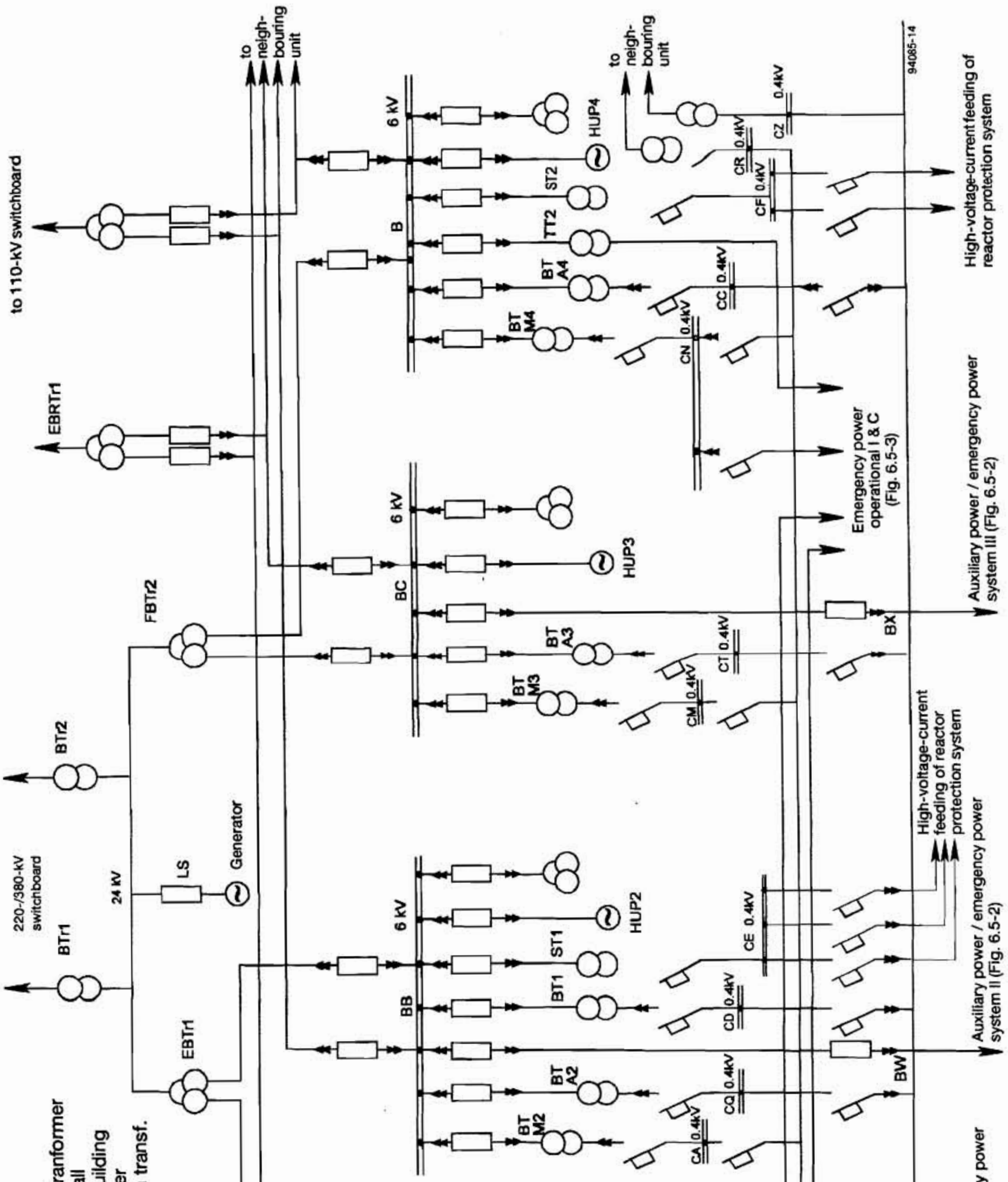



**Fig. 6.5-1** Survey diagram auxiliary power supply, 6-kV and 0.3-kV levels, turbine hall and reactor building (without general auxiliary supply)

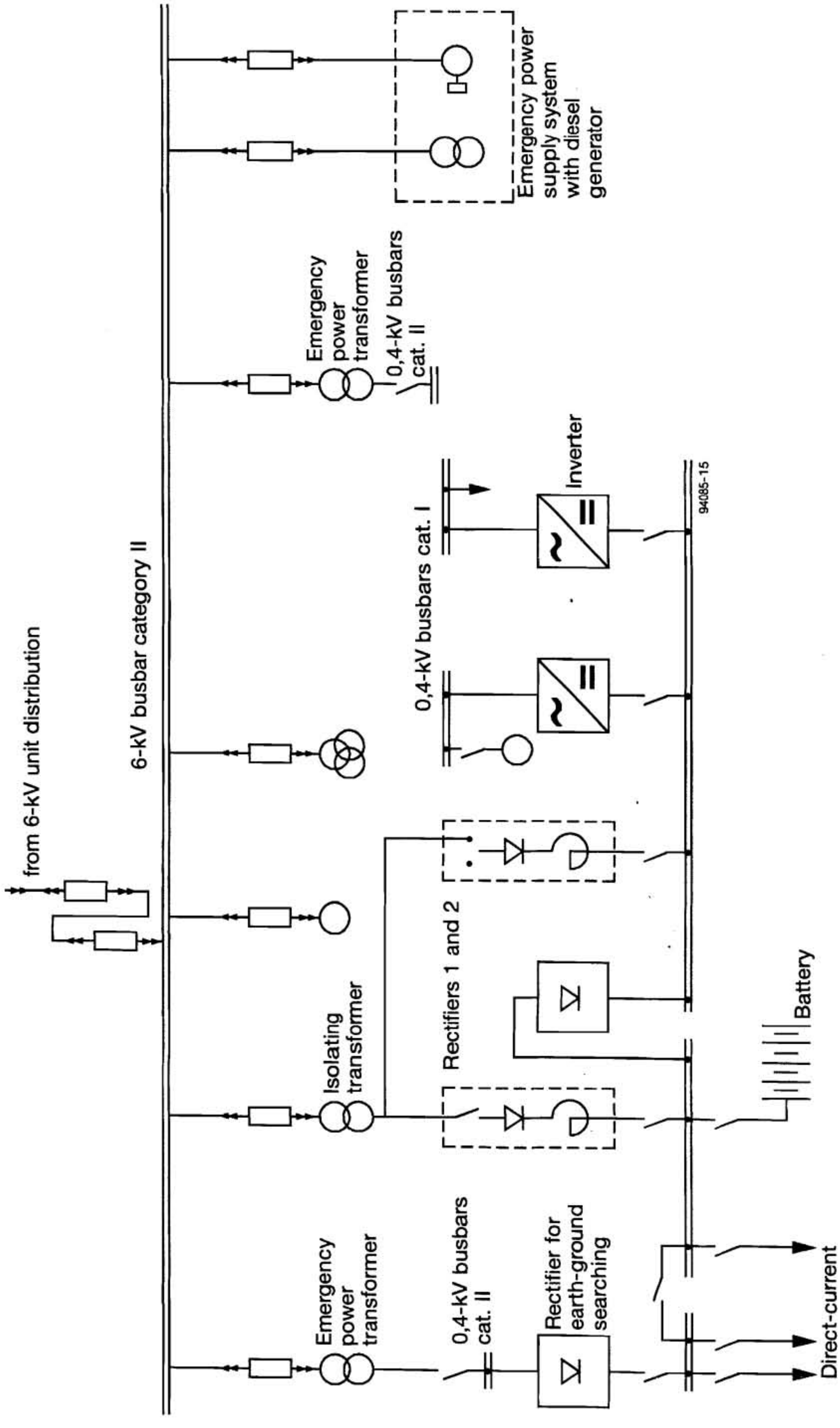



**Fig. 6.5-2** Survey diagram of emergency power supply of a train of the safety system

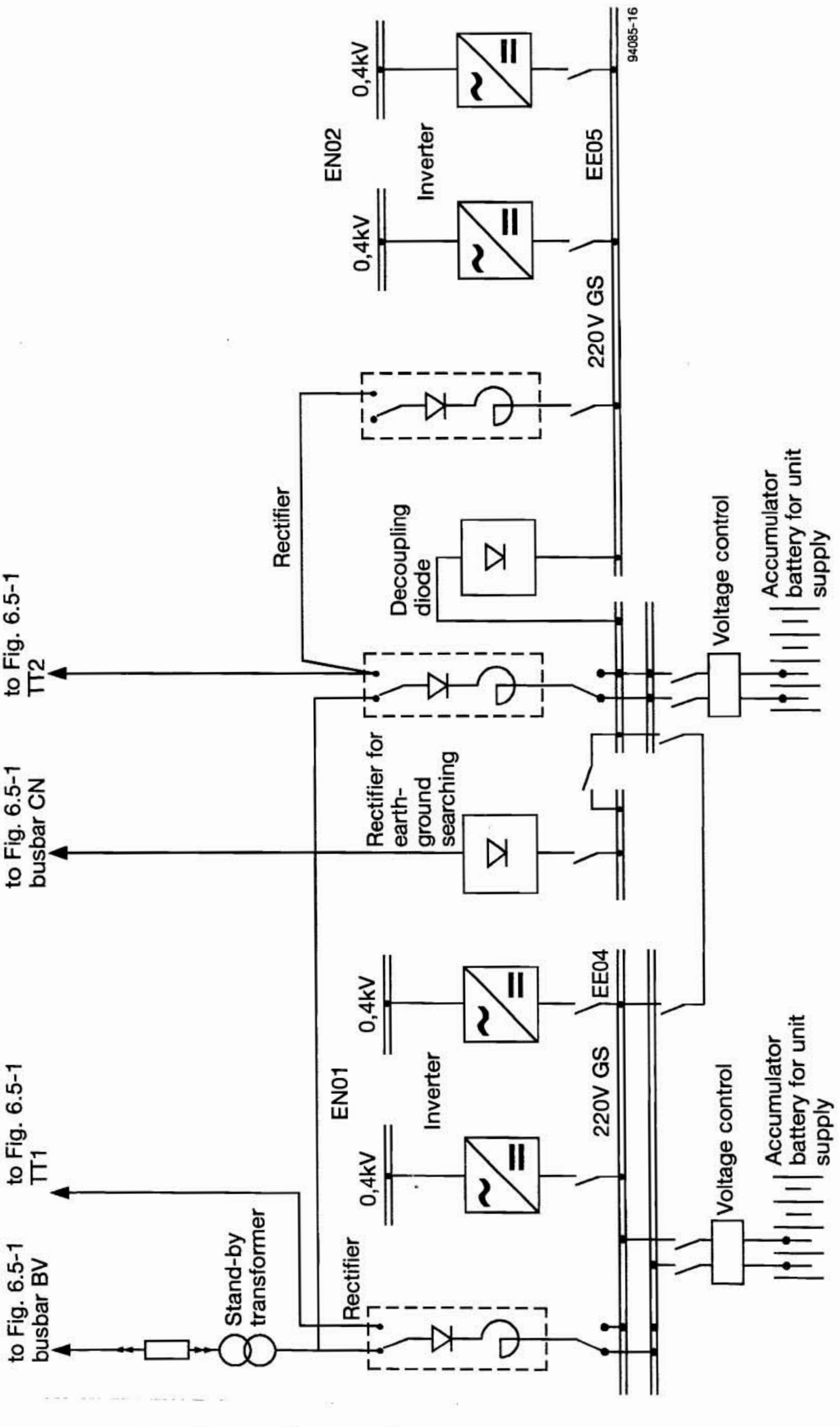



**Fig. 6.5-3** Survey diagram of the emergency power supply of the operational instrumentation and control

# 7 Civil Engineering Aspects, Spreading Impacts, Radiation Protection

## 7.1 Civil Enginering Aspects

### 7.1.1 The Reactor Building

#### 7.1.1.1 Design

The reactor building, see Fig. 5.2-1, at a height of -4.2 m is built on a jointless foundation of about 65 m x 65 m. It consists of a base, the containment and the outer surrounding of the containment built thereon. The base floor is finished off with the base plate of the containment at a height of +13.2 m. The cylindrical containment is built centrally symmetric on the base plate and finishes off with a dome-shaped ceiling. The outer surrounding of the containment with the outer dimensions 65 m x 65 m to a height of about +51 m also begins at a height of +13.2 m.

The containment was designed as a composite steel cell structure and represents a prototype. In Section 7.1.6 the composite steel cell construction technique will be dealt with in more detail. According to /TEP 81/ the containment has the following dimensions:

| | |
|---|---|
| - height of the cylindrical part | 37.4 m |
| - inside diameter of the cylindrical part | 45.0 m |
| - inside radius of the hemispheric | |
| - dome-shaped ceiling | 22.5 m |
| - wall thickness of the cylindrical | |
| - part and the dome-shaped ceiling | 1.2 m |
| - total height above surface | 74.3 m |

Neither the inside concrete internals nor the outer surrounding of the containment are connected with the cylindrical containment wall. A supplementary part of the containment is an L-shaped room, at the same time serving as emergency boron tank and sump, located below the containment base plate. Three square openings of 1 $m^2$

each in the base plate at a height of + 13.2 m connect this part with the containment /SIE 90/.

Two transfer canals, the main transfer canal at + 36.6 m and the emergency transfer canal at +19.2 m provide access to the containment from the surrounding outer building. Additionally there is a transport hatch in the containment base plate, which is positioned above the track corridor.

With the above dimensions, the gross volume of the cylindrical part and the dome-shaped ceiling of the containment, i.e. without subtraction of inside concrete internals and components is about 83,300 $m^3$.

#### 7.1.1.2 Function of the Containment

The containment (containment vessel) is designed as a single-shell full pressure containment. In particular the components of the primary system and the spent-fuel pools are located in the containment. The function of the containment is to work as a hermetic barrier, even in case of accident with releases from the primary system into the atmosphere. The containment has to withstand the internal pressures and temperatures occuring in such a case and it has to observe the specified leak rate. The containment-spray system to limit pressure and temperature or decrease them in the long-term during loss-of-coolant accidents, is installed in the containment. In addition, the containment has to absorb all external pressures.

#### 7.1.1.3 Requirements to be met by the Containment

The requirements to be met by a safety confinement in the Federal Republic of Germany are published in the Safety Criteria for Nuclear Power Plants of the BMI. In Section 8 of these Safety Criteria

- the function of the safety confinement in a nuclear power reactor
- the design principles of the safety confinement
- leaktightness examinations of the containment vessel
- the penetrations through the containment vessel and
- the heat removal from the safety confinement

are dealt with.

The system consisting of containment vessel and surrounding building, as well as auxiliary systems for retaining and filtering possible leakages of the containment vessel, is referred to as the safety confinement.

According to BMI Criterion Section 8.1, the nuclear power plant must have a safety confinement which can fulfil its function with respect to safety, in particular under accident conditions. Parts of the plant containing radioactive substances must be accommodated within the safety confinement, if an inadmissible release of radioactive substances into the environment cannot sufficiently reliably be prevented by other means. In particuar, the primary coolant system of the reactor plant under high pressure must principally be accommodated in the containment vessel. Sections of the main steam and feedwater lines as well as other lines can be excepted from this rule, if this proves to be necessary from a technical point of view and if it is ensured that their break will not lead to an inadmissible radiation exposure in the environment. A reliable and sufficiently fast isolation of the penetrations must be ensured by the containment vessel.

Refer to Section 6.3 for the assessment of safety technology.

#### 7.1.1.4 Load Assumptions for the Containment

- **Load Assumptions for Internal Loads and Examinations**

- **Preset Assumptions**

Considering the current GDR guidelines, the load assumptions for the design of the containment were principally predetermined by the Soviet general project engineer. This does not only apply to internal loads, but to all loads /BAK 85/.
According to /BAK 85/ and /TEP 81/ the design is based on the following internal loads:

- for the severe accident (2A-break of a main coolant line):
  internal pressure $P_a = 500$ kPa
  temperature $T_a = 150$ °C
  with a linear increase from normal conditions in 10 s and an impact duration of 10 hours; thereafter a pressure decrease to 100 kPa (for radiological reasons)

- for abnormal operation:

  internal pressure $P_b$ = 170 kPa

  temperature $T_b$ = 90 °C

  This load can occur up to 100 times during service life.
- preset minimum pressure after the severe accident $P_c$ = 50 kPa
- for pressure test, 1.15 x internal pressure of the severe accident
- for examination of the leakage rate, the internal pressure of the severe accident.

The performance of first pressure examinations of the safety enclosure of concrete steel and prestressed concrete for nuclear power plants according to DIN V 25 459 have the primary purpose of proving leaktightness. A pressure test as a static load test for the supporting framework of concrete, steel or prestressed concrete, including the liner and its anchoring arrangements, is not required according to DIN V 25 459.

In Federal German practice, however, pressure tests as static load tests have been performed for a prestressed concrete reactor pressure vessel as well as for a prestressed containment. These tests were combined with an examination of whether the specified leak rate was observed. The static load tests were conducted with the design pressure multiplied by about 1.05.

- **Assessment**

The above pressures and temperatures for the types of burden mentioned - severe accident, abnormal operation and calculated sub-pressure - are assessed under consideration of the Federal German regulations in Section 5.2.

The intended test pressure for the containment of the Stendal NPP corresponds to the previous Federal German practice for prestressed containments and thus meets the requirements of the German codes and standards.

According to DIN 25 436 and to the RSK Guidelines for pressurised water reactors the first examination for determining the leak rate has to be carried out with the design pressure. For the containment of the Stendal NPP this examination is also planned with the design pressure, thus here too the requirements of the Federal German codes and standards are met.

- **Jet Forces and Reaction Forces**

- **Preset Assumptions**

To protect the containment against jet forces and reaction forces according to /TEP 81/ the following measures are provided:

- At positions where the containment can be damaged by flying objects or other mechanical impacts, the containment wall is protected by protective walls (concrete steel walls or steel claddings), layers of concrete or similar structural measures.
- All large pipes, including the ones from the primary system DN 850, are equipped with whip restraints preventing a rupture during accidents and impacts on the building construction.
- The protection of pipe and cable racking components (penetrations) in the containment wall against flying objects, jet forces and other impacts normally are concrete steel protection walls, which additionally have a biological protective function and which render maintenance and inspections possible.

According to /SIE 90b/ and contrary to the statements in /TEP 81/ whip restraints to absorb reaction forces are only arranged directly at the main coolant lines. This is justified with the Soviet transition to a leak-before-break concept for connection lines to the primary system as well as for main steam and feedwater lines.

- **Assessment**

The leak-before-break concept, apart from suitable leak detection facilities, in particular requires fulfilment of the requirements of material selection, manufacture, examination etc. according to the concept of the basically safe design according to the RSK Guidelines. A proof with respect to the preconditions of this concept has, however, not been presented by the Soviet project engineer.

Break philosophy and the pipe whip limitation concept are commented on in Section 4.2 (R 4.2-13 and R 4.2-21).

No evidence has been provided by the manufacturer on the protection against impacts arising from explosions and jet forces on safety-relevant plant components in the containment. Calculations referring to jet and reaction forces were only performed in /SIE 90b/ for a 0.1A-leak, permissible with a basically safe design. If the necessary preconditions for a basically safe design are not fulfilled, a full break of the pipe concerned will have to be assumed for the calculation of jet and reaction forces (R 7.1-1).

No documents on the determination of the leak size as a function of the design features, the calculation procedures for jet and reaction forces and the accommodation of forces on tube holding devices, walls and ceilings were provided for assessment by the project engineer. Definite statements according to RSK Guideline 5.1(5) thus are currently not possible. In case of an inadequate design, the arrangement of protective walls, the strict physical separation of the safety systems and possibly the exchange of pipes remain as additional protective measures.

- **Load Assumptions for External Loads**

According to /BAK 85/ and /TEP 81/

- snow loads
- wind loads
- earthquake
- loads from external blast waves
- loads resulting from airplane crash

are to be considered as external loads.

Furthermore,

- maximum frost penetration depth
- maximum calculated outside temperature and
- maximum calculated ground water level

are to be taken into account.

- **Preset Assumptions**
  - wind and snow loads

According to /TEP 81b/ the load assumptions are:

| | |
|---|---|
| - normal load from snow | 700 $N/m^2$ |
| - standard impact pressure from wind for a height of 10 m above the ground | 550 $N/m^2$ |
| - overload factors | |
| for snow | 2.0 |
| for wind | 2.5 |

- earthquake loads

According to /TEP 81/, for the Stendal location it is assumed:

| | |
|---|---|
| - Intensity 5 | for a frequency of occurence of $10^{-2}$/year as design earthquake |
| - Intensity 7 | for a frequency of occurence of $10^{-4}$/year as maximum calculated earthquake |

According to /HAB 83/ the following ground accelerations at the foundation level are assigned to the intensities determined for the location:

| | |
|---|---|
| - Intensity 5 | $a_0 = 0.60$ $m/s^2$ (design earthquake) |
| - Intensity 7 | $a_0 = 1.30$ $m/s^2$ (maximum calculated earthquake) |

For the calculation of the containment the simultaneous effects of two horizontal components vertically directed at each other and one vertical component are to be taken into account.

More detailed information on the calculation of impacts resulting from earthquakes is given in a general form only. There are no documents indicating which procedure was used at the Stendal NPP.

- Loads from External Blast Waves

For the design of the building structures, loads from external blast waves are assumed according to /BAK 85/ and /TEP 81/. The following load-time function is preset:

- Linear increase of excess pressure within 0.1 s to 67.4 kPa, subsequent decrease of excess pressure to 30 kPa within a period of 0.1 s, and then excess pressure remaining at 30 kPa for a further 0.8 s.

- Loads from Airplane Crash

/BAK 85/ and /TEP 81/ define the loads from airplane crash as the impact of an airplane with a mass of $10^4$ kg and a speed of 750 km/h. In a guideline on the inclusion of extraordinary external impacts for special building structures of the NPP construction /HAB 83/ further statements are made. According to /BAK 85/ the impact area is assumed to be 7.0 $m^2$. A load-time diagram in the form of a step-wise linear load increase from 0 MN to 37.5 MN first and then to 75.0 MN and a subsequent linear decrease to 0 MN again, during an overall impact period of 52 ms, is to be assumed.

- **Assessment**

The load assumptions for wind and snow approximately correspond to the values set by DIN 1055. These loads normally are not decisive for dimensioning.

Loads resulting from earthquakes are location-dependent. The calculated seismic intensities at Stendal were determined by a location-related seismic expertise and the Temporary Design Standard for Nuclear Energy Plants in Seismic Areas VSN-15-78 /VSN 79/. The location-related ground accelerations are not assessed here. If necessary, further seismological expertise is to be commissioned.

The comparison with the load assumptions set forth in the BMI guideline on the protection of NPP against blast waves resulting from chemical reactions shows that with the same temporal sequence, the peak pressure according to /BAK 85/ is set 22.4 kPa higher than the 45 kPa set forth in the BMI guideline. In both cases the remaining excess pressure after decrease of the peak pressure is 30 kPa.

The load assumptions for external blast waves planned for the Stendal containment thus meet the requirements of the German codes and standards. The peak pressure value even exceeds the requirements.

The comparison of the assumptions on which the RSK Guidelines for pressurised water reactors are based, for the load case resulting from airplane crash, and the load-time diagram indicated there show:

- The mass of the impinging aircraft assumed for Stendal NPP is half as big as assumed in the RSK Guidelines.
- The impact speed with the set 750 km/h, approx. 208 m/s is slightly smaller than the 215 m/s prescribed in the RSK Guidelines.
- The impact area is the same.
- The load-time diagram with respect to the height of the load only reaches 68 % of the peak load in the RSK Guidelines and the impact duration of the load with only 52 ms is shorter than the 70 ms of the RSK Guidelines.

- **Load Combinations**

- **Preset Assumptions**

Load combinations for voltage and stability verifications are contained in the respective regulations /BAK 85/ and KTA-Rule 3401.2. An immediate comparison of the two regulations, because of the different conceptions of the containment or the safety confinement, respectively, is not as yet possible. In pressurised water reactors as operated in the Federal Republic of Germany the safety confinement consists of the steel containment and the surrounding building separated by a distance. The steel cladding absorbs loads from plant-internal accidents, while the surrounding concrete cladding accommodates impacts from external loads. A connection between containment and the surrounding building only exists in the bearing area of the containment vessel. The containment vessel and its internals are only indirectly concerned with external loads (induced vibrations).

The containment of the Stendal NPP by contrast is a one-shell construction which has to absorb internal loads as well as external impacts directly. This is especially true for the dome-shaped area, while in the cylindrical area the outer surrounding of the building functions as an outer barrier for external loads. The craneway of the polar crane via the consols is directly supported by the containment, while in Federal German pressurised water reactors it is borne by structures which are built inside the containment.

Despite the different concept, a basic comparison of the impacts comprised in load combinations is possible. In such a comparison it is to be considered that according to design regulation /BAK 85/ the calculated values of the impacts for every load case are to be determined by multiplication of the standard values with partial safety factors.

As partial safety factors

- load factor
- combination factor
- adaption factor to take into account idealisation in the framework of the assumptions made for calculations as a function of the calculation procedures
- valency factor

have been introduced.

These factors can range between 0.8 and 1.25.

A simplified summary concentrating on essential impacts of the respective combined load impacts has been compiled in Table 7.1-1.

In addition thereto, the regulations /BAK 85/ and KTA-Rule 3401.2 mention further load combinations, like the assembly case, pressure examination and in-service leak-rate examination.

- **Assessment**

A comparable procedure for considering impacts of loads can be derived from a comparison of the combination of essential impacts on which the design of the containment is based in accordance with the design regulation /BAK 85/ with the respective rules for the design of PWR containments.

This statement alone does not permit a sufficient statement on whether the design of the containment meets the Federal German requirements. For this purpose it is necessary to also compare the calculation bases and the design conditions.

The attempt to arrive at a rough assessment by a relatively simple comparison of the calculation bases was not successful. A more detailed comparison of the rules would be required which can, however, not be performed within the framework of this project.

To nevertheless make a first assessment of the present design, dominant load combinations corresponding to the Federal German codes and standards are estimated and compared with the results of the original design calculations (see Section 7.1.1.6).

#### 7.1.1.5 Constructive Peculiarities of the Composite Steel Cell Construction Type

- **Description**

The composite steel cell construction type so far has not been used in the Federal Republic of Germany. For the composite steel cell construction type, prefabricated steel cells are welded together on the construction site so that the respective wall section is located by sheets of the steel cell from the inside and the outside. After that concrete is filled in. The steel cells of the containment, normally 1.20 m thick, consist of 18-25 mm outer sheets which primarily fulfil static functions and of 12 mm inner sheets which fulfil the static function of a reinforcement as well as the sealing function of a liner. In addition to this sheet reinforcement, a conventional untensioned reinforcement is inserted in the circumferential and meridional directions of the containment. At a few, highly loaded, positions the round steel reinforcement was further strengthened. This is especially the case at the change-over from bottom plate to cylinder, in the craneway area and in the area of the penetrations.

The difference from linered concrete constructions, where a functional separation between the bearing function of the concrete with the conventional round steel reinforcement situated within it and the purely sealing function of the liner is assumed, is the intended use of the outer and inner sheets for reinforcement.

Tube penetrations, also called racking components, through the steel cell walls and ceilings, are already built into the steel by the manufacturer. Normally a rigid anchorage in the composite structure can be presumed. The weakening of the cross-sectional areas of the composite steel cell construction can be compensated by reinforcing sheets or junctions of round steel.

- **Assessment**

From the viewpoint of material savings, the composite steel cell construction type certainly represents an advantage, but at the same time high requirements are to be met by the constructive design. Referring to the arrangement of the stiffenings and anchorages, for example by dowel cleats, attention is to be paid that extension concentrations are avoided. Otherwise, especially during forced stress, the strength of the material used can be exceeded locally and the sealing function thus be impaired.

During temperature loads, which can occur in the course of an accident or of a fire, the composite steel cell construction type is to be regarded rather critically. As the supporting sheet metal is positioned at the outer surface of the stressed component, the material properties, like the modulus of elasticity, apparent limit of elasticity and tensile strength are directly influenced by temperature. During higher temperatures (> 200 °C) the strength of the steel used decreases rapidly, which in many cases is synonymous with a decrease of the carrying capacity of the component. The concrete cover of concrete supporting frameworks with round steel reinforcements is normally so big that because of the poor heat conduction of the concrete, temperature can only influence the reinforcement rods after a longer period.

With respect to the carrying capacity limit a conventional concrete construction with a liner offers greater reserves than a composite steel cell construction. If a component is loaded beyond its design limits, this is connected with great deformations. If the liner failed locally in the course of this process due for example, to stress concentration in the area of the anchorages, for a conventional construction this normally means a leak, but not the impairment of the carrying capacity of the component. For a steel cell, however, the local destruction of a metal supporting sheet can develop to a component failure.

It can be summarised that the composite steel cell construction in the Federal Republic of Germany does not represent a recognised construction type. If this type of construction is used, a general license of the Institut für Bautechnik (Civil Engineering Institute) in Berlin, or a special license from the planning department and building control office of the state government responsible, will be required in any individual case (R 7.1-2). In both cases, numerous details with respect to this construction type are still to be clarified and, if necessary, to be secured by licensing tests. The tests performed at the Bauakademie could certainly largely be used for this purpose. Possible questions to be clarified in this context are: flow and shrinkage bahaviour of the dry-out resistant concrete, thermal behaviour, behaviour in case of fire, the whereabouts of the residual water in the construction not required for the process of setting, pressure built-up in the steel cells during high thermal loads because of the formation of condensate, corrosion protection, especially for the use of this construction type for reinforced ceilings.

Independent of the above general remarks, no severe weaknesses of the constructive design of the steel cells and the construction as a whole became apparent during the assessment of the documents available.

The anchoring of the racking components (anchor studs) for the absorption of forces from the component supports possibly have to be examined at a later stage of the examinations (R 7.1.-3). As the respective values are not indicated, it is not known for which loads the racking components are to be designed. It shall, however, be mentioned that contrary to the common practice in the old Federal German Länder, the supporting walls in the area of the racking components are not reinforced by additional reinforcements or by increasing the metal sheet thickness of the steel cell. The racking components are normally only anchored by straight round steel rods vertically to the wall, partially in the zone subject to tensile forces of constructions subject to bending loads. According to the Soviet regulations, the length of the anchoring normally is the diameter multiplied by the factor 40, i.e. these are 80 cm for a diameter of 20 mm. This roughly corresponds to the value required by DIN 1045. The weld joint of the round steel horizontal to the anchor stud as well as the loads of the anchor studs in the thickness direction are still to be analysed more closely (R 7.1-3). A final assessment of the composite steel cell construction type within the framework of this project is not possible.

#### 7.1.1.6 Results of the Comparative Calculations

For the reactor building, /EIB 91/ examined whether the containment and the structures connected to it correspond to the design conditions required, according to the state of the art, or which deficits exist, respectively. The studies performed with respect to licensability basically restricted themselves to the essential supporting structures and the dominant impacts.

The Soviet project engineer was responsible for the detailed planning of most of the buildings, including the reactor building. The GDR did not examine the statics and the design of projects which were not planned in the former GDR and was only provided with execution plans. Complete and testable static verifications are not available. For the estimation performed here there was also no comprehensive set of constructional drawings.

- **Internal Impacts**

- **Calculations Performed**

In the framework of the investigations relating to internal impacts, calculations of carrying capacities have been performed on a rotationally symmetrical model of the containment with the help of non-linear finite element calculations considering realistic material models for steel, concrete and reinforcement. In this context the loads of the containment shell during a 2A break of the main coolant line were analysed. The following conservative assumptions were made: maximum internal pressure 550 kPa at a termperature of about 135°C. These values were the results of first estimates at the beginning of the investigations on the containment shell and they were confirmed conservatively in the further course of the studies (cf. Fig. 5.2-2 and 5.2-3).

Additionally, analyses on the carrying capacity limit were performed. The internal pressure here at a temperature of 135 °C was varied up to 950 kPa.

- **Results and Assessment**

The calculations showed that the loads of a 2A-break can globally be controlled by the containment (see Fig. 7.1-1). The carrying capacity limit was determined to be at an internal pressure of about 900 kPa. Whether there can be local overloads, for example in the area of the penetrations, could not be investigated in detail within the framework of this study.

Owing to the lack of information no statements can be made on the accommodation of differential pressures between the individual rooms within the containment, the impacts of jet forces upon breaks in pipes (cf. Section 7.1.1.4) and with respect to anchoring and stability of large components.

- **External Impacts**

• Design

Earthquakes, airplane crashes and explosion blast wave were considered as loads with external impacts.

The design of the containment lies in the highest category of earthquake-safety. As far as known, no dynamic verification calculations for dimensioning, customary in the Federal Republic of Germany, were performed by the Soviet Union. Static equivalent loads were used instead. Floor response spectra specifically valid for the Stendal location were also not determined.

Within the framework of the /EIB/ study no new earthquake calculations were performed for verification. But independent calculations referring to the earthquake load with the simultaneous loads of a 2A-break of a main coolant line were performed by the Bauakademie of the GDR.

The airplane crash load at the Stendal NPP belonged to the design loads. The load-time function assumed here shows clearly lower load coordinates than the RSK-function normally used in the Federal Republic of Germany, cf. Section 7.1.1.4. It was found that, using the assumption of the above-mentioned load function, the containment shell is not penetrated, while assuming the RSK-function, a penetration has to be expected. According to present knowledge, induced vibrations for verifying the design and anchoring of the equipment have not been analysed during the design. With the present dimensions of the containment shell a complete protection in accordance with the Federal German criteria does not exist.

The reactor building was designed against the load of an external blast wave. The load assumptions applied here are slightly above the loads assumed in the Federal Republic of Germany, cf. Section 7.1.1.4. Because of the constructive design of the building structures it can, according to /EIB 91/, be presumed without detailed re-calculation that loads resulting from a blast wave can probably be accommodated. Again, for this dynamic load, according to the present knowledge, no induced vibrations have been calculated.

The surrounding outer building of the containment and the base floor were built using concrete steel-cells. These are concrete-steel boards manufactured in a prefabrication plant which are installed on the building site and cast with concrete. They are connected with each other with reinforcing cages.

The surrounding outer building is separated from the actual containment by joints so that even during dynamic loads, like earthquakes or blast waves, no contact between the two building parts is established. It is assumed that the relatively rigid box-type structures withstand the latter loads. The external impact of airplane crash, however, cannot be accommodated.

- **Assessment**

It was determined by comparative calculations using a simplified model, suitable for describing the stability of the undamaged shell of the containment, that a maximum internal pressure of up to 550 kPa to be expected with a simultaneous temperature of about 135 °C can be accommodated. The carrying capacity limit of the containment is reached at about 900 kPa.

With respect to the load resulting from airplane crash it was found that, assuming the load function according to /BAK 85/ and /TEP 81/, the containment shell is not penetrated. Assuming the RSK-load function, a penetration of the containment shell must be expected.

Relating to the load resulting from earthquake, no comparative calculations were carried out. The author of /EIB 91/ did, however, have the opportunity of inspecting calculations of the Bauakademie of the GDR referring to the earthquake load case, with the simultaneous serious accident not introduced into the licensing procedure, and of performing examinations. According to these examinations it can be confirmed that the steel cell construction withstands these combined loads. The load assumptions were based on the statements made in /TEP 81b/ and /HAB 83/. No statements can, however, be made on the floor response spectra relating to the design of the equipment.

Referring to the blast wave resulting from external explosions no comparative calculations were made. Because of the constructive design of the design structures according to /EIB 91/ it can be assumed that these loads can be accommodated.

A final evaluation of the constructional design of the reactor building within the framework of construction-supervision procedures requires a complete examination of the design and the calculations (R 7.1-4).

The vibrations resulting from the load cases earthquake, airplane crash and external blast waves have not been investigated. It is recommended to determine the corresponding response spectra (R 7.1-5).

#### 7.1.1.7 Leak Test of the Containment

- **Description**

The permissible leak rate of the containment at the Stendal NPP under design pressure was established to be 0.1 Vol-%/day. To ensure this leak rate a number of requirements are to be met. To these belong according to /TEP 81/:

- the leaktight execution of all welds of the inner metal sheets of the containment and their control for leaktightness prior to and during operation
- the installation of two, in most cases three, penetration isolation valves in line between the containment and the environment
- the high requirements, with respect to the quality of assembly and post-assembly testing, of the penetrations
- the execution of all penetrations through the containment in such a way that they can be inspected for leaktightness. Leaktightness checks are performed prior to initial operation at the manufacturer's and in the nuclear power plant after assembly.

- **Assessment Criteria**

In the Federal German codes and guidelines there are no specified values relating to the permissible leak rate of containment vessels. Determinations relating to the pressure-time sequence in the containment vessel to be considered for calculating the leak rate sequence are contained in the accident calculation bases of the BMI Guidelines. In the Federal Republic of Germany it is common practice to demonstrate a leak rate of 0.25 Vol-%/day in relation to the air volume in the containment vessel as the design value for PWR containment vessels. According to RSK-Guideline 5.5(1) leak rate testing starting out from the unpressurised state has to be carried out with an increasing pressure level sequence at an in-service examination pressure of at least 150 kPa and at design pressure. In-service examinations to demonstrate leaktightness of the containment are to be performed (annually) at a pressure of 170 kPa. For this examination pressure a leak rate of 0.04 Vol-%/day may not be exceeded according to KTA-Rule 3405. Such A values are close to the verification limit and require a long measurement period (24 - 48 h). Furthermore, extensive measurements using different methods are carried out to determine local leakages.

- **Assessment**

According to /TEP 81/ a leak rate of less than 0.1 Vol-%/day related to the air volume is planned for the Stendal containment. This value meets the requirements of the Federal German state of the art.

/UVA 84/ comprises a list of leaktightness requirements to be met by penetrations. The permanent ensurance of the required leaktightness of gate and hatch of the transfer canal (size 5.0 x 11.2 m) which shall be interlocked against each other is assessed to be problematic. Their share of the integral leak rate - 2.7 $m^3$/h at 500 kPa or 1.1 $m^3$/h at 170 kPa, respectively - according to /UVA 84/ shall not exceed 1 x $10^{-4}m^3$/h at 500 kPa or 0.4 x $10^{-4}m^3$/h at 170 kPa respectively.

For underpressure safety of the containment KTA-Rule 3401.1 requires an underpressure test with the maximum underpressure multiplied by a factor of 1.5. Such A a test is not planned for the containment of the Stendal NPP (R 7.1-6).

No comparisons are made here with respect to the performance of leak rate examination, analysis of the measurement results and the evaluation of the measurement results.

#### 7.1.1.8 Summarising Evaluation of the Containment Concept

The encapsulation of the primary system of the Stendal NPP is designed as a single-shell containment. Nuclear power plants with PWRs designed and operated in the Federal Republic of Germany have a two-shell encapsulation: a containment vessel of steel and a surrounding building of concrete. The space in between is sucked off and it is thus possible to let off leakages after filtering in a controlled way. This possibility does not exist in a single-shell containment. This does not correspond to the Federal German practice. Evidence must therefore be provided that a single-shell containment is also able to provide the necessary protection against an inadmissible release of radioactive substances (R 7.1-7).

The two-shell construction type of the containment vessel for PWR is designed in such a way that loads, like pressure and temperature increases resulting from accidents are accommodated by the containment vessel (steel ball) while loads acting upon the encapsulation from outside, like e.g. loads resulting from airplane crash or blast waves of chemical reactions, are accommodated by the surrounding concrete building. These then only have secondary effects on the containment vessel.

In a single-shell encapsulation during external loads the containment function is affected directly. For the Stendal NPP this especially applies to the dome-shaped roof area. The cylindrical part of the containment to a certain degree is protected by the outer surrounding of the building.

The results of a first engineering estimate /EIB 92/ showed that, apart from the airplane crash load case, it would be possible to fulfil the Federal German regulations from the structural engineering point of view, possibly after some re-construction and upgrading measures. The induced vibrations, not examined, and the floor response spectra, not determined, could introduce additional problems.

## 7.2 Plant-Internal Spreading Impacts

### 7.2.1 Fire Protection

This section restricts itself to the consideration of internal fires, which are fires originating within buildings. Fires spreading to buildings from the outside, like, for example, fires owing to accidents in installations with large fire loads on the site of the nuclear power plant (like petrol stations and gas-storage tanks) or fires outside the power-plant site, like, for example, fires as a result of transport accidents (accidents of air, rail, road and waterborne traffic) are not considered here, as there are no documents available. Fires outside the power-plant site are to be discussed in connection with the assessment of external impacts. A comprehensive concept on the protection of the nuclear power plant against external impacts, which also considers fires outside the power-plant site, is to be provided by the applicant for assessment (R 2.7-1). During the construction of facilities with a potential for large fires, it must be ensured that inadmissible fire impacts on important safety-relevant buildings and facilities can be avoided (R 7.2-1).

#### 7.2.1.1 General

For the assessment of fire protection at the Stendal Nuclear Power Plant it must be recognised that the power plant is still in an unfinished state. Fire protection devices and fire protection components, like, for example, fire reporting devices, stationary fire fighting facilities, vents as well as fire breaks (doors, cable compartments, etc.) are normally installed at a later stage, so that only an assessment of the planned facilities on the basis of the design documents available can be performed here. These design documents normally do not go beyond the conceptional description of fire protection so that there can only be an evaluation of the fire protection concept.

The systems facilities and components to be protected from a fire protective point of view have also not been installed yet. An assessment of the intended locations, from the viewpoint of fire protection and the fire protective separations, at the present time can also only be made on the basis of the existing design documents.

Fire protective weaknesses resulting from deficiencies during assembly can in principle not be derived from the design documents. They can only be determined in the framework of an acceptance test of the respective systems. Those areas in the Stendal power plant, where it is expected that the present design data cannot be realised, because of the experiences of other WWER plants, and where weaknesses in the field of fire protection may thus occur, are dealt with in the present assessment.

Statements on in-service inspections of fire protection facilities are not made in the documents. These inspections are highly important for safety engineering. In the framework of additional tests a consistent concept has to be presented for examination (R 7.2-2).

#### 7.2.1.2 Assessment Criteria

Fire protection within the framework of this report is principally assessed in accordance with the state of the art of fire protection in Federal German nuclear power plants. Referring to nuclear-technology-related requirements the state of the art is governed by the following essential documents:

- Criterion 2.7 "Fire and Explosion Protection" of the Safety Criteria for Nuclear Power Stations with the Interpretation of November 28, 1979

- RSK-Guideline for Pessurised Water Reactors, Guideline 11 "Fire Protection", Guideline 12 "Escape Routes and Alarm Systems"
- KTA-Rule 2101-1 "Fire Protection in Nuclear Power Stations", part 1: Fire Protection Principles.

Further KTA-Rules referring to fire protection in nuclear power stations are presently being discussed. The appropriate regulations can, however, be regarded as references to the present state of knowledge and thus be used as an assessment aid:

- KTA-Rule 2101.2 "Fire Protection in Nuclear Power Plants", Part 2: Fire Protection of Constructional Plants, Draft, version of 06.91
- KTA-Rule 2101.3 "Fire Protection in Nuclear Power Plants", Part 3: Fire Protection of Machines and Electro-technical Plants, Draft, version of 11.90
- KTA-Rule 2102 "Escape Routes in Nuclear Power Plants", Draft, state 06.90.

#### 7.2.1.3 Basic Fire Protection Concept

- **Description**

The fire protection concept of the Stendal Nuclear Power Station is based on the Soviet standards as well as on the regulations in the GDR. The fire protective design was based on the following essential principles:

- The three redundancies of the safety system are structurally separated and isolated from each other in a fire-protective way (fire resistance 90 minutes).
- In room areas with a higher fire potential, fire reporting and fire fighting devices are installed.

In contrast to older WWER plants, safety-relevant facilities at the Stendal A NPP are located outside the turbine hall. Furthermore, the turbine hall is a separate building so that "fire in the turbine hall" is of minor importance for the Stendal A Nuclear Power Plant. The accident combination "external impacts (e.g. earthquake) with consequential fire" is not explicitly dealt with in the present fire protection concept. The water-spraying fire extinguishing facilities for redundant, important safety-relevant systems and cable compartments within the reactor building have, however, been designed earthquake-proof.

- **Assessment**

The design principles mentioned here essentially correspond to the fire protection principles of Federal German nuclear fire protection rules. To consider external impacts with consequential fire, a systematic treatment of this topic would, however, be necessary, as indicated in KTA-Rule 2101.1. A systematic investigation of the accident combination "External impacts with consequential fire" must be performed within the framework of further examinations (R 7.2-3).

#### 7.2.1.4 Structural Fire Protection Measures

- **Intended Measures**

- Main Buildings (Reactor Building, Turbine Hall, Degasser Extension, Electrical Extension)

With the turbine hall, the degasser extension and the electrical extension the reactor building forms one interconnected building complex. It is intended to protect the flat roofs of these buildings against spreading fires from outside with a 20 mm fire protective layer of gravel.

- Reactor Building (Containment and Surrounding Outer Building)

According to the project documents all bearing and limiting structural components within the reactor building shall have a fire resistance of at least 150 minutes. Bearing steel constructions shall be plastered to ensure the required fire resistance, but no reliable figures are indicated. The rooms for safety system components (technological rooms of the safety system) below the containment are separated by structural components having a fire resistance of at least 90 minutes.

The rooms located in the reactor building containing oil systems (e.g. the main coolant pumps) shall also have a fire resistance of at least 90 minutes. Furthermore, sills are planned in the area of the door openings of these rooms to ensure the collection of the entire oil volume. The oil tank of the main coolant system shall be provided with a discharge into the emergency oil discharge tank, which is also located in a fire resistant, separated room.

The three trains of the electrotechnical equipment of the safety system are routed separately. They are each located in separate rooms or cable channels and shafts. The walls of these rooms or cable channels and shafts shall have a fire resistance of at least 90 minutes. The cables for the general energy supply shall only be installed within one train of the safety system of one unit.

Additional air and exhaust systems are planned for the ventilation of the cable compartments, with the cable compartments of the three trains of the safety system each having independent ventilation systems. Ventilation channels routed through other rooms of the building shall have a fire resistance of at least 90 minutes. For certain rooms of the oil system an independent air and exhaust system is planned.

For battery and acid rooms, supply air and exhaust systems are intended which ensure natural ventilation of these rooms upon failure of the ventilators by bringing together the exhaust system. The lines of the exhaust system shall be designed with a fire resistance of 45 minutes and be sparkproof. Fire protection flaps shall be installed in the supply air and exhaust lines of fire-endangered rooms, which isolate the room in case of fire.

Upon actuation of the installed fire fighting equipment the supply air and exhaust systems shall be switched off automatically. After the fire has been extinguished, ventilation is started again locally to remove fumes.

- Turbine Hall, Degasser extension, Electrical extension

All bearing and room delimiting structural components within the turbine hall and the degasser extension are made of non-combustible building materials and shall principally have a fire resistance of at least 120 minutes. The wall between the electrical extension and the degasser extension shall have a fire resistance of 240 minutes. Those rooms with the lowest fire risk according to Soviet standards shall be separated by partition walls having a fire resistance of at least 45 minutes. The doors in these walls shall be fire-resistant for at least 30 minutes. The walls of the evacuation staircase, according to the present documents, shall have a fire resistance of 120 minutes. It is intended to cover the roof of the turbine hall with a hardly inflammable silicate foam insulation.

- Emergency Power Buildings

The emergency power buildings are separate building structures, each having their separate fire section.

- **Assessment**

The statements made in the different documents with respect to structural fire protection are not comprehensive, are not always unambiguous and are not free from contradictions. Terms like "feuerfest" (fireproof) are, for example, used for different periods of fire resistance. It is therefore difficult to comprehend what is, for example, meant by the different production categories and the different degrees of fire resistance. On the basis of the present information, the following essential design principles can, however, be assumed:

- The outer construction of the building and the structural building components of concrete relevant for the stability of the buildings are at least fire-resisting (fire resistance 90 minutes or more). With respect to the classification of steel-cellular composite-design structures, some individual issues are still to be clarified and a formal classification is still pending (R 7.2-4).
- If possible, only non-burnable or at least hardly inflammable building materials are used.
- The relevant buildings are sub-divided into fire lobbies or separated fire-resistant areas (fire sections). Aspects like the separation of the redundancies of important safety-relevant facilities and the protection of escape ways are taken into account in principle .
- Redundant safety-relevant systems and facilities are structurally separated and will be separated from each other in a fire-resistant way (fire resistance 90 minutes). There is a corresponding separation of the routing of the safety system cables.

- Areas with larger oil containers shall be isolated in a fire-resistant way (fire resistance 90 minutes).

These basic design principles can be agreed to, taking the rules and guidelines valid in the Federal Republic of Germany into account.

Because of the experience gained in connection with the assessment of the Greifswald plants as well as the findings of other WWER plants it must, however, be expected that the conceptual design principles are not always observed subsequently. This, in particular, applies to the structural separation of the redundant trains of the safety system (for example in the area of the intersections of cable routings and in the control room). In the area of the main steam and feedwater valves a desirable fire-protective separation is not planned. A final evaluation and identification of such a problem areas is not possible at present. These problem areas are to be identified and evaluated in the framework of a fire-hazard analysis. If necessary, additional fire-protection measures are to be taken (R 7.2-5).

There is no information on structural fire protection facilities, like fire doors, cable compartments and fire-protection flaps. Only those fire protection facilities, e.g. fire doors, cable compartments and fire-protection flaps, approved by the construction supervision authorities may be installed (R 7.2-6).

Furthermore, the concept concerning the use of fire-protection flaps in the ventilation ducts is not clearly recognisable. Ventilation ducts that run through several fire-resistant areas must be provided with fire-protection flaps in the penetration areas of the necessary fire-resistant partitions (R 7.2-7).

No information is provided on the decoupling of the emergency control room from the main control room (system decoupling see Section 6.4). Such A a decoupling (cables not to be routed over the control room area, electrical decoupling), however, is highly important for safety as well as reasons of fire protection. For reasons of fire protection a decoupling of the emergency control room from the main control room is considered to be necessary (R 7.2-8).

It is planned to bring large quantities of oil for the main coolant pumps (2 x 10 $m^3$) into the reactor building. A fire-hazard analysis must therefore be carried out and, if necessary, additional fire-protection measures are to be taken (R 7.2-9).

The routing of cables of redundant systems which do not belong to the safety system cannot be recognised in the documents. It is considered to be necessary that these cables are also physically separated (R 7.2-10).

#### 7.2.1.5 Fire-Protection Measures of the Plant

- **Fire Signalling Systems**

- Intended Measures

Manual and automatic fire detectors shall be installed for rapid fire detection and signalling. In rooms accommodating equipment endangered by fire, automatic fire detectors are to be provided everywhere. Smoke alarms are planned within the cable compartments and heat detectors in the area of the oil supply of the pumps. At all staircase entries push-botton fire signalling devices are installed. The fire shall be signalled in the permanently attended main control room. It is further intended to indicate the position of the valves for fire extinguishing water supply in the main control room. An overall fire signalling centre shall additionally be installed in the building of the fire brigade.

- **Assessment**

Automatic fire detectors in the concept are only planned for rooms with large fire potentials. As it cannot be excluded that large fire loads can be brought into rooms with a lower fire potential, it is considered to be necessary to install automatic fire detectors in all areas of safety-related importance (R 7.2-11).

No information is provided on the arrangement and the quality of the fire detectors. During the installation of the fire detectors the room dimensions, the type of combustible material as well as the ventilation conditions must be taken into account.

If the conditions are unclear, experiments to determine the smoke propagation will have to be carried out. The application of qualified type-inspected fire detectors suitable for the respective combustible material is considered to be necessary (R 7.2-12).

It has to be remarked as a positive aspect that the designer of the plant in further considerations planned the automatic control of the fire protection facilities (e.g. ventilation systems, fire extinguishing systems) via the fire detection system. A two-detector dependency is generally intended to prevent malactuations.

- **Fire Extinguishing Systems**

- Intended Measures

- Fire Extinguishing Water Network

An external extinguishing water network shall be erected on the power plant site. Extinguishing water shall be provided via two physically separated extinguishing water pump stations. It is intended to install three emergency power secured fire extinguishing pumps with a capacity of 50 % each in pump station I (1st construction stage). Apart from a pressure system (with pressure maintaining pumps), water storage tanks are planned there. Pump station II (2nd construction stage) shall have four pumps each with a capacity of 50 %, again with pressure maintaining systems as well as water storage tanks.

- Reactor Building

The intended installed fire extinguishing systems are water sprinkler fire-fighting systems subdivided into different fire-fighting zones. It is the object of these fire extinguishing systems to fight the development of fires, to prevent spreading of fires and to cool oil-containing facilities in case of fire.

An independent extinguishing water network designed as a closed circuit pipeline with outgoing sections for extinguishing water consumers is to be designed for the reactor building (see Fig. 7.2-1). The closed circuit pipeline will be connnected with the fire-extinguisting water network of the nuclear power plant via four infeeds. It is intended that two loops of the closed circuit pipeline should branch off to one gate valve chamber and feed one sprinkler fire-fighting system each. A further loop will feed three extinguishing water tanks having a volume of 72 $m^3$ each belonging to a large sprinkler fire-fighting system. It is planned to design this system earthquake-proof. The following three fire-fighting zones are planned in the reactor building:

- Sprinkler fire fighting section of the main coolant pumps consisting of four fire-fighting sections and protecting the oil supply of the main coolant pumps.
- Sprinkler fire fighting section of the make-up pumps consisting of three fire-fighting sections and protecting the oil systems of the make-up pumps.
- Sprinkler fire fighting section for redundant, important safety-relevant system and cable compartments (this system is designed earthquake-proof).

For the sprinkler fire fighting sections of the main coolant pumps and the make-up pumps, it is planned that there is water up to the valves (in the valve compartments), and thereafter a dry line, routed to the systems to be protected. The systems will be actuated automatically by the second signal of the fire alarm system of the respective fire-fighting zone. Here it is planned to open the motor-driven valve assigned to the corresponding fire-fighting zone in the valve chamber. It is intended to additionally arrange one manual valve, parallel to each motor-driven valve. It is further intended to switch off the fire-fighting system manually.

The essential cable compartments are to be protected by the sprinkler sections for redundant, important safety-relevant systems and cable compartments. As an important safety-relevant systems this fire-fighting system is designed as a three-train system (applies to tank, pump and manifold) so that all fire-fighting zones can be supplied with water by each of these three trains. Pumps, tanks, isolating valves and startup valves of each train shall be physically separated from each other.

The functioning of the system is planned by analogy to the sprinkler systems of the main coolant pumps and the make-up pumps. Differences exist in the following points:

- Extinguishing water is supplied from extinguishing water tanks fed by the extinguishing water network or by the service water system A.

- The fire-fighting system shall be actuated here when two fire detector lines in a fire room respond. There are four lines in every room.
- The three trains of the fire-fighting system are assigned to the respective trains of the safety system. It is planned that the two trains not affected by the fire are always switched in and then, if both trains inject, one train is switched off again. The third train can also be switched in manually.
- It is intended that the containment-isolation valves at the penetration through the containment are permanently open. They are closed by the reactor protection.

Besides this automatic stationary spray flooding systems, a dry rising main is planned in the reactor building. Outside the building (level 0 m) this line shall provide three infeeds for pump water tenders of the fire brigade.

On this rising main, three connections for fire hoses are planned within the dome-shaped roof of the containment.

For manual fire-fighting there are hydrants on different levels within the surrounding outer building of the reactor building which are supplied with extinguishing water via spurs from the closed circuit pipeline.

- Turbine Hall with Extensions

The intended stationary fire-fighting systems of the turbine hall are sprinkler fire-fighting systems subdivided into 19 fire-fighting zones. The individual fire-fighting zones are differentiated between "physically closed fire-fighting zones" (e.g. cable compartment, oil compartment) and "physically open fire-fighting zones" (e.g. fire fighting system of oil sections).

It is the object of the sprinkler fire extinguishing systems to fight the origins of fires, to prevent their spread as well as to cool oil systems.

For extinguishing water supply of the turbine hall a closed circuit pipeline (DN 200) is planned which is fed from three positions of the extinguishing water network of the nuclear power plant. The three ring system connections lead into three distribution rooms (valve compartments) with motor-driven valves and parallel manual valves. A conceptual change, without distribution rooms and the construction of a line from the closed circuit pipeline via isolating valves directly to the individual fire-fighting zones, is being considered. Dry lines are planned from the distribution rooms to the items to be protected.

The fire-fighting system shall be actuated automatically upon response of two fire detector lines of physically closed fire-fighting zones. In these rooms two fire detector lines with ionisation-detectors each shall be installed. It shall, however, also be possible to manually actuate these fire-fighting systems either locally or from the control room. Considereations have been given to istalling an interval switch for the fire-fighting process; two minutes fire-fighting, two minutes break.

For physically open fire-fighting zones an automatic actuation of the fire-fighting system is not normally provided. Here the fire-fighting system shall only be actuated manually, after a check patrol from the main control room or locally by opening the valves. For safe fire detection via two detection signals, an automatic actuation possibility is, however, also planned for the open fire-fighting zones.

The fire-fighting systems are to be switched off manually. An indication of the functioning of the fire-fighting systems and of the position of the valves is planned in the main control room. Upon actuation of the fire-fighting system, the ventilation of the rooms concerned shall be switched off.

Parallel to the infeed positions of the extinguishing water network in the closed circuit pipeline of the turbine hall, three emergency infeeds for pump water tenders of the fire brigade are intended. Thus, an extinguishing water supply to the fire-fighting systems located in the turbine hall is also possible during a failure of the extinguishing water network.

To manually fight fires within the turbine hall there are hydrants on different levels which are supplied with extinguishing water from the closed circuit pipeline via spurs. Additional rising mains with triple infeeds for pump water tenders of the fire brigade on the outside are provided for fighting fires on the turbine hall roof, at the turbine table area, in the degasser extension as well as for the roof of the electrical extension.

- Emergency Protection Building

In all three buildings, wall hydrants are provided. In addition, stationary sprinkler fire-fighting systems shall be installed here.

- **Assessment**

The concept described makes sense and also meets the requirements of the Federal German rules and guidelines. It is still to be examined whether sufficient pumping capacity and extinguishing water reserves are ensured for all fire-fighting zones, also allowing for additional manual fire-fighting (R 7.2-13).

Motorised isolation devices and manual isolation devices of the extinguishing water lines for several fire-fighting zones will be accommodated within the valve compartments, where the design fire loads are low. It must be determined by analysis to what extend simultaneous failure of several fire-extinguishing systems within the valve compartments is possible. Backfitting measures might become necessary (R 7.2-14).

As regards the extinguishing water supply for equipment inside the containment, it must be checked whether the containment isolation valves can be reopened after closure by a signal from the emergency cooling signal. Becuase of the possibility of closure by a spurious signal, the possibility of re-setting the valves is deemed to be necessary (R 7.2-15).

The aspect "failure of engineered safeguards owing to the admission of extinguishing water" has not been discussed. Here it must be demonstrated that it is not possible that several redundancies of safety-relevant systems or equipment are inadmissibly impaired by fire-fighting (7.2-16). Unacceptable consequences of fire-fighting measures can, for example, be avoided by the use of waterproof fire protection flaps.

#### 7.2.1.6 Operational Fire Protection

Manual fire-fighting in the plant is performed by the plant fire brigade. Binding statements on the organisation and size of the fire brigade as well as administrative regulations in case of fire do not exist. This item is not decisive for the concept. It can be assumed that the plant fire brigade will operate according to the valid rules and guidelines. In the framework of further analyses a concept relating to this must be presented (R 7.2-17).

#### 7.2.1.7 Conventional Fire Protection Requirements

Besides the nuclear-specific requirements to be met by fire protection, the conventional requirements of the construction supervisory authorities are also to be considered. At the Stendal Nuclear Power Plant, a concept relating to the formation of fire sections and fire zones and more consistent isolation of larger fire loads can be determined. The problem of human protection is thus eased. Futhermore, smoke clearance measures were being considered. The permissible lengths of escape routes according to GDR regulations are observed everywhere.

Examinations are, however, still required with respect to certain details. In particular the design and the quality of structural fire protection measures and the protection of escape routes (e.g. free passage ways, consistent structural separation of staircases) are to be investigated. In addition, special problem areas, like fire protection in the area of the oil systems, are to be examined (R 7.2-18).

#### 7.2.1.8 Summary of the Events and Recommendations

The following points at Stendal NPP A are assessed to be positive:

- the exstensive structural separation of the triple redundant design of the safety system
- presence of an emergency control room
- the fire-resistance of at least 90 minutes due to the structural separations of concrete
- the installation of fire detectors in all areas with a high fire load

- the installation of extensive stationary fire-fighting equipment in areas with cable concentrations and areas of the oil supply which are largely actuated automatically and some of which can also be re-fed with extinguishing water from the fire-fighting supply.
- the earthquake-proof design of the fire-fighting equipment for redundant important safety-relevant systems and cable compartments in the reactor building.

For areas of the fire-protective design of the Stendal Nuclear Power Plant which do not meet the current requirements or which owing to the lack of information cannot sufficiently be assessed, recommendations are given (see Section 10).

### 7.2.2 Flooding

The present section only deals with flooding from inside; flooding from outside is not discussed here because there are no relevant documents available. Because of the location of the site of the nuclear power plant 10 m above the average water level of the river Elbe, the external flood risk is assessed to be low and it is therefore not considered further.

- **Description of the Buildings**

- Turbine Hall

There are no safety-relevant components in the turbine hall so that there can be a partial flooding of the turbine hall caused by a failure of the feedwater lines, of the feedwater tanks or the main coolant lines, which will, however, not lead directly to the failure of engineered safeguards.

- Reactor Building (Apparatus House)

Below the 13.20 m ceiling, the reactor building accommodates three floors (see Fig. 4.2-1). Almost all engineered safeguards are located in this building. They are positioned below the 13.20 m ceiling and the emergency boron tank and are therefore potentially endangered by flooding. The doors of the chambers are therefore provided with seals and are locked from the outside.

The emergency control room and the pumps for the emergency feedwater and for the three trains of the HP and the LP emergency cooling system are all located in separate rooms on the lowest level (-4.20 m). The three emergency and the three pool coolers are positioned on level 0.00 m.On the 3.60 m level above the emergency feedwater pumps there are the three emergency feedwater tanks (500 $m^3$ each), in a common tub-shaped room which is accessible from the 6.60 m level via three stairs. The main control room and a large proportion of the instrumentation and control cabinets are also positioned on the 6.60 m level. On this level there is also the bottom of the L-shaped emergency boron tank (630 $m^3$) from which the HP and LP emergency cooling system are fed upon demand. The outlet lines from the emergency boron tanks have three trains (DN 600) with a single pipe each running to the emergency coolers. Each outlet line has only one motor-driven isolating valve about 10 m from the tank. Before the motor-driven isolating valve there are three lines (DN 10) to the sampling system of the primary system, one line (DN 150) for heating the boron acid solution and one line (DN 100) coming from special water treatment (SWA) to fill up the emergency boron tank. In addition three lines (DN 100) executed as single tubes with two isolating valves, lead to the special water treatment (SWA) or drain waters, respectively.

Because of the large boric acid solution and deionised water reserves and the almost unlimited backfeed capacity of the service water system A on one hand, and the safety-relevant importance of the systems endangered by flooding (emergency control room, emergency feedwater system) on the other hand, reliable measures for preventing flooding are required. A corresponding concept does not exist.

- **Assessment Criterion**

According to the BMI Accident Guidelines inadmissible effects on safety-relevant systems caused by flooding are to be prevented. The flooding risk is avoided by division into sectors, arrangement on a certain height, isolation measures, double tubes at the sump suction line and separation into chambers.

- **Assessment**

In the turbine hall, flooding does not directly lead to a failure of engineered safeguards. Flooding of an emergency power building can be tolerated because of the three-train design of the safety system. For this reason no additional measures for avoiding flooding in the turbine hall and the emergency power buildings are deemed to be necessary.

The simultaneous failure of more than one train of the individual engineered safeguards in the reactor building shall be prevented by accommodating them in chambers separated from each other. To verify the operativeness of this arrangement, it must be demonstrated that the walls between the chambers, the doors and their seals, as well as the penetrations in the walls, withstand the jet forces and water loads to be assumed (R 7.2-19).

The drains existing in the chambers are to be equipped with appropriate isolating devices to be able to prevent flooding of adjacent trains of the safety system by drainage. The isolating devices between the drain systems of the redundant systems must be safely locked in the closed position during normal operation (R 7.2-20).

To reduce the frequency of flooding events, a qualified and reliable device for leak detection must be installed to enable shift staff to take effective counter-measures (R 7.2-21).

The three drains of the important safety-relevant emergency boron tank which is a part of the containment represent a particularly severe weakness as they only consist of plain tubes and cannot be isolated directly at the tank.

In accordance with the BMI Accident Guidelines it is therefore deemed to be necessary to equip the penetrations through the containment with double-walled pipes with leak detection. Furthermore, motor-driven isolating valves must be installed as close to the sump as possible at the end of the double-walled pipes (R 7.2-22).

The important safety-relevant components and installations of which there is only one are particularly endangered. This especially applies to the emergency control room located on the lowest level, -4.20 m. It must be protected by raised thresholds and tight-fitting doors to withstand possible water loads or jet forces (R 7.2-23).

On the 20.40-m level in the containment there are the three spent fuel pools executed as double tanks. They are equipped with a leakage detection system and the supply lines are connected from above without penetrations so that flooding of the lower rooms owing to leakages of the spent fuel pools can be excluded. Any existing outlet pipes must be equipped with double isolating valves. It must be possible to prevent a siphon effect in pipes connecting from above (R 7.2-24).

- **Summarising Assessment**

Whether flooding of buildings or parts of buildings leads to safety-relevant effects on the plant as a whole, depends on the possible leak volumes, the pump capacities, the room areas concerned, the engineered safeguards installed in these rooms, the detection possibilities, as well as the possible counter-measures by the operational staff. Particularly important from the standpoint of safety are above all those events which can lead to flooding of several redundancies and thus to the failure of several trains of the engineered safeguards. Safety-relevant components of which only one exists, for example, the emergency control room must be protected particularly carefully against loss of function.

### 7.2.3 Drop of Loads

- **Description**

The following cranes and elevators or the intended locations of cranes and elevators, respectively, were subjected to a conceptual evaluation:

| No. | Crane/reference | Location/room |
|---|---|---|
| 1 | Polar crane<br>320 t/160 t<br>2 x 70 t | Containment<br>GA 701 |
| 2 | Bridge crane<br>125 t/20 t<br>2 x 100 t | Turbine hall<br>2 pcs |
| 3 | Semiportal crane/bridge crane<br>15 t | Turbine hall |
| 4 | Overhead crane<br>5 t | Turbine hall,<br>degasser extension, above feedwater tank |
| 5 | Overhead crane<br>8 t | Turbine hall,<br>degasser extension, above feedwater tank |
| 6 | Bridge crane<br>20 t/5 t | Turbine hall,<br>degasser extension,<br>feedwater tank section above feedwater pumps |
| 7 | No documents referring to this crane | Surrounding outer building<br>A 820<br>valve chamber |
| 8 | No documents referring to this crane | Surrounding outer building<br>A 911.1<br>ventilation centre |
| 9 | No documents referring to this crane | Surrounding outer building<br>A 911.2<br>ventilation centre |
| 10 | No documents referring to this crane | Emergency power building |
| 11 | Bridge crane<br>32 t/1 t | Central active workshop,<br>storage for fresh fuel elements |
| 12 | Fuel-element-handling machine | Containment,<br>reactor hall |

- **Assessment Criterion**

Cranes which can cause danger by drop of a load are subject to the requirements of KTA Rule 3902 on the design of cranes in nuclear power plants.

- **Assessment**

The essential design characteristics of the above cranes of the Stendal A plant were compared with the requirements of KTA Rule 3902.

The examination of the cranes showed that the crane systems and the fuel-element-handling machine essentially correspond to Section 3.0 (General Conditions) of KTA Rule 3902.

Corresponding to the effects of dropping loads, the polar crane 320 t/160 t/2 x 70 t in the containment and the 10-t electric hoist on the gantry crane situated on the supports of the polar bridge crane have to meet the requirements of Section 4.3 (increased demands) of KTA-Rule 3902. It is considered necessary that the cranes be upgraded in order to comply with KTA Rule 3902. The corresponding evidence will have to be presented (R 7.2-25).

The cranes indicated under Nos. 2 to 9 of the above list also have to comply with the additional requirements of KTA Rule 3902, Section 4.5, unless it is possible to avoid any transport processes during power operation of the plant completely or to limit the possible consequences of a load drop by hardware measures and restrictions of the crane's use, to such a degree that the dangers according to KTA Rule 3902, Section 4.2, need not to be applied. The demand that the additional requirements of KTA Rule 3902 be fulfilled make an upgrading of the cranes necessary. The corresponding evidence will have to be presented (R 7.2-26).

Corresponding to the effects of a load drop, the fuel element handling machine has to meet the requirements of KTA Rule 3902, Section 4.4. It is considered necessary to adapt the fuel element handling machine accordingly, unless this has already been carried out. The corresponding evidence will have to be presented (R 7.2-27).

For the other cranes, like

- the cranes (design unkown) in the emergency power building and
- the bridge crane 32 t/1 t in the central active workshop (storage of fresh fuel elements)

no additional backfitting measures or adaptations are deemed to be necessary, on the basis of the requirements of KTA Rule 3902.

## 7.3 Radiological Protection of Labour

### 7.3.1 Introduction

In the assessment of the radiation protection of the personnel, it has to be assumed that radiation protection for the Technical Project of the Stendal NPP, in accordance with the contractual agreements between the USSR and the GDR, was based on the Soviet "Standards for Radiation Protection" (NRB-76), the "Sanitary regulations for design and operation of nuclear power plants" (SPAES-79) as well as the "Basic sanitary principles for the contact with radioactive substances" (OSP-72), valid during the 70ies /NRB 76/, /SPA 79/, /OSP 72/. Furthermore, for the scope of design of the purchaser, the then valid GDR radiation protection regulations /SSV 69/, /DBS 69/, as well as the actualised versions /VOA 84/, /DBV 84/ were applied in the course of time.

The administrative and technical measures taken within the overall concept of operational radiation protection to ensure radiological protection of labour, as well as the radiation exposure of the personnel during power operation and maintenance works to be expected, are described below. The respective statements are based especially on the documents of the Stendal NPP Technical Project, the results of the expert opinion on the Technical Project by the Staatliche Amt für Atomsicherheit und Strahlenschutz (SAAS) of the GDR and other institutions, the binding offer relating to the 1st construction stage of Stendal NPP, correspondence between the SAAS and the VEB Kombinat Kraftwerksanlagenbau (KKAB), specifications of the Stendal NPP Technical Project as well as findings made during the first conceptual phase of the cooperation between Siemens AG and K.A.B. AG. A more detailed indication of the sources to all sections can be derived from the report /ACL 91/.

The entire concept of the operational radiation protection of the Stendal NPP comprises extensive organisational (e.g. division into zones, room classification) and technical measures (e.g. shielding of the most important sources of radiation, closed systems for activity retention, systems for reducing activity concentrations in fluid and gaseous media, radiation protection monitoring). As sources of ionising irradiation, the systems and the equipment of the primary and secondary system as well as the auxiliary circuits or auxiliary systems, of the NPP are taken into account and the nuclide-specific characterisation of the radiation sources are illustrated /TEP 81/.

Comparing the complex of organisational and technical measures of radiation protection introduced in the Technical Project with the radiation protection concept of an NPP meeting the requirements of the Federal German body of rules, it becomes evident that the respective overall concepts for ensuring radiation protection correspond to each other. But to arrive at a more detailed result, selected organisational and technical measures of the radiation protection concept are assessed below considering the StrlSchV (Radiation Protection Ordinance), the BMI Safety Criteria and KTA Rules. The testing laboratory building for metrology to monitor radiation protection intended at the Stendal NPP location represents a special problem in this context which is, however, not incorporated into the assessment, as the building does not belong to the power plant.

## 7.3.2 Organisational Measures

### 7.3.2.1 Division into Zones and Room Classification

- **Description**

In the Stendal NPP Technical Project all production rooms, buildings and plants are assigned to the control and monitoring sector. Rooms of the control area, in accordance with the design dose capacity $H_\gamma$, are again divided into permanently maintained rooms (with $H_\gamma \leq 12$ µSv/h), semi-maintained rooms (with $H_\gamma \leq 24$ µSv/h) and unmaintained rooms (with $H_\gamma \leq 240$ µSv/h, unattended during power operation). For any rooms in the monitored sector, $H_\gamma \leq 1$ µSv/h applies /TEP 81/.

- **Assessment Criteria**

The regulations of Sec. §§ 35, 57, 58, 60 StrlSchV (Radiation Protection Ordinance) are the assessment criteria for the division into zones. According to these regulations the radiation protection areas are to be divided into control and monitoring sectors, areas with $H\gamma$ > 3 μSv/h within the control areas are to be separated as exclusion areas. In addition thereto control and exclusion areas are to be marked with radiation signs and the addition "Control Area" or "Exclusion Area - No Entry", respectively.

Rooms are to be classified under consideration of DIN 25440. According to this standard rooms are classified into room classes corresponding to the maximum local dose to be expected in the generally accessible area (e.g. Class A up to 10 μSv/h Class B up to $10^2$ μSv/h, etc.).

- **Assessment**

The division into zones chosen by the designer principally corresponds to the division predetermined by Sec. §§ 58, 60 StrlSchV (Radiation Protection Ordinance). Slight differences result from the different criteria with respect to irradiation exposure (cf. section 7.3.4). It is, however, ensured that the radiation protection limits are not exceeded. The exclusion area defined according to Section § 57 StrlSchV (Radiation Protection Ordinance) principally corresponds to the category "unmaintained room". Here, the regulations on the design dose deviate from each other.

In this context it is recommended to mark the maintained, half-maintained and unmaintained rooms correctly according to the above criteria (R 7.3-1).

The rooms of the reactor building, the special water treatment (SWA) and the central active workshop (ZAW) have already been classified according to DIN 25440. These will, however, have to be examined with respect to the necessity of additional organisational and technical measures to observe the requirements of Section § 54 StrlSchV (Radiation Protection Ordinance) and KTA Rule 1301.1 (also cf. compare with the relevant recommendation in Section 7.3.3.1).

In addition, it is recommended to provide appropriate measures that exclude or minimise the necessity of persons entering the unmaintained containment rooms during operation (R 7.3-2).

#### 7.3.2.2 Hygiene Wing

- **Description**

The hygiene wing (transfer canal area) representing the connection between the control area and the monitoring area is designed for the complete exchange of outdoor clothing of the personnel working in the control area with wardrobes, showers, rooms to control surface contamination of the body as well as storage rooms for individual protective agents and clean or contaminated special clothing /TEP 81/. In this context it is planned to realise the entry and exit of personnel to or from the control areas of the intended four units through one common hygiene wing in the special building.

- **Assessment Criteria**

According to KTA Rule 1301.1 for the design of the hygiene wing a reserve for outside personnel of three times the plants-own personnel must be provided. Sufficient space for changing and washing as well as a sufficient number of contamination monitors must be available.

- **Assessment**

The present concept of the hygiene wing does not meet the above criteria. To ensure smooth and clearly separated passage of the new and old personnel at change of shift, the design and the equipment of the hygiene wing should be revised or enlarged, assuming a personnel demand of 300 employees from the plant and 900 employees from outside (R 7.3-3).

#### 7.3.2.3 Organisational Structure of Radiation Protection

The designer prescribed the personnel framework for the organisation of radiation protection during operation of the individual construction stages of Stendal NPP in the Technical Project. For an assessment, this organisational structure has to be specified under consideration of Sec. §§ 29, 30, 31 StrlSchV (Radiation Protection Ordinance). Special attention must be paid to the determination of responsibilites.

### 7.3.3 Technical Measures

#### 7.3.3.1 Shielding

- **Description**

By stationary shielding (biological protection) the equivalent doses in the control and monitoring area are reduced to a level which ensures observance of and values below the design doses predetermined according to the room classification (cf. Section 7.3.2.1). According to the Technical Project reinforced concrete having a density of 2.1 or 3.3 g/cm$^3$, water and different metal constructions are used. The methods for calculating the biological protection as well as the nuclide spectra and source strengths for the main radiation sources of the reactor building are described in the Technical Project. The information on shielding radiation of the radioactive systems are contained in the binding offer. In addition, the shielding design was worked out for the pipes of the pipeline bridge leading to the special water treatment building carrying radioactive media. It must be mentioned that there are test requirements in connection with the quality assurance of the shielding constructions.

Furthermore, the deviations from the project which occured during the construction of the reactor shielding are to be mentioned which, after examination by SAAS, led to a temporary standstill in the construction. The VEB KKAB in 1989 was required to determine the effects of these deviations from the project with respect to the change of the radiation field and to derive the approriate measures for ensuring radiation protection in the adjacent rooms.

- **Assessment Criteria**

According to Sec. § 54 StrlSchV (Radiation Protection Ordinance) shielding is to be dimensioned under consideration of the total access time in such a way that the radiation exposure during the course of normal operational cannot exceed one fifth of the values of plant X, Table X1, Column 2 StrlSchV (Radiation Protection Ordinance) (for example the effective dose of 10 mSv/a for personnel of category A).

Furthermore it is requested according to KTA Rule 1301.1 that the shielding walls are to be dimensioned in such a way that contribution to the local dose from the adjacent room is a maximum of 20 % of the upper room class limit. With respect to the shielding of selected workplaces (more than 1000 h/a attended rooms, hygiene wing, first-aid-room) or frequently used pathways, the local dose may not exceed the value of 5 or 10 µSV/h, respectively.

- **Assessment**

The thickness of the walls in the rooms of the controlled area must be examined as to whether they comply with the demands mentioned above; if necessary, measures must be determined to upgrade the shielding or to limit access periods (R 7.3-4).

The deviations from the project state that arose during the construction of the reactor shielding have to be analysed as to the expected changes in the radiation field (R 7.3-5).

#### 7.3.3.2 Radiation Protection Monitoring

- **Description**

According to the Technical Project, the stationary Radiation Protection Monitoring (SSÜ) would originally be realised with the AKRB-03 system, the overall concept of which is based on a variety of stationary gauges for system-related and dosimetric radiation control, with transfer of the values measured to the central dosimetric control room for all four units or to the unit-related radiation protection monitoring control desks. Apart from the measuring channels for normal operation a number of devices were intended here which should also ensure the provision of information during accidents. In the design phase of the project, the subsequent AKRB-08 system was offered which, however, only insignificantly differs from its predecessor. For human dosimetric monitoring the UI-27 system should be used.

- **Assessment Criteria**

Basic requirements to be met by the radiation protection monitoring of NPP are contained in BMI Criterion 10.1. They concern the personnel, organisational, spatial and equipment related preconditions for radiation protection monitoring in the plant and they refer to the scope of the necessary measurement equipment. The KTA Rules comprise a specification of these requirements.

In addition thereto criterion 10.2 of the BMI Safety Criteria is to be mentioned which contains the requirements with respect to activitiy monitoring in exhaust air and waste water and which is explained in the KTA Rules.

- **Assessment**

The essential disadvantages of the AKRB-03 system and its further development AKRB-08 especially concern the evidence limits of the measurement areas for recording radioactive releases during normal operation and during accidents beyond design limits, as well as deficiencies in the illustration of measured data. The measurement system UI-27 for human dosimetric monitoring also exhibits essential deficiencies. It is to be replaced by a system which, apart from recording the dose per person, also renders dose warnings, access monitoring and coupling with computer-aided assessment of data possible.

This results in the general requirement that measuring systems for the radiological monitoring of the technical system and dose rates must be modified according to the state of the art (R 7.3-6). The concept of the routine centralised monitoring of a variety of measuring parameters can partially be replaced by demand-related measurements, if the necessity for the routine access to plant rooms during power operation can be reduced. But, the development of a new concept for the use of portable measuring devices is a precondition for this.

## 7.3.4 Radiation Exposure of the Personnel

### 7.3.4.1 Radiation Protection Limits

- **Description**

The Technical Project of the Stendal NPP is based on the radiation protection limits contained in Table 7.3-1 in which the equivalent dose limits for organs of groups I, II and III for persons belonging to category A (personnel professionally exposed to radiation) and category B (personnel not exposed to radiation) are listed.

- **Assessment Criteria**

The regulations of Sec. § 49 StrlSchV represent the basis for the assessment. In particular, reference is made to the values illustrated in Table 7.3-2 (cf. also Annex X, Table X1, StrlSchV) where, in contrast to NRB-76, category B according to StrlSchV for professionally exposed personnel applies. In addition, the sum of the effective doses of persons professionally exposed to radiation, determined in all calendar years may not exceed 400 mSv (age-related dose) according to Sec. § 49, Subsec. 1 StrlSchV.

- **Assessment**

A comparison with the equivalent dose limits stated in Table 7.3-1 for different groups of organs with the body dose limits of Table 7.3-2 shows that the radiation protection limits on which the design of the Stendal NNP was based correspond to the criteria of the StrlSchV (Radiation Protection Ordinance), although it must be mentioned that an age-related dose is not determined in the Soviet radiation protection standard NRB-76. It must therefore be examined whether measures are required for special maintenance personnel to keep them within their age-related dose (R 7.3-7).

#### 7.3.4.2 Exposure to Radiation during Power Operation and Maintenance Work

- **Description**

Based on the Technical Project there is information on the irradiation situation during power operation for the rooms of the reactor building and the surrounding outer building, where a maximum leak of the primary system of 0.2 t/h is assumed. There is, however, no information on the scope of routine work to be expected during power operation or on the resulting individual and collective exposure values derived.

With respect to the execution of maintenance works the Project corresponds to the degree of specification of the respective determinations of design regulation SPAES-79 /SPA 79/. Here, in particular, the use of remote-controlled systems and of facilities and equipment which are easy to handle, have not been considered to a sufficient degree.

- **Assessment Criteria**

According to Sec. § 28 Radiation Protection any unnecessary radiation exposure of persons is to be avoided and to be kept as low as possible, even below the radiation protection limits mentioned in Section 7.3.4.1 considering the state of the art.

There are detailed requirements referring to the preventive measures to be taken during the design of the plant for radiation protection of the personnel during maintenance works in the respective IWRS Guideline as well as KTA Rule 1301.1. They especially refer to the reduction of the local dose, as well as the arrangement and design of systems and components. Furthermore, determinations for verifying respective preventive measures are stipulated in the IWRS Guideline.

- **Assessment**

Considering the principles set forth in Sec. § 28, StrlSchV (Radiation Protection Ordinance), the overall concept of the primary system has to be revised with a view to minimising the occurence of leaks (R 7.3-8). The extent of routine work during maintenance and power operation modes as well as the resulting individual and collective exposures to radiation must be analysed. Measures for a further reduction of radiation exposure are to be derived from this analysis (R 7.3-9). Evidence for preventive radiation-protection measures according to the IWRS Guideline has to be presented (R 7.3-10).

Considering the conclusions arrived at in the course of the cooperation between Siemens and K.A.B. AG, the following recommendations are also to be mentioned: For the performance of maintenance work, the latest equipment in modern inspection technology is to be used. Any work that has to be carried out under intense radiation is to be automated to the largest possible degree (R 7.3-11). Considering the experience during the design of the confinement system of the Greifswald NPP, Unit 5, storage space and temporary stores, as well as moving space for maintenance measures, are to be created by locally changing the arrangement of components and pipe routes (R 7.3-12). Furthermore, modern breathing apparatus is to be provided for maintenance work with potential inhalation dangers (R 7.3-13).

## 7.4 Radiation Protection of the Surrounding Population

### 7.4.1 Disposal with Outgoing Air

- **Description**

- Disposal Rates

According to /SIE 90/, at Stendal NPP, Unit A, the following application values were determined for the disposal of radioactive substances with the outgoing air:

| | | |
|---|---|---|
| - Radioactive gases | $2 \times 10^{15}$ | Bq/a |
| - Aerosols | $5 \times 10^{10}$ | Bq/a |
| - I 131 | $2 \times 10^{10}$ | Bq/a |

For the determination of the radiation exposure, the radionuclides tritium and carbon 14, which, as known from experience, are released by nuclear power plants with pressurised water reactors via the outgoing air have to be considered. In accordance with the pragmatical values for the disposal of H 3 and C 14 from nuclear power plants with pressurised water reactors in operation in the Federal Republic of Germany /BMI 83, BMU 86/ annual disposal rates of $3.7 \times 10^{12}$ Bq for H 3 and $5.6 \times 10^{11}$Bq for C 14 have to be assumed for every nuclear power plant unit in a conservative estimate of the radiation exposure. These values are also assumed for Unit A of the Stendal Nuclear Power Plant.

The calculation of the radiation exposure for Unit A is thus based on the following disposal rates:

| | | |
|---|---|---|
| - Radioactive gases | $2 \times 10^{15}$ | Bq/a |
| - Aerosols (half-life value > 8 d) | $5 \times 10^{10}$ | Bq/a |
| - I 131 | $2 \times 10^{10}$ | Bq/a |
| - H 3 | $3.7 \times 10^{12}$ | Bq/a |
| - C 14 | $5.6 \times 10^{11}$ | Bq/a |

According to the statements in /PLW 90/ the disposal rates are distributed to the emission sources "stack (height 100 m) of the reactor building of Unit A" and "stack (height 50 m) of the central active workshop ZAW" in the following way:

| Source of emission | Stack reactor building A | Stack ZAW |
|---|---|---|
| Radioactive gases | 100 % | - |
| H 3 | 100 % | - |
| C 14 | 100 % | - |
| Aerosols | 90 % | 10 % |
| I 131 | 80 % | 10 % |

Many years experience on the nuclide composition of the individual groups of nuclides does not exist. For this reason the calculations are based on the respective statements in /AVV 90/ for nuclear power stations with pressurised water reactors.

- Spreading Factors

To determine the effects of the disposal of radioactive substances long-term spreading and long-term washout factors were calculated using the meteorological data of /PWL 90/ according to the procedures in /AVV 90/. The values for the spreading factors for the most adverse position of reaction which was determined taking the influence of building and cooling tower into account, were;

| | $X^G_\gamma$ $s \times m^{-2}$ | $X^G$ $s \times m^{-3}$ | $W^G$ $m^{-2}$ | $W^S$ $m^{-2}$ |
|---|---|---|---|---|
| Stack reactor building | $1 \times 10^{-4}$ | $4 \times 10^{-7}$ | $1.4 \times 10^{-9}$ | $1.4 \times 10^{-9}$ |
| Stack ZAW | $3 \times 10^{-4}$ | $4 \times 10^{-6}$ | $1.4 \times 10^{-9}$ | $1.4 \times 10^{-9}$ |
| $X^G_\gamma$ | long-term spreading factor for the whole year for gamma submersion | | | |
| $X^G$ | long-term spreading factor for the whole year | | | |
| $W^G$ | long-term wash-out factor for the whole year | | | |
| $W^S$ | long-term wash-out factor for the summer term | | | |

- **Assessment Criterion**

The regulations of Sec. § 45 StrlSchV (Radiation Protection Ordinance) are the assessment criterion for the protection of the population and the environment against ionising radiation. In Sec. § 45 the dose limit values for body doses for the radiation exposure of human beings owing to the disposal of radioactive substances by air or water are determined.

In the Federal Republic of Germany the permitted disposal values for radioactive substances released with the outgoing air for nuclear power plants with pressurised water reactors having an electrical gross power of up to 1400 MW are as follows:

| Source of emission | Bq/a |
|---|---|
| Noble gases | $\leq 1.11 \times 10^{15}$ |
| I 131 | $\leq 1.63 \times 10^{10}$ |
| Aerosols (half-life value > 8 d) | $\leq 4.00 \times 10^{10}$ |

- **Assessment**

The application values for Unit A of the Stendal Nuclear Power Station with a comparable reactor power are clearly above the values permitted in the Federal Republic of Germany.

The calculation of the potential radiation exposure owing to the disposal of radioactive substances was based on the calculation procedures and parameters of the "Allgemeine Verwaltungsvorschrift zu § 45 StrlSchV" (General administrative regulation to Sec. § 45 StrlSchV (Radiation Protection Ordinance)) using the preconditions of emission rates and the nuclide compositions described above.

The following exposure pathways are considered in accordance with Appendix XI to Sec. § 45, Subsec. 2 StrlSchV for disposals by air:

- Exposure by beta radiation within the waste air
- Exposure by gamma radiation from the waste air
- Exposure by gamma radiation of the radioactive substances deposited on the ground

- Air - plant
- Air - forage plant - cow - milk
- Air - forage plant - animal - meat
- Inhaled air.

The values of the potential radiation exposure are calculated for the most adverse positions of reactions. For disposal with the outgoing air these are immediately beyond the borders of the nuclear power plant site. The following maximum values for the potential radiation exposures are obtained in detail:

| Organs of the body | Equivalent dose in $10^{-5}$Sv/a | | |
|---|---|---|---|
| | Adult | Infant | Limits accord. to Sec. § 45 StrlSchV |
| Gonad,<br>uterus,<br>red marrow | ≤ 9.0 | ≤ 8.4 | 30 |
| Surface of bones,<br>skin | ≤ 11.0 | ≤ 14.0 | 180 |
| Thyroid,<br>other organs | 17<br>≤ 9.2 | 34<br>≤ 8.7 | 90<br>90 |
| Effective equivalent dose | 9.4 | 9.1 | 30 |

The values calculated for the potential radiation exposure during normal operation of Unit A show that the dose limits of Sec. § 45 StrlSchV for the disposal of radioactive substances with the outgoing air can be observed at the Stendal location. For disposals with the outgoing air the radiation exposures are at least 2.5 times below the dose limits of Sec. § 45 StrlSchV. This shows that the Stendal location with respect to the disposal of radioactive substances with the outgoing air during normal operation is suited for the erection of a nuclear power plant unit.

### 7.4.2 Disposal with Waste Water

- **Description**

- Disposal Rates

According to /SIE 90/ the following application values for the disposal of radioactive substances were determined for the Stendal Nuclear Power Plant:

| | | |
|---|---|---|
| - Nuclide mixture without tritium | $2 \times 10^{11}$ | Bq/a |
| - Tritium | $2 \times 10^{14}$ | Bq/a |

The erection of four power plant units was planned at the location so that an average of 1/4 of the above location values is to be taken as a basis.

Experience values on the composition of nuclides over many years are only available to a limited degree. For this reason the calculations for determining radiation exposure are based on the spectrum of radioactive disposals with the waste water from nuclear power stations with light water reactors mentioned in the general administrative regulation relating to Sec. § 45 StrlSchV.

The radioactive waste water of the Stendal Nuclear Power Plant shall be discharged into the River Elbe. The average water regime in the area of the discharge position according to /PLW 90/ is:

| | |
|---|---|
| - arithmetic average of the flowrate during the entire year | 571 $m^3/s$ |
| - arithmetic average of the flowrate during summer half year | 447 $m^3/s$ |

- Radiological Bias of the Elbe

To determine potential radiation exposure resulting from radiological bias of the Elbe according to /PLW 90/ the following concentrations of radioactive substances have to be assumed conservatively:

| Nuclide | Concentration Bq/m$^3$ |
|---|---|
| Cr 51 | $2 \times 10^{-2}$ |
| Co 57 | $1 \times 10^{-3}$ |
| Co 58 | $4 \times 10^{-3}$ |
| Fe 59 | $5 \times 10^{-4}$ |
| Sr 90 | 15 |
| Tc 99 m | $5 \times 10^{-1}$ |
| I 125 | $1 \times 10^{-2}$ |
| I 131 | 250 |
| Cs 134, Cs 137 | 20 |
| Yb 169 | $1 \times 10^{-1}$ |
| Tl 201 | $8 \times 10^{-1}$ |

- **Assessment Criterion**

The regulations of Sec. § 45 StrlSchV (Radiation Protection Ordinance) to protect population and environment against ionising radiation represent the assessment criteria. In Sec. § 45 StrlSchV the dose values of body doses for the radiation exposure of human beings resulting from the disposal of radioactive substances by air or water are determined.

The approved values for the disposal of radioactive substances with the waste water from nuclear power plants with pressurised water reactors having an electrical gross power of up to 1400 MW in the Federal Republic of Germany for the

- nuclide mixture without titrium are $\leq 1.85 \times 10^{11}$ Bq/a and

- for tritium $\leq 5.0 \times 10^{13}$ Bq/a.

- **Assessment**

The application values of the Stendal Nuclear Power Plant with a comparable reactor power are significantly higher than the approved values in the Federal Republic of Germany.

The calculation of the potential radiation exposures resulting from the disposal of radioactive substances is based on the calculation procedures and parameters of the general administrative regulation relating to Sec. § 45 StrlSchV applying the preconditions to emission rates and nuclide compositions mentioned in the previous sections.

In accordance with Appendix XI of Sec. § 45, Subsec. 2 StrlSchV the following exposure pathways are considered for disposals with water:

- stay on sediment
- drinking water
- water - fish
- cattle watering tank - cow - milk
- cattle watering tank - animal - meat
- overhead irrigation - forage plant - cow - milk
- overhead irrigation - forage plant - animal - meat
- overhead irrigation - plant

The values of the potential radiation exposures are calculated for the most adverse positions of reactions. For disposals with the waste water of the Stendal Nuclear Power Station, Unit A, these are near the discharge position. The following maximum values for the potential radiation exposures are obtained in detail:

| **Organs of the body** | **Equivalent dose in $10^{-5}$Sv/a** | | |
|---|---|---|---|
| | **Adult** | **Infant** | **Limits accord. to Sec. § 45 StrlSchV** |
| Gonad,<br>uterus,<br>red marrow | ≤ 0.2 | ≤ 0.1 | 30 |
| Surface of bones,<br>skin | ≤ 0.2 | ≤ 0.1 | 180 |
| Thyroid,<br>other organs | 0.2<br>≤ 0.2 | 0.2<br>≤ 0.1 | 90<br>90 |
| Effective equivalent dose | 0.2 | 0.1 | 30 |

The concentrations of radioactive substances in the Elbe owing to the radiological bias for the potential radiation exposures via the water pathway result in the following values:

| Organs of the body | Equivalent dose in $10^{-5}$Sv/a | |
|---|---|---|
| | Adult | Infant |
| Gonad,<br>uterus,<br>red marrow | ≤ 3.1 | ≤ 1.8 |
| Surface of bones,<br>skin | ≤ 5.1 | ≤ 2.8 |
| Thyroid,<br>other organs | 26<br>≤ 1.9 | 39<br>≤ 1.4 |
| Effective equivalent dose | 2.8 | 2.5 |

Altogether the following values for the waste water pathway of the Stendal location were calculated:

| Organs of the body | Equivalent dose in $10^{-5}$Sv/a | | |
|---|---|---|---|
| | Adult | Infant | Limits accord. to Sec. § 45 StrlSchV |
| Gonad,<br>uterus,<br>red marrow | ≤ 3.3 | ≤ 1.9 | 30 |
| Surface of bones,<br>skin | ≤ 5.2 | ≤ 2.9 | 180 |
| Thyroid,<br>other organs | 26<br>≤ 2.1 | 39<br>≤ 1.4 | 90<br>90 |
| Effective equivalent dose | 2.9 | 2.5 | 30 |

The potential radiation exposures calculated for the normal operation of Unit A on the whole show that the dose limits of Sec. § 45 StrlSchV for the disposal of radioactive substances with waste water can be met at the Stendal location. Without considering the radiological bias, the radiation exposures calculated for disposals with waste water are below the dose limits of Sec. § 45 StrlSchV by a factor 30. This shows that that the Stendal location is suitable for the erection of a nuclear power plant unit with respect to the disposal of radioactive substances during normal operation. The erection of further nuclear power plant units at the Stendal location, in particular because of the relatively high bias of the drainage canal, requires further investigations with respect to the disposal of radioactive substances.

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| /SIE 90 a/ | Siemens AG<br>KKW Stendal I, Block A<br>Teil: System- und Anlagentechnik<br>Untersuchungen zu einer vorläufigen Bewertung der sicherheitstechnischen Auslegung<br>(Stendal NPP I, Unit A, Part: Systems and Plant Technology, Investigation for a Provisional Assessment of the Safety-Related Design)<br>Vol. 1, Edition 08/90 |
| /SIE 90 b/ | Siemens AG<br>KKW Stendal I, Block A<br>Teil I: System- und Anlagentechnik<br>Untersuchungen zu einer vorläufigen Bewertung der sicherheitstechnischen Auslegung<br>(Stendal NPP I, Unit A, Part: Systems and Plant Technology, Investigation for a Provisional Assessment of the Safety-Related Design)<br>Band 2, Ausgabe 08/90 |
| /SPA 79/ | Sanitäre Vorschriften für die Projektierung und den Betrieb von Kernkraftwerken, SPAES-79<br>(Sanitary Regulations for Design and Operation of Nuclear Power |

Plants)
Moskau, 1979

/SSV 69/ Verordnung vom 26.11.1969 über den Schutz vor der schädigenden Einwirkung ionisierender Strahlung - Strahlenschutzverordnung (Ordinance of November 26, 1969 on the Protection against Damaging Impacts of Ionising Radiation - Radiation Protection Ordinance)
GBl der DDR, 1969, Part II, Nr. 99, p. 627

/TEP 81/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project)
Part 5, Section 5.1, Vol. 1
Moscow 1981

/TEP 81a/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project)
Part 8, Section 3.13, Safety System
GRS-Reg.-No.: PL-WWER-91/0028-1

/TEP 81b/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project
Part 8, Section 3.14, Fire Protection Means and Signallisation)
GRS-Reg.-No.: PL-WWER-91/0028-1

/TEP 81c/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP Technical Project
Part 8, Section 3.4, Electrical Engineering Explanatory Report)
GRS-Reg.-No.: PL-WWER-91/0016-1

/TEP 81d/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project
Part 4: Water Supply and Discharge Channels, Section 4.2 Water Supply and Discharge Channels)
GRS-Reg.-No.: PL-WWER-91/0023-1

/TEP 81e/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project

Part 4: Water Supply and Discharge Channels,
Section 4.2 Fire Protection Measures Main Building)
GRS-Reg.-No.: PL-WWER-91/0023-1

/TEP 81f/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project
Part 4: Water Supply and Discharge Channels,
Section 4.2.1, Internal Water Lines and Discharge Channels, Main Building)
GRS-Reg.-No.: PL-WWER-91/0023-1

/TEP 81g/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project
Part 4: Water Supply and Discharge Channels,
Section 4.3, Stationary Automatic Fire-Fighting Installations)
GRS-Reg.-No.: PL-WWER-91/0023-1

/TEP 81i/ Atomenergoprojekt
KKW Stendal, Technisches Projekt
(Stendal NPP, Technical Project, Supplement No. 1)
GRS-Reg.-No.: PL-WWER-91/0030-1

/TEP 81h/ Teploelektroprojekt
KKW Stendal, Technisches Projekt (Stendal NPP, Technical Project
Part 4: Water Supply and Discharge Channels,
Section 4.4, Stationary Automatic Fire-Fighting Installations)
GRS-Reg.-No.: PL-WWER-91/0023-1

/TÜV 91/ Technischer Überwachungs-Verein Norddeutschland e.V.
Nuclear Technology and Radiation Protection Department
Bericht zur Konzeptbeurteilung der Hebezeuge im Kernkraftwerk Stendal
(Report on Concept Assessment of Cranes and Elevators in the Stendal Nuclear Power Plant)
Hamburg, November 1991

/UVA 84/ Kernkraftwerk Stendal, 1. Baustufe (Stendal Nuclear Power Station, 1st Construction Stage)

Unterlagen zum verbindlichen Angebot (Teil B), Band 5
(Documents on the Binding Offer (Part B), Vol. 5
VEB Kombinat Kraftwerksanlagenbau
Berlin, December 21, 1984

/VOA 84/ Verordnung über die Gewährleistung von Atomsicherheit und Strahlenschutz vom 11.10.1984
(Ordinance on Ensuring Atomary Safety and Radiation Protection of October 11, 1984)
GBl der DDR, 1984, Part 1, No. 30, p. 341

/VSN 79/ VSN-15-7
Zeitweilige Projektierungsnorm für Kernenergieanlagen in seismischen Gebieten
(Temporary Design Standards for Nuclear Energy Plants in Seismic Areas)
Moscow, 1979

## Tables, Section 7

| | | Own weight | Permanent working load | Wind and snow load | Design pressure | Resp. temperature | MRE/design earthquake | External blast wave | Airplane crash | Sub-presssure | Crane load |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Normal operation | Accord. to /BAK 85/ | X | X | X | | X | | | | | X |
| | Accord. to KTA 3401.2 | X | X | X | | X | | | | X | |
| Loss-off-coolant accident | Accord. to /BAK 85/ | X | X | | X | X | X | | | | |
| | Accord. to KTA 3401.2 | X | X | | X | X | | | | | |
| External Impacts | Accord. to /BAK 85/ | X | X | | | X | X | | | | |
| | | X | X | | | X | | X | | | |
| | | X | X | | | X | | | X | | |
| | Accord. to KTA 3401.2 | X | X | | | X | X | | | | |
| | | X | X | | | X | | X | | | |
| | | X | X | | | X | | | X | | |

**Table 7.1-1** Simplified summary restricted to the essential impacts of combined impacts of loads

| Group of Organs | Equivalent dose limit in mSv/a | |
|---|---|---|
| | Category A | Category B |
| I.<br>Whole body, gonads, uterus, red marrow | 50 | 5 |
| II.<br>Muscles, thyroid, adipose, liver, kidneys, spleen, gastro-intestinal tract, lungs, eye lens, and other organs (with the exception of organs mentioned under groups I and III) | 5.2 | 15 |
| III.<br>Skin, bony tissue, hands, lower arms, lower legs and feet | 150 | 30 |

**Table 7.3-1** Radiation protection limit for the design of the Stendal NPP (according to /NRB 76/)

| Emergency dose | Emergency dose limit per calendar year for persons professionally exposed to radiation in mSv | |
|---|---|---|
| | Category A | Category B |
| 1. Effective dose, partial body radiation dose: gonads, uterus, red marrow | 50 | 15 |
| 2. Partial body radiation dose: all organs and tissues not mentioned under 1., 3. or 4. | 150 | 45 |
| 3. Partial body radiation dose: thyroid, surface of bones, skin, if not mentioned under 4. | 300 | 90 |
| 4. Partial body radiation dose: hands, lower arms, feet, lower legs, knuckles and corresponding skin | 500 | 150 |

**Table 7.3-2** Limits of emergency doses in the calendar year for persons professionally exposed to radiation in mSv (according to the Radiation Protection Ordinance)

## Figures, Section 7

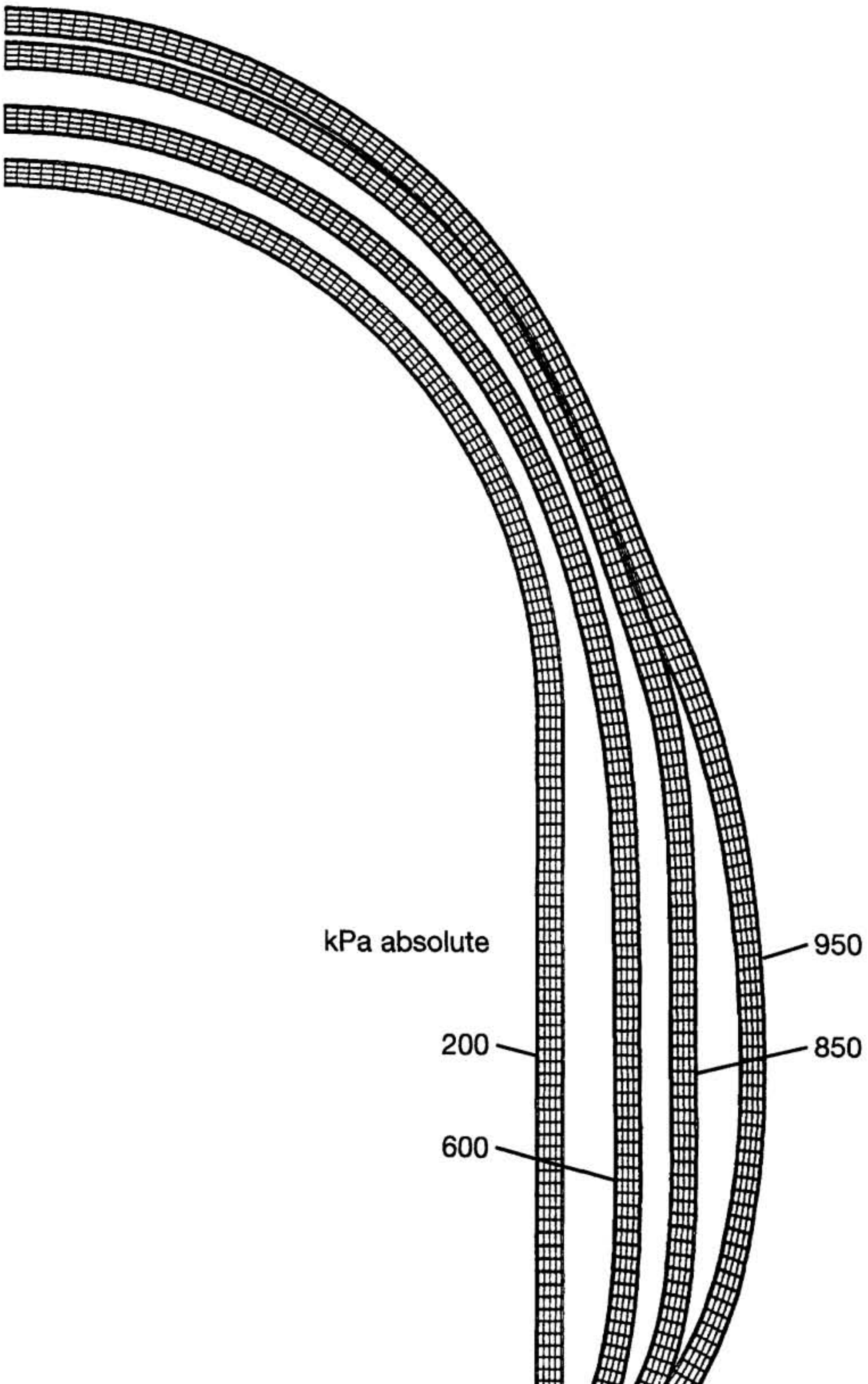
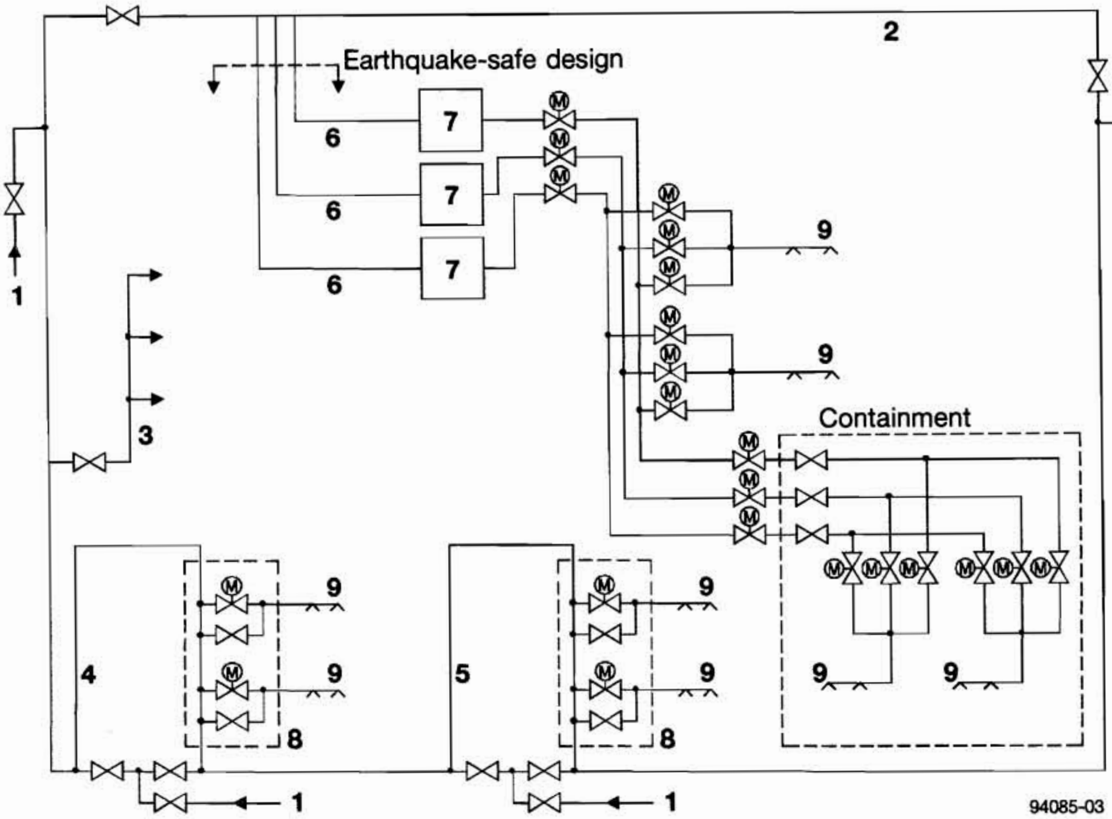



**Fig. 7.1-1** Relative deformations of the containment of Stendal NPP uopn different internal pressures




**1** from fire-fighting water system on site
**2** Circular fire-fighting water line in the reactor building
**3** Raising pipes in the staircases
**4** Spray water extinguishing area of the MCP
**5** Spray water extinguishing area of the make-up system pumps
**6** Spray water extinguishing area for redundant, safety-relevant system and cable rooms
**7** Fire-fighting water storage tank
**8** Valve chambers
**9** Extinguishing sections

**Fig. 7.2-1** Layout of the fire fighting system in the reactor building

# 8 Evaluation of Operating Experience of other WWER-1000 Plants

## 8.1 Introduction

An analysis of the operating experience of plants of the WWER-1000 reactor type was performed in the investigations for a safety assessment of the planned Stendal Nuclear Power Plant. Plants of this type are being operated in the Soviet Union and in Bulgaria with a total of about 75 reactor years. A specific investigation of the prototype (Novo-Voronesh 5), the plants of the "small series" (Kalinin 1 and 2, South Ukraine 1 and 2) and the remaining plants of the "unified series" to which the planned Stendal plant also belongs, could not be performed, as there is no detailed information on the technical differences between the individual types.

Incident reports within the framework of the "Incident Reporting System" of the IAEO (IRS) and the "ISI-System" coordinated by Interatomenergo in Moscow, which has recorded incidents in the area of the former COMECON states since 1988, are the basis of the investigations. A total of 64 incident reports from 15 plants, 59 taken from IRS (until August 1990) and 34 from ISI (until the end of 1989), 29 of which were also contained in IRS, were presented for investigation.

The evaluation of the incidents will support the concept assessment of WWER-1000, while providing hints on areas for improvements. The analysis therefore was not restricted to the present sequences of events, but furthermore general upgrading measures are recommended.

The evaluation serves to determine whether

- the frequency and types of events,
- the sequences of those events,
- the frequencies of component and systems failures

provide any information about

- design deficiencies in the combined actions of systems functions,
- defects in the design of systems and components,

- deficiencies in component reliability during operation and upon demand,
- shortcomings in plant management.

The events occuring were subdivided into classes of events for purposes of systematisation. For events which related to process-technological installations, a distinction was made between

- disturbances in the control and protection system of the reactor
- disturbances in the primary system
- failures and disturbances in the emergency cooling system (emergency core cooling and residual heat removal system),
- failures in the feedwater system,
- leakages in the primary system and
- disturbances in the pressure protection of pressuriser and steam generator.

For events concerning electrical and I & C installations it was differentiated between

- disturbances at the emergency power diesels
- failures of the power supply of important safety-relevant consumers
- failures in the instrumentation and control system

Furthermore, events where deficiencies in quality assurance could be perceived were summarized in an independent class.

## 8.2 Selected Events

Of the total of 64 existing events, 24 events selected because of their characteristic sequence will be described in detail and assessed in Section 8.3. In addition thereto, eleven other events are briefly described in the text. In addition, 14 further cases are mentioned to provide a broader impression of the present operating experience.

For the events described in detail in Section 8.3, the course of event is always described first and the causes for the failures which occured are given. After that the respective upgrading measures are stated, which in our opinion are required to remove the deficits occuring during the event described. For events which, according to the documents available, represent individual cases, upgrading measures are also suggested, if these events refer to severe safety-related deficits. In some cases additional measures are listed which the operator has taken because of the event or which he has planned to take. These measures are specially characterised as measures for the operator.

### 8.2.1 Completeness of the Existing Events

To estimate to what extent the events occuring are considered by IRS or ISI, respectively, the computer-based database Power-Reactor-Information-System (PRIS) of IAEP, Vienna, was analysed for WWER-1000.

In this database the power reductions and shutdowns of plants reported by operators to IAEO are recorded. Until the end of 1989 there were approx. 210 WWER-1000 cases referring to shutdowns and power reductions initiated by disturbances in the plants. Because of the keyword descriptions of events in the IAEO data bank there are uncertainties in the assessment of the actual safety-related relevance of these events. It can, however, be seen that apart from the 64 events reported to IRS or ISI, a large number of similar events occured. A rough estimate shows that less than 20 % of the "PRIS events" were reported to the other sources of data. For this estimate, it must also be taken into account that safety-relevant events, which occured without reduction of power during the event or where the reduction of power was possibly very small with respect to duration or size, as well as events which occured during shutdown or during the revision phase, were not reported to the PRIS database system.

These reflections show that the present 64 reports of events only represent a small fraction of the disturbances which occured. It therefore cannot be expected that all phenomena and weaknesses of the plants were considered in the analyses.

## 8.3 Evaluation of the Selected Events

In the following overview the incidents described below are arranged according to their time sequence. The section in which the events are described is indicated. It is further stated whether the case is described in summary (S) or in detail (D). The events referred to as KS are incidents in plants of the "small series" or from the prototype plant.

| Date of Event | Event | Section | Description |
|---|---|---|---|
| 11.05.1984 | Loss-of-coolant because of inadverted opening of a pressuriser safety valve (KS) | 8.3.6 | D |
| 15.06.198 | Imbalance in the thermal power of the primary and secondary system during a fast power reduction (KS) | 8.3.9.6 | D |
| 09.07.1986 | Reactor scram owing to failure of oil supply for all four main coolant pumps after initiation of the safety system | 8.3.2 | D |
| 30.12.1986 | Reactor scram after interruption of oil supply for two out of four main coolant pumps (KS) | 8.3.9.1 | S |
| 08.01.1987 | Failure of power supply for one train of the safety system | 8.3.8.4 | D |
| 13.01.1987 | Automatic power reduction by failure of power measurement in one main coolant pump | 8.3.9.1 | D |
| 25.01.1987 | Reactor scram by a wrong signal from main-steam pressure measurement (KS) | 8.3.9.2 | S |
| 09.02.1987 | Reactor scram because of wrong signal: "Water level in high-pressure preheater high" | 8.3.9.3 | D |
| 14.06.1987 | BRU-A valves stuck open (KS) | 8.3.6 | D |
| 11.01.1988 | Non-availability of a diesel generator because of decrease in battery voltage | 8.3.7 | S |
| 19.01.1988 | Failure of a HP-emergency cooling pump during in-service inspection (KS) | 8.3.3 | D |
| 30.01.1988 | Reactor scram by non-closure of the pressuriser injection valve | 8.3.9.4 | D |
| 08.02.1988 | Reactor scram after pressure increase in the containment by leakage at the pressuriser injection valve (gland) | 8.3.5 | S |
| 22.03.1988 | Reactor scram after a leak in a pulse line of the water-level measurement of the pressuriser | 8.3.10 | D |
| 27.03.1988 | Reactor scram by incorrect initiation of the bypass station and failure of turbine control | 8.3.10 | S |

| Date of Event | Event | Section | Description |
|---|---|---|---|
| 05.01.1989 | Delayed absorber rod drop after failure of two main coolant pumps (KS) | 8.3.1 | D |
| 17.01.1989 | Fault in the exitation circuit of a diesel generator | 8.3.7 | D |
| 01.02.1989 | Reactor scram owing to a defect in the turbine control and handling mistake at main feedwater pump caused by wiring fault | 8.3.10 | S |
| 19.02.1989 | Reactor scram after defect in reactor power control | 8.3.9.4 | D |
| 22.02.1989 | Reactor scram by incorrect actuation of protection of condensate pumps and generator | 8.3.8.1 | D |
| 06.03.1989 (and 10.08.1989) | Shutdown of main coolant pumps because of incorrect actuation of excess heat protection with power decrease (or reactor scram, resp.) | 8.3.9.3 | S |
| 14.04.1989 | Non-shutdown of the main feedwater pump after reactor scram | 8.3.4 | D |
| 23.04.1989 | Reactor scram after power drop and sudden increase of turbo generator load | 8.3.9.5 | D |
| 06.05.1989 | Shutdown of a diesel generator due to failure of a switch | 8.3.10 | S |
| 14.05.1989 | Reactor scram due to incomplete opening of a gate valve in the steam supply of the turbine-driven feedwater pump | 8.3.4 | D |
| 21.05.1989 | Power reduction owing to defects in the steam-generator water level control | 8.3.9.6 | D |
| 08.06.1989 | Reactor scram owing to contact separation in the emergency protection system (KS) | 8.3.1 | D |
| 07.08.1989 | Failure of a containment-spray pump during in-service inspection | 8.3.3 | D |
| 01.09.1989 | Interruption of the feedwater supply to a steam generator | 8.3.9.1 | D |
| 20.09.1989 | turbine trip because of defects in the steam-generator water level control | 8.3.9.6 | D |
| 07.10.1989 | Reactor scram during inspection due to incorrect handling of a BRU-A valve | 8.3.9.4 | S |
| 06.01.1990 | Failure of two emergency power diesels and relief of primary system water via the BRU-A valves (KS) | 8.3.5 | D |
| 05.03.1990 | Start failure of an emergency power diesel caused by temperatures in the starter air system being too low | 8.3.7 | S |
| 21.08.1990 | Failure of a train of the safety system after short circuit | 8.3.8.2 | D |

### 8.3.1 Defects in the Control and Protection System of the Reactor

- **Delayed Absorber Rod Drop after Failure of Two Main Coolant Pumps on January 5, 1989, Reactor Power 100 % (Plant belonging to the "Small Series")**

- Event Sequence

The transient was initiated by the spontaneous closure of a pneumatic isolating valve in the oil system of the main coolant pumps. In accordance with the design the main coolant pumps 2 and 4 were then switched off and reactor scram was initiated by dropping rods of one absorber rod bank. One rod then dropped with a delay; it took 150s, instead of the three to four seconds normally required. At the same time, two absorber rods of a different bank dropped spontaneously and got stuck in an intermediate position.

In the course of the transient a malfunction of one of the two pressuriser water level indicators was detected. The measurement which has its lowest point in the nozzle of the volume control line showed a steep, pulsed decrease of the water level, while the second measurement apparently showed the correct decrease of the water level.

The plant could be stabilised at 32 % power.

- Causes

The closure of the pneumatic isolating valve was initiated by a short circuit in the control cable. The cable insulation had been damaged during installation in 1984. In 1988 the cable had been covered with a fire coating. This led to a heat-up of the cable and thus to the initiation of a short circuit.

The reason for the delayed rod insertion was a jamming drive shaft in the absorber element itself. There is no information on the reasons for the incorrect insertion of the two other elements.

The malfunction of the water-level measurement of the pressuriser was not explained.

- Upgrading Measures

Because of this event the operator made the following alterations:

- Change of the actuation logic for failure of a pneumatic oil isolating valve so that only one main coolant pump is switched off (R 8.3-1)
- Introduction of in-service insertion tests for the absorber rods which are required for reactor scram
- Ensurance that the pressuriser water level control is controlled by a second measurement position during reactor scram

In our opinion it is furthermore necessary to check the insulation of all control cables in safety-relevant systems (also see Section 8.3.8.3) (R 8.3-2). In addition, the effects of the backfitted fire protection measures on the operational safety of the cables are to be checked. Temperatures are to be checked in all energy supply cables, which were extensively treated with fire-resistant coatings (R 8.3-3).

The design of the absorber rod drives must be checked as to whether the drive shaft is principally a weak point (R 8.3-4). The actuation level for the absorber rod drives is to be checked with respect to its logic as well as to its switching circuit (R 8.3-5). Both water-level measurements of the pressurisers are to be upgraded so that they both indicate the same correct water level during all operating conditions, even during major transients (also see Section 8.3.9.5) (R 8.3-6).

- **Reactor Scram Due to Contact Separation in the Emergency Protection System on June 8, 1989, Reactor Power 72 % (Plant belonging to the "Small Series")**

- Event Sequence

During operation with three main coolant pumps, reactor scram was initiated. In the main control room no signals for the reasons for the reactor scram were indicated. After insertion of the absorber rods the operator of the reactor initiated reactor scram for the turbo-generator and the turbo-feedwater pumps. In the course of this operation the fast-acting valve of a feedwater pump remained open. This led to a pressure decrease in the steam generators of 2 MPa and a temperature decrease in the primary system until the fast-acting valve could be manually closed locally.

- Causes

During an examination of the contacts in the actuation circuits it was detected that reactor scram was initiated by a contact interruption owing to an assembly fault in the manual switch for reactor scram in the emergency control room.

The fast-acting valve remained open because a switch in the control circuit had failed.

- Upgrading Measures

A signalling system has to be installed which, in the case of reactor scram, shows the operator in any case the actuation criteria that have triggered off the scram. Signalling interruptions must be as far as possible self-reporting. Regular checks of the relay contacts and the links between contacts is not sufficient. It has to be determined to what extent these requirements can be met with the existing relays on the actuation level of the emergency-cooling system. Any faults must be as far as possible self-reporting. With a relay system self-reporting can only be achieved by taking costly additional measures. (R 8.3-7).

Furthermore, an automatic actuation of turbine trip after reactor scram is missing. It must be checked whether automatic actuation can also be introduced for trip of the turbo-feedwater pumps in order to prevent sub-cooling transients. This seems particularly necessary for the protection of the steam generators (R 8.3-8).

### 8.3.2 Defects in the Primary System

- **Reactor Scram due to Failure of the Oil Supply for all four Main Coolant Pumps after Actuating the Safety System on July 9, 1986, Reactor Power 98 %**

- Event Sequence

The power supply of the "1st class" consumer (uninterrupted supply) automatically switched over to the reserve busbar. The reasons for the switch-over are not indicated in the report of the event.

Operating staff tried, by switching operations, to switch back from the reserve busbar to the normal busbar. A complete loss of voltage of a safe sub-distribution was thus caused by maloperation. As all transducers of one actuation level of the safety system are supplied by one sub-distribution, the pneumatic valves of the localisation system (penetration isolation) were closed (1 out of 2 actuation). This, among other things, led to the disconnection of the oil pumps for the main coolant pumps so that all four main coolant pumps were switched off. Because of this, reactor scram was actuated automatically and turbine trip was initiated manually.

- Causes

Initiators of the event were switching operations of the power supply not in accordance with the regulations, which led to a voltage drop in the sensors of an actuation level of the safety system.The main reason was the fact that all sensors of one actuation level were supplied by one sub-distribution.

- Upgrading Measures

    - The power supply for the sensors of one actuation level must be divided (R 8.3-9).

    - Signalling has to be improved and locks have to be installed in order to be able to prevent, as far as possible, any inadvertent switchings (R 8.3-10).

    - In order to prevent transients occuring due to wrong signals, measured-value and limit-value processing are in principle to be designed completely redundant and, if possible, in a diverse manner up to the actuation level, to avoid erroneous actuations of the containment isolation signals of components with relevant availability (oil and feedwater supplies and pump trains of the operational make-up system) (R 8.3-11).

    - The available documentation does not show clearly how the power supply for the sensors of the safety system is designed in detail. No further specific demands can therefore be derived from this area. However, an assessment of the design appears to be necessary following past operating experience (R 8.3-12).

### 8.3.3 Failures and Defects in the Emergency Cooling System (Emergency Core Cooling and Residual Heat Removal System)

- **Failure of an Emergency Cooling Pump during In-Service Testing on January 19, 1988, Reactor Power 93 % (Plant belonging to the "Small Series")**

- Event Sequence

During power operation an in-service inspection of the 2nd train of the safety system was performed.

During the inspection while (among other things) a HP-emergency cooling pump was operating in minimum flow operation, the pump jammed. During turning of the pump by hand, a wedging of the rotor has detected. The pump was repaired within 53 h, which significantly exceeded the operating instructions for the maximum time of 16 h allowed for maintenance of a train. In this respect the conditions for safe operation of the unit were violated.

- Causes

The jamming of the pump was caused by entry of dirt particles into the gap of the hydraulic axial-thrust compensation of the rotor. The exact sequence of the event is unclear. The damage was possibly initiated by blockage of a filter at the suction nozzle of the pump.

It is suspected that the present case represents a design deficiency. It can be presumed from operating experience that the reliability of the pumps is insufficient. On the other hand the blockage of the suction-filter indicates that the emergency cooling system is strongly polluted. Possibly there was so much dirt that the design values for the pumps were clearly exceeded. A final clarification of the cause is not possible on the basis of the present report.

- Upgrading Measures
  - The pumps must be upgraded, for example by improving the surface coating of the axial-thrust compensation to reduce friction (R 8.3-13).
  - A reliable pump protection must be established with respect to temperature and suction pressure (R 8.3-14).
  - Determination of temperature and operating time limits for minimum flow operation; possibly the installation of additional heat exchangers for cooling during minimum flow operation (R 8.3-15).
  - Before the emergency cooling system is taken into operation, sufficient purging has to be carried out and, if necessary, elimination of the pollution sources (R 8.3-16).

- **Failure of a Containment-Spray Pump during In-Service Inspection on August 7, 1989, Reactor Power 63 %**

- Event Sequence

During in-service inspection of all three trains of the containment-spray system the containment-spray pumps were started and operated in the minimum flow operation mode. After the start of the pump of train 3, the pump bearings reached a temperature of 40-50 °C. However, one bearing after 30 s heated to almost 100 °C. The pump was therefore switched off by the equipment protection.

The two other pumps were tested and found to be in order; the pump which had been switched off was examined and repaired.

- Causes

As the oil level had been too low, lubrication of the pump bearing had not taken place and the bearing had thus been damaged. The minimum filling level permitted was not marked on the oil-level indicator of the oil tank.

- Upgrading Measures

The actual oil level and the difference to the minimum oil level of all safety-relevant pumps must, in principle, be easily and precisely determined. This has to be checked, and backfittings have to be carried out where necessary (R 8.3.17).

### 8.3.4 Defects in the Feedwater System

- **Reactor Scram owing to Incomplete Opening of a Gate Valve in the Steam Supply of the Turbine-Driven Feedwater Pump on May 14, 1989, Reactor Power 58 %**

- Event Sequence

In the commissioning phase during testing of load rejection, main coolant pump 1 was switched off, which automatically reduced the power of the reactor and the turbogenerator. The main steam pressure of the auxiliary steam supply was also lowered. During the switch-over processes in the steam supply for the turbo-feedwater pump there was a defect in one valve. The valve got stuck in a 10 % open position, because of a defect in its electrical actuating drive during opening. The signal "valve open" caused by the defect led to further automatic switching operations in the system. This caused the steam pressure upstream of the turbine of the turbo-feedwater pump to drop, so that its performance and delivery rate decreased. The level in the steam generators then fell until "steam generator level low" was reached and reactor scram was actuated. The defect is of safety-related importance as it represents a systematic problem of steam generator feeding.

- Causes

The event was triggered off by a failure of the mechanical load transmission in the drive of the control valve. The reasons are the inappropriate design of the valve drive and insufficient control during assembly, incoming and commissioning tests as well as during maintenance.

- Upgrading Measures

  - The mechanical drive of the valve must be upgraded (R 8.3-18).

  - The logic of the locks in the feedwater control system has to be improved in order to ensure reliable and effective operation of the pumps (R 8.3-19).

- **Feedwater Pump Not Switched Off after Reactor Scram on April 14, 1989, Reactor Power 42 %**

- Event Sequence

The turbine-driven feedwater pump (TDFP)-1 operated in the minimum flow operation mode and TDFP-2 in normal operation in the "automatic" mode. During repair work, two sub-distributions in the electric auxiliary-power supply of the 0.4-kV system failed owing to a short-circuit.This caused a failure of the oil supply (pumps and standby pumps) of two main coolant pumps and as a consequence led to reactor scram. Turbine trip was actuated manually. The control room staff only switched off the turbine-driven feedwater pumps and the low-pressure auxiliary steam consumers five minutes later. By that time main steam pressure had already dropped to 5 MPa and the difference in the saturation temperature between the primary and the secondary system had increased beyond 75 °C. The actuation limit for the start of the safety system was thus reached.

Furthermore, in the course of the transient the pressure valve of the HP-emergency cooling system had to be opened manually owing to the local failure of the auxiliary-power supply.

- Causes

Owing to the short circuit on one auxiliary power supply busbar an arc flashed over to a second busbar which also failed due to short circuit. This led to a failure of the main oil pump of two main coolant pumps. The standby oil pump, because of a defect in the actuation logic, did not start. Because of an operator error, the feedwater pumps still injected water into the steam generators more than five minutes after reactor scram. It cannot clearly be derived from the present information whether there had been a danger of recriticality because of the extremely high cooldown rate until shutdown of the turbine-driven feedwater pump. In any case, the fast cooldown represents a danger for the steam generators. Because of the large cooldown gradient there was also a danger for the integrity of the primary system.

This event also shows that over-cooling transients can occur due to defects especially in the area of the turbo-feedwater pumps.

- Upgrading Measures

  - The automatic standby-activation (automatische Reserve-Einschaltung, ARE) for the oil pumps of the main recirculation pumps is to be improved (ARE apparently only responds after the simultaneous failure of all three oil pumps) (R 8.3-20).

  - Turbine trip with an automatic disconnection of the turbo-feedwater pumps after reactor scram is to be automated (R 8.3-8).

  - The emergency-cooling system has to be upgraded in such a way that injection can take place without active opening of the isolation valves (R 8.3-21).

  - The physical separation of the auxiliary power supply busbars needs to be backfitted (R 8.3-22).

  - The switch gear must be short circuit proof (sufficient selectivity) (E 8.3-23).

### 8.3.5 Leakages in the Primary System

Among the incidents analysed there were two loss-of-coolant accidents and three events with steam generator tube damage. In this section, one of the cases with damage of a steam generator tube and in Section 8.3.6, one of the two loss-of-coolant accidents (owing to the erroneous opening of a pressuriser safety valve) will be evaluated in an exemplary way. The second loss-of-coolant accident will briefly be described here, as it represents a sequence of events where the plant, despite loss of coolant, was operated for about a further 17 minutes.

The packing of the stuffing box of one pressuriser injection valve had been pressed out of its support. As the stuffing box was not connected to a suction line, the positioning pins of the bottom bush had been corroded by the accumulating boric acid. There was a drop of the primary system pressure to 15.6 MPa. At this level it could be stabilised for 17 minutes with an injection rate of 20 t/h. Ventilation isolation was actuated by the signal "containment pressure > 0.129 MPa". During shutdown of the plant, reactor scram was initiated at a power of 740 MWel, when the primary pressure had fallen below 14.8 MPa.

The selected case with the steam generator tube damage is described below.

- **Failure of Two Emergency Power Diesels and Relief of Primary System Water via the BRU-A Valves on January 6, 1990, Reactor Power 100 % (Plant belonging to the "Small Series")**

- Event Sequence

At the time of the event there were tube leaks in all four steam generators of the plant. The leak rate ranged between 0.2 l/h and 1.1 l/h and was thus below the permissible limit of 5 l/h per steam generator.

Due to a fire in one 0.4-kV-emergency supply busbar, one train of the safety system failed. As a result, the reactor was setback to the minimum controllable power and the generator was separated from the grid. In accordance with the operating instructions the emergency power diesels of the two remaining trains were then tested.

As the repair of the 0.4-kV-busbar took too long, the reactor was shut down, the diesels were switched off and the reactor was borated. During rundown the pressure relief device of one diesel casing was actuated and oil sprayed from the main drive. The second train of the safety system had thus failed. About one hour later, action was started to raise the low steam generator level to the 3500 mm level, with the help of the emergency feedwater pump.

To make repairs to the drain pipes of the main steam system possible, the interlocks of all four loops were rendered ineffective, so that, after closure of all four main steam fast-acting valves, the design disconnection of the main coolant pumps was suppressed. The plant was thus in an inadmissible state of operation.

About 2 1/2 h after the start of the emergency feedwater pumps, an activity increase in steam generator 4 from 25 x $10^4$ Bq/$m^3$ to 150 x $10^4$ Bq/$m^3$ was detected. The operational staff concluded that the steam generator 4 level measurement had failed, that the steam generator had been flooded and water had penetrated into the main steam lines. As a result, the filling of the steam generators was terminated six minutes later.

About 10 minutes later the pressure in steam generator 4 had risen from 6.9 MPa to 7.6 MPa. To reduce pressure, the operator opened the BRU-A valves. As a result, about 20 $m^3$ of water with an overall activity of about 3 x $10^9$ Bq, estimated by the operator, was released onto the roof of the turbine hall. As the roof leaked, the contaminated water also penetrated into the turbine hall.

About 3 1/2 h later the plant was shutdown to the cold state.

- Causes

The cause of the fire was poor insulation of a cable which led to a short circuit. As a fuse which turned out to be too large had been installed, the short circuit was not switched off selectively. This led to fire in a relay which later expanded to the entire busbar.

The diesel failed owing to a defect in a bearing in the drive of the auxiliary oil pump, which was intended to ensure sufficient lubrication during the fast startup sequence. Similar problems had already been experienced in two other diesels. As a result, backfitting measures were taken for the previously faultless diesel (312 starts).

The closure of the main steam fast-acting valves without prior cold shutdown of the reactor and defeating the interlocks, to prevent shutdown of the coolant pump, can be considered the reason for the release. These two measures represent a flagrant violation of the valid operating instructions. A further aggravating factor was that level measurement in steam generator 4 had failed.

The event is an example of the fact that the plants are operated with numerous, partially hidden, partially known mistakes without taking special preventive safety measures. By contrast, existing safeguards are even rendered ineffective.

- Upgrading Measures
  - The frequency of the short-circuits observed shows that a general examination, especially of the cables, is necessary (R 8.3-2).
  - Lubrication of the bearings in the oil pumps (bearing temperature monitoring) of the emergency diesels has to be improved (R 8.3-24).
  - The steam-generator water-level measurement has to be improved by installing more reliable technology (R 8.3-25).
  - The failures of important safety-relevant measurements have to be self-reporting (R 8.3-26)
  - The unlocking of actuation criteria when the plant has not been shut down must be prevented through technical measures (R 8.3-27).
  - The roof of the turbine hall and the roof of the surrounding outer building must be sealed (R 8.3-28).

### 8.3.6 Defects in the Pressure Protection of Pressuriser and Steam Generator

- **Loss of Coolant because of Erroneous Opening of a Pressuriser Safety Valve on May 11, 1984, Reactor Power 30 % (Plant belonging to the "Small Series")**

- Event Sequence

During trial operation of the reactor, the pre-controlled safety valve of the pressuriser opened due to an incorrect signal and the pressure in the primary system at a temperature of 298 °C fell below 15 MPa. Reactor scram was thus actuated.

The primary system pressure rapidly dropped further and the pressuriser water level fell. At a primary system pressure of 5.8 MPa the core flooding tanks (accumulator) of the emergency cooling system (emergency core cooling system) started to inject into the primary system.

The pressure in the containment increased to 0.128 MPa, as further primary coolant was blown down through the open safety valve of the pressuriser and the rupture disc of the relief tank performed in accordance with the design. In the control room the non-closure of the pressuriser valve was observed. The operational staff, without success, tried to close the valve by turning the keyswitch. The main coolant pumps were manually taken out of operation and emergency boron injection was activated. The pressuriser heating was switched on, the generator was taken off the grid and the turbine was switched off. The cooldown of the secondary system was accelerated and the BRU-K to the condenser opened.

The core flooding tanks (accumulator) of the emergency cooling system supplied their entire useable water supplies into the core. The steam-gas mixture (top bubble) which had accumulated below the top of the RPV was evacuated. This again led to uniform temperatures in the reactor pressure vessel (RPV) and the further accumulation of gases was prevented. The precontrol valve only closed at a pressure of 0.9 MPa after staff had managed to de-energise the opening magnet.

The pressure in the containment was relieved via filters of the ventilation systems in the exhaust stack. After the event examinations were performed in the course of which no visible damage of the pipes or other reactor equipment was detected.

- Causes

The reasons for the failure of the precontrol valve were the defective wiring of the valve, as well as a break of a cable in the earth connection ("earthing") at the actuating switch in the emergency control room.

- Upgrading Measures

    - The wiring of the safety-relevant valves and pumps must be carried out correctly and has to be checked (R 8.3-29).

    - Quality assurance (inspections on receipt, etc.) must be extended to such a degree that faults in the functioning of valves are detected before they are installed (R 8.3-30).

    - The system must be single-failure-proof (R 8.3-31).

- The priority control system between the main control room and the emergency control room has to be redesigned and upgraded (R 8.3-32).

- **BRU-A Valves Remain Open on June 14, 1987, Reactor Power 100 % (Plant belonging to the "Small Series")**

- Event Sequence

The monitoring device of the plant indicated a defect in the electro-hydraulic turbine control system. Owing to of a loss of voltage in the automatic turbine control, the system automatically switched from the electro-hydraulic to the hydraulic control system. After the defect had been removed, the operator switched back to the electro-hydraulic system. The protection device which prevents positional differences (between the electro-hydraulic and the hydraulic turbine control) of more than 30 % thereupon automatically closed the turbine trip valve as well as the turbine control valves.

After turbine trip had taken place, reactor scram was actuated manually by the operator.

The pressure in the primary and secondary system increased on turbine trip and the BRU-A (atmospheric steam dump station) as well as the BRU-K (bypass station) of the secondary system opened. The pressuriser spray system was also activated. After that pressure in both systems dropped. When the pressure in the secondary system had fallen to the required level, BRU-A (atmospheric steam dump station) and BRU-K (bypass station) valves were closed in accordance with the design. The BRU-K closed completely, but two of the four BRU-A valves remained about 20 to 25 % open. Operational staff tried to close the fast-acting valves installed upstream of the BRU-A valves. Owing to defects in the limit switch position, one of these valves also did not close completely. As loud noises could still be heard from the BRU-A area, the operational staff decided to locally close the fast-acting valve concerned manually.

Over a period of 5 minutes pressure in the secondary system had fallen to 5.6 MPa and in the primary system to 13.3 MPa; temperature in the primary system had fallen to 263 °C. The conditions for safe operation were not exceeded.

The event is safety-relevant because by this means a sub-cooling transient with the possible consequences mentioned in Section 8.3.4 was initiated. In particular, the failure of the BRU-A valves must be assessed as a common mode failure.

- Causes

The incident was initiated by an electrical failure in the turbine control system. The consequences resulted from a common mode failure of the BRU-A valves. During relief, there were strong vibrations at the relief lines which exceeded the vibration levels against which the valves were designed. This caused defects at the limit-position switches so that these were not closed.

- Upgrading Measures
  - The BRU-A (steam-dump station) including the limit-position switches must be upgraded (R 8.3-33).
  - The vibrations during pressure relief via the BRU-A are to be reduced by constructive measures (R 8.3-34).
  - Investigation of whether the use of limit-position switches without contacts is technically useful in the case of the steam-dump station is required (R 8.3-35).

### 8.3.7 Defects at Emergency Power Diesels

Apart from the diesel failures already described in Section 8.3.5 and the start failures known from the analysis of the operating experience of Greifswald 5 because of humidity input into the starter air system there was a series of other incidents which concerned the diesels. An event is described below which occured due to a defect in the excitation circuit of the diesel generator. The case is particularly interesting, as a failure for the same reason is also reported from another nuclear power plant (suspicion of common mode failure).

- **Defect in the Excitation Circuit of a Diesel Generator on January 17, 1989 (2nd incident on August 20, 1987), Reactor Power 100 %**

- Event Sequence

The three emergency power diesel generators were being ruotinely inspected. The examination took place without load suspension. At the local control desk of diesel generator DG-3 operational staff noticed that the voltage in the generator increased considerably slower than intended (23 s instead of the intended 15 s) to the preset value. After 15 s the actual value of the voltage was 5.5 to 5.8 kV. As a result this generator was taken out of the "standby" reserve in order to examine the defect. The two other diesel generators were examined without finding any defect.

- Causes

The reason for the defect was a fabrication mistake in the standby controller of the exciter which had not been detected before. There is a "normal" and a "standby" controller at every diesel generator; the so-called "standby" controller apparently also influences the excitation voltage under normal conditions. Details are not known. The preset value of the standby controller drifted and interaction with the normal controller resulted in an elongation of the time in which the output voltage reached the required value.

- Upgrading Measures

  - Due to the above mentioned suspicion of a common mode defect, the cause of the drifting of the nominal value of the backup control should in any case be found and eliminated (R 8.3-36).

- **Further Events at the Emergency Power Diesel System with the Suspicion of "Common Mode"**

For other events with defects of emergency power diesels there are also indications for common mode faults. In one case it was found that the state of the battery voltage did not suffice for starting the diesel and that there was no secure indication of the state of the battery, owing to a deficiency in the design. Monitoring of the recharging voltage of the batteries and the constant upkeep of their charging current must be improved through recurring tests (R 8.3-37).

In another case a start failure of an emergency power diesel is reported, caused by temperatures being too low in the starter air system. Diesel preheating had been switched off by official order. In addition, the response value for the alarm system for temperatures being too low had been reduced, to switch off the pending alarm signal. Preheating had already repeatedly been switched off before. In the present case this led to a non-availability of the diesels for 2 1/2 weeks. As an urgent upgrading measure a warning system for signalling low temperatures in the diesel's starter air is suggested in the report, which has to be protected against interference by the operating personnel (R 8.3-38). Here too there is a suspicion that this case represents a common mode failure in the broadest sense. A further aggravating fact is that the failure remained undetected for 2 1/2 weeks. In the report on the incident it is emphasized that the current operating instructions were violated.

### 8.3.8 Failures in the Power Supply of Important Safety-Relevant Consumers

#### 8.3.8.1 Lacking Redundancy

Several of the analysed events showed that the power supply of systems redundant in relation to each other were supplied by the same switchgear busbar. This applies to the energy supply of motors (e.g. oil pump and standby oil pump of two main coolant pumps at a 0.4-kV-busbar) as well as the supply for control. A case from this sphere is described below.

- **Reactor Scram by False Tripping of the Condensate Pump Protection and of the Generator on February 22, 1989, Reactor Power 100 %**

- Event Sequence

During cleaning and repair work in the relay room, a cleaner erroneously switched off a BMSR cabinet with a ladder (by touching the input switch). This led to signals switching off the condensate pumps and initiating generator protection. The BRU-K (main steam bypass station) was blocked because of the lack of injection water, so the BRU-A valves (atmospheric steam dump station) opened at a secondary system pressure of 7.3 MPa. One BRU-A valve remained open and only closed again at a pressure of 3.8 MPa. This led to reactor scram via the criteria "Difference in the saturation temperatures between primary and secondary system ; max. and main steam pressure low".

- Causes

The power supply of five diesel generator protection circuits for all three channels had been supplied by one switchgear cabinet. By the voltage failure the respective generator, condenser and condensate pump protection circuits were actuated.

- Upgrading Measures

  - The power supply of the three channels of the diesel generator protection system must be divided between different, physically separated busbars (R 8.3-39).

  - The power supply breakers located in cabinets must be protected against inadvertent operation (R.8.3-40).

#### 8.3.8.2 Failure of Power Switches

- **Failure of One Train of the Safety System after Short Circuit on August 21, 1990, Reactor Power 100 %**

- Event Sequence

One train of the safety system had failed due to a short circuit in a 0.4-kV emergency supply busbar. As a result the two intact trains were tested. After the test the power switch between the 6-kV-auxiliary-power busbar and the emergency supply busbar could not be closed again in a further train. The second train of the safety system thus could not be supplied with the normal auxiliary power supply. The plant was further operated without sufficient redundancy for 30 minutes. This represents a violation of the current operating instructions.

- Causes

The short circuit had been caused by loose contacts. In the respective report the failure of the 6-kV-switch is assigned to a design fault in the movable contact unit, which so far has led to numerous failures in switches of the Russian series VEh-6 and VEhS-6 in different plants.

- Upgrading Measures

    - The 6-kV switches must be replaced by appropriate switches (R 8.3-41).

#### 8.3.8.3 Cable Defects

A large number of the analysed events was caused by damages of cable insulations of the different capacities. Frequently cable connections to motors, junction boxes and cable sealing ends were concerned. This again and again led to short circuits in the most diverse areas of the plant. The reason for the defects normally were bad quality assurance during manufacture, assembly mistakes during the installation of the cables, damaging of the cables during operation as well as the use of cables unsuitable for the operating conditions and which had not been tested on electrical load. As there were cable defects in a number of events described in the foregoing sections, an individual event will not be discussed in this section (see, for example, Sections 8.3.1, 8.3.5).

#### 8.3.8.4 Layout of the Building, Condition of the Rooms

- **Failure of Power Supply for One Train of the Safety System on January 8, 1987, Reactor Power 100 %**

- Event Sequence

The outer temperature at the nuclear power plant location had fallen drastically for two days and was -20 °C. During normal operation of the reactor a fault to earth of the 6-kV-busbar supplying the first train of the safety system was indicated. The injection switch to the emergency supply busbar opened. The emergency power diesel started and supplied the busbar concerned again and the connection of load sequence was initiated. After one minute the emergency power diesel generator was disconnected from the voltage regulators, as the safety relays of the emergency power supply and the diesel busbar had actuated because of a short circuit. The first train of the safety system thus failed completely. The two remaining trains were examined and power operation was continued.

- Causes

The switchgear and controlgear cabinet of the sub-distribution concerned is located in a room which is bounded by the outer wall of the reactor building and a section in which the main steam lines are located. From this side, humidity could penetrate into the switchgear and controlgear room via a ventilation system. As the temperature within the room had dropped due to an undetected failure of room temperature control, condensed water accumulated on the outside wall. Here the temperature was +10 °C. Also undetected was water droping from the ceiling into the switchgear cabinet next to the intake opening of the ventilation system. When enough water had accumulated on the busbars and isolators, it formed a short circuit.

- Upgrading Measures

  - The penetration of humidity and water into the switch-gear rooms must be prevented through constructional measures and an appropriate design of the ventilation system (R 8.3-42).

Failures of the room-temperature control in rooms with safety-related systems must be self-reporting (R 8.3-43).

### 8.3.9 Failures of Instrumentation and Control

#### 8.3.9.1 False Tripping because of Defects in the Power Supply

- **Automatic Power Decrease by Failure of Power Measurement of a Main Coolant Pump on January 13, 1987, Reactor Power 100 %**

- Event Sequence

By an incorrect signal "Failure of Main Coolant Pump 3" power was automatically reduced by the reactor power limitation to a level of 650 MW. In the control room, the operators observed the signal "Failure of Main Coolant Pump 3", although the remaining parameters, like loss of pressure in the line and above the pump, etc. remained unchanged. It turned out that the signals "Main Coolant Pump 3 Switched Off" and "Drop of Rotational Frequency of Main Coolant Pump 3 < 48.5 Hz" were pending in all three channels of the alarm system (reactor power limitation system).

- Causes

The reason for the incorrect signal was a short circuit in one 6-kV power cable showing quality defects. As a result the 6-kV/0.4-kV transformer supplying all three channels of the sensor for power and frequency measurement both of the reactor protection system and of the alarm system (twelve sensors at one busbar) became de-energized.

- Upgrading Measures

  - The design of the voltage supply of the power and frequency measurement system of the main-coolant-pump monitoring system is to be changed by distribution over several supply busbars (R 8.3-44).

The reliability of the supply busbars including the cables, connections and contacts is to be improved (R 8.3-45).

An examination of the entire measurement and control system including the emergency cooling system should be carried out with regard to design flaws in the power supply (R 8.3-46).

- **Interruption of the Feedwater Supply to One Steam Generator on September 1, 1989, Reactor Power 100 %**

- Event Sequence

By erroneous actuation of an interlock, closing commands for the feedwater control valves and the gate valves of steam generator 4 were given. By switching off the automatic system, the operators were able to prevent a complete closure of the slowly closing gate valves. The main coolant pump 4 was switched off by the falling water-level in the steam generator and the reactor power was decreased automatically. To re-establish feedwater supply to steam generator 4, the operators incorrectly opened the main control valve instead of the startup control valve of steam generator 4. As the main coolant pump 4 was switched off, the water-level increased rapidly. In addition, the water-level in steam generator 3 now began to fall, because this steam generator is the last steam generator to be supplied by the feedwater header, while steam generator 4 is the first one. The water level in steam generator 3 fell to such an extent that the main coolant pump 3 was switched off and the reactor power fell to 50 %. The high water level in steam generator 4 led to turbine trip with subsequent reactor scram.

- Causes

A short-term voltage dip on a sub-distribution supplying all three logic units of the interlock led to an erroneous actuation of the interlock. In the report on the incident there is reference to the fact that all three logic units were supplied by the same sub-distribution to improve the signal-noise ratio towards the projected circuit (supply via different sub-distributions).

- Upgrading Measures

The automatic locking of the feeding of steam generator 4 had already responded erroneously several times on this plant. The voltage supply of all three logic units via one sub-distribution only and the inadequate signal-noise ratio represent weaknesses of the system, which can impair the heat removal via the steam generators. It therefore must be requested that the energy supply of the actuation level is changed so that the automatic locking mechanism (2 out of 3) is not activated by malfunctions in one supply busbar (8.3-47).

Earthing and configuration of the energy supply must be designed in such a way that there is a sufficient interference-voltage distance (R 8.3-48).

To avoid the above transients, the operator suggests training of the staff. The staff of the unit will mitigate transients in the feedwater area by fast manual measures. In our opinion, however, such transients must be absorbed by effective control and limits (R 8.3-49).

It is also to be investigated to what extent it is ensured, by sufficient dimensioning of the feedwater lines and by a control system that corresponds to the safety requirements, that no asymmetrical conditions can occur during steam generator feeding (R 8.3-50).

- **A Further Example of Instrumentation and Control Failures: Reactor Scram after Disruption of the Oil Supply to Two out of Four Main Coolant Lines on December 30, 1986, Reactor Power 100 % (Plant belonging to the "Small Series")**

By electrical pick-up from the 220-V supply in the 24-V and 48-V circuits of the control logic of the oil pumps for the main coolant pumps, the oil supply of two main coolant pumps was interrupted. As a result, the two main coolant pumps switched off and the reactor protection activated reactor scram. An additional failure of the BRU-K, i.e. one BRU-K valve remaining open, then occured. Therefore the fast-acting valves of the BRU-K had to be actuated.

This case too led to a sub-cooling transient.

Measures have to be taken to prevent pick-up from the 220-V supply in the 24-V and 48-V logic switching circuits (R 8.3-51).

#### 8.3.9.2 Failures of the Measured-Value Acquisition

In the documents available two common mode failures in the area of pulse lines of pressure measurement are reported. In one event the main steam pressure measuring device was frozen. The resulting spurious signals led to the incorrect opening of the relief control valves (prototype plant). The second case is described below.

- **Reactor Scram after a Leak in a Pulse Line of the Pressuriser Level Measurement on March 22, 1988, Reactor Power 100 %**

- Event Sequence

Owing to leakage at a connecting point of the common pulse line to one of the three pressuriser level indicators, the signal "Pressuriser level very low" was generated on three channels of a protection system train and reactor scram was thus actuated.

- Causes

The leakage was caused by a wrongly assembled seal of a union nut. As all three level indicators for a 2 out of 3 circuit were connected with the pressuriser via a pulse line and thus also with each other, there could be a simultaneous malactuation of all three channels.

- Upgrading Measures

All instrument channels have to be functionally separated, from the intake of the medium to the actuation signal, in order to exclude erroneous actuation by single fault (R 8.3-52). It can be seen from the report of the event that the respective investigations have already been performed by the operator of the power plant concerned.

#### 8.3.9.3 Failures of Signal Transmission

- **Reactor Scram because of Incorrect Signal "High-Pressure Preheater Level High" on February 9, 1987, Reactor Power 70 %**

- Event Sequence

During operation the signal "High-pressure preheater level high" appeared. Thereupon turbine trip was actuated by the turbine protection system and two pumps were switched off. In the incident report cooling water pumps are mentioned, but presumably feedwater pumps are meant. Reactor power was decreased by power limitation and the fast-acting valves of the BRU-A and BRU-K responded. The water levels in the steam generators fell and the temperature in the primary system increased so that pressuriser spraying was initiated. Reactor scram was actuated by the signal "Steam generator level very low".

- Causes

Investigations showed that the level-signal was initiated erroneously by a failure in the turbine protection caused by corrosion-related fluctuations of the contact resistance at a relay contact. All similar relays were then examined. The incident report does not include the results of this examination.

- Upgrading Measures

  - The alarm system (reactor power limitation) should be upgraded in such a way that such transients can be regulated without the safety systems being actuated. If this is not possible with the existing technology, reactor scram must automatically be triggered after turbine trip (R 8.3-53).

  - The contact surfaces of the relays must be made of material with sufficiently assured quality (R 8.3-54).

- **Further Events with Signal Transmission Failures**

Like in the event described above, signal transmission failures in two nuclear power plants led to decrease of the unit power and reactor scram, respectively. In both plants the equipment protection "Overheating of a bearing segment in the pressure bearing of the main coolant pump was activated at full load. Measurement of the bearing temperature is designed as a 2 out of 3 circuit. In both plants the main coolant pump concerned was switched off because of the coincidence of a cable break already detected earlier and an erroneous response of a second channel. In one unit of the reactor the reactor power was reduced to 67 %, in the unit of the other nuclear power plant reactor scram occured because of a further failure of the protection system. The events represent common mode failures. The reasons are constructional and design flaws in the cable connections of the sensors. Cable connections were destroyed by vibrations caused by the flow of oil in the pump bearings.

The defect in the emergency-cooling system which in one case led to reactor scram was also caused by signal interruption and signal interference of the 2 out of 3 measured-value acquisition for the pressure difference across a main coolant pump.

The frequency of events caused by signal transmission failures indicates that the entire instrumentation and control is not very reliable. On the basis of the evaluated operating experience and of examinations that have been carried out, a full analysis of the deficiencies of the whole instrumentation and control system must be carried out on the plant. It then has to be decided whether the existing technology can be upgraded or if the instrumentation and control equipment should be replaced to a large extent (R 8.3-55).

#### 8.3.9.4 Faults on the Logic Level

Events where adjustments and controls could be influenced simultaneously from two positions, because of a lack of priority control were observed. In one case during a test, an OPEN command for a BRU-A valve was given by the engineer on duty, without coordination with the operator, who at the same time gave a CLOSED command from a different position, not indicated in the report. By these commands the valve was opened too wide.

In the present reports further events are described which are attributed to an insufficient interlocking logic.

- **Reactor Scram after Failures in the Reactor Power Control on February 19, 1989, Reactor Power 70 %**

- Event Sequence

After a fault to earth in a control and monitoring unit of the 6-kV-auxiliary power supply busbar BA the shift leader "Electric", without prior coordination with the control room, started repair measures. During repair he effected further failures which led to a grid disconnection of the unit. The busbars BB, BC, BD switched over to the standby transformer; busbar BA could not be switched over because of the breakdown of the control unit and became de-energised. As a result the connected pumps, main coolant pump 1, cooling water pump 1 (probably the main cooling water pump is meant) and the service water pump 1 failed among other things. The resulting transient led to the actuation of the system for a fast power reduction by dropping an absorber rod bank. As the auxiliary power supply busbar BA was de-energized, the corresponding emergency power diesel started. The components of the safety system connected with train 1 were activated in accordance with the design.

As the operator had not noticed the actuation of the system for fast power reduction, he tried to reduce reactor power to 37 % via the touch switch "Preventive protection 1" with the alarm system (reactor power limitation system). After one minute he realised that reactor power had already fallen to 7 % and primary system pressure to 14.7 MPa. Thereupon he released the touch switch, withdrew the inserted rods and switched off the pumps of train 1 of the safety system. The primary system could be stabilised at a power of 10 % and 16.0 MPa.

As further steam was withdrawn from the main steam header to operate the turbo-feedwater pumps, the primary system cooled down further. As a result, the shift leader had the unit connected with the auxiliary steam system of the overall plant for additional steam but, as the pressure of the auxiliary steam system was 0.4 MPa lower than in the unit main steam system more steam was drwan off, so that the unit was cooled down even faster. This led to reactor scram via "Difference in the saturation temperature between the primary and the secondary system > 75 K and pressure in the main steam collector < 4.9 Mpa".

- Causes

The operator reduced the reactor power manually with the "preventive protection 1" having a higher priority than the automatic controller. He operated the touch switch too long and therefore reduced the reactor power too much. The evaporation of the main steam system into the auxiliary steam system was possible, because an isolating device preventing maloperations in this area is missing.

- Upgrading Measures

The interaction of the individual power controllers and the options for manual intervention in power control by the operators must be checked (R 8.3-56).

An isolating device has to prevent an uncontrolled evaporation from the secondary system into the auxiliary steam network. Process-based measures are also required, like, e.g., decoupling via check valves or control valves which can prevent maloperations during equalisation of pressure (R 8.3-57).

In the following event description deficienies on the logic level led to unnecessary actuations of reactor scram and of the safety system.

- **Reactor Scram by Non-Closure of the Pressuriser Injection Valve on January 30, 1988, Reactor Power 100 %**

- Event Sequence

Without consultation with the shift leader responsible for operation, a boron concentration compensation was carried out in the primary system. The electrical protection of the pressuriser injection valve drive was activated and thus prevented the closure of the open valve. The interlock for closing the subsequent isolating valve was not actuated so that there was a pressure decrease in the primary system. The valve also could not be closed by the individual control. At a primary system pressure of 14.8 MPa this led to reactor scram. The HP-emergency cooling pumps were started via the signal "Difference between primary system teperature and saturation temperature < 10 K" and operated in the minimum flow mode. In due course, the pressuriser injection valve could be closed manually from the control room. Turbine trip was initiated and the generator was separated from the grid after two minutes.

- Causes

Apart from administrative deficiencies, which led to an arbitrary compensation of the boron concentration without prior coordination with the shift leader, there were the following deficiencies:

The pressuriser injection valve failed because of a manufacturing defect in the thyristor amplifier of the corresponding voltage supply. In addition, the position indication of the valve was defective. The interlock only permitted a closure of the subsequent isolating valve via the positioning signal of the injection valve. For this reason the isolating valve was not closed, despite the pressure decrease. It is unclear why the individual control also failed.

The signal "Difference in temperature < 10 K" occured erroneously, the actual temperature difference was 25 K. Because of the design of the measurement there can be measurement errors which reach the permissible bandwith of the temperature difference.

- Upgrading Measures

The actuation logic of the protective interlock for the isolating valve of the pressuriser injection line has to be enlarged in such a way that the injection valve can be operated independent of the reset position (R 8.3-58).

The actuation logic for the formation of the signal "Difference between primary system temperature and saturation temperature below 10 K" must be improved so that the measurement error is clearly lower than the admissible deviation range of the measured value (R 8.3-59).

In addition, the entire reactor protection logic must be checked as to whether other actuation criteria use reset positions too exclusively and whether measurement errors lie within the range of the distances from the normal parameters to the activation limit value (R 8.3-60).

#### 8.3.9.5 Turbine Control

In several of the events analysed there were defects in turbine control. The transients caused by these defects partially indicate severe deficiencies in the operational and safety systems. In Section 8.3.6 one incident is already described where an operational mistake during the change from the hydraulic to the electro-hydraulic controller had resulted in a wrong position of the turbine control valves. In the course of the transient two relief valves remained open.

An event which resulted in a strong power oscillation is described below.

- **Reactor Scram after Power Decrease and Sudden Increase of the Load on the Turbo-Generator on April 23, 1989, Reactor Power 100 %**

- Event Sequence

At the end of cycle, the plant was operating with a boron concentration of 60 ppm and thus with a strongly negative coolant temperature coefficient. A fast closure of the turbine control valves, owing to an incorrect signal, actuated a steep power decrease at the turbo-generator from 1000 MW to 3 MW. The corresponding increase in main steam pressure led to an increase of the mean coolant temperature and thus to an increase of the pressuriser water level. The automatic power controller (ARM) switched to the main steam pressure control mode (T-Regime) and started a reduction of the reactor power. As the power was not reduced fast enough, the pressure limits for the actuation of pressuriser injection were reached.

Owing to the increasing main steam pressure, the BRU-K and BRU-A opened at a time when the electrical power had fallen to 100 MW. The alarm system (reactor power limitation) decreased the reactor power to 85 %.

The closure of the turbine control valves resulted in a pressure decrease in the turbine bled steam and thus in the boiling of the heating steam condensate. The protection system for filling-level monitoring switched off the high-pressure preheater.

Eight seconds after the power decrease there was an uncontrolled power increase at the turbine caused by the sudden opening of the turbine control valves, leading to a turbine power of 1046 MW after a further seven seconds. This resulted in a strong decrease of the main steam pressure (from about 7.4 MPa to about 5.6 MPa) of the mean coolant temperature and the pressuriser water level (figures are not available).

The pressuriser heating was switched off about 40 s after the start of the event by the signal "Water level very low". The signals for reactor scram and switching off the heating are actuated by two different measurement arrangements; the measurement arrangement for the switch off of the heating initiating an incorrect signal. As a result and owing to the slow behaviour of the ARM in the T-regime the pressuriser water level fell to the limit and reactor scram was actuated after about 60 s. 13 s after reactor scram turbine trip was actuated via "main steam pressure < 5.2 MPa".

Because of the false signal "Pressuriser water level very low" the shift engineer switched off the controls for the make-up system (coolant letdown rate, pressure upstream of the letdown valves, pressuriser water level) to isolate the coolant letdown and to increase the make-up rate to a maximum. This led to an injection of cold water (60 - 70 °C) into the primary system and according to the accident report, to a temperature decrease of a total of 18 °C within 20 s in the cold legs.

As the operator wrongly attributed the cooldown of the primary system to a cooldown by the secondary system only, he closed the main steam fast-acting valves. By this measure the main coolant pumps were also switched off. After that the plant could be stabilised in natural circulation at a primary system pressure of 13 MPa and a temperature of 270 °C.

- Causes

The sudden closure of the turbine control valves at the beginning of the event probably had been actuated by a short-term blockage of a control valve in the electro-hydraulic converter of the turbine control. After an inspection of the valve, a pollutant of a diameter of about 1 mm was found. The release of this particle apparently only eight seconds later led to a renewed opening of the control valves and the resulting power increase.

The pressuriser heating was switched off due to false signals of a pressuriser water level measurement, which also led to problems in the event described in Section 8.3.1. The lower pulse line connection is located near the inlet of the volume control (surge) line in the pressuriser. Upon a rapidly falling filling level this apparently leads to false measurements.

The main reason for sub-cooling of the primary system according to the accident report was the injection of cold make-up water. In addition thereto the intake of steam by the further operation of the turbine after reactor scram and the failure of the high-pressure preheater contributed to the cooldown of the primary system. In the incident report, it is pointed out that because of the highly negative coolant temperature coefficient there was a danger of recriticality.

- Upgrading Measures

Adequate filters must be provided in the oil circuits of the turbine-control system in order to avoid pollution (R 8.3-61).

The turbine control system has to be upgraded to such an extent that extreme loads, as in this case, are excluded. This can, for example, be achieved by installing two electro-hydraulic converters with fast synchronisation control and consecutive MIN-selection (R 8.3-62).

The arrangement of the water-level measurement lines of the pressurisers has to be changed in such a way that correct measurements can be ensured even during fast changes of the water level (R 8.3-6).

After reactor scram, turbine trip must be actuated automatically to prevent sub-cooling transients (R 8.3-8).

#### 8.3.9.6 Steam Generator Water Level Control

- **Turbine Trip because of Failures in the Steam Generator Water Level Control on September 20, 1989, Reactor Power 92 %**

- Event Sequence

A leak in a seal of the oil circuit led to a spontaneous closure of a pneumatic isolation valve in the return line of the oil system of the main coolant pumps 1 and 3. As the OPEN as well as the CLOSED limit switch had failed due to pollution, the shift staff did not notice the closure of the valve. The isolation of the oil return from the drainage tank led to a decrease of the filling level of the oil storage tank and furthermore to the failure of the main and standby oil pumps, which both draw from the same oil tank. In accordance with the design, the two main coolant pumps were switched off.

As a result, fast shutdown to a power of 50 % was initiated. This resulted in an increase of the water level of steam generator 3, the closure of the feedwater control valve and the opening of the startup control valve for steam generator 3. Because of design flaws in the interaction between two interlocks, which are controlled by the steam generator water level, and differently adjusted limits for main and startup controllers, there was an increase of the water level in steam generator 3 up to the third limit and thus to turbine trip. The exact sequence of the transient cannot be derived from the document.

During the course of the transient it was detected that it is difficult to compare the indications of the measured values of the control channels and the safety channels under dynamic conditions. The turbine operator has to interpret the measured values indicated, the different absolute values for the steam generator water level and different changes of the measured values correctly.

- Causes

The two interlocks relate back to signals of different measurement transformers. These measurement transformers also supply the different signals for the turbine operator. As these apparently differ strongly, there could be an almost simultaneous actuation of the two interlocks which finally led to turbine trip.

- Upgrading Measures

A plant-state-signalisation system has to be introduced to help recognise more easily the failure of position indicators on valves (R 8.3-63).

The filling level in the oil tanks of the main coolant pumps' oil circuits must be monitored and equipped with warning devices (R 8.3-64).

Steam generator water level measurement and limit-value adjustment must be improved by technical measures. In particular it must be ensured that gauges measuring in the same measurement units are synchronised (R 8.3-65).

- **Power Reduction owing to Deficiencies in the Steam Generator Water Level Control on May 21, 1989, Reactor Power 75 %**

- Event Sequence

To test the pulse valves of the turbo-feedwater pump 1, the main feedwater control valves, according to a work program which deviated from the standard program, were taken out of automatic control. As a result, the automatic system controlled the water level in the steam generators with the startup control valves. The water level hereupon increased by 150 cm beyond the control range of + 50 cm. Because of a zero shift of level measurement and an actuation limit which was adjusted 30 cm too low, the main steam fast-acting valve was closed on steam generator 3 and the main coolant pump 3 was switched off.

As a result the reactor power limiter reduced the reactor power to 67 %.

- Causes

There could be an increase of the water level because of the change-over to the very slow startup control.

- Upgrading Measures

The testing possibilities must principally be determined or automated in such a way that there will be no undesired transients (R 8.3-66).

Provisions are to be made such that switching of the steam generator water level control to start-up control is avoided during power operation (R 8.3-67).

Adjustments of zero point and limit values must be monitored either by inspections or through self-reporting (R 8.3-68).

- **Imbalance in the Thermal Power of the Primary and Secondary System during a fast Power Reduction on June 15, 1985, Reactor Power 100 % (Plant Belonging to the "Small Series")**

- Event Sequence

During work on the 750-kV switchyard, there was a false actuation of the emergency protection system (kind of protective actuation unkown). This resulted in fast power reduction from 1000 MW to 300 MW with an increase back to 600 MW. This fast transient led to an increase of the main steam pressure with opening of the BRU-K (bypass station).

A few seconds after the beginning of the transient reactor scram was actuated by a false signal "Water level low" in one of the four steam generators. Operating staff initiated turbine trip manually.

During the following switch-over of the electric auxiliary-power supply, there was a failure of a 6-kV power switch in the supply, of a train of the safety system. The appropriate diesel started but it did not switch in. The reasons for this could not be found by an examination of the diesel. This led to a complete failure of this train of the safety system.

- Causes

The reasons for the false measurement of the steam generator water level cannot be derived from the present document.

- Upgrading Measures

Steam generator water level measurement must operate reliably, including in the case of rapid changes in main steam pressure (R 8.3-69).

Referring to 6-kV power switches and diesel generators, upgrading measures are indicated in Sections 8.3.8.2 and 8.3.7.

### 8.3.10 Deficiencies in Quality Assurance

Deficiencies in quality assurance are highly important for the evaluation of operating plants. It can already be derived from the limited number of the present incident reports that such deficiencies frequently led to disturbances or they influenced the sequence of disturbances. Deficiencies were detected in the quality assurance during the manufacture of the components as well as during assembly and especially during maintenance in the power plant. Examples are the failure of turbine control during a transient owing to a missing locking screw in the limit switch of the electromagnetic switch-over device (manufacturing fault) and the failure of a switch which was not permitted for use in the nuclear power station and which had been installed because of an insufficient incoming inspection. A further example is an assembly fault, which led to maloperation, i.e. the cross connection of indication and handling devices for two main feedwater pumps in the control room.

The events show that it is highly important to require evidence of detailed quality assurance from the plant manufacturers of all components that are used in safety-relevant plant areas; in addition, separate comprehensive inspections must be carried out (R 8.3-70).

It has been found that, as there is no systematic quality assurance there does not seem to be a systematic acquisition of deficiencies in the area of quality assurance. The above deficiencies therefore could only be reconstructed from the reported events and thus only represent spot findings.

### 8.3.11 Other Events

- April 19, 1983, (plant belonging to the "small series") destruction of a main coolant pump, admission of 38 kg abraded matter, into the primary circuit, increase of coolant flow rate through defective pump, increase of thermal reactor power by 45 MW, increase of fuel element temperature by 10 to 15 % combined with an increase of the coolant temperature at the outlet of five fuel elements by 1 to 2 °C above the permissible temperature. The defective pump is switched off 9 h after the power increase.
- October 26, 1984, (plant belonging to the "small series") failure of main feedwater pumps, start of emergency feedwater pumps, decrease of the steam generator water level. As a result increase of the primary coolant temperature in two loops (cold leg) to 320 °C (= core outlet temperature) with failure of the two main coolant pumps, pressure increase in the secondary system, actuation of the BRU-K, pressure decrease in the primary system and reactor scram.
- November 5, 1987, humidity in the plug of an absorber rod drive, false insertion of the absorber rod, failure of steam generator water level controls and failure of the controller of the BRU-K (bypass station), water level of two steam generators high, water level of two steam generators low, reactor scram.
- December 11, 1987, fog precipitate on isolators of unit transformer, fault to earth, turbine trip, reactor scram, start of the emergency power diesels and of the safety system.
- February 23, 1988, failure of a main coolant pump after maloperation during testing of the steam generator water level control. There was no automatic test mechanism.
- April 6, 1988, manual reactor scram after defects in the HP-preheaters because of wrong line routings.
- August 14, 1988, short circuit during repair in a control unit of the steam generator water level control (constructional fault 220-V contacts) led to the

failure of two main coolant pumps, as two channels of the protection circuit of the main coolant pumps are fed by one supply.

- January 4, 1989, malactuation of the fire extinguishing system as wrongly dimensioned flue gas detectors responded to humidity, admission of water into the cable shaft of two channels, false signal (poor insulation of cable, high-voltage test had not been carried out prior to installation) for failure of main coolant pumps results in reactor scram.

- February 6, 1989, (plant belonging to the "small series") voltage fluctuations in the third train of the interruption-free emergency power network; as a result actuation of several automatic systems; after switch-over to standby injection, failure of reactor protection boards, reactor scram, no signalisation of voltage failure on reactor protection boards.

- April 9, 1989, failure of a 0.4-kV-emergency supply busbar because of charred contacts at a switch, leads to voltage failure at reactor protection boards as both infeeds start out from one inverter; reactor scram.

- June 3, 1989, storm causes short circuit in one phase of the 500-kV overhead line. Subsequent failure of a main cooling water pump because of fault to earth (poor insulation of the power cable, contact with power switch casing), reduction of power to 60 %, switch-over of the steam supply for turbo-feedwater pumps from reheat to auxiliary steam is delayed; as a result of decrease of power in feedwater pumps, steam generator water level low leads to failure of two coolant pumps.

- June 18, 1989, absorber rod drive not connected with absorber rod, further operation with reduced power, later shutdown because of leak in a neutron flux measurement nozzle.

- August 15, 1989, non-opening of a deionate injection valve (operating fault) leads to the failure of the make-up pumps and to the interruption of the coolant pump seal water flow, further operating fault (isolation of seal water discharge at the shaft seals of the main coolant pumps) leads to a failure of all main coolant pumps and reactor scram.

- October 2, 1989, LP-emergency cooling pump operated 52 minutes with the pending signal "temperature high" (caused by the inflow of water into bearing) bearing molten, shaft damaged.

## 8.4 Summary

There were reports on 64 incidents from the "Incident Reporting Systems" of the IAEO and the "ISI" system of Interatomenergo Moscow for the reporting period from 1983 - 1990 for assessing the operating experienceof WWER-1000 type nuclear power plants. The data base available is too restricted for a final evaluation of this type. Nevertheless, characteristic features with respect to the design of the plant, the techical equipment, the quality of the components employed and the transient behaviour can clearly be recognised. The assessment shows that all transients proceed in clearly shorter periods than in the WWER-440/W-230 and W-213 plants. This can be explained by the high power density of the reactor and the low water reserves in the primary and secondary system in relation to power. Operating experience shows that the limits employed work too slowly for controlling fast transients and for preventing the actuation of protective actions. Operational staff are frequently unable to take manual measures for limiting incidents within the periods available.The degree of automation and the effectiveness of the control, limitation and protective devices should be increased significantly. The Konvoi plants, having about the same power density and similar specific volumes of water, have a higher number of fast-acting functional group controls and limitations.

The present incident reports show that the course of a transient in many cases was influenced by faults which occured in addition to the initiating event. These were often faults which had occured already some time ago, but which had remained undetected. Even if the number of the present incident reports is relatively small, the damage that occured gives the impression that the plants are in a poor maintenance state. The inspection concepts of the plants do not seem to be well suited for detecting faults and failures at the important safety-relevant systems at an early stage. Furthermore, the plants do not seem to be easily inspectable.

It can principally be said that the safety engineering of the plants is not consistently designed to be single-failure proof. The functional and physical separations of the redundancies have not been realised consistently. A complete examination of the control system seems to be inevitable. The functional separation of main and emergency control room with a reversible priority control is also missing.

The extreme turbine power oscillations observed during some incidents show that in the secondary system there are important design deficiencies in the area of control, limitation and protection systems.

The number of sub-cooling transients is also clearly excessive. Here the missing automatic actuation of turbine trip after reactor scram, as well as the further operation of the turbine feedwater pumps (also after turbine trip), become apparent.

The number of faults in steam generator feeding seems to be too high, as these are potential predecessors of severe plant disturbances.

Apart from that further systematic weaknesses of the plants can be perceived:

- Components which do not reach the required degree of reliability (6-kV power switches, emergency boron injection pumps ("small series").
- The quality of the power and control cables employed and of their connections is inadequate.
- System weaknesses impairing the safety of the plant (water level measurement of pressuriser, steam generator water level measurement, turbine control, feedwater control, missing signalisation and self-monitoring, missing priority control, missing redundancy in the area of the sensor).
- Actuation of transients because of missing redundancy in operational systems (two main coolant pumps on one oil circuit).
- Incidents can be actuated because of the adverse environmental conditions in the rooms and buildings of the plant.
- Missing or defective functional group controls and automatic inspection and test units, frequently necessitating manual interference and leading to maloperations.

# 9 Summary

The safety assessment of the nuclear power plant Stendal A of the type WWER-1000/W-320 has been performed by GRS on the basis of the current safety guidelines and technical regulations that apply in the Federal Republic of Germany. It largely consists of an evaluation of the design of the plant. The weaknesses perceived have been listed; in some cases possible solutions for upgrading measures are suggested.

The documentation of the Stendal nuclear power plant is incomplete and not always consistent, which reduces the reliability of the findings of the present design evaluation. Important information, for example referring to quality assurance, to the proper functioning of the planned components and piping, to accident analysis or to the concept for controlling external impacts, was not available to a sufficient degree. Therefore, the provision of the necessary precautions against possible damage arising from basic design and from operation of the plant could only be verified to a limited extent. In this respect, a final assessment of the concept of the WWER-1000/W-320 can only be made after additional documents have been presented and reviewed.

Despite the insufficient documentation, a definite safety assessment could be performed for essential areas. This holds especially for the systems analysis, where the operating experience of other WWER-1000s could be utilized for the assessment of the Stendal plant, and for the analysis of loss-of-coolant accidents, where incomplete documentation was supplemented by some calculations performed by GRS. As a result it was stated that although the plant partially meets the requirements of the German regulations, it has considerable weaknesses in the design. In cases where the Stendal plant does not meet the requirements of the German regulations, technical investigations were conducted to explore whether a deficit in terms of safety arises and which measures could possibly be taken to eliminate these deficiencies.

Investigations relating to accident management measures were not performed.

The WWER-1000 plants do not have the advantageous safety-related characteristics of the WWER-440 series such as low core power density, large water volume in the primary system, large water volume on the secondary side of the steam generators and the possibility of isolating main coolant loops. This results in higher safety-related requirements on the components and systems as well as on the operation compared with reactor plants of the WWER-440 series.

The most important results of the safety evaluation are subsequently presented, following the order of the structure of this report.

Referring to the **core design**, modifications are required which refer to the core loading, the control of the core power density as well as to the instrumentation.

The fuel assemblies designed for the three-year-cycle appear to be suited in principle. However, the loading of the core should be optimized by introduction of a low-leakage loading scheme. This requires the use of burnable absorbers like gadolinium in the fuel assemblies. The low-leakage loading at the same time results in a reduction of the neutron irradiation and thus in a reduced neutron induced embrittlement of the reactor pressure vessel wall.

The core power density distribution should be automatically controlled. An automatic insertion limitation has to be provided for the control elements. The operational use of the control elements has to be optimized to avoid the initiation of xenon instabilities.

The instrumentation of the core has to be enhanced as a prerequisite for an effective limit control of the core power density and for an improved power density monitoring. Regular testing of the power distribution detectors should be performed. Furthermore, with a reliable instrumentation, derived values such as low DNBR, can be introduced for reactor scram.

Referring to the **pressurized components,** three problem areas became evident: The neutron induced embrittlement of the reactor pressure vessel wall close to the core, the missing proof of exclusion of component ruptures in the primary and in the secondary systems and damage which occurred during normal operation loads in the cold collectors of the steam generators.

During the assessment of the reactor plant inadequate information was available with respect to the influence of the relatively high nickel content on the neutron embrittlement of the pressure vessel material. Therefore, measures for a long-term preservation of the present safety reserves, e.g. the use of shielding elements at the edge positions, or low leakage loadings of the reactor core are required until the respective documentation is available. After the examination of the first series of surveillance specimens in the reactor pressure vessel for monitoring the state of the material, a new decision has to be made. Pressure loads in a cold state of the plant have to be excluded by technical measures.

During the design and the manufacturing of the components of the primary and the secondary systems, insufficient measures were provided for excluding a rupture of these components as it is for instance suggested by the "basic safety" concept. This holds especially for the selected combination of materials in the secondary circuit. Here, the conditioning of the circulating water is restricted such that pit corrosion with subsequent stress corrosion cracking of the steam generator heater tubes and erosion corrosion of low-alloy steels cannot be avoided at the same time. Due to the physically close arrangement of the main steam and main feedwater lines in the penetration zone through the containment, subsequent ruptures of pipes cannot be excluded after the rupture of one single pipe.

All in all, 36 steam generators had to be replaced until the end of the year 1991 in nuclear power plants of WWER-1000 design. Cracks with a total length of more than 1 m have developed in the cold collector of the steam generators between the SG-tubes which are all fixed to the collector by an explosion technique. The failures occurred under operational conditions. So far, no solution for the problem could be found by modification of the manufacturing technology. In-depth analyses are required here, since the rupture of a collector can have significant radiological consequences.

Investigations are necessary to clarify the possible consequences of a complete failure of the collector on the integrity of the steam generator shell and on the containment, respectively.

The quality of the available **accident analyses** is insufficient. For instance, initial and boundary conditions which are not further described, outdated nuclear data and outdated actuation signals for the actuation of the reactor scram and of the safety system are frequently used. In numerous cases the simulation time of accident calculations is too short.

It is generally recommended to repeat the entire accident analysis for nuclear power plants of WWER-1000 design as a part of an up-dated safety analysis report with state-of-the-art computer codes using actual data for the reactor core, the reactor protection system and the process engineering subsystems of the safety system. In addition, accidents have to be investigated that are either specific to WWER plants, e.g. the rupture of the head of the steam generator collector, or accidents which have not been considered so far, e.g. ATWS, and accidents from plant shut down conditions. Furthermore, main steam or feedwater line ruptures subsequent to initiating steam or feedwater line ruptures in the vicinity of the penetration through the containment have to be analyzed. For some cases, e.g. reactivity accidents and ruptures in the main steam system, three-dimensional core dynamics codes have to be used. In the accident analyses to be performed, conservative assumptions have to be used systematically for the boundary conditions, e.g. the single failure criterion, the repair case, the second reactor scram signal etc.

To cope with leaks from the primary circuit to the secondary side, suitable accident procedures have to be developed on the basis of specific accident analyses still to be performed.

It can already be concluded from the available accident analyses that the main steam relief valves (BRU-A) and the pressurizer safety valves have to be upgraded for the impact of two-phase mixtures.

Incidents and accidents in WWER-1000 plants should be systematically evaluated with the aim to recalculate with advanced accident codes those cases which are well documented and which are well suited for code qualification.

According to GRS-analyses, the radiological consequences of the rupture of a primary circuit instrumentation line outside the containment, of damage to a fuel assembly caused by inadequate handling or of a double-ended rupture of a main coolant line are for some cases significantly below the accident planning levels in the Federal Republic of Germany. In the case of steam generator collector damage, e.g. in the collector head area, which leads to discharge of large amounts of primary coolant through main steam relief valves to the atmosphere, it was estimated that the radiological consequences to the environment will exceed the accident planning levels of the Radiological Protection Ordinance. Therefore, this group of accidents also has to be analyzed with respect to radiological consequences. Furthermore, hard-ware measures have to be provided to exclude leaks or to minimize leak cross-sections at the collector head of the steam generators.

Because of its steel-cellular-composite design, the **containment** of the nuclear power plant Stendal is a prototype. Its characteristics are supposed to correspond to a containment of the prestressed concrete construction type, as realized in all other WWER-1000 units. The design pressure is 500 kPa at a temperature of 150 °C. These values have been affirmed by a calculation performed by GRS for the double-ended break of a main coolant line, additionally including the secondary inventory of one steam generator and considering a safety margin of 15 % on the calculated excess pressure.

Differential pressures between the compartments of the containment, as well as jet and reaction forces were not examined in detail. The basic safety of the primary and secondary system components - which has not been documented so far - has to be proven as well as the capability of the containment to withstand the differential pressures and the jet and reaction forces.

The design load for the single shell containment with respect to airplane crash is considerably lower than the load specified in Germany. Backfitting is practically impossible.

The designed leakage rate of 0.1 Vol.% per day at design pressure is smaller than the common practice in Western Europe. The lack of suction from the containment annulus, which is common practice in Western reactors, could partly be compensated by an additional leakage suction system to be installed at the containment penetrations.

The engineered safeguards **systems** are physically separated to a large extent. They have a design capacity of 3 x 100 %. There are only few exceptions. However, it can be concluded from GRS-calculations that the emergency core cooling system cannot cope with a loss-of-coolant accident after a double-ended break in a main coolant line with unfavourable break location and a simultaneous occurence of a single failure with a repair case.

An independent, sheltered emergency system with additional water supply, as demanded by the German and also by the more recent Soviet regulations, is not existing. This system must be demanded. In addition, a component cooling system to the nuclear service water system must be demanded in order to avoid the release of radioactivity into the cooling ponds and thus to the environment, if the emergency cooler is leaking.

An essential weakness is the design of the three drain pipes of the emergency boric acid storage tank as simple pipes up to the isolating valves. If one of these pipes is leaking, the containment function is lost since the emergency boric acid storage tank is a part of the containment. Furthermore, the water supply is endangered for the high pressure and the low pressure emergency core cooling system and the spray system of the building. An improvement could be achieved by double-walled pipes with leak detection and isolation valves, which would have to be positioned as close as possible to the emergency boric acid storage tank.

In addition, the following essential backfitting measures are proposed:

- protective measures against missiles and fire at the 29.0 m level outside the containment in the area of the main feedwater and main steam lines as well as the main steam relief valves (BRU-A),
- installation of isolation valves (with an emergency power supply) upstream of the main steam relief valves to the atmosphere,
- keeping open the first isolation valves in the high pressure and low pressure injection lines of the emergency core cooling system during power operation.

Referring to **instrumentation and control (I&C)**, weaknesses have been detected to such an extent that it is recommended, as planned for the nuclear power plant Temelin, to replace the entire I&C system by a modern one. In this context the following recommendations should be considered:

- improvement of the insufficient control concept for dynamic transient processes (e.g. xenon instabilities in the reactor core),
- introduction of the missing control rod insertion limitations,

- introduction of a single-failure-proof and fault-self-revealing I&C for the safety system and for safety-related systems,
- qualification of the equipment according to international standards,
- installation of accident instrumentation,
- ensuring the independence of the emergency control room from the main control room,
- replacement of the online computer system which is too slow,
- replacement or upgrading of the core instrumentation (calibration of the neutron flux measurement, temperature measurement at the fuel assembly exit).

The concept of **electrical engineering** is accepted; however, the following improvements have to be performed, e.g. relating to

- quality assurance of cables,
- reliability of the circuit breakers,
- selectivity against short circuit in the emergency power system,
- additional connection to the grid by underground cable,
- qualification of the equipment of the emergency power system according to international standards.

**Internal hazards** like fire, flooding and crash of heavy loads have been analyzed with respect to conceptual deficiencies of their design. The requirements are fulfilled to a large extent by physical separation, by division into fire zones and by redundancy; however, analytical proof is frequently lacking.

Essential recommendations for improvements are:

- the emergency control room has to be separated from the main control room in a way that ensures its operability in the case of a fire in the main control room,

- cables of redundant systems which do not belong to the safety system must also be physically separated for fire protection reasons,
- a qualified leak detection system has to be provided for all rooms where safety-related systems are installed,
- the location of the emergency control room at the lowest level (-4.20 m) should be changed because of the flooding risk.

The current design of the cranes and of the fuel assembly reloading machine leads to handling restrictions which, however, can be suspended by backfitting measures.

The risk of a multiple pipe rupture resulting from pipe whip can only be limited by the leak-before-break concept. This applies to the steam generators installed in pairs as well as to the multiple penetrations through the containment at the 29.0 m level.

**External impacts** like earthquakes and blast wave loads as well as flooding have to be analyzed site-specific with respect to the design loads. They were not investigated here in detail.

The design loads for aircraft crash - which are insufficient according to the German regulations - have already been dealt with in the paragraph relating to the containment design.

A concept on external impacts has to be presented for assessment. This concept should comprise e.g. lists of the external-impact-safe facilities and a description of the measures to cope with damage following external impacts.

As the site of the nuclear power plant is 10 m above the average water level of the River Elbe, the flooding risk is assessed to be relatively low.

The investigations referring to **radiation protection** during normal operation showed that in accordance with the regulations, the release to the environment is far below the legal limits. The radiological protection of the staff should be improved especially in the following respect:

- the measurement systems for technological and dosimetric radiation protection control have to be upgraded in accordance with the state-of-the-art,
- the use of modern testing and remote control technology for radiation intensive activities has to be extended during maintenance work.

The evaluation of the **operating experience** of other WWER-1000 units showed a large number of weaknesses of certain components and systems. These findings have retroactive effects on the general assessment of components and systems. The main proportion of the GRS recommendations results from disturbances in instrumentation and control (41), followed by mechanical systems (13) and the house load power supply (11). Deficiencies in structural engineering (3) and in organisation, quality assurance and supervision (2) are of a minor importance because of their low frequency of occurrence.

The analyses which in addition are necessary for a **final evaluation** of reactors of the type WWER-1000/W-320 should be provided in the context of a comprehensive safety assessment of a plant which is already in operation or almost completed. A predominant part of these investigations should also be the quality assurance with regard to the project phase, the manufacturing, the construction and installation, as well as to the start-up and the operation of the plant according to the regulations. To assess whether the safety design of the plant is well-balanced, more use should be made of the evaluation of the operation experience. In addition, it is recommended to make use of probabilitistic methods.

# 9. Zusammenfassung

Die sicherheitstechnische Bewertung des Kernkraftwerks Stendal, Block A, vom Typ WWER-1000/W-320 wurde von der GRS anhand der in der Bundesrepublik Deutschland geltenden Sicherheitsrichtlinien und technischen Regeln durchgeführt. Sie erstreckt sich weitgehend auf eine Beurteilung der Anlagenkonzeption. Erkannte Schwachstellen sind aufgeführt; teilweise werden Lösungsmöglichkeiten für Ertüchtigungsmaßnahmen vorgeschlagen.

Der Unterlagenstand zum Kernkraftwerk Stendal ist unvollständig und nicht immer konsistent. Dies mindert die Belastbarkeit der im Rahmen der vorliegenden Konzeptbeurteilung gewonnenen Erkenntnisse. Wichtige Informationen beispielsweise zur Qualitätssicherung, zur Funktionstüchtigkeit der vorgesehenen Komponenten und Rohrleitungen, zur Störfallanalyse oder zum Konzept zur Beherrschung von Einwirkungen von außen lagen nur in unzureichendem Umfang vor. Deshalb konnte die Gewährleistung der erforderlichen Vorsorge gegen Schäden durch die Errichtung und den Betrieb der Anlage nur eingeschränkt überprüft werden. Insofern kann eine abschließende Aussage zum Konzept des WWER-1000/W-320 erst nach der Vorlage und Prüfung ergänzender Unterlagen erfolgen.

Trotz unzureichenden Dokumentationsstandes konnte jedoch für wesentliche Teilbereiche eine eindeutige sicherheitstechnische Bewertung durchgeführt werden. Dies gilt insbesondere für die Systemanalyse, bei der auch die Betriebserfahrungen anderer WWER-1000 für die Anlage Stendal genutzt werden konnten, und für die Analysen von Kühlmittelverluststörfällen, wo unzureichende Unterlagen in einigen Fällen durch eigene Rechnungen ergänzt wurden. Im Ergebnis wurde festgestellt, daß die Anlage zwar teilweise den Anforderungen des bundesdeutschen Regelwerkes entspricht, andererseits jedoch wesentliche konzeptionelle Schwachstellen vorhanden sind. In Fällen, in denen die Anlage Stendal den Anforderungen des bundesdeutschen Regelwerks nicht genügt, wurde durch ingenieurmäßige Untersuchungen geprüft, ob hierdurch ein sicherheitstechnisches Defizit besteht und welche Ersatzmaßnahmen gegebenenfalls zum Ausgleich möglich sind.

Es wurden keine Untersuchungen zu Accident-Management-Maßnahmen durchgeführt.

Die sicherheitstechnisch vorteilhaften Eigenschaften der Baulinie WWER-440 - wie z.B. geringe Leistungsdichte des Reaktorkerns, großes Wasservolumen des Primärkreislaufes, großes Wasserinventar der Sekundärseite der Dampferzeuger und Absperrbarkeit der Hauptumwälzleitungen - sind beim WWER-1000 nicht gegeben. Hieraus resultieren höhere sicherheitstechnische Anforderungen an Komponenten, Systeme und Betriebsführung im Vergleich zu den Reaktoranlagen der Baulinie WWER-440.

Im folgenden werden die wichtigsten Ergebnisse der Sicherheitsbewertung in der Reihenfolge der Gliederung des Berichtes dargestellt.

Bei der **Kernauslegung** sind Änderungen bezüglich der Kernbeladung, der Leistungs- und Leistungsdichteverteilungsregelung sowie der Instrumentierung erforderlich.

Die für den Dreijahreszyklus vorgesehenen Brennstoffkassetten erscheinen grundsätzlich geeignet. Die Kernbeladung ist durch Einführung einer Low-Leakage-Beladung zu optimieren. Dies erfordert den Einsatz von abbrennbaren Absorbern wie Gadolinium in den Brennstoffkassetten. Die Low-Leakage-Beladung führt gleichzeitig zur Verringerung der Neutronenbestrahlung und damit zur geringeren Neutronenversprödung in der Wand des Reaktordruckgefäßes.

Die Leistungsdichteverteilungsregelung ist zu automatisieren. Für die Steuerelemente ist eine automatische Einfahrbegrenzung vorzusehen. Der betriebliche Einsatz der Steuerelemente ist zu optimieren, um die Anregung von Xenonschwingungen zu vermeiden.

Die Kerninstrumentierung ist als Voraussetzung für die wirkungsvolle Leistungsdichtebegrenzung und -überwachung zu verbessern. Eine regelmäßige Überprüfung der Leistungsverteilungsdetektoren ist vorzusehen. Bei einer zuverlässigen Instrumentierung können auch abgeleitete Größen wie DNB-Werte zur Reaktorschnellabschaltung eingeführt werden.

Bei den **druckführenden Komponenten** sind drei Problemkreise deutlich geworden: Die Neutronenversprödung der kernnahen Wand des Reaktordruckgefäßes, der fehlende Nachweis des Bruchausschlusses für Komponenten des Primär- und des

Sekundärkreislaufes und bei normalen Betriebsbelastungen entstandene Schäden in den kaltseitigen Kollektoren der Dampferzeuger.

Während der Beurteilung der Reaktoranlage standen keine ausreichenden Informationen zum Einfluß des relativ hohen Nickelgehaltes auf die Neutronenversprödung des Reaktordruckgefäßwerkstoffes zur Verfügung. Deshalb sind bis zum Vorliegen entsprechender Unterlagen Maßnahmen zur langfristigen Erhaltung der vorhandenen Sicherheitsreserven notwendig, wie z. B. der Einsatz von Abschirmkassetten auf den Randpositionen bzw. Low-Leakage-Beladungen des Reaktorkernes. Nach der Prüfung der ersten Serie von Einhängeproben im Reaktordruckgefäß zur Überwachung des Werkstoffzustandes ist dann erneut zu entscheiden. Druckbelastungen im kalten Anlagenzustand sind durch technische Maßnahmen auszuschließen.

Bei der Auslegung und Fertigung der Komponenten des Primär- und des Sekundärkreislaufes wurden keine hinreichenden Maßnahmen vorgesehen, die einen Bruch dieser Komponenten ausschließen, wie sie z. B. das Basissicherheitskonzept vorschlägt. Dies gilt insbesondere für die ausgewählte Werkstoffkombination im Sekundärkreislauf. Hier sind der Konditionierung des Kreislaufwassers enge Grenzen gesetzt, so daß Lochkorrosion mit nachfolgender Spannungsrißkorrosion der Dampferzeugerheizrohre und Erosionskorrosion der niedriglegierten Stähle nicht gleichzeitig vermieden werden können. Beim Versagen auch nur einer Leitung können in Folge der räumlich konzentrierten Anordnung der Frischdampf- und der Speisewasserleitungen im Bereich der Durchführung durch das Containment schwerwiegende Folgeschäden nicht ausgeschlossen werden.

In Kernkraftwerken der Baulinie WWER-1000 mußten bis Ende 1991 insgesamt 36 Dampferzeuger ausgetauscht werden. Bei Betriebsbelastung waren in den Stegen zwischen den eingesprengten Heizrohren der kaltseitigen Kollektoren Risse bis über 1 m Gesamtlänge entstanden. Das Problem konnte durch Veränderung der Herstellungstechnologie bisher nicht gelöst werden. Hier sind vertiefende Analysen erforderlich, da der Bruch eines Kollektors erhebliche radiologische Konsequenzen haben kann.

Zur Klärung von möglichen Auswirkungen eines totalen Kollektorversagens auf die Integrität des Dampferzeugermantels und gegebenenfalls des Containments sind Untersuchungen erforderlich.

Die Qualität der vorliegenden **Störfallanalysen** ist unzureichend. So werden häufig nicht mehr näher bezeichnete Anfangs- und Randbedingungen, überholte nukleare Daten und nicht mehr aktuelle Anregesignale für das Havarieschutzsystem und für die verfahrenstechnischen Einrichtungen des Sicherheitssystems verwendet. In etlichen Fällen ist die Simulationsdauer der Störfallrechnungen zu kurz.

Generell wird empfohlen, die gesamte Störfallanalyse für Kernkraftwerke der Baulinie WWER-1000 als Teil eines aktualisierten Sicherheitsberichts mit fortschrittlichen verifizierten Rechenprogrammen unter Verwendung aktueller Daten für den Reaktorkern, das Havarieschutzsystem und die verfahrenstechnischen Einrichtungen des Sicherheitssystems erneut durchzuführen. Dabei müssen auch WWER-spezifische Störfälle, wie z. B. der Abriß des Kollektordeckels, sowie bisher noch nicht berücksichtigte Störfälle, wie z. B. ATWS und solche im abgeschalteten Zustand der Anlage, untersucht werden. Des weiteren sind Folgebrüche von Frischdampf- und Speisewasserleitungsbrüchen im Bereich der Durchführung des Containments zu analysieren. Für einige Fälle wie z. B. Reaktivitätsstörfälle und Brüche im Frischdampfsystem sind auch dreidimensionale Kerndynamik-Programme einzusetzen. In den durchzuführenden Störfallanalysen sind systematisch konservative Annahmen für die Randbedingungen wie die Berücksichtigung des Einzelfehlers, des Reparaturfalls, des zweiten Reaktorabschaltsignals usw.zu treffen.

Zur Beherrschung von Lecks vom Primärkreislauf zur Sekundärseite sind auf der Basis durchzuführender Analysen geeignete Störfallprozeduren zu entwickeln.

Es ergibt sich bereits aus den vorliegenden Störfallanalysen, daß die Frischdampf-Abblaseregelventile (BRU-A) und die Druckhalter-Sicherheitsventile für die Beaufschlagung mit Zweiphasengemisch auszulegen sind.

Störfälle in Anlagen der Baulinie WWER-1000 sollten systematisch mit dem Ziel ausgewertet werden, die gut dokumentierten und für eine Code-Qualifizierung ergiebigen Fälle mit fortschrittlichen Störfall-Codes nachzurechnen.

Die radiologischen Auswirkungen beim Bruch einer primärkühlmittelführenden Meßleitung außerhalb des Containments, bei einer Brennstoffkassettenbeschädigung durch Handhabungsfehler und beim 2F-Bruch einer Hauptumwälzleitung bleiben nach eigenen Analysen zum Teil deutlich unter den bundesdeutschen Störfallplanungswerten. Für den Fall eines Dampferzeugerkollektor-Schadens, z. B. im Deckelbereich, mit dem Austrag von großen Mengen des Primärkühlmittels über die Frischdampf-Abblaseregelventile in die Atmosphäre wurde abgeschätzt, daß die radiologischen Auswirkungen auf die Umgebung die Störfallplanungswerte der Strahlenschutzverordnung überschreiten werden. Diese Störfallgruppe ist daher auch hinsichtlich ihrer radiologischen Auswirkungen zu untersuchen. Unabhängig davon sind konstruktive Maßnahmen zum Leckausschluß bzw. zur Minimierung der Leckquerschnitte im Bereich des Dampferzeugerkollektors vorzusehen.

Das **Containment** des Kraftwerkes Stendal ist wegen der Stahlzellenverbundbauweise ein Prototyp. In seinen Eigenschaften soll es einem Containment in Spannbetonbauweise entsprechen, wie es bei allen anderen WWER-1000-Blöcken verwirklicht ist. Als Auslegungswert wird ein Innendruck von 500 kPa absolut bei einer Temperatur von 150°C zugrunde gelegt. Die Einhaltung dieses Wertes wurde durch eine GRS-Rechnung zum doppelendigen Abriß einer Hauptumwälzleitung auch bei zusätzlicher Entleerung eines Dampferzeugers und der Berücksichtigung eines Sicherheitszuschlages von 15 % auf den berechneten Überdruck bestätigt.

Differenzdrücke zwischen den Räumen des Containments sowie Strahl- und Reaktionskräfte wurden nicht detailliert untersucht. Die bisher nicht belegte Basissicherheit der Primär- und der Sekundärkreislaufkomponenten sowie die Lastabtragung der Differenzdrücke und der Strahl- und Reaktionskräfte muß nachgewiesen werden.

Die Auslegung des einschaligen Containments gegen Flugzeugabsturz sieht eine Belastung vor, die deutlich geringer ist als in Deutschland vorgeschrieben. Eine Nachrüstung ist praktisch unmöglich.

Die vorgesehene Leckrate beim Auslegungsdruck ist mit 0,1 Vol-% pro Tag kleiner als in Westeuropa üblich. Das Fehlen der in westlichen Reaktoren üblichen Absaugung aus dem Ringraum könnte durch ein zusätzlich zu installierendes

Leckabsaugesystem an den Containmentdurchdringungen zum Teil kompensiert werden.

Die **Systeme** der Sicherheitseinrichtungen sind weitgehend räumlich getrennt und mit einer Kapazität von 3 x 100 % ausgelegt. Nur in wenigen Fällen gibt es Ausnahmen. Aus eigenen Rechnungen zur Störfallanalyse ist zu folgern, daß das Havariekühlsystem jedoch einen Kühlmittelverluststörfall nach einem 2F-Bruch in der Hauptumwälzleitung mit ungünstiger Bruchlage bei gleichzeitigem Auftreten eines Einzelfehlers und eines Reparaturfalles nicht beherrschen kann.

Ein unabhängiges, verbunkertes Notstandssystem mit zusätzlichen Wasservorräten, wie es das bundesdeutsche und auch das neuere sowjetische Regelwerk vorsehen, ist nicht vorhanden. Dieses System ist ebenso zu fordern wie ein Zwischenkühlkreislauf zum nuklearen Nebenkühlwassersystem, der bei Undichtigkeiten im Havariekühler den Austrag von Radioaktivität in die Kühlteiche und damit in die Umgebung verhindert.

Eine wesentliche Schwachstelle sind die drei Ablaufleitungen des Havarieborbehälters, die als einfache Rohrleitungen bis zur Absperrarmatur ausgeführt sind. Bei Undichtigkeit in einer dieser Leitungen geht sowohl die Containmentfunktion verloren, da der Havarieborbehälter Teil des Containments ist, als auch die Wasservorräte für das HD- und das ND-Notkühlsystem und das Gebäudesprühsystem. Durch ein doppelwandiges Rohr mit Leckdetektion und eine Absperrarmatur, die möglichst dicht am Havarieborbehälter anzuordnen ist, könnte eine Verbesserung erreicht werden.

Weitere wesentliche Nachrüstmaßnahmen werden vorgeschlagen:

Schutzmaßnahmen gegen Bruchstücke und Feuer auf der 29,0-m-Ebene außerhalb des Containments bei der Konzentration der vier Speisewasser- und der vier Frischdampfleitungen sowie der vier Abblaseregelventile (BRU-A)

Installation von notstromversorgten Absperrventilen vor den Abblaseregelventilen

Offenhaltung der Erstabsperrung in den Einspeiseleitungen des HD- und ND-Notkühlsystems während des Leistungsbetriebes.

Bei der **Leittechnik** wurden Schwachstellen in einem solchen Umfang gefunden, daß vorgeschlagen wird, wie auch beim KKW Temelin vorgesehen, die gesamte Leittechnik gegen eine modernere auszutauschen. Die folgenden Empfehlungen sollten dabei berücksichtigt werden:

- Verbesserung des mangelhaften Regelkonzeptes bei dynamischen Übergangsprozessen (z.B. Xenon-Schwingungen im Kern)
- Einführung der fehlenden Steuerstabfahrbegrenzungen
- Einführung einer fehlerselbstmeldenden und einzelfehlerfesten Sicherheitsleittechnik und Leittechnik für die sicherheitsrelevanten Systeme
- Qualifikation der Ausrüstungsteile entsprechend internationalen Standards
- Installierung einer Störfallinstrumentierung
- Sicherstellung der Unabhängigkeit der Block- von der Reservewarte
- Austausch des zu langsamen Blockrechners
- Austausch bzw. Ergänzung der Kerninstrumentierung (Kalibrierung der Neutronenflußmessung, Temperaturmessung am Brennstoffkassettenaustritt).

Das Konzept der **Elektrotechnik** wird akzeptiert, allerdings müssen Verbesserungen durchgeführt werden, z.B.:

- Qualitätssicherung bei Kabeln
- Zuverlässigkeit der Schalter
- Selektivität gegen Kurzschluß im Notstromsystem
- Zweiter Netzanschluß als Kabelanschluß
- Qualifikation der Ausrüstungsteile des Notstromsystems entspreched internationalen Standards.

Anlageninterne **übergreifende Ereignisse** wie Brand, Überflutung und fallende Lasten wurden hinsichtlich konzeptioneller Auslegungsmängel untersucht. Durch räumliche Trennung, Einteilung in Brandabschnitte und Redundanz werden die Anforderungen weitgehend erfüllt, jedoch fehlen oft die analytischen Nachweise.

Wesentliche Verbesserungsvorschläge sind:

- Die Reservewarte ist von der Blockwarte so zu entkoppeln, daß bei einem Brand in der Blockwarte die Funktion der Reservewarte erhalten bleibt.
- Kabel von redundanten Systemen, die nicht zum Sicherheitssystem gehören, sind ebenfalls brandschutztechnisch voneinander zu trennen.
- Ein qualifiziertes Leckageerkennungssystem muß in allen Räumen, in denen sicherheitsrelevante Systeme installiert sind, vorhanden sein.
- Die Anordnung der Reservewarte auf der untersten Ebene (- 4,20 m) sollte aus Gründen der Überflutungsgefahr geändert werden.

Bei der jetzigen Auslegung der Krane und der Brennstoffkassetten-Umlademaschine ergeben sich Beschränkungen bei der Handhabung, die sich jedoch durch Nachrüstungen aufheben lassen.

Die Gefahr des mehrfachen Rohrbruchs durch Rohrschlagen läßt sich nur durch das Leck-vor-Bruch-Konzept begrenzen. Das gilt sowohl für die paarweise aufgestellten Dampferzeuger als auch für die Mehrfachdurchdringungen des Containments auf der 29,0-m-Ebene.

**Einwirkungen von außen** (EVA) wie Erdbeben und Druckwellenbelastung sowie Überschwemmung sind hinsichtlich der Lastannahmen standortspezifisch zu betrachten. Sie wurden hier nicht detailliert untersucht.

Die nach den bundesdeutschen Regeln zu geringen Lastannahmen beim Flugzeugabsturz wurden bereits unter Containmentauslegung behandelt.

Zu einer Beurteilung der anlagentechnischen Maßnahmen zur Beherrschung der Einwirkungen von außen ist die Vorlage eines EVA-Konzeptes erforderlich. Dieses muß z. B. Listen der EVA-sicheren Einrichtungen und eine Beschreibung der Maßnahmen zur Beherrschung von EVA-Folgeschäden enthalten.

Die Gefahr durch Überschwemmung wird als gering eingeschätzt, da das Kernkraftwerksgelände 10 m über dem mittlerem Elbwasserspiegel liegt.

Die Untersuchungen zum **Strahlenschutz** bei bestimmungsgemäßem Betrieb haben ergeben, daß die Freisetzung in die Umgebung weit unterhalb der gesetzlichen Grenzwerte liegt. Der radiologische Arbeitsschutz sollte besonders in den folgenden Punkten verbessert werden:

- Die Meßsysteme zur systemtechnischen und dosimetrischen Strahlenschutzüberwachung sind entsprechend dem Stand von Wissenschaft und Technik zu ändern.
- Zur Durchführung von Instandhaltungsarbeiten ist der Einsatz moderner Prüftechnik sowie Fernbedientechnik für strahlenintensive Tätigkeiten zu erhöhen.

Die **Auswertung der Betriebserfahrungen** von anderen Blöcken der Baulinie WWER-1000 hat unter anderem eine große Anzahl von Schwachstellen an bestimmten Komponenten und Systemen aufgezeigt. Dies hat Rückwirkungen auf die grundsätzliche Bewertung von Komponenten und Systemen. Die überwiegende Zahl der GRS-Empfehlungen resultiert aus Störungen auf dem Gebiet der Leittechnik (41), gefolgt von der Maschinentechnik (13) und der Eigenbedarfsversorgung (11). Mängel bei der bautechnischen Ausführung (3) und der Organisation, Qualitätssicherung und Kontrolle (2) treten zahlenmäßig in den Hintergrund.

Für eine **endgültige Beurteilung** von Reaktoranlagen des Typs WWER-1000/W-320 sollten die noch erforderlichen Analysen und Nachweise im Rahmen einer ausführlichen Sicherheitsbewertung möglichst am Beispiel einer in Betrieb befindlichen oder nahezu fertiggestellten Anlage erstellt werden. Teil dieser Untersuchungen sollte vordringlich auch eine Bewertung der Qualitätssicherung in Bezug auf die Projektierung, die Fertigung, die Montage bzw. die Errichtung, die Inbetriebsetzung und den bestimmungsgemäßen Betrieb der Anlage sein. Zur Bewertung der sicherheitstechnischen Ausgewogenheit der Anlagenauslegung sollte die Auswertung der Betriebserfahrung verstärkt genutzt werden. Außerdem wird empfohlen, probabilistische Methoden einzusetzen.

# 9 Резюме

Оценка безопасности АЭС Стендаль, блока А типа ВВЭР-1000 была проведена по поручению федерального управления по радиационной безопасности (BFS) на основе действующих в ФРГ руководящих указаний по безопасности и технических норм. Она распространяется главным образом на оценку концепции установки. В ней приводятся выявленные уязвимые места и частично предлагаются возможные решения с помощью модернизации.

Существующая документация по АЭС Стендаль некомплектна и частично противоречива. Таким образом это снижает ценность выводов, полученных в рамках оценки концепции. Важная информация такая, как например по контролю качества. по работоспособности предусмотренного оборудования и трубопроводов, по анализам аварийных ситуаций или по концепции защиты против внешнего воздействия была представлена в недостаточном объёме. В связи с этим проверку гарантий по обеспечению безопасной эксплуатации АЭС можно было проверить только с ограничениями. Таким образом окончательные выводы по концепции ВВЭР-1000/В-320 могут быть сделаны только после представления и проверки дополнительной документации.

Несмотря на недостаточный объём документации была проведена однозначная оценка безопасности отдельных участков установки. Это особенно касается системного анализа, где были использованы и результаты опыта эксплуатации других установок типа ВВЭР-1000, и анализа аварийных ситуаций, где недостаточные документации, в некоторых случаях, были дополнены собственными расчётами. В результате было установлено, что эта установка хотя частично удовлетворяет требованиям регламентов ФРГ, с другой стороны в ней имеются принципиально слабые места. В тех случаях, когда установка Стендаль не удовлетворяла требованиям регламентов ФРГ, были проведены инженерные исследования с целью, возникает ли в этом случае дефицит по обеспечению безопасности и какие эквивалентные меры возможны в этом случае.

Не были проведены исследования по мерам по устранению последствий аварий.

Положительные качества присущие ВВЭР-440, такие как например: низкое энерговыделение в активной зоне, большой водный объём в первом контуре, большие резервы воды во втором контуре парогенераторов и отсекаемость петель главного циркуляционного контура, они недействительны для установок ВВЭР-1000. Поэтому для компонентов, систем и эксплуатации реакторных установок ставятся более высокие требования по обеспечению безопасности по сравнению с установками типа ВВЭР-440.

Ниже приведены важнейшие результаты оценки безопасности том же порядке, как и в оглавлении.

**При разработке активной зоны** нужно будет выполнить некоторые изменения в отношении загрузки, управления мощностью и распределением энерговыделения, а также приборного обеспечения.

Трехлетний цикл топливной загрузки признается принципиально пригодным. Загрузку нужно будет оптимировать с помощью загрузки с малой утечкой нейтронов. Это требует применения выгораемых поглотителей из гадолиния в топливных сборках. Загрузка с малой утечкой ведёт одновременно к снижению нейтронного облучения и таким образом меньшему охрупчиванию стенок корпуса реактора.

Управление распределением энерговыделения в активной зоне необходимо автоматизировать. Для управляющих элементов нужно предусмотреть автоматическое ограничение по вводу в зону. Режим работы регулирования управляющими элементами нужно организовать таким образом, чтобы избегать возникновения ксеноновых колебаний.

В качестве предпосылки для эффективного ограничения плотности нейтронного потока в активной зоне и его контроля нужно улучшить приборное обеспечение контроля активной зоны. Нужно предусмотреть регулярную проверку детекторов распределения нейтронного потока в активной зоне. При использованиии надёжной техники возможно использование производных параметров для аварийной защиты реактора, таких как например запас до кризиса кипения.

Для **оборудования находящегося под давлением** выявлены три проблемные области:
охрупчивание стенок корпуса реактора со стороны активной зоны в результате нейтронного облучения;
отсутствия доказательства исключения разрывов на оборудовании первого и второго контура;
повреждения "холодного" коллектора парогенератора при нормальной эксплуатации.

Во время проведения оценки реакторной установки не имелось достаточной информации по влиянию сравнительно высокого содержания никеля на охрупчивание материала корпуса реактора. Поэтому нужно до получения соответствующей документации реализовать мероприятия по долгосрочному сохранению существующих резервов по обеспечению безопасности, такие как например: использование экранирующих кассет на краю активной зоны и реализация загрузки с малой утечкой. После исследования первой серии образцов для контроля охрупчивания материала корпуса реактора вынести соответствующее решение. Исключить нагрузки давлением в холодном состоянии с помощью технических мер.

При проектировке и изготовлении компонентов первого и второго контура не были предусмотрены в достаточной мере мероприятия, которые бы исключали разрывы этих компонентов, как это предлагается в концепции "базисной безопасности". Это относится к комбинациям материалов во втором контуре. Здесь ставятся сложные условия по поддержанию водного режима в контуре, так что язвенная коррозия и последующая коррозия в виде трещин трубчатки парогенераторов и эрозивная коррозия низколегированных сталей не могут одновременно исключаться. Концентрация трубопроводов свежего пара и питательной воды в районе проходок из защитной оболочки может привести при повреждении даже одного труборовода к тяжелым последствиям.

До конца 1991 г. на АЭС с ВВЭР-1000 должна быть проведена замена 36 парогенераторов. В результате нормальных эксплуатационных нагрузок возникли трещины длиной более 1 метра на холодных коллекторах парогенераторов в области между запрессоваными трубками. До сих пор эта проблема не могла быть решена на основе изменения технологии изготовления. Следует провести углубленные исследования, вследствии возможного радиологического воздействия на окружающую среду в случае разрыва коллектора.

Необходимо провести исследования по выяснению возможного влияния полного отказа коллектора на целостность корпуса парогенератора и в отрицательном случае - контейнемента.

Качество представленных **анализов безопасности** является недостаточным. В них часто были использованы не четко определенные исходные и граничные условия, устаревшие нейтронно-физические характеристики, устаревшие сигналы по срабатыванию аварийной защиты и системы безопасности. Для многих случаев расчёта аварийных процессов длительность симуляции их была слишком короткой.

Поэтому рекомендуется провести снова весь анализ безопасности АЭС типа ВВЭР-1000 заново, в качестве актуализированной части отчёта безопасности, с помощью современных верифицированных расчётных программ с использованием актуализированных данных по активной зоне, по аварийной защите и системам обеспечения безопасности. При этом должны быть исследованы аварийные ситуации специфичные для реакторов типа ВВЭР, как например отрыв крышки коллектора и такие, как например ATWS-аварии с отказом аварийной защиты реактора и аварийные ситуации для отключенного реактора. Кроме этого нужно проанализировать вторичные разрывы трубопроводов, как последствия разрывов трубопроводов свежего пара и питательной воды в районе проходок из защитной оболочки. Для некоторых случаев, таких как например: аварии связанные с реактивностью и разрывы в системе свежего пара рекомендуется использование трехмерных расчетных программ по динамике активной зоны. При проводимых анализах нужно систематически применять консервативные предположения для таких граничных условий, как учёт единичной ошибки, ремонта, второго сигнала по отключению реактора и т.д.

Для освоения течи из первого во второй контур следует разработать соответствующие меры по устранению аварии на основе проводимых анализов. Из представленных анализов уже сейчас видно, что клапаны БРУ-А и предохранительные клапаны компенсатора давления должны проектироваться с учетом влияния двухфазной смеси.

Аварийные ситуации происшедшие на установках типа ВВЭР-1000 должны систематически исследоваться с целью хорошего документирования и для последующего расчёта аварийных процессов на базе современного кода, которые могут быть использованы для улучшения кода.

Радиологическое воздействие на окружающую среду при разрыве измерительного трубопровода с водой первого контура вне защитной оболочки, при повреждении топливной кассеты при манипуляции с ней и при двухстороннем разрыве трубопровода главного циркуляционного контура находится, по собственному анализу, частично значительно ниже проектных значений ФРГ для аварий. Для случая повреждения коллектора парогенератора, как например в области крышки коллектора, была проведена оценка выброса среды первого контура через сбросные клапаны свежего пара в атмосферу, при которой радиологическое воздействие на окружающую среду превышает проектные значения регламента по радиационной защите. Для этой группы аварий нужно провести исследования в отношении влияния на окружающую среду. Независимо от этого следует предусмотреть меры по исключению течи или минимизации сечения течи в части коллектора парогенератора.

**Контейнемент (защитная оболочка)** АЭС Стендаль является прототипом вследствии конструкционного исполнения в виде стальных секций. Его свойства должны соответствовать контейнементу из напряженного железобетона, как это было выполнено на всех других блоках с ВВЭР-1000. В качестве проектного значения было принято давление 500 кПа (абс.) при температуре 150 град. С. Соблюдение этого значения подтверждается расчётом GRS для двухстороннего разрыва трубопровода главного циркуляционного контура при дополнительном опрожнении парогенератора и учёта фактора по запасу в 15% для рассчитанного избыточного давления.

Не было проведено тщательное исследование разности давлений в помещениях контейнемента, а также струйных и реактивных сил. До сих пор недоказанная базисная безопасность оборудования первого и второго контура должна быть доказана, а также компенсации возникающей нагрузки на основе перепадов давлений струйных и реактивных сил.

**Проектное решение** контейнемента с защитой против падения самолёта в виде одной защитной оболочки предусматривает нагрузку,которая составляет, примерно, только половину значения, которое предписывается в Германии. Дооснащение в этом случае практически не возможно.

Предусмотренные протечки при расчётном давлении величиной 0,1% объёма в день меньше, чем обычно в Западной Европе. Отсутствие отсосов из кольцевого помещения, как это принято для западных реакторов, можно частично компенсировать с помощью дополнительной системы отсоса протечек на проходках в контейнементе.

**Системы** устройств обеспечения безопасности расположены значительным образом в отдельных помещениях и разработаны с резервированием 3*100%. Только в некоторых случаях существуют исключения. Исходя из собственных расчетов по анализу безопасности следует, что аварийная система охлаждения не осваивает аварию с двухсторонним разрывом трубопровода главного циркуляционного контура с неблагоприятным местом разрыва, одновременной единичной ошибкой и случаем ремонта.

Установка не имеет независимой аварийной системы бункерного исполнения с дополнительными запасами воды, как это требуется нормами ФРГ и также новыми советскими нормами. Кроме того требуется промежуточный контур охлаждения для системы технической воды, который бы исключал возможность выноса радиоактивности в пруды-охладители и таким образом в окружающую среду при повреждениях телообменника аварийного расхолаживания.

Важным слабым звеном являются трубопроводы подачи из баков аварийного запаса бора, которые выполнены в виде обычных трубопроводов до отсечной арматуры. При разуплотнении этих трубопроводов теряется функция контейнемента, так как бак аварийного запаса бора является сам частью контейнемента, так и потеря запасов воды для аварийных систем высокого и низкого давления и спринклерной системы. Улучшение ситуации может быть достигнуто с помощью двойной трубы с индикацией течи и отсечной арматурой, которая устанавливается в непосредственной близости от бака аварийного запаса.

Предлагаются дополнительные существенные меры по модернизации:

- Защитные мероприятия против обломков и пожара на отметке 29,0 м., вне контейнемента в том месте, где сконцентрированы 4 трубопровода питательной и 4 трубопровода свежего пара, а также 4 сбросных клапана БРУ-А;

- Запорную арматуру с надёжным электропитанием перед регулирующими клапанами сброса пара;

- Открытое состояние первых отсечных арматур в трубопроводах подачи от систем высокого и низкого давления во время работы на мощности.

В **технике автоматического управления и контроля** было найдено много слабых мест и таким образом предлагается чтобы, как и случае АЭС Темелин , эта техника была полностью заменена на более современную. При этом должны быть учтены следующие рекомендации:

- Улучшение концепции регулирования при переходных процессах (например ксеноновые колебания активной зоны);
- Введение отсутствующих ограничений по перемещению регулирующих стержней;
- Реализация техники управления и контроля для систем безопасности устойчивой против единичной ошибки и с самоиндикацией помехи и автоматики для систем важных с точки зрения обеспечения безопасности;
- Квалифицирование оборудования согласно международным стандартам;
- Установка приборов регистрации параметров во время аварии;
- Реализация независимости блочного и резервного щита управления;
- Замена медленной блочной ЭВМ;
- Замена или дополнение приборного обеспечения контроля активной зоны (калибрация измерений нейтронного потока, измерение температуры на выходе из кассет).

**Концепция электротехники** является приемлемой, но должны быть выполнены некоторые изменения, как например:

- Гарантия качества кабелей;
- Надёжность переключателей;
- Селективность в случае короткого замыкания в системе надёжного питания;
- Второе подключение к сети в виде кабельного соединения;
- Квалификация оборудования системы надёжного электропитания в соответствии с международными стандартами.

Было произведено исследование событий внутри установки, таких как пожар, затопление и падение груза, которые пересекают границы систем, в отношении недостатков проектного решения. Поставленные требования большей частью выполняются на основе разделения по помещениям, разделения на пожаростойкие участки и разделения редундантного оборудования, но при этом часто отсутствуют аналитические доказательства.

Ниже приводятся существенные рекомендации по уязвимым местам:

- Резервный щит нужно отделить от блочного щита управления таким образом, чтобы при пожаре на блочном щите, функция резервного щита оставалась незатронутой;

- Кабели редундантных систем, которые не относятся к аварийным системам, следует тоже разделить с точки зрения пожарной безопасности;

- Во всех помещениях с системами важными с точки зрения безопасности должна быть установлена высококачественная система опознания течи;

- Расположение резервного щита управления на нижней отметке (-4,20 м.) должно быть заново продумано из-за опасности затопления.

Существующее проектное решение кранов и перегрузочной машины ведёт к ограничениям при их применении, которые можно устранить с помощью модернизации.

Опасность многократных разрывов трубопроводов вызванных ударами труб можно ограничить только на основе концепции "течь перед разрывом". Это действительно как для парогенераторов установленных парами, так и для многократных проходок через защитную оболочку отметке 29,0 м.

**Внешние воздействия** такие, как землетрясения, ударная волна и наводнение, должны рассматриваться в отношении возникающих нагрузок специфически в зависимости от местоположения. Здесь они не были тщательно исследованы.

Слишком малая нагрузка, исходя из соответствующих норм ФРГ, на защитную оболочку в результате падения самолёта уже была обсуждена выше.

Для оценки технических мер против внешнего воздействия требуется соответствующая концепция. Она должна содержать например: перечень устройств устойчивых против внешнего воздействия и описание мер по устранению последствий его.

Опасность затопления оценивается, как незначительная, так как площадка АЭС находится на 10 м. выше среднего уровня Эльбы.

Исследования по **радиационной защите** при нормальной эксплуатации привели к выводу, что выбросы в окружающую среду находятся значительно ниже граничных значений, предписанных законом. Радиационная техника безопасности должна быть улучшена особенно по следующим пунктам:

- Нужно изменить измерительные системы радиационного и дозиметрического контроля в соответствии с совеменными требованиями науки и техники;

- Для проведения ревизий и ремонтных работ нужно увеличить объём применения современной техники контроля и дистанционных манипуляторов при работах связанных с ионизационным облучением.

Обработка результатов **опыта эксплуатации** других блоков типа ВВЭР-1000 указала наряду с другим большое число недостатков определенных оборудований и систем. Эти факты оказали влияние на принципиальную оценку оборудований и систем. Большая часть рекомендаций GRS следует из неполадок в части техники управления и контроля (41), за ними следуют неполадки машинной техники(13) и системы собственных нужд (11). Недостатки по строительной части(3) и организационного порядка, контроля и гарантии качества (2) играют численно незначительную роль.

Считается целесообразным провести необходимые анализы и доказательства для окончательной оценки реакторных установок ВВЭР-1000 в рамках тщательной оценки безопасности, по возможности, на примере работающей или почти готовой к пуску установки. Часть этих исследований должна провести оценку контроля качества в отношении проектирования, изготовления, монтажа и сооружения, пуско-наладочных работ и эксплуатации установки. Для проверки проектного решения обеспечения безопасности установки должен быть усиленно использован опыт эксплуатации. Кроме того рекомендуется привлечь вероятностные анализы безопасности.

# 10 Recommendations

**Important recommendations are marked with an asterisk (*)**

**Recommendations derived from Chapter 2: Description of the nuclear power plant**

R 2.7-1* It is recommended that a consistent concept for the buildings and plant layout is worked out to control accidents that are caused by external events.

R 2.7-2* It has to be demonstrated that the engineered safeguards in the reactor building are not damaged to an inadmissible degree by vibrations caused by an airplane crash.

**Recommendations derived from Chapter 4: Core design and pressurised components**

R 4.1-1 A complete core-design report must be presented for the three-year life of the fuel elements.

R 4.1-2* A low-leakage core-loading strategy is recommended.

R 4.1-3* A limitation for control-rod insertion must be implemented.

R 4.1-4 It must be demonstrated for all admissible operating conditions that shutdown leads to a sub-critical state of at least 1 %, even with the failure of the most effective control element, until the sub-critical state is ensured by the liquid-poison systems.

R 4.1-5 Part-sized control elements must not be used.

R 4.1-6* Distribution control of the power and the power density as well as xenon control is to be automated.

R 4.1-7 It has to be demonstrated that the boron-injection systems fulfil their function as a second shutdown system, rendering the reactor core sufficiently subcritical, when also taking a single-failure criterion into account.

R 4.1-8 For the second shutdown system, it must be demonstrated by calculations that shutdown reactivity is 1 % when the neutron flux and the absorber concentration are monitored; without the monitoring measures, it must be 5 %.

R 4.1-9 Operating experience with the system for measuring the power-density distribution must be evaluated.

R 4.1-10* The concept of in-core instrumentation must be examined in order to supplement the existing power-density-distribution detectors with an additional system for calibrating and testing (cf. R 6.4-5).

R 4.1-11* In-core instrumentation must not be used for power-density-distribution monitoring alone but must be extended, through a link with the control-element-control system, to an automatic power-density-limitation system.

R 4.1-12* The determining transient for the definition of the minimum permissible Departure from Nucleate Boiling (DNB) values, the complete failure of the main coolant pumps or the single failure of one individual recirculation pump must be examined, taking the possible most unfavourable starting conditions into account.

R 4.1-13 A description of the experimental background of the Departure from Nucleate Boiling (DNB) correlation including a justification of the accuracy and the tolerance limit must be provided.

R 4.1-14* It must be examined whether a power-density-limitation system including a DNB-signal for reactor scram, derived from core instrumentation, is necessary for safety-related reasons.

R 4.1-15 The basic materials used and the additional welding materials must be assessed according to their material specifications, particularly with regard to their carbon content, taking operating experience into account.

R 4.1-16 The calculations for the verification of the toughness of the reactor pressure vessel internals in normal operating and accident conditions must be checked.

R 4.1-17 Operating experience relating to the fuel elements must be compiled, including the causes of any fuel-element damage that has been observed.

R 4.1-18 The existing differences between ZrNb1 and Zircaloy must be assessed under consideration of accident loads

R 4.1-19 The integrity of the core internals during normal operation is to be ensured for their entire service life, taking into account various operating modes.

R 4.1-20 It must be demonstrated that the core internals are designed in such a way that design limits (e.g. maximum fuel-rod-cladding temperature less than 1200 °C), required by emergency core cooling according to RSK-Guideline 22.1, are not exceeded under accident conditions.

R 4.2-1* The influence of the integral neutron fluence and the content of nickel in the basic material and the welding material of the reactor pressure vessel as well as the effect of the neutron-flux density on the neutron-embrittlement sensitivity, must be investigated.

R 4.2-2* Detailed examinations are necessary concerning the crack-formation tendency of the transitional area between basic material and cladding in the root area of the weld seams in the main coolant lines.

R 4.2-3 Documents about the testing and qualification of the material 06Ch12N3DL for the lower part of the main coolant pumps must be presented for an evaluation.

R 4.2-4 It is recommended to provide stress calculations for the components with the corresponding life-time analyses for loads resulting from operational transients and accidents, including oscillations caused by earthquakes, airplane crashes and blast waves.

R 4.2-5 The applicability of ultra-sound tests for primary-system components must be investigated. In particular, the number of existing restrictions for non-destructive tests is to be reduced by optimizing the conditions at the place of examination (e.g. level-grinding of excess weld material) and improving examination methods. If not even the adapted examination methods prove to be sensitive enough for fault detection, changes in the design must be considered.

R 4.2-6 For the assessment of the welds in the area of the pipe connections and the reactor pressure vessel head it is necessary to evaluate the manufacturing documentation due to the existing restrictions with regard to the possibility of ultra-sound tests.

R 4.2-7* An examination concept must be worked out for the connections and the perforated area of the reactor pressure vessel head, taking into account the way the connections are manufactured and built in.

R 4.2-8 A leak-monitoring system for localising leakages must be installed at the RPV-head penetrations.

R 4.2-9 A testing concept for in-service inspections, based on the eddy-current test method, must be worked out for the steam-generator tubes; it must also be able to detect early any possible operationally induced damages in the bend areas.

R 4.2-10 Possibilities of pollution in the primary system must be analysed and, if existing, eliminated by technical measures (e.g. by installation of resin catchers).

R 4.2-11 Automatic measuring devices must be installed for the monitoring of chemical parameters in the primary system and in the make-up system as well as in the secondary system.

R 4.2-12* The material concept of the secondary system must be revised with a view to preventing local corrosion of the steam-generator tubes and erosion-corrosion in the condensate and feedwater areas supported by improved water chemistry.

R 4.2-13* According to the present state of knowledge (material specification, documentation about installation and routing), a break and consecutive damage of the main-steam, feedwater and emergency-feedwater systems outside the containment cannot be excluded. These events must be included in the accident analyses.

R 4.2-14 Until a status report on neutron embrittlement of the RPV-material is available, shielding elements must be used at the outer positions of the fission zone in order to maintain a safe distance to prevent brittle fracture of the reactor pressure vessel.

R 4.2-15* The validity of examination results taken from suspended samples in the reactor pressure vessel must be checked with regard to the influence of the neutron-flux density and the irradiation temperature.

R 4.2-16* Administrative measures and technical installations for the prevention of cold pressure overloads on the primary system must be examined and introduced where necessary.

R 4.2-17* Available examination methods must be adapted to their respective tasks on the reactor pressure vessel; any remaining examination restrictions must be assessed as to their safety relevance.

R 4.2-18* The knowledge of the damage mechanism leading to crack formation in the walls of crosspieces between the holes, leading to leaks in the cold collectors of the steam generators, must be intensified. Measures to prevent such damage must be worked out and implemented. Additionally, a non-destructive test method for the early detection of initial cracks must be upgraded. The influence of these initial cracks on the integrity of the collectors must be analysed.

R 4.2-19* The effects on the steam-generator wall at the failure of the steam-generator collector and fast depressurisation of the primary system must be examined. If necessary, the effects of radiation and reaction loads on neighbouring steam generators as well as the effects on the containment integrity must be analysed.

R 4.2-20 A static calculation of the piping system of the main coolant lines must be provided for the evaluation of stress level and peak stresses.

R 4.2-21 The qualification of the supports of components and pipes as pipe-whip limiters in case of pipe breaks must be verified.

**Recommendations derived from Chapter 5: Accident Analysis**

R 5.1-1 It is recommended that in accordance with the RSK-Guideline 21.1 (3) and with regard to the design of the emergency-cooling system an analysis of the RPV-leak of 20 cm2 located below the upper core edge is carried out.

R 5.1-2 It is recommended to consider an early shutdown of the secondary side by use of appropriate automatic criteria in the case of leak accidents, especially in order to be able to use the water reservoirs of the emergency-cooling system more effectively.

R 5.1-3* As none of the existing accident analyses for loss-of-coolant accidents meets the requirements of the German regulations without reservations, it is recommended that in the case of a licensing procedure for the Stendal NPP the accident analyses must be newly performed with an advanced thermohydraulic code, following the assumptions of the RSK-Guideline 22.1 and the safety criteria of the BMI for the accident spectrum according to the accident guidelines. These analyses must be comprehensively documented. In this context, the analyses must be based on the finally determined set values of the safety system.

R 5.1-4* The emergency-cooling system must be designed in such a way that the requirements of the BMI safety criteria for the fulfilment of the safety functions are met even at the consideration of single failure and simultaneous repairing. As a substitutional measure, repair-time limits that are narrowly defined and justified may be provided (cf. R 6.4-12).

R 5.1-5* It is recommended to analyse WWER-typical accidents, like e.g. the rupture of the collector head in the steam generator.

R 5.1-6* It is recommended to install isolating valves in the BRU-A (cf. R 6.3-13)

R 5.1-7* It is recommended to provide constructive measures to exclude leaks and/or minimise leak cross-sections in the area of the collector in the steam generator.

R 5.1-8* It is recommended to develop on the basis of analyses appropriate accident procedures for controlling the whole spectrum of steam-generator-tube leaks and large leaks from the primary into the secondary system (e.g. the rupture of the collector head in the steam generator); these accident procedures must take the following points into account:

- acceptable time criterion for manual measures
- automatic reactor scram
- automatic shutdown of the secondary side
- automatic primary-side depressurisation at sufficient sub-cooling
- isolation of the defective steam generator; here, the pipes and valves must be designed to sustain possible two-phase flow
- additional borating of the primary system
- ensuring sufficient quantities of borated water for the primary system

R 5.1-9* It is recommended to analyse anew the entire spectrum of reactivity accidents under conservative boundary and initial conditions with verified computer codes and the nuclear data of the respective core loadings. For some cases, e.g. for rod ejection, appropriate 3D core-dynamics codes should be applied.

R 5.1-10* It is recommended to carry out further analyses of breaks of main-steam lines, using validated models for the mixing of coolant. In this context it must be ensured that the most unfavourable combination for the sub-cooling of the primary coolant is covered by systematic variation of location and size of the leak. Any analyses of breaks of main-steam lines from the initial hot zero power state that have not yet been performed must be carried out. It must be examined whether there is any re-criticality. For the analysis of the spectrum of breaks in main-steam lines, 3D core models are also to be applied.

R 5.1-11 It is recommended to carry out analyses to control the consequences of the accident category "break of a main-steam line in the area between the containment penetration and the isolating valve with simultaneous leaks or breaks in the steam-generator collector". For the verification of the basic safety of the collector it still has to be verified with analytical tools that the rupture of the collector head is controlled, if necessary under consideration of constructive measures for mitigating the consequences of a rupture of the head. An alternative would be presented by constructive measures for the exclusion of leaks in main-steam lines in the area between the containment penetration and the isolating valve.

R 5.1-12* It is recommended to analyse consequential breaks resulting from breaks in main-steam and feedwater lines in the closer vicinity of these lines near the containment penetrations. These analyses serve for the verification of accident control, unless the pipes are sufficiently protected from each other by dividing walls.

R 5.1-13* So far, there are no accident analyses available dealing with leaks and breaks in the feedwater system. It is recommended to carry out such analyses.

R 5.1-14 Due to the use of the new reactor-protection signal "Difference of the saturation temperatures between primary and secondary system high at low main steam pressure" instead of the old reactor-protection signal "Pressure decrease in the main steam collector high" it is recommended to check all analyses of breaks and leaks in the main steam system and, if necessary, to perform new analyses with up to date reactor-protection signals.

R 5.1-15 It is recommended to examine the stability behaviour of the reactor core by using the final core data.

R 5.1-16 It is recommended to analyse anew the entire spectrum of operating transients in accordance with the BMI List of Notes for a standard safety-analysis report, using the finally determined set values of the reactor-protection system or the protection system for controlling the safety system.

R 5.1-17* It is recommended to evaluate systematically the incidents that have occurred in other WWER-1000-type plants, with the aim of recalculating with an advanced accident code those cases that are well documented and suitable for code qualification.

R 5.1-18* It is recommended to analyse operating transients with presumed failure of reactor scram (ATWS) according to RSK-Guideline 20. The verification objectives here are the adherence to allowable stresses in the primary system, the ensurance of long-term residual-heat removal, and the safe shutdown of the reactor.

R 5.1-19* So long as it cannot be excluded on the basis of new analyses that the pressuriser safety valves are affected by two-phase mixture in the course of ATWS-accidents, the pipes concerned, the pressuriser safety valves and the relief tank are to be accordingly designed.

R 5.1-20* It is recommended to provide an efficient additional borating system for shutting down the reactor and ensuring long-term sub-criticality during ATWS-accidents. Its dimensions must be justified by analyses.

R 5.1-21* It is recommended to carry out analyses regarding accidents during shutdown states, start-up and shutdown procedures, and beyond-design-basis accidents.

R 5.2-1 Contrary to the design concept of the plant, long-term sub-pressure in the containment cannot be reached after a 2A-break of a main coolant line. It is therefore deemed necessary that there be further investigations concerning long-term accident management.

R 5.2-2 In order to determine the pressures that are to be expected in the case of secondary system breaks in the containment, it is recommended to carry out detailed analyses of the locking mechanisms and the control of the secondary system's isolating valves, of the expected break dimensions in the secondary-system pipes and within the steam generators, of the heat removal from the primary system via the remaining steam generators, etc.

R 5.2-3 It is recommended to carry out detailed analyses of the loads resulting from pressure differences during loss-of-coolant accidents in the containment and of the building's capability of absorbing such loads.

R 5.3-1* The group of "steam generator collector damage accidents with leaks between primary and secondary system" must be examined with regard to the radiological effects on the environment.

**Recommendations derived from Sections 2 and 3 of Chapter 6: Analysis of the safety systems**

R 6.2-1 As a basis for an evaluation, it is necessary that detailed documentation of the reactor-scram system is made available and that test results and operating experience are evaluated, with regard to the wide range of dropping times of the scram elements given in the technical project.

R 6.2-2 As a basis for an evaluation, it is necessary that detailed documentation of the ensured functioning of the HP emergency boron injection system, based on operating experience, is made available.

R 6.2-3* A concept for pressuriser spraying with the HP emergency boron injection pumps must be worked out and realised.

R 6.2-4* A concept for using the HP emergency boron injection system as an independent safety system (automatic actuation, extension of capacity to ATWS-accidents) must be developed and realised.

R 6.2-5 For better control of the "steam generator tube rupture" accident, the operational make-up system must be upgraded as a short-term measure (e.g. by automating the spray function necessary for this). In the long term, a solution according to R 6.2-3 should be found.

R 6.3-1 Evidence of a sufficiently large water reservoir in the sump must be given for all accident phases by a water balance for sump operation of the emergency-cooling system during loss-of-coolant accidents.

R 6.3-2 Evidence must be given under consideration of the definitions in KTA-Rule 3301 for the operating reliability and the efficiency of the sump cover (grids) and the filter devices at the outlets.

R 6.3-3* Basic safety must be proved for the pipes connecting to the emergency boron storage-tank as well as for the tank itself, so that a loss of water under accident conditions could be excluded. However, going further than the proof of basic safety of the connecting lines, it is recommended according to the state of the art to install double-walled pipes with leak detection between the tank and the isolating valve. The isolating valve should be located as close as possible to the emergency-boron storage-tank (cf. R 7.2-22).

R 6.3-4* For the residual heat removal chain and for residual heat removal from the spent fuel pool, an intermediate nuclear component-cooling system must be installed.

R 6.3-5* The locked injection valves of the HP- and LP-emergency-core-cooling systems must be interlocked in "open" position.

R 6.3-6 Monitoring of the leaktightness of the check valves in all injection lines of the emergency cooling system as well as their accessibility for function tests must be verified.

R 6.3-7 A systematic examination of the operating reliability of all pumps of the emergency cooling system and the containment spray system in other WWER-plants must be performed.

R 6.3-8 Experimental evidence is required of the effectiveness of the sprinkler nozzles for all accident conditions, including the design pressure of the containment.

R 6.3-9 A technical solution must be provided for periodic function tests of the containment spray system up to the last check valve during power operation of the unit; the test cycles for the sprinkler nozzles must be determined.

R 6.3-10* Physical separation of the 3 x 500 $m^3$ emergency feedwater tanks must be implemented if, in the case of a leak in an emergency feedwater tank, the functioning of the remaining system cannot be ensured.

R 6.3-11* Evidence must be given of the basic safety of the main-steam and feedwater lines in room A 820 of the surrounding outer building (height: 29.0 m) to exclude consecutive failures in case of a pipe rupture.

R 6.3-12* Room A 820 of the surrounding outer building, which houses the atmospheric steam-dump stations BRU-A and the steam-generator safety valves, must be designed to withstand external impacts and, if basic safety of the main-steam and feedwater lines is not ensured, must be divided into sections.

R 6.3-13* Remote-controlled isolation valves with emergency-power supply must be installed upstream of the BRU-A (cf. R 5.1-6).

R 6.3-14* Evidence must be given of sufficient water resources in the spray ponds of the nuclear service water system A during design basis accidents. If this is not possible, the additional water supply must be designed in accordance with the KTA-Rules for safety-related supply systems.

R 6.3-15 Cross-over points of pipes from the nuclear service-water system A of the three trains located outside must be made safe to withstand external impacts (only applies to multi-unit plants).

R 6.3-16 Evidence must be given of the resistance to external impacts throughout the nuclear service-water system A.

R 6.3-17* An emergency standby system must be backfitted.

R 6.3-18* A pressuriser relief valve that can be isolated must be installed which is also suitable for discharging two-phase mixtures and water.

R 6.3-19 The operating reliability of the pressuriser safety valves during the discharge of two-phase mixtures and water must be verified. In case these valves are newly installed, the principle of diversity is to be applied.

R 6.3-20* A leak-suction system is to be installed for all containment penetrations, for a controlled and filtered discharge of leakages.

R 6.3-21 Evidence must be given of the effectiveness and the operating reliability of the ventilation system.

R 6.3-22* If evidence cannot be given that the $H_2$-flammability limit is not exceeded during normal operation as well as during an accident, measures must be implemented to prevent the formation of flammable hydrogen concentrations. Independent of such measures, a monitoring system must be installed.

R 6.3-23 It must be verified that even during longer periods of recirculation by the HP emergency cooling pumps there is no need to cool the recirculated water, i.e. that the design temperature of the HP emergency cooling pumps is not exceeded.

The majority of the recommendations have been derived from the differences between the requirements demanded by the German body of rules and the Stendal NPP as it is described in the project. In this context there arise the two different areas of lack of verification, especially of the effectiveness of the safety systems, and demands for changes to the plant. The following table lists the recommendations with their references to the corresponding German rules.

**Recommendations of Sections 6.2 and 6.3 with their references to the corresponding German rules**

| Nr. of the Recommendation | Reference to the corresponding rules and guidelines | | |
|---|---|---|---|
| | RSK- Guidelines | Accident Guidelines | KTA-Rules |
| R 6.2-1 | 3.1.2, 20 (1) | | 3501 |
| R 6.2-2 | | | |
| R 6.2-3 | | Table 1.2 | |
| R 6.2-4 | 20 | | |
| R 6.2-5 | | | |
| R 6.3-1 | 22.1.2 (6, 14), 22.1.3 (3) | | 3301/Item 4.4.1 |
| R 6.3-2 | | | 3301/Item 6.2.2.2 |
| R 6.3-3 | 4.2, 22.1.2 (7) | | 3301/Item 5.2.2.2, 6.2.2.3 |
| R 6.3-4 | 22.1.2 (5, 6) | | 3301/Item 5.4.2 |
| R 6.3-5 | | | 3301/Item 7.1.3 (2) |
| R 6.3-6 | 21.1 (4), 22.1.2 (13), 5.6 (1) | | 3301/Item 7.1.2, 7.2.2 |
| R 6.3-7 | | | |
| R 6.3-8 | | | |
| R 6.3-9 | 22.1.2 (13) | | |
| R 6.3-10 | 19.4 | | 3301/Item 5.2.2.2, 5.2.2.4, 6.3 |
| R 6.3-11 | 5.2 (5) | | 3301/Item 5.2.4, 5.3 |
| R 6.3-12 | | | 3301/Item 3.3 |
| R 6.3-13 | | | |
| R 6.3-14 | 22.1.2 (14) | | 3301/Item 4.4.3 (by analogy) 6.4.1, 6.4.4 |
| R 6.3-15 | 19.4, 22.1.2 (1) | | 3301/Item 6.4.2 (d), 6.3 |
| R 6.3-16 | | | 3301/Item 3.3 |
| R 6.3-17 | 22.2 | | 3904, 3301/Item 4.4.1 (1), 6.3 |
| R 6.3-18 | 3.1.4 | | |
| R 6.3-19 | | | 3301/Item 4.3.4 |
| R 6.3-20 | 5.6 (9) | | |
| R 6.3-21 | | | 3601/Item 3.5 |
| R 6.3-22 | 24 | | |
| R 6.3-23 | | | |

**Recommendations derived from sections 4 and 5 of Chapter 6: Analysis of the safety system - instrumentation and control and electric-power supply**

R 6.4-1 As the unit cannot be operated in power-operation mode without the control computer in the long term, it must be thoroughly examined if and for how long power operation is admissible.

R 6.4-2* The solution concerning the decoupling and preferential switching between the main control room and the standby control room must be analysed in detail and evaluated as to its admissibility.

R 6.4-3 The transmission of information to the standby control room upon entry of operating personnel must be examined as to its correctness, by use of further documentation.

R 6.4-4* The cycles of acquisition of the analogue and binary signals from the control computer are too slow. The computers in use do not conform to international standards. There are no statements available on the reliability of either the hardware or the software. It is therefore recommended to install modern computer technology from the start, should construction of the power plant be resumed.

R 6.4-5* The concept of core instrumentation as it was introduced in the technical project of 1981 should be thoroughly revised. In this context it should be extended by a power-limitation system as well as a reliable calibration system (cf. R4.1-10).

R 6.4-6 The reliability of the instrumentation and control system is inadequate. This concerns the actuations, the position indicators and the limit-position switches of all isolating and control valves.

R 6.4-7 The gauges for pressure and differential pressure should be qualified.

R 6.4-8* Following negative operating experience in other WWER-1000 units, the I&C concept for the control of dynamic transition processes should be revised.

R 6.4-9 There is no diversity in the equipment in the two trains of the emergency-protection system for reactor scram. No evidence is available that this is compensated by special technical and/or organisational measures. Such evidence should be given.

R 6.4-10* Except in the neutron flux measurement system, there appears to be no self-monitoring available in the emergency protection system. Self-monitoring should be backfitted.

R 6.4-11* It is possible that the limit values of the neutron flux measurement system as well as those of the gauges of the actuation criteria related to process-engineering may readjust themselves without being noticed. It is recommended to eliminate this deficiency by technical measures.

R 6.4-12 A case where there is maintenance work going on in one train of the emergency-protection system and a failure occurs simultaneously which renders the entire second train ineffective (e.g. through external or internal impacts) cannot be controlled. It must be examined if and for how long one train may be taken out of operation for maintenance purposes (cf. R 5.1-4)

R 6.4-13 A control-element-insertion limitation must be backfitted for ensuring shutdown reactivity.

R 6.4-14 As a conclusion from the operating experience in other, operational WWER-1000 units (cf. Chapter 8), it is recommended to revise the reporting and inspection concept of the emergency protection system.

R 6.4-15* It is recommended to backfit complete self-monitoring of the protection system.

R 6.4-16* An unnoticed readjustment of the limit values in the gauges of the protection system is possible. It is recommended to eliminate this deficiency by technical measures (see also R 6.4-11).

R 6.4-17 There is no evidence that manual protective measures for accident control do not become necessary before 30 minutes have elapsed. For such manual protective measures, safety-hazard reportings according to KTA 3501 should be backfitted.

R 6.4-18 It is recommended to provide evidence that the protection system does not initiate any safety-significant transients during power cuts.

R 6.4-19* It is recommended to provide evidence of type inspections conforming to international standards for all pieces of equipment used. Wherever this is not possible, the technical equipment should be replaced.

R 6.4-20* It is recommended to provide evidence that the requirements of KTA-Rule 3502 concerning accident instrumentation are met by the existing equipment. Backfitting must be carried out where no such evidence exists.

R 6.5-1 In the first construction phase the grid connection is only carried out via a 220-kV switchyard which also feeds the 110-kV switchyard. In case of a defect in the 220-kV switchyard all other grid connections may possibly fail. It is therefore recommended to build a second switchyard, e.g. a 380-kV switchyard, in order to provide a redundancy.

R 6.5-2* It is recommended to backfit an emergency grid connection, which so far is not available, by way of an underground cable.

R 6.5-3* It can be derived from operating experience in other operational units of the same type, that the quality assurance particularly of the cables and switches is poor. Cables and switches should be replaced by approved ones (cf. R 8.3-41).

R 6.5-4* In the auxiliary power system, sufficient selectivity to prevent short circuits and protection against consequential spreading impacts between the individual 0.4-kV and 6-kV busbars must be backfitted (cf. R 6.5-8).

R 6.5-5 Since there is no below-frequency actuation of the diesel generator, it should be backfitted.

R 6.5-6 It is not possible to switch the electricity supply of the safety system from emergency power back to normal power supply as long as there are still any process-based actuation criteria in effect. Therefore a synchronising device for each diesel generator should be backfitted to make a switch back possible.

R 6.5-7* Evidence should be provided that the discharge time of the batteries of the emergency power system is kept > 2 h.

R 6.5-8 It can be derived from operating experience in other operational units of the same type (cf. Chapter 8), that the cable and switch concept must be revised in connection with the ensurance of selectivity in the case of short circuits.

R 6.5-9* The components used in the emergency-power systems must be of approved types.

R 6.5-10 As it can be assumed that, as a result of upgrading of the safety system, the number of the consumers to be supplied with emergency power will increase, more powerful emergency diesels should be used.

**Recommendations derived from Chapter 7: Civil-engineering aspects, internal and external impacts, radiation protection**

R 7.1-1* Evidence must be provided that the necessary characteristics, according to the RSK-Guidelines for Pressurised Water Reactors for admission of a 0.1A-leak assumption, exist in the calculation of the jet and reaction forces. Furthermore, it must be verified that the pressure differences and the jet and reaction forces in the containment can be absorbed.

R 7.1-2 The steel-cellular composite design does not correspond to the state of the art in Germany. This type of construction would therefore require a special license from the Institut für Bautechnik (Civil Engineering Institute) in Berlin or from the planning department and building control office of the state government responsible in any individual case.

R 7.1-3 The anchoring of the racking components (anchor studs) for the absorption of forces from the component supports is to be examined. In particular, the welds of the round steel horizontal to the anchor stud in the direction of the thickness is also to be analysed in detail.

R 7.1-4 A final evaluation of the constructional design of the reactor building, in the framework of construction-supervision procedures, requires a complete examination of the design and the calculations.

R 7.1-5* For the determination of the external loads resulting from the load cases earthquake, airplane crash and external blast waves it is recommended to determine the corresponding response spectra.

R 7.1-6 An underpressure test must be carried out at 15 kPa (maximum underpressure multiplied by a factor of 1.5).

R 7.1-7* Evidence must be provided that a single-shell containment is also able to provide the necessary protection against an inadmissible release of radioactive substances in accordance with the requirements of the German body of rules.

R 7.2-1 In the framework of the construction on the power-plant site of facilities with potential for large fires, like petrol stations and gas-storage tanks, it must be ensured that inadmissible fire impacts on safety-relevant buildings and facilities can be excluded.

R 7.2-2 In the framework of additional tests, a concept has to be presented for recurring tests of fire-protection facilities.

R 7.2-3 The accident combination "external impacts with consequential fire" must be systematically investigated in the framework of further analyses.

R 7.2-4 Some individual issues must still be clarified for the classification of steel-cellular composite-design structures in a fire-resistance scale.

R 7.2-5 In the framework of a fire-hazard analysis, a final assessment and identification has to be carried out of the areas where a consistent physical separation of the redundant trains of the safety system has not been applied. Additional fire-protection measures have to be carried out if necessary.

R 7.2-6 It must be ensured that no fire-protection measures except those approved by the authorities for construction supervision, like e.g. fire doors, cable compartments and fire-protection flaps, are installed.

R 7.2-7 The concept concerning the use of fire-protection flaps in the ventilation ducts is not clearly recognisable. Ventilation ducts that run through several fire-resistant areas must be provided with fire-protection flaps in the penetration areas of the necessary fire-resistant partitions.

R 7.2-8* The emergency control room should be decoupled from the main control room for reasons of fire protection.

R 7.2-9 For the oil supply of the main coolant pumps, a fire-hazard analysis must be performed. Additional fire-protection measures have to be carried out if necessary.

R 7.2-10* Cables of redundant systems that do not belong to the safety system must be physically separated for reasons of fire protection.

R 7.2-11 Automatic fire detectors have to be installed in all rooms with safety-related equipment.

R 7.2-12 It is necessary to employ qualified type-inspected fire detectors for the respective types of combustible material. When the fire detectors are installed, the room dimensions, the type of the combustible material and the ventilation conditions must be taken into account,

R 7.2-13 It must be checked whether sufficient pumping capacity and water reserves are ensured for all fire-fighting zones, also allowing for manual fire-fighting measures.

R 7.2-14 It must be determined by analysis if simultanous failure of several fire-extinguishing systems within the valve compartments of the spray-water fire-fighting system is possible. Backfitting measures may be necessary.

R 7.2-15 As regards the water supply for equipment inside the containment it has to be checked whether the containment isolation valves can be reopened after an erroneous actuation by the emergency cooling signal. The possibility of re-setting the valves is deemed to be necessary.

R 7.2-16 It has to be verified that during fire-fighting activities it is not possible that several redundancies of safety-relevant systems or equipment are adversely affected by the water.

R 7.2-17 In the framework of further analyses, a concept must be presented of the organisation and size of the plant's fire brigade as well as of the administrative regulations in the case of a fire.

R 7.2-18 In the framework of further analyses, a detailed examination of conventional fire-protection requirements, like e.g. the provision of safe escape routes, must be carried out.

R 7.2-19 The walls between the different chambers of the reactor building below the 13.2-m ceiling, the doors to these chambers, and the penetrations in the walls must be verified to be able to withstand jet forces and water loads.

R 7.2-20 The drains existing in the chambers are to be equipped with appropriate isolating devices. The isolating devices between the drain systems of redundant systems must be safely locked in the closed position during normal operation.

R 7.2-21* Accident-proof and reliable leak detectors must be installed in the reactor building.

R 7.2-22* The three sump drains are to be installed as double-walled pipes with leak detection. Motor-driven isolating valves must be installed as close to the sump as possible at the end of the double-walled pipes (cf. R 6.3-3).

R 7.2-23* The emergency control room must be particularly protected against possible flooding, e.g. through failing pipes or erroneous actuation of the fire-protection system, by the installation of raised thresholds, tight-fitting doors, etc.

R 7.2-24 The outlet pipes of the spent-fuel pools must consist of double-walled pipes and double valves. It must be possible to prevent the draining of the pools through siphon effect in the pipes connecting from above.

R 7.2-25 Corresponding to the effects of dropping loads, the polar crane 320t/160t/2x70t in the containment and the 10-t electric hoist on the gantry crane situated on the supports of the polar bridge crane have to meet the requirements of Section 4.3 (increased demands) of KTA-Rule 3902. It is considered necessary that the cranes be upgraded in order to comply with KTA-Rule 3902. The corresponding evidence will have to be presented.

R 7.2-26 The cranes in the turbine building, the wing housing the feedwater tank, and the surrounding outer construction also have to comply with the additional requirements of KTA-Rule 3902, Section 4.5, unless it is possible to avoid completely any transport processes during power operation of the plant or to limit the possible consequences of a load drop by hardware measures and restrictions of the crane's use to such a degree that the dangers according to KTA-Rule 3902, Section 4.2, need not be applied. The demand that the additional requirements of KTA-Rule 3902 be fulfilled make an upgrading of the cranes necessary. The correspon-ding evidence will have to be presented.

R 7.2-27 Corresponding to the effects of a load drop, the fuel-element-handling machine has to meet the requirements of KTA-Rule 3902, Section 4.4. It is considered necessary to adapt the fuel element handling machine accordingly, unless this has already been carried out. The corresponding evidence will have to be presented.

R 7.3-1 The rooms of the exclusion area that are designated as maintained or half-maintained are to be marked with radiation signs and "Control Area" labels; the rooms that have not been maintained must be marked with radiation signs and "Prohibited Area - No Entry" labels.

R 7.3-2 Measures must be provided that exclude or minimise the necessity of persons entering the containment rooms that have not been maintained during operation.

R 7.3-3 The design and the equipment of the hygiene wing should be revised, to be basically suited for staff numbers of 300 employees from the plant and 900 workers from outside.

R 7.3-4 The thickness of the walls in the rooms of the controlled area must be examined as to whether they comply with the demands of § 54 StrlSchV and KTA-Rule 1301.1; if necessary, measures must be determined to upgrade the shielding or to limit access periods.

R 7.3-5 The deviations from the project state that arose during the construction of the reactor's shielding have to be analysed as to the expected chan-ges in the field of radiation.

R 7.3-6* The measuring systems for the radiological monitoring of the technical system and dose rates must be modified according to the state of the art.

R 7.3-7 It has to be checked whether measures are required for special maintenance personnel to keep them within their age-related dose of 400 mSv.

R 7.3-8 The overall concept of the primary system has to be revised with a view to minimising the occurrence of leaks.

R 7.3-9 The extent of work required during the maintenance and power-operation modes as well as the resulting exposures to radiation have to be analysed. Measures for a further reduction of radiation exposure are to be derived from this analysis.

R 7.3-10 Evidence for preventive radiation-protection measures according to the IWRS-Guideline has to be presented.

R 7.3-11* For the performance of maintenance work, the latest equipment in modern inspection technology is to be used. Any work that has to be carried out under intense radiation is to be automated to the largest possible degree.

R 7.3-12 Storage space and temporary stores, as well as moving space for maintenance measures, are to be created by locally changing the arrangement of components and pipe routes.

R 7.3-13 Modern breathing apparatus is to be provided for maintenance work with potential inhalation dangers.

**Recommendations derived from Chapter 8: Evaluation of operating experience from other WWER-1000 plants**

The deficiencies identified during the analysis of the individual results lead to the demand for backfitting measures in the following areas:

A. Mechanical systems

B. Instrumentation and control

C. Auxiliary-power supply

D. Building structures

E. Plant organisation, operating instructions, quality assurance

The demands based on events that occurred in plants of the "small series" (kleine Serie) are marked by the letters KS.

**A. Mechanical systems**

R 8.3-4 The design of the absorber-rod drives is to be checked as to whether its drive shaft is principally a weak point (Section 8.3.1).

R 8.3-13 The HP-emergency cooling pumps must be upgraded (e.g. improvement of the surface coating of the axial-thrust compensation) to reduce friction (Section 8.3.3, KS).

R 8.3-14 For the HP-emergency cooling pumps, a reliable pump protection must be established as regards temperature and suction pressure (Section 8.3.3).

R 8.3-15 Temperature and operating-time limits for minimum flow operation must be determined for the HP-emergency cooling pumps; the installation of additional heat exchangers may possibly be required for cooling during minimum flow operation (Section 8.3.3, KS).

R 8.3-16 Before the emergency cooling system is taken into operation, sufficient purging has to be carried out. Pollution sources must be identified and, if necessary, eliminated (Section 8.3.3).

R 8.3-18 The mechanical drives of the turbine-control valves of the turbo-feedwater pumps must be upgraded (Section 8.3.4).

R 8.3-21 The emergency-cooling system has to be upgraded in such a way that injection can take place without active opening of the isolation valves (Section 8.3.4).

R 8.3-24 Lubrication of the bearings in the oil pumps (bearing temperature monitoring) of the emergency diesels has to be improved (Section 8.3.5, KS).

R 8.3-33 The steam-dump station (BRU-A) including the limit-position switches must be upgraded (Section 8.3.6, KS).

R 8.3-34 The vibrations during pressure relief via the BRU-A are to be reduced by constructive measures (Section 8.3.6, KS).

R 8.3-35 It has to be checked whether the use of limit-position switches without contacts is technically useful in the case of the steam-dump station (Section 8.3.6, KS).

R 8.3-50 Through sufficient dimensioning of the feedwater lines and through a control system that corresponds to the safety requirements it has to be ensured that no asymmetrical conditions can occur during steam-generator feeding which might lead to reactor scram (Section 8.3.9.1)

R 8.3-57 An isolating device has to prevent an uncontrolled evaporation from the secondary system into the auxiliary steam network. Process-based measures are also required, like e.g. decoupling via check valves or control valves which can prevent maloperations during equalisation of pressure (Section 8.3.9.4).

**B. Instrumentation and Control**

R 8.3-1 The actuation logic for the failure of a pneumatic oil-isolation valve in the oil circuit of the main coolant pump is to be changed in such a way that only one main coolant pump is switched off in this case (Section 8.3.1, KS).

R 8.3-5 The actuation level for the absorber-rod drives is to be checked as to its logic as well as to its switching circuit (Section 8.3.1).

R 8.3-6 The two water-level measurements of the pressurisers are to be upgraded so that they both indicate the same correct water level during all operating conditions, even during major transients (Section 8.3.1 (KS) and Section 8.3.9.5).

R 8.3-7 A signalling system has to be installed which, in the case of reactor scram shows the operator in any case the actuation criteria that have triggered off the scram. Signalling interruptions must be as far as possible self-reporting. Regular checks of the relay contacts and the links between contacts is not sufficient. It has to be determined to what extent these requirements can be met with the existing relays on the actuation level of the emergency-cooling system. Any faults must be as far as possible self-reporting (Section 8.3.1, KS).

R 8.3-8* After reactor scram, turbine trip must be automatically actuated. It must be checked whether automatic actuation can also be introduced for trip of the turbo-feedwater pumps in order to prevent sub-cooling transients. This seems particularly necessary for the protection of the steam generators (Section 8.3.1 (KS), Section 8.3.4 and Section 8.3.9.5).

R 8.3-10 Signalling by the measured-value transmitters of the power supply has to be improved and locks have to be installed in order to be able to prevent as far as possible any inadvertent switchings within the auxiliary-power network (Section 8.3.2).

R 8.3-11 In order to prevent transients occurring due to wrong signals, measured-value and limit-value processing are in principle to be designed completely redundant and, if possible, in a diverse manner up to the actuation level, to avoid erroneous actuations of the containment isolation signals of components with relevant availability (oil and feedwater supplies and pump trains of the operational make-up system) (Section 8.3.2).

R 8.3-12* The available documentation does not show clearly how the power supply for the sensors of the safety system is designed in detail. No further specific demands can therefore be derived from this area. However, an assessment of the design of the measurement points' power supply appears to be necessary following past operating experience (Section 8.3.2).

R 8.3-17* The actual oil level and the difference to the minimum oil level of all safety-relevant pumps must in principle easily and precisely determined. This has to be checked, and backfittings have to be carried out where necessary (Section 8.3.4)

R 8.3-19* The logic of the locks in the feedwater control system has to be improved in order to ensure reliable and effective operation of the pumps (Section 8.3.4).

R 8.3-20 The automatic standby-activation (automatische Reserve-Einschaltung, ARE) for the oil pumps of the main recirculation pumps is to be improved (ARE apparently only responds after the simultaneous failure of all three oil pumps) (Section 8.3.4).

R 8.3-25 The steam-generator water-level measurement has to be improved by installing more reliable technology (Section 8.3.5, KS).

R 8.3.-26 The failures of important safety-relevant measurements have to be self-reporting (Section 8.3.5., KS).

R 8.3-27 The unlocking of actuation criteria when the plant has not been shut down must be prevented through technical measures (Section 8.3.5, KS).

R 8.3-32 The priority-control system between the main control room and the emergency control room has to be redesigned and upgraded (Section 8.3.7, KS).

R 8.3-36 Due to suspicion of common mode defect, the cause of the drifting of the nominal value of the backup control of the diesel generator should be eliminated (Section 8.3.7).

R 8.3-37 Monitoring of the recharging voltage of the batteries and the constant upkeep of their charging current must be improved through recurring tests (Section 8.3.7).

R 8.3-38 A warning system for signalling low temperatures in the diesel's starter air has to be installed; it has to be protected against interference by the operating personnel (Section 8.3.7).

R 8.3-43 Failures of the room-temperature control in rooms with safety-related systems must be self-reporting (Section 8.3.8.4).

R 8.3-46 An examination of the entire measurement and control system including the emergency cooling system should be carried out with regard to design flaws in the power supply (separation into different supply busbars) (Section 8.3.9.1).

R 8.3-47 The energy supply of the actuation level of the automatic locking of steam-generator feeding is to be changed so that the automatic locking mechanism (2 out of 3) is not activated by one supply busbar due to e.g. an insufficient signal-noise ratio (Section 8.3.9.1).

R 8.3-48 Earthing and configuration of the logic level's energy supply must be designed in such a way that there is a sufficient interference-voltage distance (Section 8.3.9.1)

R 8.3-49 On the basis of evaluated operating experience, the effects of transients initiated by malfunction or malactuation of the feedwater system must be absorbed by the provision of more effective control and limitation (section 8.3.9.1).

R 8.3-51 Measures have to be taken to prevent pick-up from the 220-V supply in the 24-V and 48-V logic-switching circuits (Section 8.3.9.1, KS).

R 8.3-52 All instrument channels have to be functionally separated, from the intake of the medium to the actuation signal, in order to exclude erroneous actuation by a single fault (Section 8.3.9.2).

R 8.3-53 The alarm system should be upgraded in such a way that any operational deviations from individual process parameters like, e.g. RPV water level or pressure and temperature in the loops, can be regulated without the safety systems being actuated. If this is not possible with the existing technology, reactor scram must automatically be triggered after turbine trip (Section 8.3.9.3.).

R 8.3-54 The contact surfaces of the relays must be made of material with sufficiently assured quality (Section 8.3.9.1).

R 8.3-55* On the basis of evaluated operating experience and of examinations that have been carried out, a full analysis of the deficiencies of the entire instrumentation and control system must be carried out in the plant. It then has to be decided whether the existing technology can be upgraded or if the instrumentation and control equipment should be replaced to a large extent (Section 8.3.9.3)

R 8.3-56 The interaction of the individual power controllers and the options for manual intervention in power control by the operators must be checked (Section 8.3.9.4).

R 8.3.-58 The actuation logic of the protective interlock for the isolating valve of the pressuriser injection line must be extended, so that the injection valve can be operated independent of the reset position (Section 8.3.9.4).

R 8.3-59 The actuation logic for the formation of the signal "Difference between primary system temperature and saturation temperature below 10 K" must be improved so that the measurement error is clearly lower than the admissible deviation range of the measured value (Section 8.3.9.4).

R 8.3-60 The entire reactor protection logic must be revised where only reset positions are used as actuation criteria and whether measurement errors lie within the range of the distances from the normal parameters to the activation limit values (Section 8.3.9.4).

R 8.3-61 Adequate filters must be provided in the oil circuits of the turbine control system in order to avoid pollution (Section 8.3.9.5).

R 8.3-62 The turbine control system has to be upgraded to such an extent that any cases of extreme loads are excluded. This can, for example, be achieved by installing two electro-hydraulic converters with fast synchronisation control and consecutive MIN-selection (Section 8.3.9.5).

R 8.3-63 A plant-state-signalisation system has to be introduced to help recognise more easily the failure of position indicators on valves (Section 8.3.9.6).

R 8.3-64 The filling level in the oil tanks of the main coolant pumps' oil circuits must be monitored and be equipped with warning devices (Section 8.3.9.6).

R 8.3-65 Steam generator water level measurement and limit-value adjustment must be improved through technical measures. In particular it must be ensured that gauges measuring in the same measurement units are synchronised (Section 8.3.9.6).

R 8.3-66 The testing possibilities must principally be determined or automated in such a way that there will be no undesired transients (Section 8.3.9.6).

R 8.3-67* Provisions are to be made such that switching of the steam generator water-level control to start-up control is avoided during power operation (Section 8.3.9.6).

R 8.3-68* Adjustments of zero point and limit values must be monitored either by inspections or through self-reporting (Section 8.3.9.6).

R 8.3-69 Steam generator water level measurement must operate reliably, including in the case of rapid changes in main steam pressure (Section 8.3.9.6, KS).

**C. Auxiliary-power supply**

R 8.3-2 The insulations of all control cables used in safety-relevant systems for systems control or power supply have to be checked (Section 8.3.1 and Section 8.3.5, KS).

R 8.3-3 The effects of the backfitted fire-protection measures on the operational safety of the cables should be examined. Temperatures must be checked on all power-supply cables of which large surfaces were treated with fire-resistant coatings (Section 8.3.1, KS).

R 8.3-9 The power supply for the sensors of one actuation level must be divided (Section 8.3.2).

R 8.3-23 The switch gear must be short-circuit-proof (sufficient selectivity) (Section 8.3.4).

R 8.3-29* The wiring of the safety-relevant valves and pumps must be carried out correctly and has to be checked (Section 8.3.6, KS).

R 8.3-31 The auxiliary-power system must be single-failure-proof (Section 8.3.6, KS).

R 8.3-39 The power supply of the three channels of the diesel generator protection system must be divided between different, physically separated busbars (Section 8.3.8.1).

R 8.3-40 The power-supply breakers located in cabinets must be protected against inadvertent operation (Section 8.3.8.1).

R 8.3-41 The 6-kV switches must be replaced with appropriate switches (cf. R 6.5-3) (Section 8.3.8.2).

R 8.3-44 The design of the voltage supply of the power- and frequency- measurement system of the main coolant pump monitoring system is to be changed by distribution over several supply busbars (Section 8.3.9.1).

R 8.3-45 The reliability of the supply busbars including the cables, connections and contacts is to be improved (Section 8.3.9.1).

**D. Building structures**

R 8.3-22 The physical separation of the auxiliary-power-supply busbars needs to be backfitted (Section 8.3.4).

R 8.3-28 The roof of the turbine hall must be sealed (Section 8.3.5, KS).

R 8.3-42 The penetration of humidity and water into the switch-gear rooms must be prevented through constructional measures and an appropriate design of the ventilation system (Section 8.3.8.4).

**E. Plant organisation, operating instructions, quality assurance**

R 8.3-30 Quality assurance (inspection on receipt etc.) must be extended to such a degree that faults in the functioning of valves are detected before they are installed (Section 8.3.6, KS).

R 8.3-70 Evidence of detailed quality assurance is to be required from the manufacturers of all components that are used in safety-relevant plant areas; in addition, separate comprehensive inspections must be carried out (Section 8.3.10).

## Appendices

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| Appendix 1 | List of applied German rules and guidelines |
| Appendix 2 | List of recommendations following the standard structure of TÜV/GRS assessments of nuclear power plants with pressurised or boiling water reactors |

# Appendix 1

## List of applied German rules and guidelines

- **General Assessment Criteria**

- Allgemeine Verwaltungsvorschrift zu § 45 Strahlenschutzverordnung: Ermittlung der Strahlenexposition durch die Ableitung radioaktiver Stoffe aus kerntechnischen Anlagen oder Einrichtungen vom 21. Februar 1990
  Bundesanzeiger Nr. 64 a vom 31. März 1990

- Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken mit Druckwasserreaktoren gegen Störfälle im Sinne des § 28 Abs. 3 StrlSchV,
  \- Störfall-Leitlinien -
  Bekanntmachung des Bundesministers des Innern vom 18. Oktober 1983,
  Bundesanzeiger Nr. 245 vom 31. Dezember 1983

- Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor
  Bekanntmachung des Bundesministers des Innern vom 26. Juli 1976
  Gemeinsames Ministerialblatt Nr. 26 vom 30. August 1976

- Interpretationen zu den Sicherheitskriterien für Kernkraftwerke vom 17. Mai 1979, vom 28. November 1979 und vom 2. März 1984
  Bekanntmachung des Bundesministers des Innern vom 17. Mai 1979 (Gemeinsames Ministerialblatt 1979, S. 161), vom 28. November 1979 (Gemeinsames Ministerialblatt 1980, S. 90) und vom 10. Mai 1984 (Gemeinsames Ministerialblatt 1984, S. 208)

- Sicherheitskriterien für Kernkraftwerke, verabschiedet im Länderausschuß für Atomkernenergie am 22. März und am 12. Oktober 1977
  Bekanntmachung des Bundesministers des Innern vom 21. Oktober 1977,
  Bundesanzeiger Nr. 206 vom 3. November 1977

- Zusammenstellung der in atomrechtlichen Genehmigungs- und Aufsichtsverfahren für Kernkraftwerke zur Prüfung erforderlichen Informationen (ZPI), verabschiedet im Länderausschuß für Atomkernenergie am 7. September 1982
  Bekanntmachung des Bundesministers des Innern vom 20. Oktober 1982,
  Bundesanzeiger Nr. 6 a vom 11. Januar 1983

- Reaktor-Sicherheitskommission
  RSK-Leitlinien für Druckwasserreaktoren
  3. Ausgabe vom 14. Oktober 1981, Bundesanzeiger Nr. 69 vom 14. April 1982, mit Berücksichtigung der Änderungen gemäß Bundesanzeiger Nr. 106 vom 10. Juni 1983 und Bundesanzeiger Nr. 104 vom 5. Juni 1984

- Reaktor-Sicherheitskommission, Strahlenschutzkommission
  Störfallberechnungsgrundlagen für die Leitlinien des BMI zur Beurteilung der Auslegung von Kernkraftwerken mit DWR gemäß § 28 Abs. 3 StrlSchV
  Gemeinsame Empfehlung, Bekanntmachung des Bundesministers des Inneren vom 18. Oktober 1983, Bundesanzeiger Nr. 245 vom 31. Dezember 1983

- Reaktor-Sicherheitskommission
  Rahmenspezifikation Basissicherheit
  Stand: 25. April 1979, 2. Anhang zum Abschnitt 4.2 der RSK-Leitlinien für Druckwasserreaktoren, Bundesanzeiger Nr. 167 vom 6. September 1979

- Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV)
  Neufassung vom 30. Juni 1989, Bundesgesetzblatt I, Nr. 34, vom 12. Juli 1989, mit Berücksichtigung der Berichtigungen und Änderungen bis zur zweiten Änderung gemäß Bundesgesetzblatt II, Nr. 35, vom 28. September 1990

- **Guidelines**

- Richtlinie für den Schutz von Kernkraftwerken gegen Druckwellen aus chemischen Reaktionen durch Auslegung der Kernkraftwerke hinsichtlich ihrer Festigkeit und induzierter Schwingungen sowie durch Sicherheitsabstände (Stand: August 1976)
  Bekanntmachung des Bundesministers des Innern vom 13. September 1976, Bundesanzeiger Nr. 179 vom 22. September 1976

- Institut für Bautechnik, Berlin
  Ergänzende Bestimmungen zu den "Richtlinien für die Bemessung von Stahlbetonteilen von Kernkraftwerken für außergewöhnliche äußere Belastungen - Fassung Juli 1974 - "
  Fassung November 1975

- Institut für Bautechnik, Berlin
  Richtlinien für die Bemessung von Stahlbetonteilen von Kernkraftwerken für außergewöhnliche äußere Belastungen (Erdbeben, äußere Explosionen, Flugzeugabsturz)
  Fassung Juli 1974

- Richtlinie für den Strahlenschutz des Personals bei der Durchführung von Instandhaltungsarbeiten in Kernkraftwerken mit Leichtwasserreaktor: Die während der Planung der Anlage zu treffende Vorsorge, (IWRS-Richtlinie), verabschiedet im Länderausschuß für Atomkernenergie am 10. Mai 1978
  Rundschreiben des Bundesministers des Innern vom 10. Juli 1978, Gemeinsames Ministerialblatt Nr. 28 vom 31. August 1978

- VGB Technische Vereinigung der Großkraftwerksbetreiber e. V.
  VGB-Richtlinie für das Wasser in Kernkraftwerken mit Leichtwasserreaktoren,
  - VGB-R401J -
  Zweite Ausgabe, 1988

- **KTA-Rules**

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|---|---|
| KTA 1301.1 | Berücksichtigung des Strahlenschutzes der Arbeitskräfte bei Auslegung und Betrieb von Kernkraftwerken; Teil 1: Auslegung<br>Fassung 11/84 |
| KTA 1501 | Ortsfestes System zur Überwachung von Ortsdosisleistungen innerhalb von Kernkraftwerken<br>Fassung 6/91 |
| KTA 1501.1 | Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken; Teil 1: Kernkraftwerke mit Leichtwasserreaktor<br>Fassung 6/86 |
| KTA 1502.1 | Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken mit Leichtwasserreaktoren<br>Fassung 6/86 mit Berücksichtigung der Korrektur vom 6. Oktober 1986 |
| KTA 1503.1 | Messung und Überwachung der Ableitung gasförmiger und aerosolgebundener radioaktiver Stoffe; Teil 1: Messung und Überwachung der Ableitung radioaktiver Stoffe mit der Kaminabluft bei bestimmungsgemäßem Betrieb<br>Fassung 2/79 |
| KTA 1504 | Messung flüssiger radioaktiver Stoffe zur Überwachung der radioaktiven Ableitungen<br>Fassung 6/78 |
| KTA 1506 | Messung der Ortsdosisleistung in Sperrbereichen von Kernkraftwerken<br>Fassung 6/86 mit Berücksichtigung der Korrektur vom 25. November 1986 |
| KTA 2101.1 | Brandschutz in Kernkraftwerken; Teil 1: Grundsätze des Brandschutzes<br>Fassung 12/85 |
| KTA 2101.2 | Brandschutz in Kernkraftwerken; Teil 2: Brandschutz an baulichen Anlagen; Regelentwurfsvorlage<br>Fassung 6/91 |
| KTA 2101.3 | Brandschutz in Kernkraftwerken; Teil 3: Brandschutz an maschinen- und elektrotechnischen Anlagen; Regelentwurfsvorlage<br>Fassung 11/90 |
| KTA 2102 | Fluchtwege in Kernkraftwerken; Regelentwurf<br>Fassung 6.90 |
| KTA 2201.1 | Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 1: Grundsätze<br>Fassung 6/90 |

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| KTA 2201.3 | Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 3: Auslegung der baulichen Anlagen; Regelvorlage<br>Fassung 6/91 |
| KTA 3101.1 | Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 1: Grundsätze der thermohydraulischen Auslegung<br>Fassung 2/80 |
| KTA 3101.2 | Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 2: Neutronenphysikalische Anforderungen an Auslegung und Betrieb des Reaktorkerns und der angrenzenden Systeme<br>Fassung 12/87 |
| KTA 3103 | Abschaltsysteme von Leichtwasserreaktoren<br>Fassung 3/84 |
| KTA 3201.1 | Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 1: Werkstoffe<br>Fassung 6/90 |
| KTA 3201.2 | Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 2: Auslegung, Konstruktion und Berechnung<br>Fassung 3/84 |
| KTA 3201.3 | Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 3: Herstellung<br>Fassung 12/87 |
| KTA 3201.4 | Komponenten des Primärkreises von Leichtwasserreaktoren; Teil 4: Wiederkehrende Prüfungen und Betriebsüberwachung<br>Fassung 6/90 |
| KTA 3203 | Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren<br>Fassung 3/84 |
| KTA 3204 | Reaktordruckbehälter-Einbauten<br>Fassung 3/84 |
| KTA 3211.1 | Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 1: Werkstoffe<br>Fassung 6/91 |
| KTA 3211.2 | Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 2: Auslegung, Konstruktion und Berechnung; Regelvorlage<br>Fassung 3/91 |

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| KTA 3211.3 | Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 3: Herstellung<br>Fassung 6/90 |
| KTA 3211.4 | Druck- und aktivitätsführende Komponenten von Systemen außerhalb des Primärkreises; Teil 4: Wiederkehrende Prüfungen; Regelentwurf<br>Fassung 6/90 |
| KTA 3301 | Nachwärmeabfuhrsysteme von Leichtwasserreaktoren<br>Fassung 11/84 |
| KTA 3303 | Wärmeabfuhrsysteme für Brennelementlagerbecken von Kernkraftwerken mit Leichtwasserreaktoren<br>Fassung 6/90 |
| KTA 3401.1 | Reaktorsicherheitsbehälter aus Stahl; Teil 1: Herstellung<br>Fassung 11/86 |
| KTA 3401.2 | Reaktorsicherheitsbehälter aus Stahl; Teil 2: Konstruktion und Berechnung<br>Fassung 6/85 |
| KTA 3402 | Schleusen am Reaktorsicherheitsbehälter von Kernkraftwerken - Personenschleusen<br>Fassung 11/76 |
| KTA 3403 | Kabeldurchführungen im Reaktorsicherheitsbehälter von Kernkraftwerken<br>Fassung 10/80 |
| KTA 3404 | Abschließung der den Reaktorsicherheitsbehälter durchdringenden Rohrleitungen von Betriebssystemen im Falle einer Freisetzung von radioaktiven Stoffen in dem Reaktorsicherheitsbehälter<br>Fassung 9/88 |
| KTA 3405 | Integrale Leckratenprüfung des Sicherheitsbehälters mit der Absolutdruckmethode<br>Fassung 2/79 |
| KTA 3407 | Rohrdurchführungen durch den Reaktorsicherheitsbehälter<br>Fassung 6/91 |
| KTA 3409 | Schleusen am Reaktorsicherheitsbehälter von Kernkraftwerken - Materialschleusen<br>Fassung 6/79 |
| KTA 3413 | Ermittlung der Belastungen für die Auslegung des Volldrucksicherheitsbehälters gegen Störfälle innerhalb der Anlage<br>Fassung 6/89 |
| KTA 3501 | Reaktorschutzsystem und Überwachungseinrichtungen des Sicherheitssystems<br>Fassung 6/85 |

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|---|---|
| KTA 3502 | Störfallinstrumentierung<br>Fassung 11/84 |
| KTA 3601 | Lüftungstechnische Anlagen in Kernkraftwerken<br>Fassung 6/90 |
| KTA 3701.1 | Übergeordnete Anforderungen an die elektrische Energieversorgung des Sicherheitssystems in Kernkraftwerken; Teil 1: Einblockanlagen<br>Fassung 6/78 |
| KTA 3702.1 | Notstromerzeugungsanlagen mit Dieselaggregaten in Kernkraftwerken; Teil 1: Auslegung<br>Fassung 6/80 |
| KTA 3703 | Notstromerzeugungsanlagen mit Batterien und Gleichrichtergeräten in Kernkraftwerken<br>Fassung 6/86 |
| KTA 3704 | Notstromanlagen mit Gleichstrom-Wechselstrom-Umformern in Kernkraftwerken<br>Fassung 6/84 |
| KTA 3901 | Kommunikationsmittel für Kernkraftwerke<br>Fassung 3/81 |
| KTA 3902 | Auslegung von Hebezeugen in Kernkraftwerken<br>Fassung 11/83 |
| KTA 3904 | Warte, Notsteuerstelle und örtliche Leitstände in Kernkraftwerken<br>Fassung 9/88 |

- **DIN-Standards**

| | |
|---|---|
| DIN 1045 | Beton und Stahlbeton; Bemessung und Ausführung<br>Fassung 7/88 |
| DIN 1055<br>Teil 4 | Lastannahmen für Bauten; Verkehrslasten, Windlasten bei nicht schwingungsanfälligen Bauwerken<br>Fassung 8/86 |
| DIN 1055<br>Teil 4 A1 | Lastannahmen für Bauten; Verkehrslasten, Windlasten bei nicht schwingungsanfälligen Bauwerken, Änderung 1, Berichtigungen<br>Fassung 6/87 |
| DIN 1055<br>Teil 5 | Lastannahmen für Bauten; Verkehrslasten, Schneelast und Eislast<br>Fassung 6/75 |
| DIN 8556<br>Teil 1 | Schweißzusätze für das Schweißen nichtrostender und hitzebeständiger Stähle; Bezeichnung; Technische Lieferbedingungen<br>Fassung 5/86 |
| DIN 17440 | Nichtrostende Stähle; Technische Lieferbedingungen für Blech, Warmband, Walzdraht, gezogenen Draht, Stabstahl, Schmiedestücke und Halbzeug<br>Fassung 7/85 |
| DIN 25436 | Integrale Leckratenprüfung des Sicherheitsbehälters mit der Absolutdruckmethode; Sicherheitstechnische Anforderungen<br>Fassung 7/80 |
| DIN 25440 | Klassifikation der Räume des Kontrollbereichs von Kernkraftwerken nach Ortsdosisleistung<br>Fassung 11/82 |
| DIN V 25459 | Sicherheitsumschließung aus Stahlbeton und Spannbeton für Kernkraftwerke; Vornorm<br>Fassung 4/90 |

## Appendix 2

**List of recommendations following the standard structure of TÜV/GRS assessments of nuclear power plants with pressurised or boiling water reactors**

- **Structure of the list**

| | |
|---|---|
| **1** | **Power plant** |
| **1.1** | **Design requirements** |
| **1.2** | **Quality assurance** |
| **1.3** | **Civil-engineering structures** |
| **1.4** | **Containment** |
| **1.5** | **Reactor core** |
| **1.6** | **Primary circuit with reactor pressure vessel** |
| 1.6.1 | Reactor pressure vessel |
| 1.6.2 | Reactor pressure vessel internals |
| 1.6.3 | Main coolant pumps |
| 1.6.4 | Main coolant lines |
| 1.6.5 | Steam generator |
| 1.6.6 | Pressuriser and steam-dump system |
| 1.7 | Reactor auxiliary facilities |
| 1.7.1 | Shutdown installations |
| 1.7.2 | Emergency core cooling and residual heat removal |
| 1.7.2.1 | Emergency core cooling and residual heat removal system, building spray system |
| 1.7.2.2 | Emergency feedwater system, emergency standby system |
| 1.7.2.3 | Secondary-side residual heat removal |
| 1.7.3 | Handling and cooling of fuel elements |
| 1.7.4 | Other reactor auxiliary facilities |
| **1.8** | **Ventilation-related installations** |
| **1.9** | **Secondary circuit** |
| **1.10** | **Cooling-water systems** |
| **1.11** | **Power plant auxiliary facilities** |
| **1.12** | **Electrotechnical installations** |
| **1.13** | **Installations for measuring, instrumentation and control** |

| | |
|---|---|
| **1.14** | **Emergency protection system, protection system for the control of the safety system** |
| **1.15** | **Fire protection** |
| **2** | **Radioactive materials and radiological-protection measures** |
| **2.1** | **Radiation and shielding** |
| **2.2** | **Release of radioactive materials and radiation exposure of the environment** |
| **2.3** | **Radiation monitoring** |
| **2.4** | **Radiological protection of the personnel during maintenance work** |
| **3** | **Accident analysis** |
| **3.1** | **Reactivity accidents** |
| **3.2** | **Interruptions of heat removal without loss of coolant** |
| **3.3** | **Loss-of-coolant accidents** |
| **3.4** | **Other plant-internal accidents** |
| **3.5** | **External impacts** |
| **3.6** | **Radiological accident consequences** |

- **Classification of the recommendations**

| | | |
|---|---|---|
| **U** | = | **further documents required** |
| **N** | = | **verification, examination required** |
| **Ä** | = | **changes recommended** |
| * | = | **important recommendation** (printed in front of the number of the recommendation) |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| 1 Power plant | | | |
| 1.1 Design requirements | | | |
| * R 2.7-1 | + | | |
| * R 2.7-2 | | + | |
| R 7.2-5 | | + | |
| R 8.3-31 | | | + |
| 1.2 Quality assurance | | | |
| R 8.3-30 | | | + |
| R 8.3-70 | | | + |
| 1.3 Civil-engineering structures | | | |
| R 7.1-4 | | + | |
| * R 7.1-5 | + | | |
| R 7.2-19 | | + | |
| R 8.3-28<br>R 8.3-42 | | + | + |
| 1.4 Containment | | | |
| R 5.2-3 | | + | |
| * R 6.3-3<br>* R 7.2-22 | | + | + |
| * R 6.3-20 | | | + |
| * R 6.3-22 | | + | + |
| * R 7.1-1 | | + | |
| R 7.1-2 | | + | |
| R 7.1-3 | | + | |
| R 7.1-6 | | + | |
| * R 7.1-7 | | + | |
| R 7.2-15 | | + | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **1.4 Containment** (continued) | | | |
| R 8.3-11 | | | + |
| **1.5 Reactor core** | | | |
| R 4.1-1 | + | | |
| * R 4.1-2 | | | + |
| * R 4.1-3 | | | + |
| R 4.1-4<br>R 4.1-7<br>R 4.1-8 | | + | |
| R 4.1-5 | | | + |
| * R 4.1-6 | | | + |
| R 4.1-9 | + | | |
| * R 4.1-10<br>* R 4.1-11 | | | + |
| R 4.1-13 | + | | |
| * R 4.1-14 | | + | |
| R 4.1-17 | + | | |
| R 4.1-18 | | + | |
| R 4.1-19 | | + | |
| R 4.1-20 | | + | |
| **1.6 Primary circuit with reactor pressure vessel** | | | |
| R 4.2-10 | | + | + |
| R 4.2-11 | | | + |
| * R 4.2-16 | | | + |
| R 4.2-21 | | + | |
| R 7.3-8 | | + | |
| R 8.3-66 | | | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| 1.6.1 Reactor pressure vessel | | | |
| * R 4.2-1 | | + | |
| R 4.2-4 | | + | |
| R 4.2-5 | | + | |
| R 4.2-6 | | + | |
| * R 4.2-7 | | + | |
| R 4.2-8 | | + | |
| R 4.2-14 | | | + |
| * R 4.2-15 | | + | |
| * R 4.2-17 | | | + |
| 1.6.2 Reactor pressure vessel internals | | | |
| R 4.1-15 | | + | |
| R 4.1-16 | | + | |
| 1.6.3 Main coolant pumps | | | |
| R 4.2-3 | | + | |
| R 4.2-4 | + | | |
| R 4.2-5 | | + | |
| R 7.2-9 | + | | |
| R 8.3-1 | | | + |
| R 8.3-20 | | | + |
| R 8.3-44 | | | + |
| R 8.3-64 | | | + |
| 1.6.4 Main coolant lines | | | |
| * R 4.2-2 | | + | |
| R 4.2-4 | | + | |
| R 4.2-5 | | + | |
| R 4.2-20 | + | | |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| 1.6.5 Steam generator | | | |
| R 4.2-4 | | + | |
| R 4.2-5 | | + | |
| R 4.2-9 | | + | |
| * R 4.2-18 | | + | + |
| * R 5.1-7 | | | + |
| R 8.3-25<br>R 8.3-65<br>R 8.3-69 | | | + |
| R 8.3-47<br>R 8.3-48 | | + | + |
| R 8.3-67 | | | + |
| 1.6.6 Pressuriser and steam-dump system | | | |
| R 4.2-4 | | + | |
| R 4.2-5 | | + | |
| * R 5.1-19 | | | + |
| * R 6.3-18 | | | + |
| R 6.3-19 | | + | |
| R 8.3-6 | | | + |
| R 8.3-58 | | | + |
| **1.7 Reactor auxiliary facilities** | | | |
| 1.7.1 Shutdown installations | | | |
| R 4.1-4<br>R 4.1-7<br>R 4.1-8 | | + | |
| * R 5.1-20 | | | + |
| R 6.2-1 | + | | |
| R 6.2-2 | + | | |
| * R 6.2-3 | | | + |
| * R 6.2-4 | | | + |

| No. of Recommendation | Classification | | |
| --- | --- | --- | --- |
| | U | N | Ä |
| 1.7.1 Shutdown installations (continued) | | | |
| R 6.2-5 | | | + |
| R 8.3-4 | | + | |
| R 8.3-5 | | + | |
| 1.7.2 Emergency core cooling and residual heat removal | | | |
| 1.7.2.1 Emergency core cooling and residual heat removal system, building spray system | | | |
| * R 5.1-4 | | + | + |
| R 6.3-1 | | + | |
| R 6.3-2 | | + | |
| * R 6.3-3<br>* R 7.2-22 | | + | + |
| * R 6.3-5<br>R 8.3-21 | | | + |
| R 6.3-6 | | + | |
| R 6.3-7 | | + | |
| R 6.3-8 | | + | |
| R 6.3-9 | | + | |
| R 6.3-23<br>R 8.3-15 | | + | |
| R 8.3-13 | | | + |
| R 8.3-14 | | | + |
| R 8.3-16 | | | + |
| 1.7.2.2 Emergency feedwater system, emergency standby system | | | |
| * R 6.3-10 | | + | + |
| * R 6.3-17 | | | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| 1.7.2.3 Secondary-side residual heat removal | | | |
| * R 5.1-6<br>* R 6.3-13 | | | + |
| * R 6.3-11 | | + | |
| * R 6.3-12 | | | + |
| R 8.3-33<br>R 8.3-34<br>R 8.3-35 | | | + |
| 1.7.3 Handling and cooling of fuel elements | | | |
| R 7.2-24 | + | | |
| R 7.2-27 | | + | |
| 1.7.4 Other reactor auxiliary facilities | | | |
| R 7.2-20 | | | + |
| * R 7.2-21 | | | + |
| **1.8 Ventilation-related installations** | | | |
| R 6.3-21 | | + | |
| R 7.2-7 | + | | + |
| R 8.3-42 | | + | |
| **1.9 Secondary circuit** | | | |
| * R 4.2-12 | | + | |
| R 5.1-2 | | | + |
| R 5.1-11 | | | + |
| * R 5.1-12 | | | + |
| R 8.3-18 | | | + |
| R 8.3-50 | | | + |
| R 8.3-57 | | | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **1.9 Secondary circuit** (continued) | | | |
| R 8.3-61 | | | + |
| R 8.3-62 | | | + |
| R 8.3-66 | | | + |
| **1.10 Cooling-water systems** | | | |
| * R 6.3-4 | | | + |
| * R 6.3-14 | | + | + |
| R 6.3-15<br>R 6.3-16 | | + | + |
| **1.11 Power plant auxiliary facilities** | | | |
| R 7.2-25 | | | + |
| R 7.2-26 | | | + |
| **1.12 Electrotechnical installations** | | | |
| R 6.5-1 | | | + |
| * R 6.5-2 | | | + |
| * R 6.5-3<br>R 8.3-41<br>R 8.3-45 | | | + |
| * R 6.5-4<br>R 8.3-23 | | | + |
| R 6.5-5 | | | + |
| R 6.5-6 | | | + |
| * R 6.5-7 | | + | |
| R 6.5-8 | | | + |
| * R 6.5-9 | | + | |
| R 6.5-10 | | | + |
| R 8.3-2<br>R 8.3-3 | | | + |

| No. of Recommendation | Classification | | |
| --- | --- | --- | --- |
| | U | N | Ä |
| **1.12 Electrotechnical installations** (continued) | | | |
| R 8.3-10 | | | + |
| R 8.3-22 | | | + |
| R 8.3-24 | | | + |
| R 8.3-29 | | + | |
| R 8.3-36 | | | + |
| R 8.3-37 | | | + |
| R 8.3-38 | | | + |
| **1.13 Installations for measuring, instrumentation and control** | | | |
| * R 4.1-3<br>R 4.1-13 | | | + |
| * R 4.1-6<br>R 8.3-56 | | + | + |
| R 4.1-9 | + | | |
| * R 4.1-10<br>* R 4.1-11 | | | + |
| * R 4.1-14 | | + | |
| R 4.2-11 | | + | + |
| R 6.4-1 | | + | |
| * R 6.4-2<br>R 8.3-22 | | + | + |
| R 6.4-3 | | + | |
| * R 6.4-4 | | | + |
| * R 6.4-5 | | | + |
| R 6.4-6<br>R 8.3-55 | | + | + |
| R 6.4-7 | | | + |
| * R 6.4-8 | | | + |
| * R 6.4-20 | | + | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **1.13 Installations for measuring, instrumentation and control** (continued) | | | |
| * R 7.2-8 | | | + |
| * R 7.2-23 | | | + |
| R 8.3-2 | | | + |
| R 8.3-7 | | | + |
| * R 8.3-8 | | | + |
| * R 8.3-12 | + | + | |
| * R 8.3-17 | | | + |
| * R 8.3-19<br>R 8.3-50 | | | + |
| * R 8.3-29 | | + | |
| R 8.3-39 | | | + |
| R 8.3-40 | | | + |
| R 8.3-43 | | | + |
| R 8.3-46 | | + | |
| R 8.3-51 | | | + |
| R 8.3-53 | | | + |
| R 8.3-54 | | | + |
| R 8.3-62 | | | + |
| R 8.3-63 | | | + |
| * R 8.3-67 | | | + |
| R 8.3-68 | | | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **1.14 Emergency protection system, protection system for the control of the safety system** | | | |
| R 6.4-9 | | + | |
| * R 6.4-10<br>* R 6.4-15<br>* R 8.3-68 | | | + |
| * R 6.4-11<br>* R 6.4-16<br>* R 8.3-68 | | | + |
| R 6.4-12 | | + | |
| R 6.4-14 | | | + |
| R 6.4-17 | | | + |
| R 6.4-18 | | + | |
| * R 6.4-19<br>* R 8.3-55 | | + | |
| R 8.3-2 | | | + |
| R 8.3-9 | | | + |
| R 8.3-12 | + | + | |
| R 8.3-27 | | | + |
| R 8.3-46 | | + | |
| R 8.3-52 | | + | |
| R 8.3-59 | | | + |
| R 8.3-60 | | | + |
| **1.15 Fire protection** | | | |
| R 7.2-1 | | + | |
| R 7.2-2 | + | | |
| R 7.2-3 | | + | |
| R 7.2-4 | | + | |
| R 7.2-5 | | + | |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **1.15 Fire protection** (continued) | | | |
| R 7.2-6 | | + | |
| R 7.2-7 | + | | + |
| R 7.2-8 | | | + |
| R 7.2-9 | + | | |
| R 7.2-10 | | | + |
| R 7.2-11 | | | + |
| R 7.2-12 | | + | |
| R 7.2-13 | | + | |
| R 7.2-14 | | + | |
| R 7.2-17 | + | | |
| R 7.2-18 | | + | |
| **2 Radioactive materials and radiological-protection measures** | | | |
| **2.1 Radiation and shielding** | | | |
| R 7.3-5 | | + | |
| **2.2 Release of radioactive materials and radiation exposure of the environment** | | | |
| - | | | |
| **2.3 Radiation monitoring** | | | |
| * R 7.3-6 | | | + |
| **2.4 Radiological protection of the personnel during maintenance work** | | | |
| R 7.3-1 | + | | |
| R 7.3-2 | | | + |
| R 7.3-3 | | | + |
| R 7.3-4 | | + | + |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **2.4 Radiological protection of the personnel during maintenance work** (continued) | | | |
| R 7.3-7 | | + | |
| R 7.3-8 | | + | |
| R 7.3-9 | | + | |
| R 7.3-10 | | + | |
| * R 7.3-11 | | | + |
| R 7.3-12 | | | + |
| R 7.3-13 | | | + |
| **3 Accident analysis** | | | |
| * R 5.1-5 | | + | |
| * R 5.1-21 | | + | |
| R 6.4-17 | | + | |
| **3.1 Reactivity accidents** | | | |
| * R 5.1-9 | | + | |
| **3.2 Interruptions of heat removal without loss of coolant** | | | |
| * R 4.1-12 | | + | |
| R 5.1-15 | | + | |
| R 5.1-16 | | + | |
| * R 5.1-17 | | + | |
| **3.3 Loss-of-coolant accidents** | | | |
| * R 4.2-13<br>* R 5.1-12 | | + | + |
| * R 4.2-19<br>* R 5.1-5 | | + | |
| R 5.1-1 | | + | |
| * R 5.1-3 | | + | |
| * R 5.1-8 | | | + |
| * R 5.1-10 | | + | |

| No. of Recommendation | Classification | | |
|---|---|---|---|
| | U | N | Ä |
| **3.3 Loss-of-coolant accidents** (continued) | | | |
| R 5.1-11 | | + | + |
| * R 5.1-13 | | + | |
| R 5.1-14 | | + | |
| R 5.2-1 | | + | |
| R 5.2-2 | | + | |
| R 5.2-3 | | + | |
| **3.4 Other plant-internal accidents** | | | |
| * R 5.1-18 | | + | |
| see Section 1.15 | | | |
| R 7.2-16 | | + | |
| R 7.2-19 | | + | |
| R 7.2-13 | | | + |
| R 7.2-24 | + | | |
| **3.5 External impacts** | | | |
| R 2.7-1 | + | | |
| R 2.7-2 | | + | |
| **3.6 Radiological accident consequences** | | | |
| * R 5.3-1 | | + | |

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