Gesellschaft für
Reaktorsicherheit (GRS) mbH

Safety Assessment of the Nuclear Power Plant Greifswald, Units 1 to 4

A Documentation of Hitherto Existing Investigations made by GRS
- Gesellschaft für Reaktorsicherheit mbH -
and SAAS
- Staatliches Amt für Atomsicherheit und Strahlenschutz -

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Annotation

This report represents the translation of the documentation "Sicherheitsbeurteilung des Kernkraftwerks Greifswald, Block 1 - 4, Eine Dokumentation der bisherigen Untersuchungen", GRS-report GRS-77, which has been published in June 1990. In cases of doubt report no. GRS-77 is the factually correct paper.

GRS gives thanks to the Commission of the European Communities and to the Federal Ministry for Environment, Nature Conservation and Nuclear Safety for the realization of this translation.

Keywords

PWR, GDR, operational experience, cooling system, safety device, reactor safety, material problems

KURZFASSUNG

Der vorliegende Bericht enthält den ersten und zweiten Zwischenbericht zur Sicherheitsbewertung der Blöcke 1 bis 4 des Kernkraftwerks "Bruno Leuschner" Greifswald. Weiterhin enthält der Bericht Stellungnahmen sowjetischer Experten zum ersten und zweiten Zwischenbericht.

Der erste Zwischenbericht befaßt sich vorrangig mit einer Beurteilung der druckführenden Komponenten des 1. Kreislaufs, insbesondere mit der Materialversprödung der Reaktordruckbehälter. Darüber hinaus werden erste Einschätzungen zur sicherheitstechnischen Auslegung der Anlagen vorgenommen.

Der zweite Zwischenbericht gibt einen Überblick über den Stand der weitergeführten Untersuchungen zur sicherheitstechnischen Auslegung und zur Auswertung der Betriebserfahrungen.

Der Bericht enthält eine zusammenfassende Beurteilung, in der die in den Fachkapiteln aufgeführten Empfehlungen in drei Kategorien von Maßnahmen zur Ertüchtigung der Blöcke gegliedert und bewertet werden.

ABSTRACT

The present report consists of the first and second interim reports on the safety evaluation of unit 1 through 4 of the nuclear power plant "Bruno Leuschner" at Greifswald. The appendices to the report contain comments of U.S.S.R. experts to the first and second interim reports.

The first interim report primarily deals with an evaluation of the pressurized components of the primary loop, especially with the embrittlement of the reactor pressure vessel material. In addition, first estimates concerning the safety design of the plants are made.

The second interim report reflects the state of further studies relating to the safety design and the evaluation of operational experiences.

The report includes a summarized assessment in which the recommendations cited in the technical chapters are evaluated and subdivided into three categories of backfitting measures.

FIRST INTERIM REPORT CONCERNING
THE SAFETY ASSESSMENT OF
THE GREIFSWALD NPP
UNITS 1 - 4

CONTENTS

1. INTRODUCTION

In the framework of cooperation with the German Democratic Republic to warrant safety of nuclear power plants, the Federal Minister for Environment, Protection of Nature and Reactor Safety (BMU) charged the Society for Reactor Safety (GRS) with the safety assessment of the nuclear power plant Greifswald, units 1 to 4, in a letter of 22nd of January, 1990.
The necessary work was undertaken in cooperation with the National Office for Nuclear Safety and Radiation Protection (SAAS) of the German Democratic Republic and was assisted by representatives from the Nuclear Power Plant "Bruno Leuschner" at Greifswald.
The work was intiated by a seminar convened by SAAS on 25./26. January 1990 in the Nuclear Power Plant at Greifswald. Participants from the Federal Republic of Germany were delegates from BMU, GRS, MPA (materials testing office) Stuttgart, and from the technical supervising associations (TÜV).
SAAS invited representatives from NPP "Bruno Leuschner". Besides, some experts from Siemens participated in the consultations, invited by NPP Greifswald.
The seminar was intended in the first place for information on the safety related backfitting measures, deemed necessary by SAAS. The 2-years work programme on this subject, planned by SAAS in cooperation with NPP Greifswald and to be completed til end of 1991, was explained. According to this programme, it has to be checked, in an on the-spot action, whether backfitting measures are possible and useful. In the first place, materials and strength problems of the pressure boundary, especially the reactor pressure vessel, have to be treated. Starting from the results of this assessment, further decision bases on backfitting measures are to be worked out.
During an inspection of the plant in the course of the seminar, equipments of the primary circuit, the feedwater-steam circuit, the main and auxiliary control room, the machine hall and the essential service system were examined.

Four working groups were created:

AG 1: Pressurized components, primary circuit
AG 2: System design
AG 3: Confinement and common cause events
AG 4: Accident analysis, efficiency of emergency cooling

where details of the investigations to be considered, of the necessary documentation and of the further proceeding were discussed.
This document contains first comments on safety problems concerning units 1 to 4 of the NPP Greifswald. Bases of these short-term comments are the statements made and information given during the seminar and during a meeting on 1st of February 1990 by SAAS and representatives of the NPP, as well as ideas gained from documents which were available or quickly made available. Besides, there was an expert meeting with MPA on questions concerning materials and strength.
The comments concerned mainly a first assessment of actually available knowledge of the pressure components of the primary circuit, especially of the reactor pressure vessel of the units 1 to 4 (section 3). In the further sections of the assessment, single aspects of the safety design of the plant are treated (section 5 and 6) as well as aspects of common cause events (section 7). An assessment of the importance of these

single aspects for the safety of the plant as a whole have to be reserved to further investigations. As far as possible, investigations necessary for the further assessment (e.g. transient behaviour, efficiency of emergency cooling) are discussed in the respective sections. For the pursuit of the investigations, further steps have been agreed with SAAS.

2. DESCRIPTION OF THE PLANT

In this short description we intend to draw the attention on important design characteristics, for better understanding of the first assessments in the following sections.
The units 1 to 4 of the NPP Greifswald are equipped with Soviet designed pressurized water reactors of the type WWER-440/W230.
In a final stage, 4 other units (units 5 to 8) are planned at the site of Lubmin/Greifswald, equipped with pressurized water reactors of the type WWER-440/W213. Compared to the W-230 plants, there are improved safety features in the W-213 plants.
In unit 5, the test operation began last year. After an incident which occurred on 24th November 1989, the unit was shut down on 29th November 1989. The units 6 to 8 are under construction.
The following overview refers only to the most essential technical characteristics and safety equipment of the units 1 to 4 with the pressurized water reactors WWER-440/W-230.

2.1 Basic plant layout

Fig. 2-1 shows a vertical cross-section of the reactor WWER-440/W-230.
Each unit has two circuits, the primary circuit (circuit 1, cf. fig. 2-2) and the feedwater-steam system (circuit 2, cf. fig. 2-3).
The installations of circuit 1 are surrounded by a so-called confinement, the installations of the circuit 2 are located in the machine hall.
WWER nuclear power plants of the type 440 are normally built as double-unit plants. The arrangement of both reactor units in a common reactor hall is characteristic for the construction of these double units. Both reactors have as well independent and separate as commonly used operational and safety systems.
The primary circuit consists of a water cooled water moderated power reactor with 6 main loops (HUL). Each loop has a main coolant pump (HUP), a steam generator (DE), two main block valves (HAS) with electrical drive to block the main loops of nominal diameter 500 mm at the reactor pressure vessel (RDB).
In order to equalize fluctuation of pressure and volume, the primary circuit has a pressurizer (DH), which is connected to a loop by two pipe lines of nominal diameter 200 mm and which is equipped with 2 safety valves (of the firm Sempell). These valves exhaust steam into a quench tank.
The second circuit comprises the steam side of the steam generator, the turbine generators, the feed water and condenser water systems and the related auxiliary equipment and systems. To each unit, two turbine generators of 220 MW each are attached, which operate with saturated steam of 44 bar.

Fig. 2-1: WWER-440/W-230, vertical cross-section

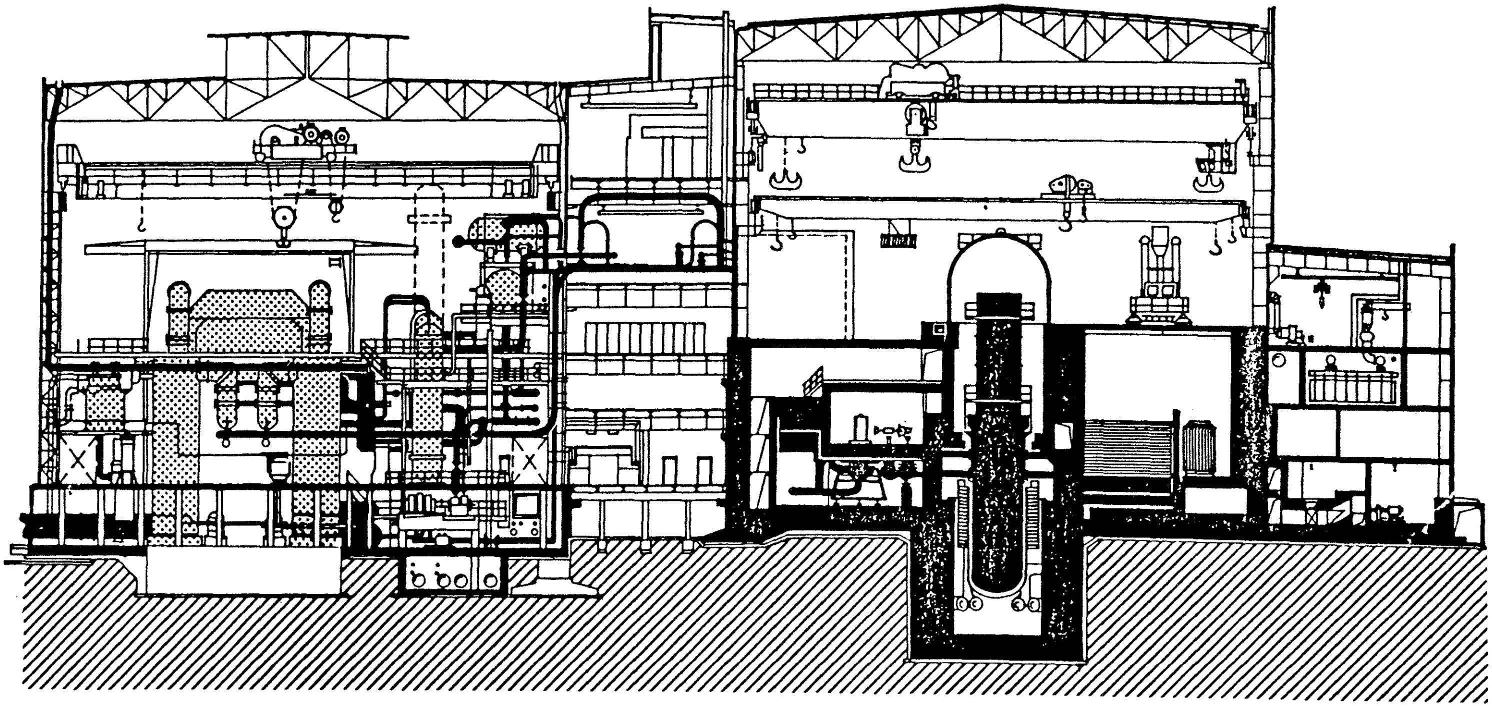
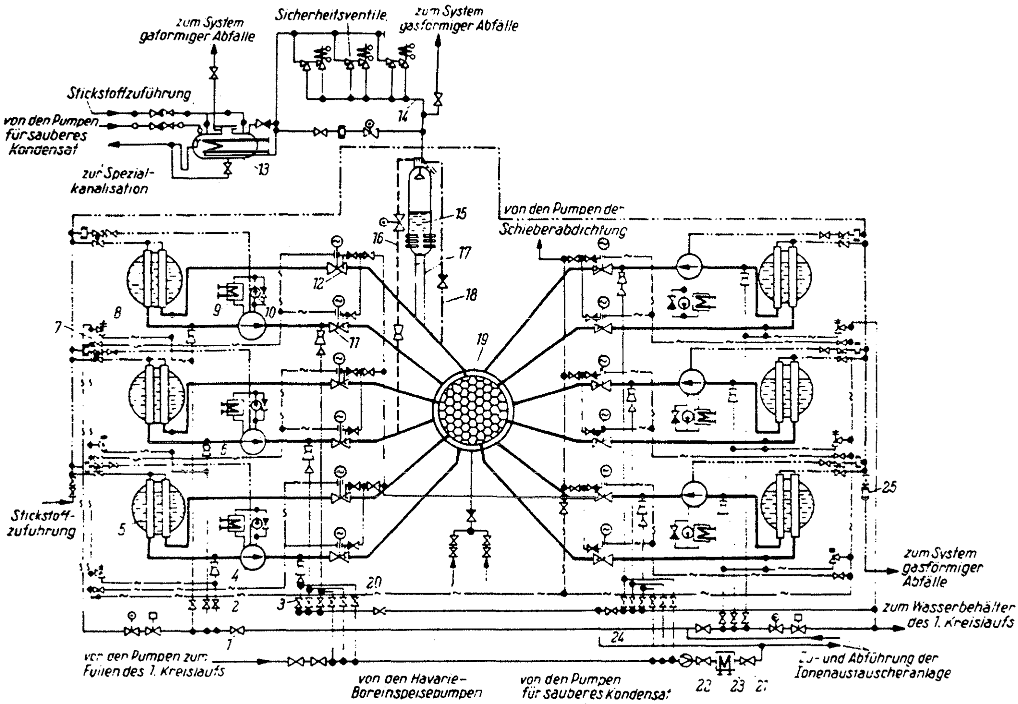
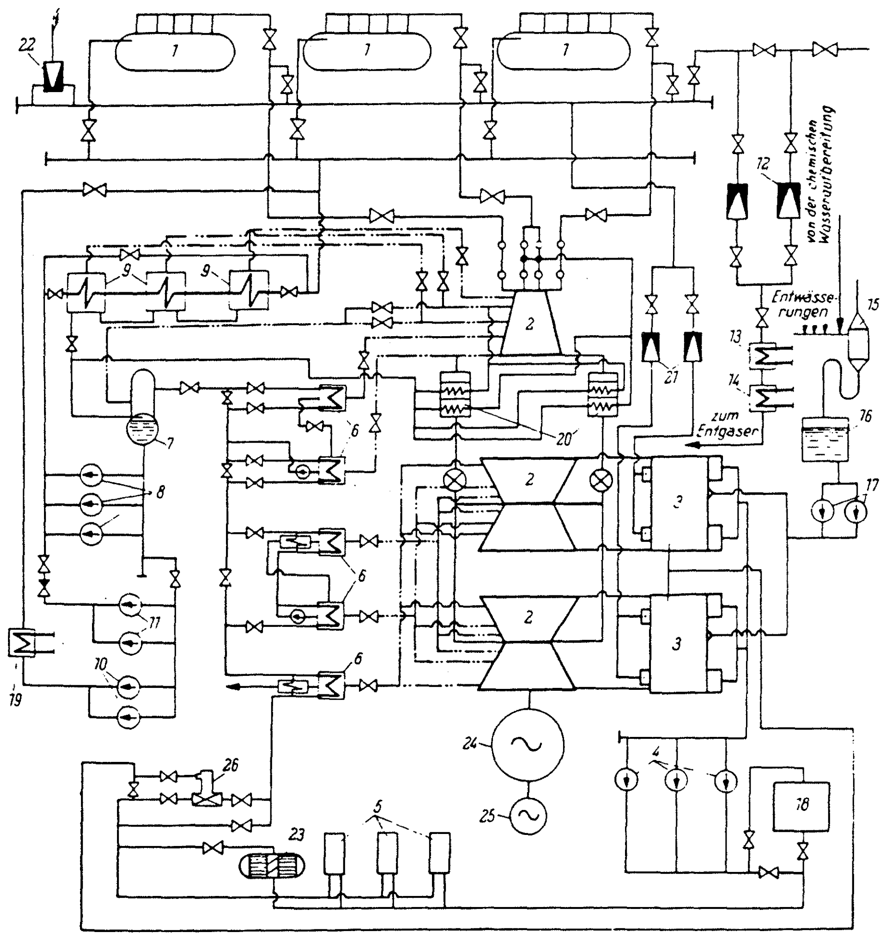


Fig. 2-2: Basic diagram of circuit 1 of the reactor WWR-440



1.2: Valves for the inlet of water into the circuit
3: Valves for water letdown
4: Main coolant pump
5: Steam generator
6: Reducing valve
7: Valve for controlled water leaks
8: Safety valve
9.10: Heat exchanger and pump of the autonomous circuit of the main coolant pumps (HUP)
11.12: Main block valves in the cold and hot legs
13: Quench tank
14: Steam collector
15: Volume control (pressurizer)
16: Injection line
17: Connection pipes between pressurizer and that part of the circuit which cannot be shut off
18: Overflow line
19: Core
20: Valve for heating and cooling of one loop
21.22: Block valves of the heat exchanger for heating and cooling
23: Heat exchanger
24: Valve for water letdown
25: Leak indicator

Fig. 2-3 Basic diagram of circuit 2



1:Steam generator
2:Turbine
3:Condensers
4:Pump for condensed water
5:Main ejectors
6:Low pressure preheater
7:Deaerator
8:Feedwater pumps
9:High pressure preheater
10:Emergency feedwater pump
11:Cooling pump
12:Reducing valve for the cooling
13:Technological condenser
14:Cooler of condensed water
15:Depressurisation of drain lines
16:Drain water tank
17:Drain water pumps
18:Purification of condensed water
19:Heat exchanger for warming up and cooling
20:Superheater - water separator
21:Relief valve of steam discharge into the turbine condenser
22:Pressure relief valve for steam into the atmosphere
23:Ejector of turbine joints
24:Main generator
25:Generator for onsite needs
26:Valve to feed condensed water

The electric power supply to the distribution grid is realized at 220 kV or 380 kV respectively; the distribution for internal needs has tensions of 6 kV, 380/220 V and 220 V DC. Besides, 200 MW of thermal energy are fed into a heat distribution grid.
The supply of cooling water and service water is realized by means of cooling with seawater from the "Greifswalder Bodden" through an inlet structure.

2.2 Safety design

The safety features of units 1 to 4 (WWER-440/W-230) correspond to those safety requirements, which had been the bases for design and construction of the WWER-440/W-230 NPPs. These safety principles are briefly presented and some system design features are explained.

2.3 Basic safety principles of the initial design of the units with WWER-440/W-230 (reproduced from /1/)

- Development of the units with pressurized water reactors was made in the USSR in the sixties.

- Main design principles:
  . To avoid burst and breach of large pipings by high quality of the material used (warm-ductile austenitic steel).
  . To exclude breach of a loop (HUL) and failure of the reactor pressure vessel.
  . To define as design basis accident (most severe accident of the project): Breach of a piping 100 mm in diameter, connected to the part of the primary circuit which can be blocked (equivalent diameter 32 mm by flow restriction) with simultaneous complete loss of electrical power.
  . To guarantee safe confinement by a pressure compartment system equipped with pressure relief valves, designed for the breach of a piping of 200 mm diameter and which stay closed up to a leak of 32 mm diameter.
  Remark: The basic idea of that was, that through the valves only the activity in the primary loop could be released (100 Ci/m^3 at about 240 m^3), and that fuel element damage occurs only after the valves have been closed.
  . To exclude severe accidents with core melt.

- Design characteristics of the units with W-230:
  . 6 primary loops.
  . Each loop can be blocked by main block valves (HAS).
  . Horizontal steam generators with big secondary water volumes.
  . In the final stage 8 units on a site with various possibilities of utilization of unit-bound systems for neighbouring units.
  . Common machine hall (length about 1000 m) for all 8 units.

2.4 Safety systems

. Emergency injection system of borated water

There are 6 pumps for injection into the primary circuit, which are installed next to each other. Three pumps at a time work as a unit and take in through a pipe from a tank of borated water, which serves all pumps. They inject into the primary circuit through two collectors. Besides the boron injection pumps, there are three spray pumps with two coolers each; also those pumps take in from the tank of borated water mentioned above; they spray, in case of loss of coolant accidents, borated water into the system of pressure resistant rooms of the containment building, in order to limit pressure build-up.

. Isolation of building - confinement

The pressure compartment system is a closed space enclosing as a confinement the main components of the primary circuit (reactor pressure vessel, steam generator, main coolant pumps, main isolation valves, pressurizer, emergency boration tank, primary feed water system). These rooms have a net volume of about 14.000 m^3 and are designed for an overpressure of 1 bar. Between accessible rooms, not accessible rooms and atmosphere, a ventilation system maintains a graduated underpressure. Lost heat is evacuated by cooled air circulators. In the box of steam generators and main coolant pumps, there are 3 lines of a spray system (pressure reduction, binding of iodines). The system of pressure resistant rooms is connected to the environment by 9 flaps.
In case of the most severe design basis accident (breach of a piping of 100 mm nominal diameter which is connected to the primary circuit, with flow restriction to 32 mm nominal diameter) and admitting effectiveness of the spray system, these flaps will not open. If the spray does not work during a design basis accident, a flap will open. Opening of all flaps protects the pressure resistant rooms in case of a breach of the largest connecting tube having 200 mm nominal diameter to the primary circuit in the unblockable area.

. Emergency feed water system

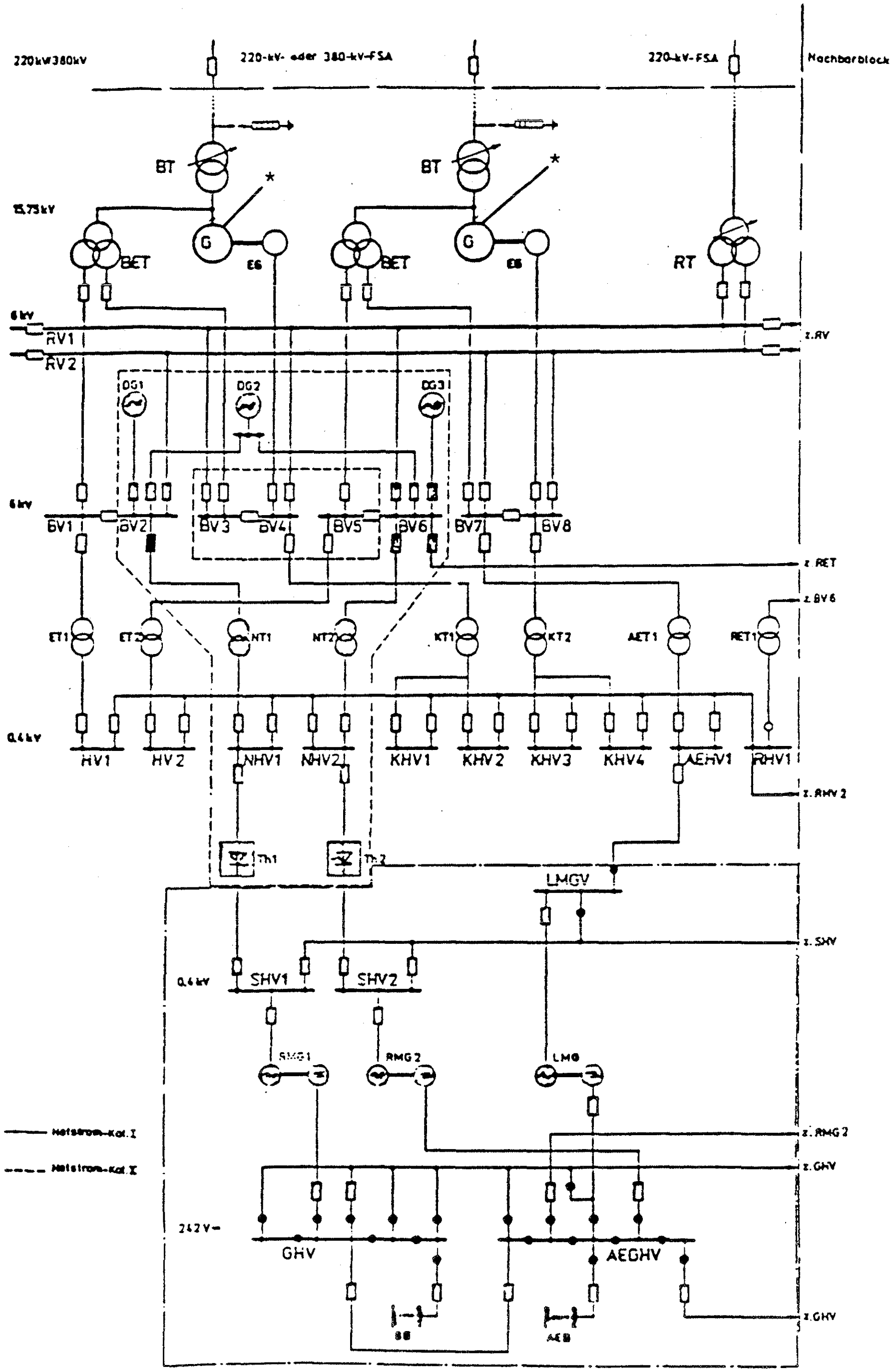
The emergency feed water system consists of two pumps which are installed in the machine hall next to the main feed water pumps. The pipings are in common both at the suction side and to the pressure side of the pumps.

. Essentail service water system
There are 5 pumps for a double unit, which are installed in a common room and which are connected at the pressure side, through a collector to all consumers of a unit (process and safety systems) which they supply with cooling water.

. Electricity supply
(cf. diagram on figure 2-4)

- Grid connexion: there is a grid connection for start-up. For the main grid connectors, generator switches have been added so that they can be used to cover onsite power needs. The connections of the different units feed into the 220 kV- and the 380 kV-lines.

Fig. 2-4: Overview diagram of the electrical installation of the 440 MW-unit, North I-II.



- Nuclear station service supply: connections are provided with the neighbouring unit. The rest energy of the turbo generators is used for limited operation of the main coolant pumps, e.g; in the emergency power case.

- Emergency power supply: the emergency power supply consists principally of two trains, but this is not strictly observed. At different levels (Diesel generators, switch gears, direct current installations) there are interconnections.

2.5 Comparison of safety level (taken from /1/)

For the planning of necessary or possible backfitting measures, investigations to compare different safety levels were executed by the nuclear power plant "Bruno Leuschner". According to /1/, some key-issues are presented here concerning recognized safety deficits. In table 2-1 from /1/ a safety level comparison is made to reactors of more recent design, using selected criteria.
"The actual safety level of units with W-230 reactors; main safety deficiencies compared to world level".

- Comparison of the level of nuclear safety by means of main criteria: cf. table 2-1.

- Start of operation of GDR-units: cf. table 2-2.

- Main deficiencies of the W-230 units compared to world level:
  a) design basis accident (rupture of a tube of nominal diameter 100 mm with flow limitation equivalent to a nominal diameter of 32 mm.
  b) due to a): emergency cooling system without accumulators and long term emergency cooling
  c) pressure resistant rooms with flaps (1.6 bar and 1.8 bar), no tight confinement
  d) diesel emergency power supply underdimensioned in power and reliability,
  e) heat sink of the primary circuit (essential service water system 1) not strictly separated, partly with interconnections)
  f) heat sink of the secondary circuit, emergency feedwater system 2)
  g) separation of safety systems.

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1) especially no emergency cooling system

2) only two pumps

Table 2-1
Comparison of safety levels of WWER-reactors by means of selected criteria

Nr	Criteria	W 230 before backfitting	W 213	WWER-1000	BIBLIS B
1	max. design basis accident (LOCA)	nominal 100 (32) mm with station black-	2 F primary circuit with station blackout	2 F primary circuit with station blackout	2 F primary circuit with station blackout
2	Important project incidents: - breach of a steam line - loss of feedwater supply - pressurizer safety valve stays open - steam generator leak - loss of power for main coolant pumps	 no (1) yes yes yes (3) yes (4)	 yes (2) yes yes yes yes	 yes yes yes yes yes	 yes yes yes yes yes
3	Redundancy of safety systems	2 x 100% (not strictly)	3 x 100%	3 x 100%	4 x 100% (5)

(Numerotation changed as compared to the original German text)

1) supplementary shut down measures in units 2 and 4

2) in case of leak, no pipe whips

3) with operator actions

4) utilisation of rest energy of turbo-generators

5) remark by GRS, not valid in every case

Table 2-1 continued
Comparison of safety levels of WWER-reactors by means of selected criteria

Nr	Criteria	W 230 before backfitting	W 213	WWER-1000	BIBLIS B
4	confinement type	System of pressure rooms with pressure relief valves	System of pressure rooms with scrubber system	Full pressure containment	Full pressure containment (6)
5	Protection against ex-external events	No	limited	yes	yes
6	Fire protection	insufficient no stand-by control-room	fire-proof materials, segregation, automatic fire fighting system, (stand-by control-room exists)		
7	Accident instrumentation and diagnosis	Obsolete instrumentation, range only for DBA	not up to date automation, measurement range limited	not up to date automation, measurement range limited	high automation application of computers, extended measurement range

6) Remark by GRS

Table 2-2

Nuclear Power Plants in GDR

First operation	Plant			
1966	KKW Rheinsberg		70 MWe	
1973	KKW Greifswald	Unit 1	440 MWe	
1974	"	Unit 2	"	
1977	"	Unit 3	"	
1979	"	Unit 4	"	
1989	"	Unit 5	"	
under constr.	"	Units 6 to 8	"	each
under constr.	KKW Stendal	Unit 1	1000 MWe	
under constr.	"	Unit 2	1000 MWe	

3. Assessment of pressure components of the primary circuit

3.1 Fabrication of the primary circuit

The pressure walls of containers of the primary circuit (RPV,SG, pressurizer) are in ferritic steel. For the RPV a low alloy Cr - Mo-Vn - steel is used, for the steam generator and pressurizer walls carbon steel was chosen. The pipings for the connection of these containers and the housings of valves and pumps are made in titanium stabised stainless Cr - Ni - steel (similar to CrNiTi 18 10).

This is also true for the steam generator tubes (boiler tubes). The heavy parts are mainly forged and consist, if possible, in seamless belts with reinforcements in the nozzle regions.

The solicitation of the structural parts of the primary circuit is, in normal operation, comparable to that of western reactors, partly it is even lower.

Figure 3-1 shows the typical design for the RPV. It consists of forged belts connected by ring welds. Inlet and outlet nozzles of the 6 circuits are found in two successive nozzle belts. The nozzle belts are strengthened and have integrated nozzle roots. The nozzle roots are welded by means of electrodes with a high Ni-content (~ 25%); the weld is executed in several layers. The joining piece is made in stainless steel and is welded on-site. The spheroidal bottom and coverparts consist of pressed parts which are connected by longitudinal welds.

The RPVs of units 1 and 2 are unplated, those of units 3 and 4 are plated with two layers. Plating was executed using band of 60 mm breadth and 9 + 2 mm thickness.

For the RPVs of the design WWER-440/W-230 the materials given in table 3 - 1 are normally used.

3.2 State of the components of the primary circuit

Concerning the state of components used, there is no documentation about chemical analysis, mechanical-technological characteristics, ductility data and procedures and testing comparable to that common in NPPs in the FRG. Such records or documents do not exist on the site. It is, therefore, necessary to rely on general information and especially on the results of periodical non destructive tests and pressure tests. The test interval is 4 years for the primary circuit, the first one taking place 2 years after start of operation.

Concerning pressure tests and surface flaw detection in the region of the more embrittled welds and inner nozzle edges, the test frequency of the RPV is higher than for western RPV designs.

From the whole of the inspection results presented, there is no indication of stress corrosion in the wet parts of the containers.

Figure 3-1 Reactor pressure vessel WWER-440 [DOE 87]

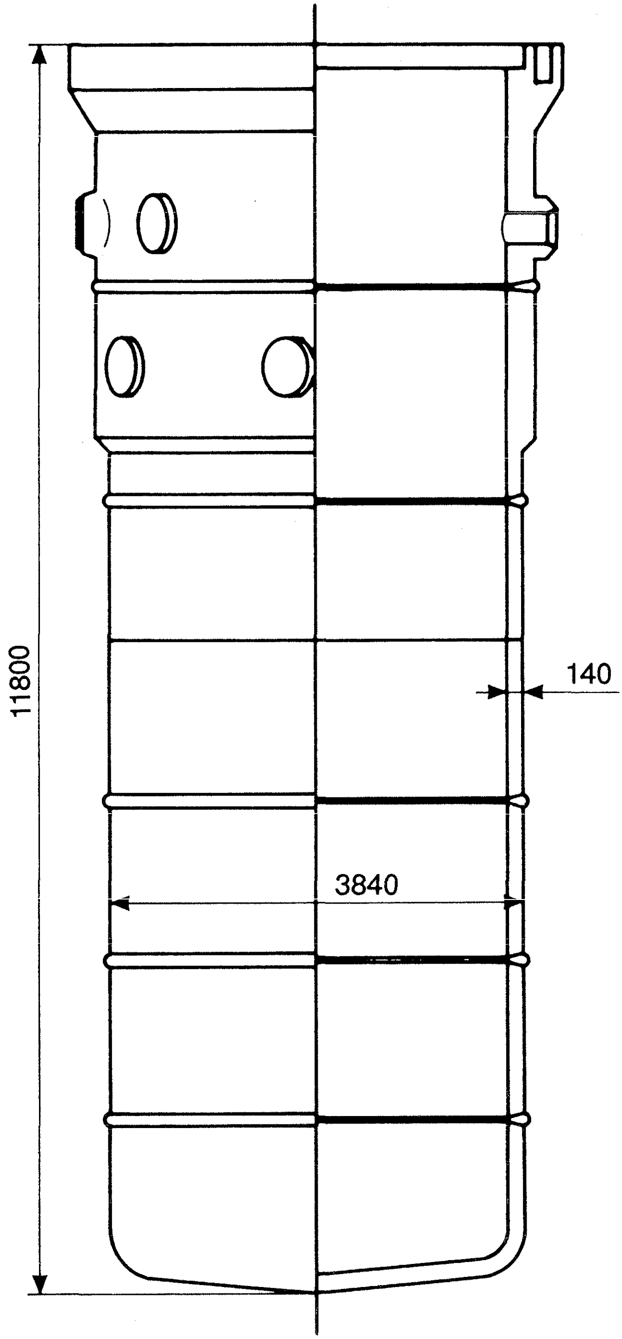


Table 3-1: Chemical composition of steels and welds for the manufacture of WWER-440 reactor pressure vessels

Massengehalte in %

Material	C	Si	Mn	S max.	P max.	Cr	Ni	Mo	V	Ti	Cu
25Cr3MFA	0.22-0.27	0.17.037	03.-0.6	0.025	0.025	2.5-3.0	0.4	0.6-0.8	0.25-0.36	-	-
15Cr2MFA	0.11-0.21	0.17-0.37	0.3-0.6	0.025	0.025	2.0-3.0	0.4	0.6-0.8	0.25-0.35	-	0.20±0.05
Sv-10CrMFT	0.07-0.12	0.15-0.35	0.4-0.7	0.03	0.03	1.4-1.8	0.3	0.4-0.6	0.20-0.35	0.05-0.12	0.05-0.10
Sv-10CrM	max 0.12	0.15-0.35	0.4-0.7	0.03	0.03	0.8-1.1	0.03	0.4-0.6	0.20-0.35	-	-
Sv-08CrMF	0.07-0.10	0.12-0.36	0.4-0.7	0.03	0.03	1.0-1.4	0.03	0.6-0.8	0.15-0.35	-	-
Sv-13Cr2FT	0.10-0.15	= 0.36	0.4-0.7	0.03	0.03	1.7-2.2	0.03	0.6-0.8	0.15-0.35	0.15-0.35	-

/DOE 87/ DOE/NE-0084, Revision 1 "Overall Plant Design Description VVER"

Further essential information, on which the following statements about the state of the components is based, are given in the tables 3-2, 3-3 and 3-4.

They were collected in discussions with the operator of the NPP and with SAAS. According to these data the state of the components can be characterized as follows:

- Unit 1

  On the unplated inner surfaces of the RPVs corrosion was detected during former periodic inspections. The reason for this damage was declared to be unconditioned, oxygen-saturated water in connection with boric acid during longer shut-down periods. The corrosion pittings had a diameter of a coin and a depth of up to 7 mm and were all removed by grinding. After grinding, flaws were not detected.
  The results of the last periodic inspections showed that the RPV has no flaws according to the magnetic powder method and no supplementary corrosion attack. Isolated indications of the ultrasonic test in the connecting welds of the belts were not long (20 to 40 mm).
  On the pressurizers or steam generator coatings no flaws were detected.
  On the pipings between main circulation loops and the pressurizer flaws have been detected during operation, some of which produced leaks. Further flaws and leaks were discovered during the pressure test and the non-destructive tests.
  The region close to the core of the RPV of unit 1 was annealed in 1988. We may assume, therefore, that the actual embrittlement of the material is comparatively low. Information on the heat treatment and on the inspection carried out or foreseen are presented in table 3-5.
  According to the actual state of the art, annealing is in principle a suitable method to treat embrittlement encountered during operation time and/or to eliminate it.
  According to our actual knowledge, there is no sufficient quantitative evidence of the degree of recovery of the ductility of a material. Additional investigations on specimens are foreseen, but the results are not yet available.
  In the steam generator tubes (boiler tubes) flaws by stress corrosion have been detected. Flaws on inner collectors have also been communicated. As far as we know, they did not lead to important operational occurrences.

Table 3-2: Information on the number of operating hours since first start-up, prognosis of neutron fluences at the inner wall of the RPV, and calculated transition temperatures for the transition to brittle fracture for beginning of operation (T_ko), for 1989 (T_k) and for the maximum value (T_k EOL).

Block No	Fluenz in cm^{-2} Neutronenenergie > 0,5 MeV				ermittelte Sprödbruchübergangstemperatur					
Betriebsstunden	1989		EOL[b)]		T_{KO}		T_K 1989		T_K EOL	
	SG	GW_{max}	SG	GW_{max}	SG	GW	SG	GW_{max}	SG	GW_{max}
Block 1 105 700 h	$4,5x10^{19}$	$7x10^{19}$	$5,7x10^{19}$	a	(+46 °C)	(0 °C)	Wert nach Glühung wird noch bestimmt	c	(150) d	c
Block 2 101 700 h	$6,4x10^{19}$	$8,2x10^{19}$	$6,4x10^{19}$	a	(+ 4 °C)	(0 °C)	(>147 °C)	c	(147)	c
Block 3 90 300 h	$5,5x10^{19}$	$7x10^{19}$	$6,5x10^{19}$	a	(+23 °C)	(0 °C)	(152 °C)	c	(163)	c
Block 4 73 300 h	$5x10^{19}$	$5,9x10^{19}$	$10,8x10^{19}$	a	(-13 °C)	(0 °C)	(124 °C)	c	(163)	c

a. values for the basic material in the middle of the core (GW max) depend on the future load scheme.
b. EOL-value = maximum admissible value for DBA.
EOL-value will be fixed again after the recovery annealing above T_k.
c. values are fixed at T_k < 70°C after 30 years.
d. value obtained before heat treatment 186°C.
() values obtained from empirical relationship.

Table 3-3: Information on the fractions of Cu and P in the chemical analysis and the coefficients AF^2 obtained from there for the estimation of the shift of the temperature of brittle fracture during irradiation

		Werte Cu, P Kernnahe Schweißnaht Grundwerkstoff (GW) Schweißgut (SG)			Koeffizient $(A_F)^2$ für die Abschätzung der Verschiebung der Sprödbruchübergangstemperatur	
		GW 1 %	SG %	GW 2 %	GW 2	SG
Block 1	Cu		0,086 - 0,104 [1] 0,109 - 0,123			
		(0,17)	(0,12)	(0,13)		
					(11,2)	33 (35,5)
	P		0,032 - 0,034 [1] 0,035 - 0,036			
		(0,010)	(0,036)	(0,012)		
Block 2	Cu		(0,18)		(11,2)	(37,7)
	P		(0,032)			
Block 3	Cu		(0,12)		(11,2)	(34,8)
	P		(0,035)			
Block 4	Cu		(0,16)		(15,6)	(37)
	P		(0,035)			

() estimated values referring to material heats fabricated in the production period.
1 values from 2 measurements
GW1 basis material below the weld close to core
GW2 basis material above the weld close to core
2 for basis material $A_F(265°C) = 1200\ (P + 0{,}1\ Cu) - 19$
for weld material $A_F(265°C) = 1200\ (P + 0{,}1\ Cu) - 16$
3 approximations found in literature, as examples; often changed in the past.

Table 3-4: Extent of the last periodic examination of the RPV (interval 4 years)

Last pressure test 1)			non-destructive tests 4)	findings
unit	year	wall temperature		
1	1987	170°C	Surface flaw test by magnetic powder method on welds close to core, basis material and inner edge of nozzles	none
			Surface flaw test with dye check and magn. test outer side of nozzles	none
			ultrasonic test of welds *	yes 2)
2	1986	≈110°C	Surface flaw test by magn. p. method at welds close to core, basis material and inner edge of nozzles	none
	1990	still to be fixed	Surface flaw test by dye check and magn. test, outside of nozzles	none
			ultrasonic test of welds *	similar to unit 1
3	1988	115°C	Surface flaw test by dye check inside, mainly plated area close to core and nozzles	none
			Surface flaw tests by dye check and magn. test outside	none
			Ultrasonic test of welds *	yes 2) 3)

Table 3-4: Extent of the last periodic examination of the RPV (interval 4 years) (continued)

Last pressure test 1)			non-destructive tests 4)	findings
unit	year	wall temperature		
4	1989	150°C	Surface flaw test by dye check inside, mainly plated area close to core and nozzles	none
			Surface flaw tests by dye check and magn. test outside	none
			Ultrasonic test of welds *	yes 2) 3)

* Since 1987, ultrasonic test with central pylon and manipulator by KWU; before that, Skoda material used.

1) 1,25 calculated pressure (operation 12 MPa, calculated 14 MPa)

2) a few indications, 20 to 30 mm long; close to core no findings

3) flaws under plating and dross inclusions in plating

4) usually before pressure test, random tests afterwards

Table 3-5

Information on the heat treatment of the RPV of unit 1

Velocity of heating up	22 K/h
Annealing temperature	475 ± 10°C
Temperature kept constant during	152 h
Velocity of cooling down	30 K/h

The region annealed was the ring weld close to the core with basis material, above and below it, over a total height of 70 cm.
Cuttings were analysed for Cu and P (cf. table 3-3)
Hardness was measured on the inner side of the RPV.

Remark: For a supplementary control, 4 plate specimens were taken by spark erosion from the inner side of the annealed region; size of specimens: 30 x 70 x 5 mm, 3 specimens from weld, one from basis material. The test results from these specimens are not yet available.
Charpy tests and tensile tests were foreseen as well as supplementary metallurgical analyses.

- Unit 2

Concerning corrosion attack, the statements made for unit 1 are valid.
The RPV was analysed in 1986 for flaws on the inner surface; no flaws were detected. The next inspection and annealing of the region close to the core is foreseen for spring 1990. We assume, that the embrittlement of the material is comparable to that of unit 1 before the heat treatment.
Concerning the other containers (pressurizer and steam generator) and concerning the steam generator tubes the statements for unit 1 are valid. Informations about possible cracks on pipings between pressurizer and main circulation loops have still to be collected.

- Unit 3

The RPV of unit 3 has on its inner surface a welded band plating of 3 layers; according to the operator, the plating has quality deficiencies. Besides, periodic inspections show dross inclusions in the plating and flaws below the plating. The surface flaw examinations by dye penetration tests during the last periodic inspection did not show any flaws.
Concerning embrittlement it can be expected, that the material is about in the state as that of unit 1 before the heat treatment.
Statements on the state of the other components correspond to those for unit 2 and unit 1. Damage to the steam generator tubes is, however, much less important.

- Unit 4

  The RPV of unit 4 has also an austenitic plating. According to the operator, the plating had been ground away locally to the ferritic basis material already before delivery. The affected regions were repaired by application of a special welding procedure.
  Besides, the periodic inspections showed the same findings as for unit 3.
  The last inspections were executed in 1989. The inspection results correspond to those of unit 3.
  For the other components, the statements for units 2 and 3 are valid also here. The amount of damage at the steam generator tubes is similar to that of unit 3.
  For the actual material of the RPV, a somewhat lower embrittlement should be assumed compared to units 2 and 3.
  The following safety assessment is limited to the RPV, as information on the pressurizer, the steam generator, the housings and the pipings are not yet sufficient.

3.3 Safety assessment of the reactor pressure vessels

The safety assessment of the reactor pressure vessel demands proof of integrity for all operational, incidental and accidental states. Proof has normally to be given, that:
- the operation temperature is higher than the brittle fracture transition temperature;
- for all operational states, there is no unstable crack propagation, when a postulated fault with a depth of one quarter of the wall thickness is assumed;. The safety factors to be applied differ in the different (national or international) rules.
- for incidental or accidental states the integrity of the RPV is preserved; also here the proof is usually given for a postulated fault of a depth of one quarter of the wall thickness in a cylindrical region. In recent years there is, however, international agreement, that for loads connected to rapid cooling down of the walls, cracks which are less deep have also to be investigated together with unsymmetrical cooling conditions. The aim of that is to take better into account the influence of the fact that the ductility of the material depends on the temperature and on the gradient of irradiation embrittlement in the wall, on the crack propagation and on local thermal stress.

For the RPVs of the units 1 to 4 we can make the following preliminary statements on the basis of the information gotten up to now:
- The values obtained for the brittle fracture transition temperature of the regions close to the core (weld, basis material) at the beginning of operation and now are given in table 3-2. When assessing values for the shift of the transition temperature of brittle fracture, it is necessary to make a supplementary assessment of eventual uncertainty bands on the basis of our actual knowledge. In this way, lack of information about the fraction of copper and phosphorus in the material of the regions concerned of the pressure vessel can be overcome by taking samples, as has been done already for unit 1. New results (cf. figure 3-2) and new understanding which indicate an increase of

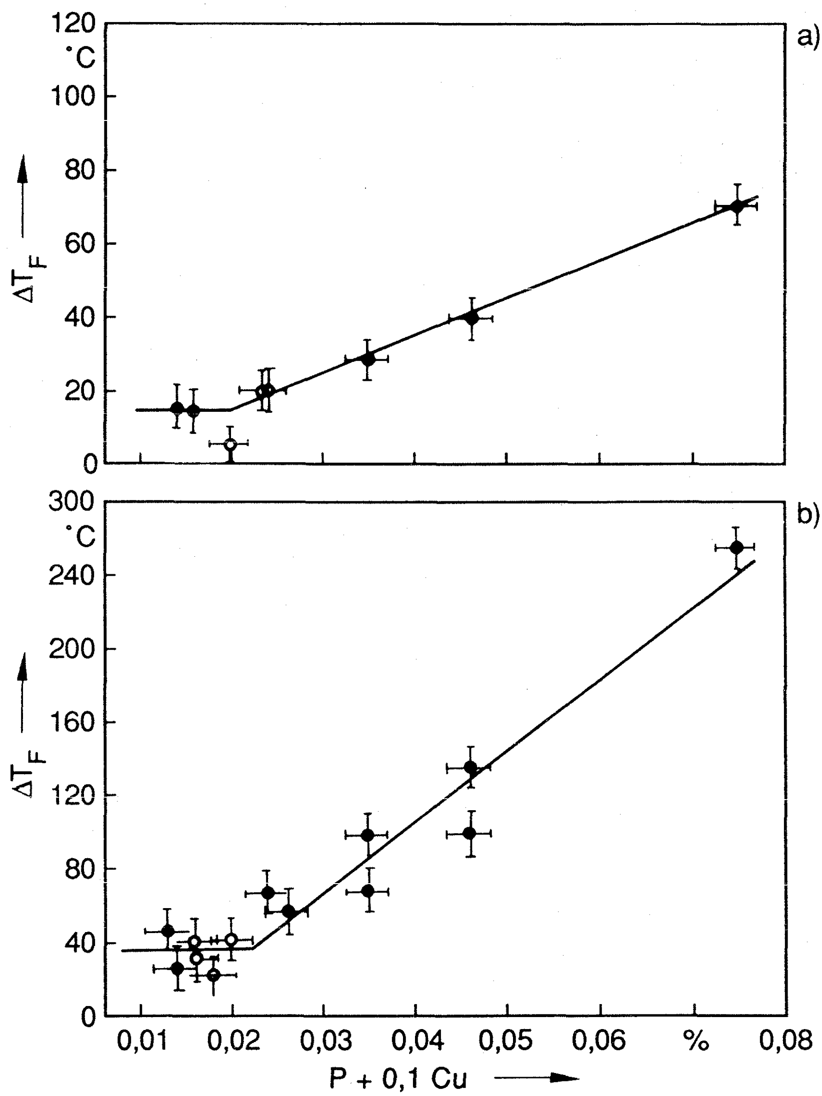
the temperature shift due to an effect of flux density, cf. figure 3-3, must be checked whether they can be considered valid for the actual case.

- In total, it can be expected that the transition temperature to brittle fracture is below the operation temperature of 265°C for all RPVs in normal operation.
- The loading of the RPVs in normal operation is comparable and sometimes lower than in RPVs of western design.
- Loadings during the start-up and run down phase and during operational transients need to be analysed thoroughly in order to allow possibilities of deviations from the permitted load-diagram. The informations necessary to do that are not available. Especially it has to be assessed whether it is possible, that pressures admitted for the respective temperature range can be considerably exceeded if a single failure is assumed; if this is true, a danger exists for the integrity of the pressure vessel by unstable crack propagation.
- For the assessment of effects of loadings during incidents and accidents we must first consider the design basis accident with an equivalent leak cross section of 10 cm2. For LOCA accidents, safety evidence has been produced by the operator, considering the expected shift of the brittle fracture transition temperature, however without the uncertainty consideration mentioned above. Admissible crack sizes are for the analyses in question in the region of 10 to 20 mm crack depth according to the load, assuming half-elliptical crack shape (crack length/crack depth = 3).
- Recording and assessment of possible faults in the RPV is the task of the periodic inspections. If cracks are identified, it has to be shown, that their sizes are below those to be postulated for the proof of integrity, taking into account safety factors. Kind, scope and results of the tests are presented in table 3-4.

In safety assessment also the frequency of leaks and their occurrence have to be considered on the basis of operational experience. Leaks of the size of a design base leak have not been reported up to now for NPPs of the type WWER-440. There have been leaks during operation in the Greifswald installations. The effective cross section was always of the order of a few cm^2.
For an exact assessment of the situation in question it is necessary to evaluate operational experience in a more detailed way; e.g. the inspections executed at the potentially dangerous places have to be collected and the sensitivities of the equipment of leak control have to be analysed. Deficiencies of knowledge in the mentioned areas have to be overcome.

Further incidents can derive from a failure of the closing function of a primary safety valve which had been opened before. For this case, the leak cross section is close to the design basis accident. In sections 5 and 7, possibilities to open the primary safety valve are discussed; there are discussed also further incidents by subcooling transients.
A LOCA accident can be caused also by leaks in tubes of the steam generators. These tubes have in fact a high share of cracks due to stress corrosion (uncontrolled water chemistry on the secondary side during the first years of operation); a complete breach of one of the 33000 tubes was not reported, however, up to now.

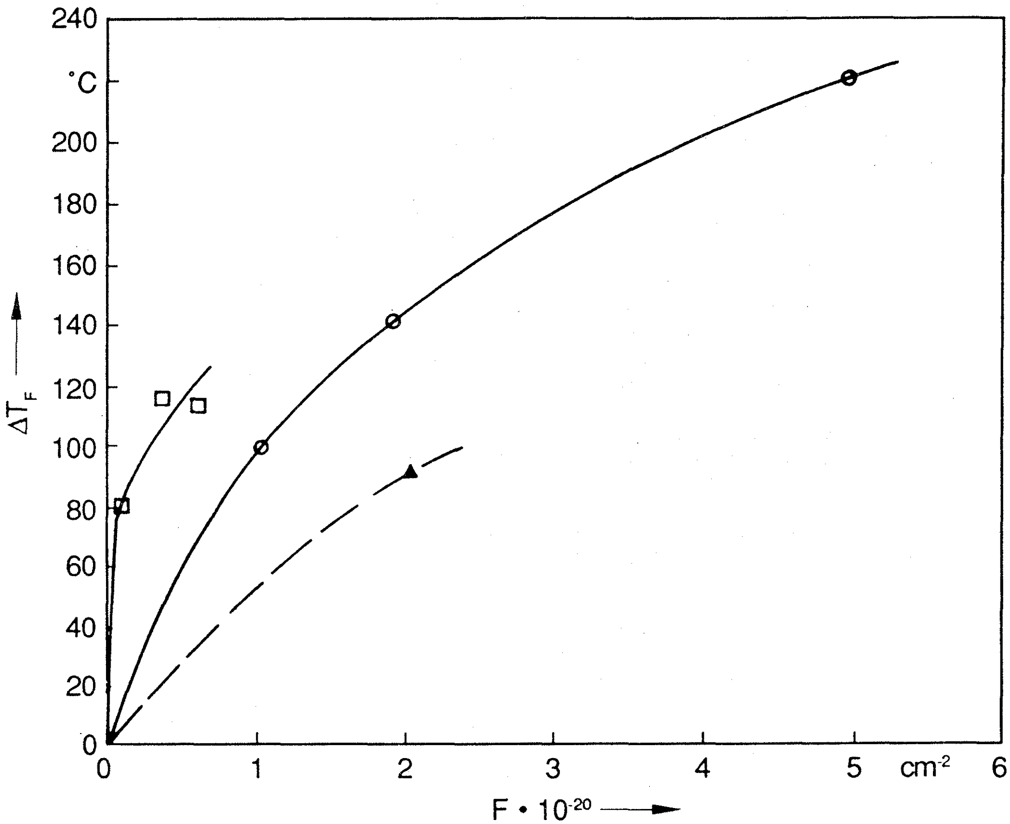
Fig. 3-2: Shift of the transition temperature of brittle fracture



T_F shift of the transition temperature
∘ welds
• steel 15Cr2MFA with different Cu and P contents
a) samples from NPP Rowensk
b) samples from Armenian NPPs

Source: Amajev,A.D., Krjukow,A.M., Sokolov,M.A.
Neutron embrittlement of RPV materials of the VVER-440 from experimental results of test samples.

Fig. 3-3: Irradiation embrittlement of weld material for an environmental temperature of 270°C



□ $\phi \sim 4 \cdot 10^{11}$ $cm^{-2} \cdot s^{-1}$

o $\phi \sim 4 \cdot 10^{12}$ $cm^{-2} \cdot s^{-1}$

Δ $\phi \sim 7 \cdot 10^{12}$ $cm^{-2} \cdot s^{-1}$

Source: Amajev,A.D., Vichrov,V.I., Krjukow,A.M., Sokolov,M.A. Irradiation embrittlement of materials of the VVER-440 pressure vessels.

Damage to the boiler tubes is highest in the steam generators of units 1 and 2. Up to now, 1.6% of the tubes of unit 1 have been closed, and 1.5% of the tubes of unit 2.
In units 3 and 4 the fraction of closed tubes is smaller, about 0.5%.
The examinations made up to now show that the typical crack is the longitudinal crack which in case of wall penetration causes only a small leak rate during operation. According to this damage type, appearance of larger leaks in steam generator tubes is to be considered rather in connection with pressure relief on the secondary side. It can be assumed, that a similarly big quantity of tubes comparable to that already closed, is damaged to leak, at differential pressure.
Assuming equal distribution of the crack lengths in the range up to 25 mm, where only a small allowance exists till the critical crack length is reached, one obtains a mean leak area of about 0.6% of the tube cross section. For units 1 and 2 one would obtain from this consideration leak cross sections of 8 and 5 cm^2 respectively, if the secondary side is completely depressurized. For the units 3 and 4, the values would be about 1 cm^2.
In order to estimate the potential for such leak events, also for leaks in the collector region, careful analyses are necessary taking into account the reliability of valves. Here also is to be taken into account that fast block valves were backfitted already in units 1 and 3 between steam generator and steam collector. For the units 2 and 4 this installation is foreseen for the next shut-down period.

For the leak events described above, water at a temperature of about 55°C is fed into the cold legs of the primary circuit before and after the main circulation pumps, from the emergency boron system. The backfitting for feed into the hot legs is part of the planned backfitting measures. For the assessment of the effects of the feed-in of cold water on the loading of postulated cracks in different regions of the RPV, a thorough comparison of the methods of analysis has to be made, in order to see in how far the statements on thermal shock analysis are reliable. The cracks postulated for the analysis of these vessels have to be discussed in the context of the methods applied described above for the examination of surface cracks.

3.4 Preliminary valuation

Regarding the limitations mentioned above concerning lack of information and problems still to be solved and analyses of details to be made, the following qualification can be made:

- State of materials of the reactor pressure vessel.
  For the RPVs of units 2 and 3, a high value is expected for the transition temperature for brittle fracture of the weld close to the core (170 to 200 °C).
  Considering the imposed supplementary examinations for the estimation of the uncertainties, an immediate solution is necessary. For unit 4, the shift of the transition temperature may be supposed to be small up to now. The examinations of the samples taken from the RPV of unit 1 have to be executed immediately in order to get additional information to evaluate the success of the heat treatment.
- The results available for the RPVs.
  The RPV of unit 1 was inspected in 1987; the annealed region was reexamined in 1988.
  The examination of unit 2 was made 4 years ago (1986). According to table 3-4 the examined surfaces of the RPVs of the units 1 and 2 have no flaws. Unit 3 was examined in 1988 and unit 4 in January 1989. For units 3 and 4 there are limitations in the inspection results due to quality deficiencies of the platings.
- Possibility of leak events.
  The existing deficiencies of knowledge about operation experience and about the range and sensivity of leak surveillance have to be redressed without delay. Event sequences have to be analysed.
  According to the procedures applied in the Federal Republic of Germany and to the circumstances as they are known up to now, we would expect that for units 2 and 3 a temporary shut-down would be considered permitting to clear the questions raised and to execute the necessary inspections. For all units, the leak surveillance has to be amended without delay.

4. <u>Efficiency of emergency cooling and transient behaviour</u>

The behaviour of the pressurized water reactors WWER-440 during transients is characterized by a low power density in the reactor core. Besides, the WWER-440 has relatively big water volumes, on the primary side as well as on the secondary side, compared to other PWR types. After complete loss of heat removal from the system, a period of about 6 hours is available after scram, before the primary coolant evaporates to such an extent, that the water level in the system falls below the top of core. This relatively long period of tim can, in principle, be used for internal accident management measures, in order to reestablish the heat removal, before core damage can arrive. Such measures are e.g. the reestablishment of power supply or supporting measures from the neighbour unit.
The possibilities for such measures should be examined systematically. In connexion with appropriate training of the operating safety of these installations can be increased by these measures. For the determination of the frequency of transients

of reactivity and the assessment of the behaviour of related systems, evaluations of operational experience and detailed analyses of incidents are necessary.
The safety related equipment installed on the basis of the initial design principles for the reactors WWER-440 do not correspond to current safety requirements. In this context one should mention especially:

- The design basis accident is assumed to be a loss of coolant accident (LOCA) in the primary system with the breach of a service pipe with a lowest diameter of 32 mm. The reasoning for this is that all pipings up to a nominal diameter of 100 mm have flow limitation corresponding to an equivalent diameter of $\leq$ 32 mm.
- As there are no accumulators nor efficient low pressure injection systems, the complete breach of a main circulation line (nominal diameter 500 mm) cannot be controlled. For this case, early core melt-down has to be expected.
- For leak events, an analysis of related loads of RPV internals by pressure waves does not exist.
- Tube ruptures in the steam generator cannot be evaluated currently.

The backfitting of the WWER-440/W-230 units with emergency cooling systems for the control of leak events up to a double breach of a main circulation line (nominal diameter 500 mm) is in itself not a big problem from the the thermohydraulical point of view alone. Such backfitting measures have to be seen, however, together with the loads during a LOCA to components of the primary circuit, mainly the RPV, and to the confinenent.

5. Systems design of safefy relevant equipment

- Primary Circuit

Reactor Scram

The reactor has 37 control and shut-down rods, the so called automatic control boxes. They consist of a control rod and a follower fuel element and are moved by a gear by means of a rack and a motor. The control rods are connected to the rack by a magnetic clutch and are inserted within 12 seconds when disengaged. During the discussions of operational experience, two deboration incidents were reported. These incidents should be taken as an occasion to examine whether the existing protection criteria are sufficiently covering and diversified.

Main circulation pumps

The main in circulation pumps are encapsulated pumps without packing boxes, which do not have gyrating mass for a prolongation of the slowing down time. The slowing down time in case of loss of electric power is about 3 seconds. Every two of these pumps are powered by two directly coupled generators of the two turbine sets. Because of the long slowing down time of the turbine set, electrical power supply is guaranteed for about 100 seconds.

As design basis accident the maximum loss of three main circulation pumps within 3 seconds is foreseen. In case a singel main circulation pump is lost, there is an automatic switch-over to reduced power operation.
During start-up it is possible to power more than two pumps from the external grid. In this state there is higher probability of simultaneous loss of more than two pumps. Because of the interconnection of guaranteed power supply by the separated generators and by grid supply, also during full power operation a failure of several main circulation pumps cannot be excluded completely. An indication of the frequency of occurence is actually not possible. The simultaneous loss is important for the pressure load of the primary circuit. This case should be investigated in connection with a transient analysis and its expected frequency should be evaluated.

Pressurizer, pressure control

As distinguished from the installations in the Federal Republic of Germany, the pressurizer has an inlet pipe of a nominal diameter of 200 mm, besides the two connecting pipes towards the hot side of the main circulation loop of the same diameter, which is connected to the vapour volume of the pressurizer. A check valve in this piping is to prevent backstreaming of the steam into the hot main circulation line.
The additional duct for equilibration serves to realize pressure equalization in the system in case of possible pressure transients. If e.g. by quick shut-down of all main circulation pumps a strong pressure transient is created, the pressure equilibration between RPV and pressurizer cannot be ascertained only by means of the two connecting lines. The pressure is controlled by spraying into the pressurizer. For this purpose, water is taken from the cold leg after the main circulation pumps through a control valve. There exists an additional spray through the injection pumps, but it is now used only during cool down operation, because damage to the spray ducts was stated during normal operation. The reason was straining by temperature change, provoked by the additional spray. The frequency of operation of the two safety valves on the pressurizer, and thus the frequency that they stick open is not known actually.

- Emergency borated water injection system
Each service pipe to the main circulation loop contains a Venturi nozzle of nominal diameter 32 mm, which is to reduce the quantity of outstreaming water in case of a breach of an injection pipe.
The pumps of the emergency borated water injection system consist of auxiliary and main pumps, which are driven by an electrical motor via a gearbox. The pump motors in units 1 and 2 are weak and cannot cover the full capacity of the pumps at different inlet pressures.
It is, therefore, necessary to limit the injection quantity by a control valve into each loop, in order to prevent overload and overheating in the motor. The pumps and their gear boxes need an operating oil supply with back cooling, which

exist for every pump. All pumps are put together in a single room, without separation.
In the presented documents, the emergency injection system of borated water is given low reliability; that is why a backfitting is urgently needed.
The emergency borated water injection system is the only system for emergency cooling. Long term cooling by this system is not possible, because the system does not have heat exchangers, and a suction line from the primary circuit for circulation by the pumps does not exist. The decay heat must therefore always be removed trough the secondary circuit. In case of LOCA, leak water and injection water is collected in a sump and flows directly back to the emergency boron container. This container is preheated to 55 to 59 °C, in order to avoid influences of thermal shocks. In case of a LOCA, the water entering the container from the sump would heat the container up. By means of the spray system it can be cooled. This function is started automatically. The emergency boron pumps are designed for an inlet temperature of 75 °C. For the further investigations it should be examined, which inlet temperatures can be reached during longer operation of the injection in case of larger LOCA.

- Spray system
  Potassium lye and sodium thiosulfate are added to the spray water, in order to settle radioactive material, mainly iodine, in the confinement. The spraywater pumps are in the same room in the reactor building as the emergency injection pumps. The spray water heat exchanger is cooled by means of the essential service water system, which contains sea water. Leaks in the essential service water system or in the suction pipes to the emergency injection tank lead to flooding of the room and to a relatively quick break down of spray system and emergency injection system.

- Service lines to the primary circuit
  In the service rooms, safe electrical subdistributions are installed. They serve for a decentralized supply of consumers, among others the motor operated valves. In connection with further investigations it has to te examined in how far unduly open service lines of the primary circuit can lead to steam release into these service rooms, and in how far isolation possibilities are affected by loss of electricity distributions.

- Ventilation systems
  The ventilation dampers in the confinement exist only as single shutters. It could not be clarified, in how far these dampers are designed and suitable for the dynamic loads which can occur. This should be examined. Of special importance for the leak tightness of the confinement are the pressure relief valves, which open in case of a larger leak in the primary circuit and must close again from the radiological point of view. These relief valves have not been tested dynamically, as far as we know and they are not inspected periodically either. During closing tests by means of weight loading and at elevated pressures defects were found in the FRG.

- Intermediate cooling system
An evaluation of the safety significance of this system is actually not possible.

- Essential service water system
The system is strongly interconnected. The pumps are situated in the inlet building together with the main cooling water pumps. All pumps of a double unit are in a single room. Here exists a rather high probability for flooding, e.g. by maintenance faults, as operational experience also from plants in the FRG shows. Operational experience indicates problems by corrosion, mud accumulation and mussel growth in coolers and pipings. Corrective measures were tried by change of the operation modes and by supplementary cleaning.

- Emergency feed water system
There is an interconnection between main feed water and the emergency feed water system, mainly in the area of steam generator feed. Valves and pumps of both systems are close to each other in the machine hall. The emergency feed water system is also used for start-up and shut-down. As a consequence of the incident in 1975, an emergency piping was installed between the collectors of the double unit.

- Flesh steam station
Three steam generators are combined on one fresh steam collector. Each steam generator has two safety valves controlled by own medium by means of weight loaded pilot valves. On the collector there is a pressure reducing valve for relief into the atmosphere. Blocking of the pressure reducing valve is realized by blocking each steam generator. The two fresh steam collectors are connected by block valves. For the delivery of fresh steam in case of turbine scram there are two pressure relief valves which remove steam into the turbine condenser. The capacity of this pressure relief valve is 70 % of the nominal steam quantity. For cooling down of the whole installaticn there is a separate cooling loop on the secondary side. To that end, three steam generators are completely filled with water. Causing loadings on the primary circuit, possibilities and frequency of subcooling transients, which are a consequence of large steam releases through leaks or incorrectly open valves in the fresh steam system, ought to be investigated and assessed.

- Electrotechnical installations

Installations for internal needs
The electrical energy distribution and supply occurs both from the 220 kV and the 380 kV lines. Since short, generator switches have been added in the connections between generator and machine transformer. Automatic switching by the unit protection has actually been realized only for the units 1 and 2. The units 3 and 4 have to be switched manually. To this end, there is a double supply as well through the mains as through the reserve grid.
The reliability of this supply is considered relatively high by the operator. From operation experience, only one case

of a loss of electrical energy supply has been reported in the early life of the units, which happened during shut-down.
The reason were wire oscillations due to ice-loads, which are avoided meantime by distance pieces. This incident could happen only in case of very special climatic conditions. As an internal loss of electrical power, here was the incident of 1975, which was initiated by a wrong connection of an earth switch. It led to a major term loss of the whole grid for internal needs and also of major parts of the emergency power supply.

- Emergency power supply
  The two emergency power bus bars are fed by three diesel generators. The bus bars are, therefore, connected with each other. The third diesel generator joins, with a delay of 10 seconds, either a bar which was not supplied up to then or a preselected bar. There is a possibility that too many consumers are connected to one bar and that a single diesel generator is overloaded.
  There are system interconnections.
  The main bars are installed in the intermediate building between reactor hall and machine hall. As far as we could state during inspection, there is no effective separation. The lack of electrical functional segregation and separation is regarded as a subject of backfitting.
  As could be seen from what was said during the presentation of these systems, the reliability of the preferred electrical power supply is limited. The cause is, among others, that there are special conditions when switching motor-generators on or over between direct current power supply and alternating current supply. These motor-generators must supply the direct current bars with direct current during normal operation and take over the supply of alternating current for the preferred current bars in case of loss of internal power supply. Per double unit there are three batteries; one battery supplies one unit, the third battery is attached to both units. They supply also process systems, e.g. the turbine. The total capacity is sufficient for about half an hour. This time appears to be short compared to the possibilities of accident management measures.

- Emergency shut-down system (reactor protection system) instrumantation and control
  The emergency shut-dcwn system has several levels which differ in the speed of intervention. The whole installation is constructed in relais-technics. In the documents there are only few details. The system has partly two trains in a two out of three arrangement. Which criteria are applied for shut-down is only partly known. The scope seems, however, lower than in installations in the FRG. There is no criterion "distance to departure from nucleate boiling" nor is there a scram due to "low level in steam generator". This latter criterion is of safety significance.
  The operator does not judge favourably the reliability of the control and protection system as a whole. During backfitting a complete replacement of this equipment is envisaged, also due to age and lack of spare parts.

- Preliminary assessment
  The spatial layout of the whole plant appears narrow and unfavorable concerning spatial separation. For this reason pipings and cables are passing connecting floors and throught the rooms.
  Segregation of ducts to single components of different redundance or loops does often not exist. Power lines, control cables and protection cables are apparently assembled on the same cable train, partially even on the trays. Repair work in incident conditions is difficult.

  On the building damages were visible, e.g. leaks into the building from outside and perviousness through walls. Effects following from this are to be investigated.
  The different systems are strongly interconnected. Most of the safety systems are not designed against single failures. In some cases this can have considerable consequences, as the incident of 1975 demonstrates, when the incident spread towards unexpected areas because of existing interdependencies and insufficient spatial segregation.
  The control room itself is constructed according to technical standards of 1950/60. It contains a very limited number of components with remote control. A continuous registration of messages was not observed. The recognition of incident initiation and the reconstruction of incidents may be difficult depending on the sequence and the development of counteractions may possibly be impeded.
  The possibilities for functional tests do not exist to the same extent as in the FRG. For this reason, limitations of availability of parts of systems have to be accepted during functional tests. Due to the kind of measures necessary for repair and tests with the interconnected systems, there is a special potential for human errors with incident initiating consequences. For further assessment of the plant, the operational experience has to be taken into account.
  The safety systems of a double unit the can be used for the operation of each of the units alone. By this, the degree of redundancy is increased. There are possibilities to limit disadvantageous effects of interconnection by fixed attribution sub-systems. In this way of operation safety is increased. Considerations on emergency measures for the feed of the steam generators and for the essential service water system can further improve safety.

- Backfitting measures foreseen
  Backfitting measures have been proposed for the different safety systems. The aim of these measures is on the one hand to broaden the control of accidents and on the other hand to reach a more or less consequent two train redundancy. A fraction of the backfitting shall be housed in the existing buildings, others in new buildings.
  A position cannot yet be taken with respect to these measures. Given the situation of the plant, also concerning the buildings, it seems reasonable, however, to install the complete backfitting into a separate building as a self sufficient unit. Then the systems of both the primary and the secondary side should be included. Possibilities for long term heat removal in case of a

leak ought to be included. The instrumentation and control and surveillance of all systems ought to be completely renewed and improved.

6. Confinement as safe containment

When assessing the confinement as safe containment for the main components of the primary circuit (described in section 2), one has to distinguish between

- the retention of radioactive materials during power operation and
- the retention of radioactive materials in case of LOCAs.

Concerning the retention function of the containment during power operation one has to consider its high leak rate. The admissible leak rate of the pressure resistent room system is, related to a test pressure of 125 kPa, 3600 m^3/h. This corresponds to about 600 volume percent a day. Measured values are between 1200 and 2600 m^3/h. This means, the retention of radioactive materials from contamination of the included air is not realised by the barrier function of the containment but is caused by maintaining an under-pressure (-0.2 to -0.3 kPa in the controlled areas) by means of ventilation systems and rejection through aerosol filters and active charcoal filters (degree of retention > 99%) towards the stack (height 100 m). Hence the function of the special ventilation system W2 (for the inaccessible areas) is important concerning the minimisation of the release of radioactivity to the environment. As the reasons for the high leak rate are known (weaknesses at doors, cable penetrations and shaft penetrations of ventilation systems) and are apparently not caused by leaks in the liner (6 mm steel plates), leak reduction measures seem possible. From the radiological point of view, however, the retention function of the safe containment during operation is less important than during accidents.

Concerning the retention of radioactive materials in case of a LOCA one must state that the design concept of the plant (assumptions of fracture, design of the safe containment) is completely different from that in western nuclear power plants. A containment comparable to western plants, which guarantees safe containment after large leaks (double breach of the primary pipings), does not exist for plants of the type WWER-440/W-230. In the following we have to distinguish between the design basis accident (breach of a service line diameter 100 mm with a flow limitation of equivalent diameter 32 mm), an accident with the breach of a service line of diameter 200 mm and beyond design basis leaks.

The efficiency of a safe containment for the retention of radioactive materials at a breach of a service line of diameter 100/32 mm depends on

- the reliable operation of the spray system (pressure reduction) and
- the leak rate of the system of pressure resistant rooms.

After loss or partial loss of the spray deluge system the pressure relief valve will open (at a pressure of 1.6 bar) with release of cooling water activity to the environment. But also when the spray deluge system operates correctly and the relief valve stays closed, the high leakage rate results in release from the containment into the surrounding rooms, from where it passes to the environment via the ventilation system (partially equipped with filters) and the stack. An improvement of the retention in case of this DBA is possible by reduction of the leakage rate and optimization of the spray deluge system. Such measures are part of the backfitting proposals made by the operator. A radiological assessment of the release of cooling water activity for different conditions (e.g. opening of the relief valve) has not been made at this first assessment.

In case of the DBA, according to SAAS, limiting values of radiological exposure in the environment will not be exceeded due to the admissible leakage rate.

After a breach of a service line of nominal diameter 200 mm, 8 other bigger relief valves will open at 1.8 bar, even if the spray deluge system operates correctly. From the radiological point of view, the moment at which damage to fuel elements occurs and the amount of this damage are important, but also the question, whether and when the relief valves close again. These conditions have to be analysed further.

Larger leaks represent beyond design basis events. For such a case it has to be stated, independently of the efficiency of the emergency cooling system, that the building structures of the system of pressure resistant rooms cannot stand the loads generated during pressure buildup (even if the relief valves are open), according to calculations made in USSR and by the operator.

Reflexions about possibilities to reinforce the pressure resistant rooms against breach of a primary circulation line with wet condensation in an auxiliary building lead to necessary overflow cross sections of 25 to 50 m^2, which seem difficult to realize, even without the problems of pressure difference in these rooms.

The backfitting deliberations of the operator, therefore, aim again to support the assumption that a breach of the primary coolant piping can be excluded; a breach of a service pipe equivalent to a nominal diameter of 200 mm can be controlled (condensation equipment) in a way, that retention is guaranteed so that exposure (inhalation, thyroid) stays below 0.3 Sv for an individual at a distance of 1500 m. An assessment to such a procedure could not be made for the time being.

7. Consequences of global and external events

Hereby we understand events which can affect large areas of the plant overlapping redundancies and systems. Such events can produce mechanical or thermal loads on structures, components and systems or flood of plant areas. In the following we distinguish internal effects by fire, flooding, failure of components in the turbine hall (pressurized parts, turbines) and external events as earthquakes, floods, lightnings, airplane crash and pressure waves due to explosions outside the plant. In assessing such events, their frequency of occurrence under the special situation of the plant or the site is an essential characteristic. For the present assessment, quantitative data of the frequency of events were generally not available.

As the type of impact of the different events is different and as the possibilities for improvement are of diverse nature, a differentiated look is given to the events in the following: Terrorism and sabotage are not considered.

7.1 Internal global events

7.1.1 Fire

From what we saw during our visit and from discussions with the operator and staff members of the SAAS, the following weakpoints of fire protection can be given:

- From the concept, segregation of redundant systems and cable connections for fire protection are missing in nearly all areas of the plant, expect e.g. the emergency Diesel building.
- Structural fire protection measures in the individual control rooms are missing, including the respective cable distributions and electrical installations from where all safety related functions start, in case emergency control rooms do not exist.
- Arrangement of major fire loads (lubricants) close to important safety related systems (e.g. feed water supply including emergency feed water supply) in the turbine hall, which is a coherant open building of about 1000 m length and in which the turbinegenerators of all units are located longitudinally.

Compared to actual requirenents of fire protection for new plants in the FRG (of KTA 2101) the following essential differences exist, apart from the fact that segregation of safety relevant systems and cable connections for fire protection is missing:

· Fire protection measures in the buildings

Consequent separation of building parts into fire areas and fire fighting sections with fire resistant walls is missing for the whole plant.
Where there is separation in the buildings the fire resistance is undetermined. Capability of fire resistance, as required in KTA 2101, e.g. 90 minutes for fire walls, does not exist, especially in the area of doors, cable and pipe penetrations and penetrations for ventilation. In the emergency Diesel building, there are spatial separations, according to the operator, between the Diesel generators and the fuel storage containers, and there exist also additional CO_2 extinguishers Punctual backfitting neasures were taken to protect some redundant cable connections (sand bed, asbestos plates, cable compartments, but their fire resistance capability is not known. These protection measures were taken after the fire incident in 1975.

- Fire protection measures in the plant

A fire alarm system with automatic fire alarm units exists for essential areas of the plant, with exceptions (the turbine hall). The areal density of fire alarm units is not yet known. In several areas with safety related systems (e.g emergency cooling system, feed water supply, service water system), there are no automatic fire alarm units, as far as we have seen at our visit. Automatically fire suppression systems exist only in very few areas of the plant, e.g. CO_2 systems started by melting solder in the emergency Diesel building. Water deluge systems exist in certain cable trays and cable rooms and at certain transformers and oil containers. The start of these systems is executed manually by the fire brigade on site. There is a ring main for fire suppression with hydrants on site and with ascending pipe lines and hydrants on the walls in staircases. The supply of water for fire suppression is not backed by emergency power. Measures for pressure increase (e.g. for the improvement of fire supression on the roof of the turbine hall) are under way.

- Operational fire protection measures

There is a fire brigade specially trained for fire suppression on the site. About the number of people and their equipment information is still missing. We may assume, however, that fire suppression on site is mainly done by this fire brigade and not by operating personnel. Fire existinguisher of different kind (water, powder, halon) can be found at several places in the plant. Their number appears to be small, corresponds however to GDR standards. Concerning examination of fire protection equipment during the construction there is no information available. Periodical tests are carried out according to conventional standards of the GDR and are made at least once a year in the frame of a revision.

- Preliminary assessment

For an assessment of fire protection of the plant, partial requirements according to the KTA-rule, especially concerning the priority of structural measures and the separation of redundant components of safety systems cannot be decisive criteria. As for older plants in the FRG and in other western countries, the assessment should in such a case be oriented towards the objective, as the structural situation does not allow a strict application of the KTA requirements, e.g. concerning spatial separation. Missing structural measures are compensated in older plants in the FRG and in other western countries by

- a complete surveillance of all safety relevant areas of the plant by means of automatic fire detectors, which guarantee fire detection in the very beginning,
- increased use of fixed fire installations for early fire fighting (water deluge systems, CO_2, halon) if possible, by automatic start or start from the control room,
- the existence of emergency systems, separated with respect to fire protection, including the emergency control room. These systems are used in case of failure of redundant safety systems due to a fire in the plant.

Comparable measures exist only partly in the units 1 to 4. It cannot be excluded, therefore, that a fire initiades incidents, in consequence of which the evacuation of decay heat is endangered. Due to a fail safe function, failure of the scram system has a low probability. Transients caused by redundancy overlapping fires with loss of feedwater supply including emergency feed water, of the supply of essential service water (safety related auxiliary cooling water) and of the intermediate cooling circuit have to be considered. If there is a fire in a cable distribution of the electrical power supply or close to switching stations, such failures can expand to other systems and can cause the loss of the control room or of parts of it (instrumentation, control). As long as such failures lead to transients without primary leaks or without pressure relief on the secondary side and the turbine hall is not affected, there are real possibilities to control these transients by accident management, due to the large water volume of the steam generators, for instance by supporting of the emergency feed water supply from other units by means of existing pipelines (up to 6 hours after the beginning of an accident). Primary leaks can occur, however, if safety valves do not close again after opening by the transient. In case of an incident "primary safety valves stay open", accident management measures are successful only, if the emergency cooling systems are not impaired (cf. incident of 1975). In case of fire initiated transients with secondary depressurisation (e.g. because valves are not in correct positions due to fire damage) or of overlapping fire in the turbine hall, the possibilities for accident management measures are probably very much reduced.

Due to the situation of fire protection and the discussed consequences, backfitting of the fire protection system is absolutely necessary.

Extent and kind of this backfitting depend on the concept of the complete backfitting programme and on the intended operational life. Due to the existing structures and the plant design, it is preferable, from a fire protection point of view, to build a separate emergency system (feedwater supply, eventually safety significant functions of emergency cooling, emergency control room, reactor protection with instrumentation and control, eventually power supply), which should be placed into a supplementary building. This system ought to be integrated into the existing plant at a suitable place in such a way, that the backfitting of the fire protection is performed in a few areas in the existing buildings. The installation of such an emergency system is reasonable only if the construction time is short compared to the expected remaining life time. If a separate emergency system is not installed, a detailed investigation in the plant is thought to be necessary, with the elaboration of specific fire protection measures (e.g. extension of automatic fire detectors and of stationary fire suppressing systems, some structural separation). Until an emergency system is installed or until specific backfitting measures are operational, a program of ad hoc measures should be prepared, in order to reduce the fire risk during this period. Such measures could be post firemen to extend inservice inspections with shorter intervals, among others at oil conducting systems, and to elaborate fire specific emergency procedures (partly already existing, prepared by the operator).

7.1.2 Internal flooding

It cannot be excluded that after failure of water filled pipelines with safety related functions in several areas of the plant can be flooded in a redundancy overlapping way. This is especially possible in the pump building. The coolant pumps of two units and the pumps for supply of essential service water (safety related auxiliary cooling water) are installed at equal height in a building in such a way that in case of a large leak of a coolant pipe, all pumps can be flooded. Possibilities of flooding in the area of the equipment building (reactor building) have not been investigated. On the whole, a deeper investigation of possibilities of events and incident scenarios with internal flooding seems to be necessary.

7.1.3 Failure of components in the turbine hall and related impact on safety relevant areas of the plant (missiles, pressure wave)

In case of a turbine failure, high energetic fragments can be formed, which could destroy components in their trajectory. Therefore, in western countries turbines of nuclear plants are situated in such a way that safety related components and equipment are not located in the region attainable by turbine fragments. The disposition of turbines in the turbine hall is unfavourable as far as risk to safety related systems and to the reactors is concerned. As thick walls were erected in front of the piping of the primary circuit, damage to it by missiles is considered unlikely. Safety relevant systems in the machine hall, in the pump building and in the swith gears are endangered, however. There are no data on failures of turbines of the type used here. We think it is necessary to consider the event "turbine failure" in the frame of a probabilistic safety analysis. This is also true for failure possibility of containers with high energy potentional, e.g. HP-preheaters, degassers (feedwater tanks) concerning the effects of a shock wave and the effect of missiles.

7.1.4 Global impacts deriving from events occuring in other units

As important events, consideration of fire, interconnection of systems and radiological exposures due to release of radioactive materials in other units is necessary.

Impacts of fires and system connections were mentioned already in sections 7.1.1 and 5. Concerning radiation exposure of operating personnel in case of a release of radioactive material into other units, precautional measures related to persons have been prepared. In the frame of the backfitting proposals made by the operator, this aspect will be taken into account.

7.2 External events

Due to the characteristics of the site and according to a first qualitative estimation of their frequency external events seem to be of no minor importance for the risk evaluation, compared to internal overlapping events and of system oriented incidents. In detail, the situation is the following:

- Earthquakes

The site of Greifswald is in a region of very low seismic activity. An analysis recently published (1988) shows that the maximal earthquake to be expected is of an intensity < class 5 (MSK) with a frequency of occurrence of about 10^{-4} per year. For this intensity, danger for the plant (also without special design measures) must not be feared.

- External floods

The cooling water for all units is brought in via an embanked canal. If the embankment fails, water is still available for safety related functions after shutdown of the plant (supply of essentiel service water).
Because disposition of the equipment houses (reactor buildings) and the turbine hall a risk from an external flood is unlikely for these areas of the plant. Safety relevant pumps in pump buildings could be endangered by extreme floods. About the frequency of occurrence of such water levels, no knowledge is available for the moment. The deliberations made by the operator for backfitting include a better protection of important pumps.

- Lightnings

Incident initiating effects from lightnings and possible scenarios were not considered in the first investigations up to now. Existing lightning protections are sufficient according to conventional GDR requirements. First reflexions concerning the influence on the frequency of the loss of electrical power, of the loss of emergency power and of instrumentation and control allow to recognise that it is necessary to consider these influences more deeply.

- Airplane crash

Since a few years it is forbidden to fly over the site covering a radius of 2 km and a height of 2000 m. There is a military airport close to the site. When the NPP was constructed, the landing runway was turned in such a way, that it is not directed towards the plant. Since short, data on the frequency of plane crashes are available to the SAAS. These data shall be used for a probabilistic risk analysis. Starting from the load assumptions for the design of NPPs against airplane crash, valid in the FRG, a protection of the reactor buildings of units 1 to 4 is not assured. Similar to older plants in the FRG without special design provisions, also here an assessment should be made in the frame of a PSA.

- Pressure waves from explosions

Pressure waves from explosions outside the plant are not expected on the basis of the site characteristics, according to actual knowledge. We have to consider, however, the possibilities of explosions on the site. Installations (pipes, stores) of the hydrogen supply of the generators do not present a problem according to the view of the operator. SAAS carried out an investigation on the energy and effects of an explosion pressure wave starting from the hydrogen store. In the frame of a PSA a further investigation seems due.

8. <u>Overall assessment</u>

In the frame of a stepwise safety analysis a first safety assessment of the units 1 to 4 of the nuclear power plant Greifswald was carried out. This first assessment concerns mainly the pressurised components of the primary circuit, especially the reactor pressure vessels. Besides, items of the safety design of the plant and of overlapping events were treated. An assessment of the significance of these single elements for the safety of the whole plant must be reserved to further investigations. As far as possible, investigations necessary for the further assessment are discussed.

The assessment is based on the insights gained from the visit of the plant on 25th January 1990, the discussions which took place on 25th and 26th January 1990 with SAAS and the operator and on further experts' discussions.
Documents were used only to a limited extent. Concerning the safety related equipment available in the plant, first examinations were performed. Compared to the requirements actually valid in the FRG, safety deficiencies are stated in nearly all areas, to a varying extent.

In how far existing deficiencies can be compensated by technical advantages of the plant (large water volumes available in the steam generators, which can be used as safety reserves for emergency measures), by short term safety improvements and by organisational and administrative requirements cannot be judged for the moment.

Of basic significance for the safety assessment is the question, whether the integrity of the primary circuit can be guaranteed with a sufficient safety margin for further operation of the units. The assessment considers, therefore, mainly the pressurised components of the primary circuit, above all the reactor pressure vessel.
The area of the RPV close to the core of unit 1 had been annealed and inspected in 1988. During the periodic inspection, flaws were not detected. According to the state of the art, the procedure of heat treatment is in principle suitable to anneal loss of ductility of the material incurred during operation. The annealing of unit 1 was valuated by SAAS in a sense, that a sufficient degree of annealing was reached. The extent of recovery of the ductility by the heat treatment is, however, not yet completely proven. Further quantification is reached by the ongoing complementary analyses of samples taken. These analyses have to be carried out with no delay. Due to the actually assumed values of the transition temperatures to brittle fracture of the welds of the reactor pressure vessels of units 2 and 3 and considering the uncertainties in material data, inspection results and possible initiating events, an interruption of operation is to be recommended, in order to clarify the open questions.

For unit 4, the brittle fracture transition temperature of the weld close to the core is lower due to shorter operation time.

An operation of the units 1 to 4 subject to certain conditions would require that sufficient precautions are taken against the specific vulnerabilities of the plants, e.g. by fire. As the assessment shows, a shut-down of units 2 and 3 can make available safety installations of these units for the units 1 and 4 to improve their safety. Details for this complementary safety support have still to be examined in detail. Furthermore, short-term backfitting of instrumentation and control systems are thought to be necessary, especially the installation of a system for leak surveillance.

The examinations continue. Results will be given in further reports.

APPENDIX 1

Comments by Soviet Experts
to the First Interim Report
on the NPP Greifswald

COMMENTS BY SOVIET EXPERTS TO THE FIRST INTERIM REPORT ON THE NPP GREIFSWALD

Proposals and supplements to the "First interim report on the safety assessment of the NPP Greifswald, units 1 to 4
(WWER -440 / V230)
Cologne, 15th February 1990

1. Chapter 2 "Description of the plant"

1.1 The diagram of the primary circuit is out-of-date. Actually e.g. pure condensate and boron are not supplied directly into the RPV, except the circuit for heating/cooling of a single loop.

A diagram has to be added, which corresponds to the actual status of units 1 to 4.

1.2 In section 2.3, after the definition of the maximum design basis accident, add:

"In the course of accidents up to the most severe design basis accident, a reduction of heat transfer is not assumed at the hot spot of the fission zone, taking into account possible deviations of parameters and lack of precision of the calculation formulae."

1.3 The fifth paragraph of section 2.3 should be formulated: "The warrant for safety is the utilisation of a system of pressure resistant rooms (boxes) with flaps, which are designed for the breach of a piping of 200 mm nominal diameter or for the designed leak of 32 mm nominal diameter in the case of deficiency of the spray system."

1.4 Section 2.4, point "Buildings - system of localisation"

After the sentence "Between the unsurveyed (and also the surveyed) rooms and the atmosphere, an under-pressure is to be maintained by ventilation systems" add: "An under-pressure of 200 mm water column in the steam - generator box and of 5 mm water column on the main circulation pumps at nominal power of the NPP is maintained by the ventilation systems W-2, P-4, W-4. In case of an incident, the ventilation is stopped when the differential pressure reaches 30 mm of water column, the localisation valves are then closed."

1.5 In section 2.4, point "Electrical energy supply", paragraph 3 "Emergency power system" replace the second and third sentence by "The system is designed such, that two lines do not cross. The third reserve Diesel generator is switched on one of the sections only (on

which the operating generator was not switched).
The calculated load of the emergency section is not higher than the permissible load of the Diesel generator (DG).
The power of the batteries is calculated for half an hour.
In practice, the discharge is not used longer than 45 seconds (time needed to start the DG).
After start-up of the DG, the reversible motor-generator switches over and the supply of alternate current, which is guaranteed by the battery in case of a loss of electrical power, is furnished by the DG and the batteries are switched over to charging."

In figure 2-4, the positions of the switches for the conduct from the generator to the transformator of the unit should be indicated. For the given position, the rest energy of the main generator may not be utilised, what contradicts the concept of the project.

1.6 Section 2.5 "Comparison of safety levels", point "Main differences compared to world level"

In Paragraph b) cancel "and a long-term emergency cooling".

For the design basis accident (breach nominal diameter 100 mm, limitation to 32 mm diameter) in the project, long-term cooling of the fission zone is assured by feed of borated water by means of the emergency injection pumps; heat is removed via the heat exchangers of the spray system.

Paragraph d) should be cancelled, as for such a conclusion the reasons are not given.

2. Chapter 3 "Analysis of the state of the pressurised components of the primary circuit"

2.1 Section 3.1

The last sentence should read: "The trade-mark composition of the materials used for the RPVs for WWER-440 given in table 3-1."

2.2 Section 3.2

Replace the sentence "According to the state of the art, the recovery of the ductility by annealing is not yet proven quantitatively" by "On the basis of the actual knowledge, the degree of recovery of the critical temperature of brittle fracture after annealing is sufficiently investigated as a function of temperature and cool down time."

Corresponding materials were given to the GDR, according to the contracts for the annealing of the RPVs of units 1 to 3.

2.3 Section 3.3

Figure 3-3 is to be replaced by figure 1 in the annex "Reason of the influence of the neutron flux density on the resistance against embrittlement."

2.4 Section 3.4

The expected critical temperature for brittle fraction in the weld next to the fission zone of units 2 and 3 should be 160 to 165 °C instead of 170 to 200°C.

In the conclusion of the chapter, the sentence concerning the shut-down of units 2 and 3 is to be replaced by: "The measures agreed between the GDR and the USSR to warrant strength against brittle fraction, including annealing of the RPV of unit 2 in the planned refuelling period and that of unit 3 are to be realised in the coming years".

2.5 In the report, the real data for the wall material and for the weld in the fission zone of the RPVs of units 1 to 4 of the NPP North should be given. (The corresponding data are presented as information 8002.00.05.319D1).

2.6 In table 3.3, replace foot-notes 2 and 3 by "The coefficients of irradiation embrittlement were determined according to Russian "Standards for the calculation of the strength of equipment and pipings of power plants (Moscow, Energoatomisdat 1989), using the equation

$A_F = 800\ (P\% + 0{,}07\ Cu\%)$"

3. Chapter 5: "Assessment of the standards of the technological systems"

3.1 The section "Scram", "System of emergency shut-down" should be corrected and amended using the following information:

"The emergency protection of the reactor WWER-440 (type 230) exists as HS1, HS2, HS3, HS4 and is triggered at signals set beforehand for warning and for actions against accidents.

The scram, leading to a reactor in the subcritical state from all operational states, is realised by HS1. The time of injection of all absorber (boxes) of the emergency protection is according to the design 8,5 - 13 seconds. The integral effectiveness of all boxes of the emergency protection, supposing the most effective absorber to stick in the upper position, and the velocity for the displacement of the SUS-absorbers downward to the bottom stop, warrants the reliable elimination of the chain reaction and the conversion of the reactor into the subcritical state, starting from any state of the reactor.

It is guaranteed that the limit values of thermal load of each fuel element are not exceeded, guaranteeing that fuel and canning tubes stay intact.

The response of the driving mechanism of a SUS-absorber in regimes HS1 and HS2 occurs by loss of current to electrical motor and the electromagnet, which cuts the gear box of the motor from the mobile parts; after that the ARK-boxes fall down by gravity with the mobile parts of the driving mechanism.

The construction of the driving mechanisms is sufficiently reliable. During the whole operating time of reactors of this type (since 1971) not one failure of a mechanical part of the driving mechanism was observed in the 14 units.

The emergency protection reacts on the velocity of the increase and on the level of the neutron flux density, on emergency signals in case of a leak in the primary circuit, on switching off of more than two main circulation pumps, on reaching emergency levels of the parameters of the primary circuit, and others. A list of protections and locks exists in the NPP.

The emergency protection consists of two autonomous systems. The control of the neutron flux in each of them occurs in 3 measurement ranges (the source range, the intermediate range and the power range) the signals of which go to the reactor protection system.
The signals from the emergency protection concerning neutron flux, the technological parameters of the primary circuit, the shut-down of safety relevant equipment have 3 measuring channels. The emergency protection system works according to the "two out of three" principle.

3.2 Section "Main circulation pumps"

Design basis accidents are: "Sudden loss of electrical power of the motors of 2 main circulation pumps" (such a regime can arrive in case of a short circuit in one of the sections serving the main circulation pumps) and "station black out", if the power for the 4 main pumps comes from the generators for proper needs (each 2 supplied by 1 generator), and the power for the remaining main circulation pumps comes from the main generators with the probable loss of one power supply.

With 3 or 4 independent sources of electrical power for the main pumps (generators for proper needs which are connected to the axis of the main generators and the main generators themselves) damage to fuel elements can occur under the condition, that the axis of both turbine-generators is blocking. But even in this unlikely case, that all primary pumps are stopped, the pressure in the primary circuit will not rise beyond 16 MPa, according to calculation.

Combining the electrical supply of the main pumps from the generator of own needs and of a part of these pumps from the grid, the permissible power of the reactor is limited according to the "table of permissible states of operation". In this way cooling of the fission zone is guaranteed in an accident situation.

3.3 Section "System of emergency injection of borated water"

For the design basis accident (breach of a piping of 100 mm nominal diameter with flow limitation to 32 mm), long term evacuation of the residual heat is realised according to the design by compensation of the leak with borated water by means of the pumps of the emergency injection system, and by heat removal via the heat exchangers of the spray system.

3.4 Section "ventilation systems"

As dampers (systems for localisation) block valves with nominal diameter 250 mm are used, which are mounted in the walls of the pressure resistant areas. The dampers are designed for 1 kp/cm^2; their time of response is 7 seconds. They are powered by the emergency power supply with an interruption of only 1 second. The dampers are not redundant. In order to raise the reliability, we propose: Add 2 fast closing valves at the exhaustion in supplement to the existing dampers; at the air inlet leave the situation as designed, as this valve is always closed during operation of the unit at power.

3.5 Section "Emergency power supply"
cf. remark 1.5

3.6 Section "System of emergency shut-down (protection system of the reactor" cf. remark 3.1

The fact that the signal for engagement of the emergency protection after the criterion "margin to (boiling) crisis" is missing, is compensated for the different states of operation by the HS-signals generated in case of a temperature increase at the outlet of a loop, by switching off primary pumps, by pressure reduction and power increase.

4. Chapter "conclusions"

It should be added in the text, that the investigations of irradiated and annealed samples carried out in the USSR show 100% recovery of the upper shelf and 80% recovery of the impact resistance of the material.

Instead of recommending definite shut-down of units 2 and 3 for additional investigations of samples, write "the timely realisation of the measures agreed between GDR and USSR in order to avoid brittle fracture, including annealing of the reactor pressure vessels of the units 2 and 3, is a necessary condition for safe operation".

ANNEX

Reason for the influence of the neutron flux density on the resistance against destruction by brittle fraction.

The formula

$A_F = 800\ (P\ \% - 0{,}07\ Cu\ \%)$ (1)

is used in the determination of the critical temperature for brittle fracture of the metal of the weld of the reactor pressure vessel WWER-440.

This function was gained from experimental data from examinations of samples irradiated in nuclear reactors (WWER-440) and in research reactors for fluxes above $10^{12}\ n \cdot cm^{-2} \cdot s^{-2}$.

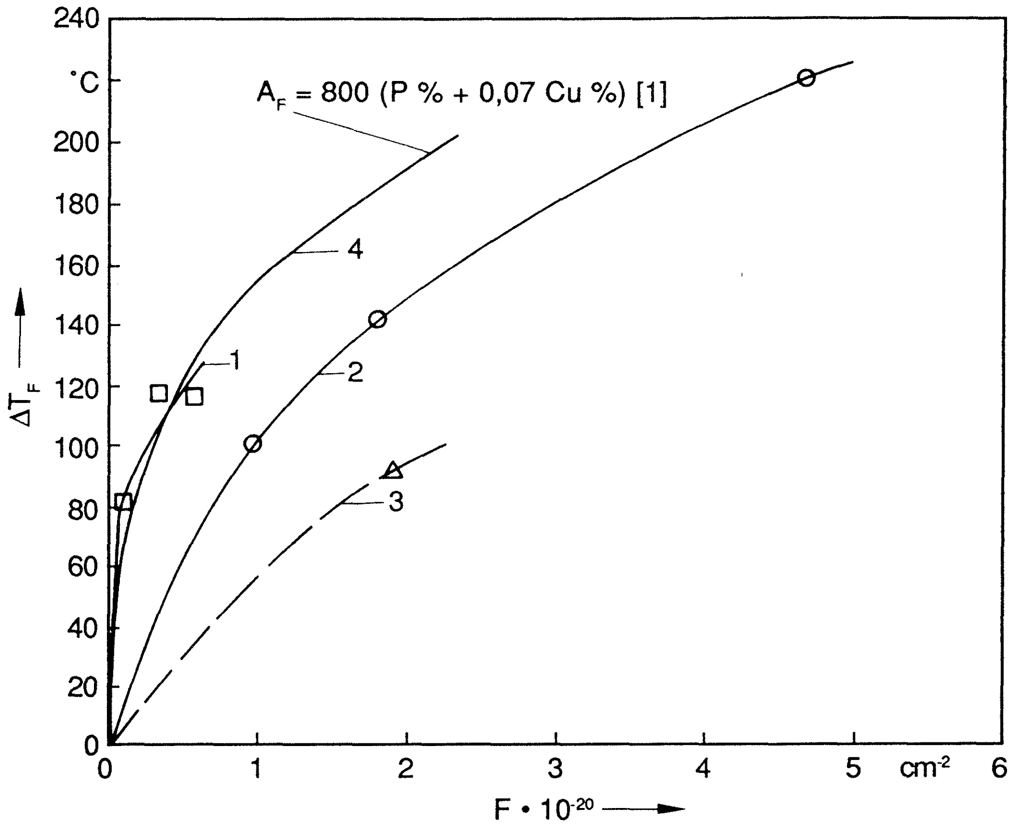
Recently test results were obtained from samples which had been irradiated in unit 1 of NPP Rovensk with shielding boxes at reduced neutron flux density (cf. figure 1).

The figure shows a comparison of experimental values of T_F obtained for different fluxes to the curve obtained from equation 1. This curve was drawn corresponding to the real content of metallic phosphorus and copper, i.e. 0,028% P and 0,18% Cu.

From the figure we can see that the data obtained for the irradiation of samples with a flux density of about $1 \cdot 10^{12}\ n \cdot cm^{-2} \cdot s^{-1}$ are clearly below the normal dependence (1) and that the data obtained for the irradiation with fluxes of about $4 \cdot 10^{11}\ n \cdot cm^{-2} \cdot s^{-1}$ coincide with the normal curve. We have to remark, however, that experimental values of ΔT_F by 10°C are above the normal curve in two cases. This deviation is compatible with the uncertainty of the determination of the critical temperature of brittle fracture of steel.
It is interesting to state that for the neutron doses of $3.4 \cdot 10^{19}\ n \cdot cm^{-2}$ and $5.10 \cdot 10^{19}\ n \cdot cm^{-2}$ equal values for ΔT_F were obtained:; the results for the irradiation of samples of $5.10 \cdot 10^{19}\ n \cdot cm^{-2}$ were situated below the normal curve. It is possible that this fact indicates a saturation tendency of ΔT_F as a function of the neutron fluency at a neutron flux of $4 \cdot 10^{11}\ n \cdot cm^{-2} \cdot s^{-1}$.

(1) Standards for the calculation of the strength of equipment and pipelines of nuclear installations, Moscow, Energoisdat, 1989, p.106

Fig. 1: Irradiation embrittlement of the weld metal, irradiated at a temperature of 270°



1 = 4 . 10^{11} cm^{-2} s
2 = 4 . 10^{12} cm^{-2} s
3 = 7 . 10^{12} cm^{-2} s
4 = the curve calculated from formula (1)

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8002.00.05.319 ДI

Information about characteristics of plates and welds in the fission region of the RPVs of the units 1 to 4 of NPP "Nord"

8002.00.05.319 DI

1990

В настоящей информации представлены данные по химическому составу и механическим свойствам обечаек и шва активной зоны корпусов реакторов блоков I - 4 АЭС "Норд".

This information presents the data on the chemical composition and mechanical properties of plates and welds of the reactor pressure vessels of units 1 to 4 of the nuclear power plant "North".

Главный инженер
Haupting.

Начальник ОТК
Leiter der HTK

Изм.	Лист	№ докум.	Подпись	Дата	8002.00.05.319 ДI			
Разраб.				14.03.9.	Информация Свойства обечаек и шва активной зоны корпусов реакторов блоков I-4 АЭС "Норд"	Лит.	Лист	Листов
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Утв.				20.03.9				

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Table 1: Chemical composition of the plates in the fission zone

Таблица I

Химический состав обечаек активной зоны

Element Элемент / Деталь Detail	Углерод C	Кремний Si	Марганец Mn	Хром Cr	Никель Ni	Медь Cu	Молибден Mo	Ванадий V	Сера S	Фосфор P
Норд бл.I (Altschmelze)										
III2.0I.0I.03I (плавочный)	0,I6	0,30	0,46	2,78	0,I8	0,I7	0,63	0,27	0,0I3	0,0I0
III2.0I.0I.032 (плавочный)	0,I7	0,25	0,43	2,68	0,I3	0,I3	0,70	0,28	0,0I8	0,0I2
Норд бл.2										
III2.0I.0I.03I (плавочный)	0,I6	0,28	0,44	2,6	0,I8	–	0,66	0,28	0,0I2	0,0I3
III2.0I.0I.032 (плавочный)	0,I7	0,25	0,47	2,5I	0,I7	0.I6	0.67	0,25	0,0I2	0,0I0
Норд бл.3										
II39.0I.0I.03I (плавочный)	0,I3	0,26	0,39	2,85	0,I5	–	0,68	0,I9	0.0I2	0,0I2
II39.0I.0I.032 (плавочный)	0,I5	0,23	0,4I	2,65	0,I3	–	0,64	0,28	0,0I3	0,0I0
Норд бл.4										
II39.0I.0I.03I (плавочный)	0,I7	0,27	0,47	3,00	0,I5	0,I2	0,63	0,29	0,0I4	0,0I6
II39.0I.0I.032 (плавочный)	0,I7	0,25	0,43	2.72	0,I3	0,I2	0,63	0,20	0,0I	0,0I

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8002.00.05.3I9 ДI

Формат А4

Table 2: Chemical composition of the weld N4

Химический состав сварного шва № 4

Таблица 2

Element Элемент Шов Naht	Угле-род C	Марганец Mn	Кремний Si	Фосфор P	Сера S	Хром Cr	Никель Ni	Молибден Mo	Титан Ti	Ванадий V	Медь Cu	Азот N	Мышьяк As	Алюминий Al	Вольфрам W
Норд бл.I Block 1															
Св-08А	0,09	0,38	0,0I	0,0I9	0,023	0,08	0,08				0,I2				
Св-I0ХМТФ	0,09	0,50	0,29	0,0I5	0,0I2	I.78	0,29	0;40	0,08	0,I9	0,I2				
Норд бл.2 Block 2															
Св-08А	0,I0	0,54	0,02	0,0I6	0,0I8	0,03	0,05				0,06				
Св-I0ХМТФ	0,08	0,63	0,I7	0,0II	0,0I3	I,50	0,I9	0,46	0,06	0,I7	0,I8				
Норд бл.3 Block 3															
Св-08А	0,09	0,45	0,02	0,0I0	0,025	0,06	0,08				0,II				
Св-I0ХМФТ ж Св-I0ХI6Н25АМ6	0,09	0,59	0,2I	0,0I4	0,0I3	I,69	0,I8	0,47	0,07	0,27	0,I2	0,0I2	0,08	0,05	0,06
Норд бл.4 Block 4															
Св-08А	0,09	0,4	-	0,0I	0,03	0,06	0,08				0I4			0,009	
Св-I0ХМФТ	0,08	0,56	0,I9	0,0I2	0,0I4	I,73	0,24	0,52	0,I0	0,2I	0,I6				
Св-I0ХI6Н25АМ6	0,II	I,85	0,32	0,024	0,0I5	I5,74	26,43	5,99			0,I9	0,I9			

Anmerkungen:

Примечания. I.Таблица 2 содержит сертификатные данные провслоки.

2.Облицовка шва выполнена, фактические данные отсутствуют.

3.Обозначение Св-I0ХМТФ или Св-I0ХМФТ принято на период изготовления.

I. Tab.2 beinhaltet Zertifikatangabe des Drahtes

2.Es erfolgte die Abdeckung d.Naht, die tatsächlichen Daten fehlen

3. die Bezeichnung des Materials wurde für die Herstellungsperiode

8002.00.05.3I9 ДI

Лист 4

Формат А4

Table 3: Mechanical properties of plates in the fission zone

Таблица 3

Механические свойства обечаек активной зоны

Деталь Detail	+20°C					+325°C				
	σ_B кгс/мм²	$\sigma_{0,2}$ кгс/мм²	δ %	ψ %	a_H тип I кгсм/см²	σ_B кгс/мм²	$\sigma_{0,2}$ кгс/мм²	δ %	ψ %	a_H тип I кгсм/см²
I	2	3	4	5	6	7	8	9	I0	II
Новд бл.I										
III2.0I.0I.03I (после закалки с отпуском) (nach Anlaßhärten)	7I,8 68,9 69,5	65,0 62,5 62,5	20,0 20,0 20,4	74,7 74,2 74,0	20,7 24,5 23,5 24,2 24,0 23,7	6I,4 62,5 60,0 59,3 58,8 6I,2	53,6 53,8 5I,4 5I,0 49,0 5I,5	I5,8 I6,0 I6.0 I7.0 I5,0 I6.0	7I,3 69,5 72,8 74,2 7I,0 74,0	3I,2 30,8 23,7 26.2 25,5 I8,8
III2.0I.0I.032 (после закалки с отпуском) (nach Anlaßhärten)	74,0 72,8 70,2	65,8 65,0 6I,2	2I,0 2I,0 I9,0	75,2 74,2 75,2	24,9 25,5 25,2 25,5 27,8 22,9	64,3 63,0 63,0 63,8 60,6 60,5	56,7 55,5 55,7 56,0 5I,5 5I,3	I6,0 I5,6 I8,0 I6,0 I6,4 I6,0	73,2 73,2 74,0 73,0 73,8 73,5	23,8 24,3 25,0 25,0 24,5 24,0

Продолжение табл. 3
Table 3 continued

I	2	3	4	5	6	7	8	9	I0	II
<u>Норд бл.2</u>										
III2.0I.0I.03I	62,4	52,5	26,6	76,0	30,8 3I,2	5I,5 5I,5	42,8 45,2	I8,8 I8,8	72,0 76,0	34,8 32,2
III2.0I.0I.032	58,0	47,0	22,4	74,5	32,8 32,8	50,4 50,5	4I,7 42,0	I9,0 20,0	75,2 75,5	36,2 36,2

Инв. № подл.	Подпись и дата	Взам. инв. №	Инв. № дубл.	Подпись и дата

Продолжение табл. 3
Table 3 continued

I	2	3	4	5	6	7	8	9	IO	II
						+350°C				
Норд бл.3										
II39.0I.0I.03I	65,0	53,7	2I,0	75,5	тип I 25,9 29,0 25,2 тип IУ 22,6 I7,4 22,2	58,0	49,3	I5,6	72,2	
	68,7	56,3	2I,2	75,5	тип I 25,5 27,8 27,8 тип IУ I9,8 I3,4 22,0	53,0	43,0	I7,6	73,2	
II39.0I.0I.032	65,7	54,0	22,2	76,2	тип I 28,5 29,9 29,9 тип IУ 25,9 27,8 25,2	54,3	46,5	I6,0	74,0	
	63,0	5I,3	22,8	78,3	тип I 33,3 33,3 30,8 тип IУ 27,8 29,9 27,4	53,3	45,0	I6,2	75,4	

Продолжение табл.3

Table 3 continued

I	2	3	4	5	6	7	8	9	I0	II
Норд бл.4						+350°C				
II39.0I.0I.03I	65,0	53,I	24,2	78,0	тип I 33,4 33,4 34,9 тип IУ I6,8 I7,3 I6,8	5I,5 5I,0	43,3 4I,7	I7,2 I6,6	76,0 76,0	25,7 28,3 3I,5
	64,4	5I,5	24,2	78,7	тип I 34,9 36,2 35,I тип IУ I8,0 I7,6 I9,I	50,8 5I,5	43,2 43,2	I5,8 I6,0	78,6 77,6	36,2 34,8 33,8
II39.0I.0I.032	64,4 63,0 63,5 63,5	52,5 5I,4 52,4 5I,4	23,6 23,6 22,0 22,0	77,2 77,4 77,4 74,4	тип I 35,8 35,0 3I,I 3I,I 35,9 35,4 тип Iу 30,5 29,6 3I,0 32,8 29,4	5I,2 50,7 5I,3 5I,8	43,5 43,6 44,4 43,5	I7,0 I6,0 I6,6 I6,6	75,5 75,8 72,5 74,8	

Table 4: Mechanical properties

Таблица 4

Механические свойства сварного шва №4

№ шва Nr. der Naht	Металл шва Nahtmetall +20°C				Schlagzähigkeit Ударная вязкость, кгсм/см²				Schweiß-verbindung +325°C				Сварное соединение +20°C
					по оси шва Längs-Schweißnaht		по зоне термического влияния im Gebiet des thermischen Einflusses						
	$\sigma_в$ кгс/мм²	$\sigma_{0,2}$ кгс/мм²	δ %	ψ %	тип VI Typ VI	тип IX Typ IX	тип VI Typ VI	тип IX Typ IX	$\sigma_в$ кгс/мм²	$\sigma_{0,2}$ Ein- кгс/мм²	δ %	ψ %	$\sigma_в$ кгс/мм²
I	2	3	4	5	6	7	8	9	I0	II	I2	I3	I4
Норд бл.I													
шов №4 Naht N4	59,6	52,7	20,6	60,3	I7,9				48,6	42,2	20,0	69,6	
	60,0	53,0	20,6	60,0	2I,6				49,7	42,7	20,0	66,6	
	55,4	46,3	26,0	72,0	I6,9				49,3	42,7	I9,0	65,4	
									+325°C				
Норд бл.2													
шов №4 Naht N4	53,0	44,2	26,6	72,0	23,8				47,5	40,3	I0,3	59,4	
	55,3	46,I	25,0	75,2	2I,0				45,0	34,8	2I,7	66,7	
	58,5	5I,5	25,0	66,6	I5,8				46,0	39,0	22,6	64,5	
	59,3	53,7	25,0	7I,7	27,3								
	54,4	48,I	25,0	73,5	2I,0								
	57,6	47,7	2I,6	72,0	I9,0								

Table 4 continued

Продолжение табл. 4

1	2	3	4	5	6	7	8	9	10	11	12	13	14
									+350°C				
Норд бл.4													
шов №4 Naht N4	62,2 64,0 63,2	50,5 51,3 49,8	24,0 24,0 25,0	72,5 71,8 69,7	11,5 13,3 16,3 16,3 14,5 12,9	3,4 5,5 3,8	17,6 21,6 18,6	17,6 20,4 17,2	52,8 48,6 51,2	45,1 43,0 41,8	19,0 16,7 18,4	64,0 78,5 64,5	61,6 61,0 58,5
Норд бл.3													
шов №4 Naht N4	65,0 63,8 65,0	51,4 51,4 53,0	22,7 24,0 22,7	66,4 70,7 70,2	15,9 17,3 15,5	6,7 7,6 3,0	27,1 25,2 30,2	22,0 20,6 21,0	51,7 51,5 51,3 51,3 53,3 54,8	39,8 40,8 39,0 39,8 42,7 44,3	19,3 20,6 15,7 19,0 18,0 19,3	68,0 66,5 64,7 72,5 67,0 66,5	58,7 61,8 61,8 61,8

Инв. № подл.	Подпись и дата	Взам. инв. №	Инв. № дубл.	Подпись и дата

Table 5: Critical temperature of brittle fracture

Таблица 5

Критическая температура хрупкости

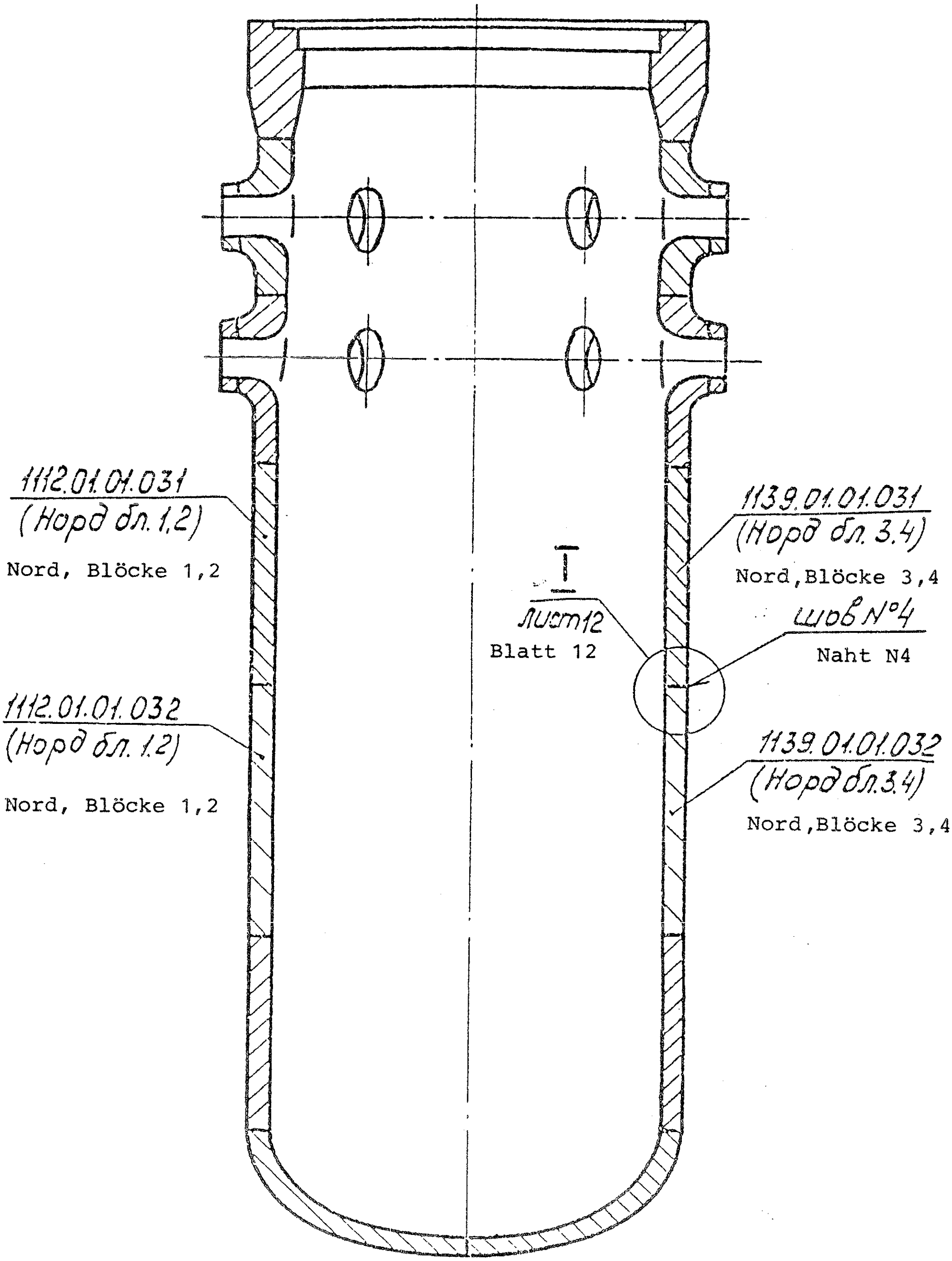
N des Details oder der Schweißnaht

Номер детали или шва	$a_н$	%	$a_н$	%	$a_н$	%	$a_н$	%	$a_н$	%	$a_н$	%	$a_н$	%	$a_н$	%	$T_{ко}$
Nord Block 4 Норд бл.4																	
II39.0I.0I.03I	-40°C		-30°C		-20°C		-25°C		+5°C		+I0°C						-20°C
	7,3 6,7 I,4	23 I7 5	8,I 6,I 3,0	23 I7 5	I5,5 5,4 7,6	46 I7 I7	8,8 I7,3 5,4	3I 69 I6	I8,3 3,0 20,6	90 I5 90	I9,0 I9,0 20,2	I00 I00 I00					
шов №4 Naht N4	-50°C		-35°C		-20°C		-5°C		+I0°C		+25°C		+45°C		+70°C		
	0,6 0,5	0 0	0,7 0,7	0 0	I,6 0,6	0 0	3,0 0,8	0 0	4,0 3,6	I0 I0	3,8 5,4	30 30	6,6 6,0	50 50	9,0 I0,8	90 90	

% - процент волокна - % der Faser

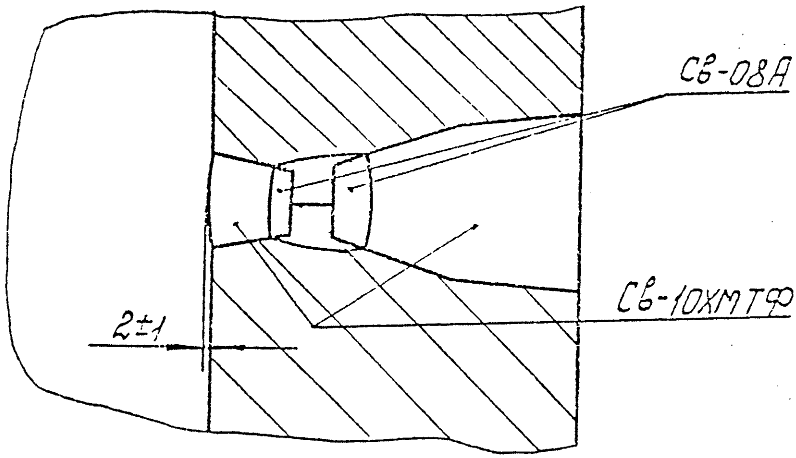
$a_н$ - ударная вязкость, образцы тип IX, кгсм/см^2
Schlagzähigkeit, Muster Typ IX

Корпус Reactor pressure vessel



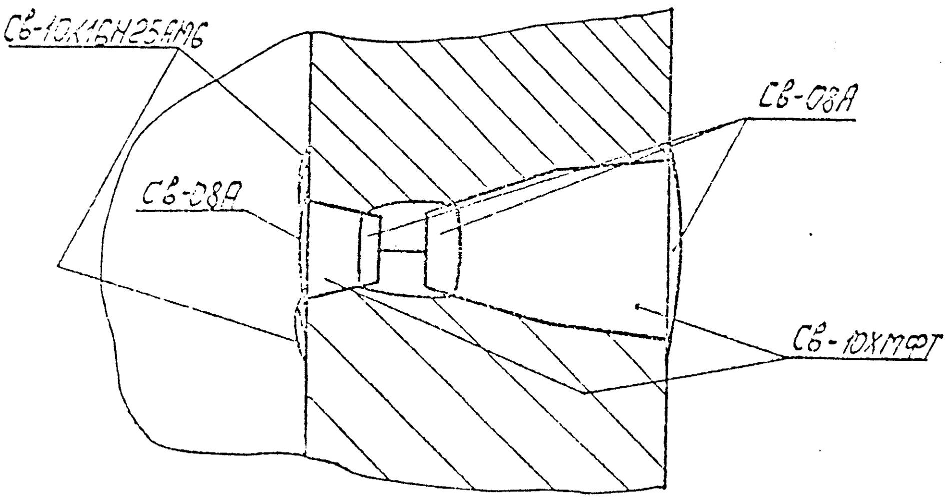
I (Норд бл.1.2), лист 12

Nord, Blöcke 1,2, Blatt 12



I (Норд бл.3,4), лист 11

Nord, Blöcke 3,4 , Blatt 11



На АЭС Норд бл.3,4 выполнена механическая обработка сварного шва заподлицо с основным металлом

In NPP Nord, units 3 and 4, the weld and basis material touch each other

APPENDIX 2

List of nuclear power plants in COMECON countries
in operation, under construction
or under development.

NPPs in the USSR

Pressurized water reactors with WWER

Site	Unit	Model	Power MWE	First operation	Remarks
Novovo-ronesh	1	WWER-200	210	9/64	decommissioned in 87
	2	WWER	365	12/69	decommission in 90
	3	WWER-440/W-230	440	12/71	
	4	WWER-440/W-230	440	12/72	
	5	WWER-1000	1000	5/80	
Kolsk (Kola)	1	WWER-440/W-230	440	6/73	
	2	WWER-440/W-230	440	12/74	
	3	WWER-440/W-213	440	3/81	
	4	WWER-440/W-213	440	10/84	
Armenien	1	WWER-440/W-230	440	12/76	decommissioned in 89
	2	WWER-440/W-230	440	1/80	decommissioned in 89
Rovensk (Rovno)	1	WWER-440/W-213	402	12/80	
	2	WWER-440/W-213	416	12/81	
	3	WWER-1000	1000	12/86	
Nikolajev Südukraine	1	WWER-1000	1000	12/82	
	2	WWER-1000	1000	12/85	
	3	WWER-1000	1000	12/86	
Kalinin	1	WWER-1000	1000	5/84	
	2	WWER-1000	1000	12/86	
Saporoshje	1	WWER-1000	1000	12/84	
	2	WWER-1000	1000	7/85	
	3	WWER-1000	1000	12/86	
	4	WWER-1000	1000	12/87	
Balakovsk	1	WWER-1000	1000	12/85	
	2	WWER-1000	1000	10/85	
	3	WWER-1000	1000	12/87	

Site	Unit	Model	Power MWE	First operation	Remarks
Chmelmizk	1	WWER-1000	1000	12/87	
<u>Boiling water reactors</u>					
Dimitroff-grad	1		50	2/65	
<u>Pressure tube reactors</u>					
Obninsk			5	6/54	decommissioned
KKW Sibiri-en Troitsk	1	RBMK	100	9/58	
	2	RBMK	100	1/59	
	3	RBMK	100	1/60	
	4	RBMK	100	1/60	
	5	RBMK	100	1/61	
	6	RBMK	100	1/63	
Belojarsk	1	RBMK	100	4/64	
	2	RBMK	200	12/67	
Bilibinsk	1	RBMK	12	12/73	
	2	RBMK	12	12/74	
	3	RBMK	12	12/75	
	4	RBMK	12	12/76	
Leningrad	1	RBMK 1000	1000	12/73	
	2	RBMK 1000	1000	7/75	
	3	RBMK 1000	1000	2/80	
	4	RBMK 1000	1000	2/81	
Kursk	1	RBMK 1000	1000	12/76	
	2	RBMK 1000	1000	12/78	
	3	RBMK 1000	1000	12/83	
	4	RBMK 1000	1000	12/85	
Tschernobyl	1	RBMK 1000	1000	9/77	
	2	RBMK 1000	1000	12/78	
	3	RBMK 1000	1000	10/83	
	4	RBMK 1000	1000	12/85	accident 4/86

Site	Unit	Model	Power MWE	First operation	Remarks
Smolensk	1	RBMK 1000	1000	11/82	
	2	RBMK 1000	1000	5/85	
Ignalinsk	1	RBMK 1500	1500	12/83	
	2	RBMK 1500	1500	8/87	

Fast breeder reactors

Site	Unit	Model	Power MWE	First operation	Remarks
Dimitroffgrad		BOR-60	12	12/69	
Schewschenko		BN-350	150	7/73	
Belojarsk		BN-600	600	4/80	

Heating reactors AST

Voronesh	1	AST	500_{th}	89
	2	AST	500_{th}	89
Gorky	1	ASt	500_{th}	90
	2	AST	500_{th}	90

23 NPP units with WWER 1000 are under construction.
The construction of 4 units RBMK was stopped.

NPPs in the GDR

Pressurized water reactors

Site	Unit	Model	Power MWE	First operation	Remarks
Rheinsberg	1	WWER	70	5/66	decommissioning in 92
Greifswald (KKW-Nord)	1	WWER-440/W-230	440	12/73	
	2	WWER-440/W-230	440	12/74	
	3	WWER-440/W-230	440	11/77	
	4	WWER-440/W-230	440	8/79	
	5	WWER-440/W-213	440	11/89	
	6	WWER-440/W-213	440	under construction since 80	
	7	WWER-440/W-213	440	under construction since 81	
	8	WWER-440/W-213	440	under construction since 81	
Stendal	1	WWER-1000		under construction since 84	
	2	WWER-1000		under construction since 84	

NPPs in Bulgaria

Pressurized water reactors

Site	Unit	Model	Power MWE	First operation	Remarks
Kozloduy	1	WWER-440/W-230	440	6/74	
	2	WWER-440/W-230	440	8/75	
	3	WWER-440/W-230	440	12/80	
	4	WWER-440/W-230	440	5/82	
	5	WWER-1000	1000	11/87	
	6	WWER-1000	1000	11/87	
Belene	1	WWER-1000	1000	under construction since 83	
	2	WWER-1000	1000	under construction since 85	

NPPs in Poland

Pressurized water reactors

Site	Unit	Model	Power MWE	First operation	Remarks
Zarnowic	1	WWER-440/W-213	440	under construction since 82	
	2	WWER-440/W-213	440	under construction since 82	
	3	WWER-440/W-213	440	under construction since 88	
	4	WWER-440/W-213	440	under construction since 88	

NPPs in Czechoslovakia (CSFR)

Pressurized water reactors

Site	Unit	Model	Power MWE	First operation	Remarks
Bohunice	1	WWER-440/W-230	440	4/79	
	2	WWER-440/W-230	440	1/81	
	3	WWER-440/W-213	440	5/85	
	4	WWER-440/W-213	440	3/86	
Dukovany	1	WWER-440/W-213	440	8/85	
	2	WWER-440/W-213	440	9/86	
	3	WWER-440/W-213	440	5/87	
	4	WWER-440/W-213	440	12/87	
Mochovce	1	WWER-440/W-213	440	10/89	
	2	WWER-440/W-213	440	under construction since 83	
	3	WWER-440/W-213	440	under construction since 85	
	4	WWER-440/W-213	440	under construction since 85	
Temelin	1	WWER-1000	1000	under construction since 84	
	2	WWER-1000	1000	under construction since 85	
	3	WWER-1000	1000	under construction since 85	
	4	WWER-1000	1000	under construction since 85	

NPPs in Hungary

Pressurized water reactor

Site	Unit	Model	Power MWE	First operation	Remarks
Paks	1	WWER-440/W-213	440	12/82	
	2	WWER-440/W-213	440	9/84	
	3	WWER-440/W-213	440	9/86	
	4	WWER-440/W-213	440	8/87	
	5	WWER-1000	1000	planning cancelled	
	6	WWER-1000	1000	planning cancelled	

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Nuklear-sicherheitstechnische Ziele und Bewertung der Rekonstruktion der Reaktorblöcke mit W-230
Vortrag, Arbeitsseminar Greifswald, 25. und 26. 1. 1990

SECOND INTERIM REPORT
ON THE SAFETY ASSESSMENT OF
THE GREIFSWALD NUCLEAR POWER PLANT
UNITS 1 - 4 (WWER-440/W-230)

CONTENTS

1. INTRODUCTION

As part of the collaboration with the German Democratic Republic in the field of the safety of nuclear power plants, the Federal Minister for the Environment, Nature Protection and Nuclear Safety (BMU) commissioned the Company for Reactor Safety (GRS) to carry out a safety assessment of the Greifswald nuclear power plant, Units 1-4.

The corresponding work is being undertaken in conjunction with the State Office for the Safety of Atomic Installations and Radiation Protection (SAAS) of the German Democratic Republic, with the participation of experts from the competent institutions in the USSR and the assistance of representatives of the Greifswald nuclear power plant. French experts from the Nuclear Safety and Protection Institute (IPSN) have taken part in various consultative meetings.

The GRS enlisted the services of experts from the Technical Supervisory Authorities (TÜV) of Bavaria, of Northern Germany, and of Northrhine-Westfalia as well as from the Stuttgart Materials Testing Institute (MPA) to handle special tasks forming part of the programme.

Work commenced in January of this year (1990). A comment on the safety-related problems to be investigated in the course of this work was submitted in a short term in a first interim report of 15.2.1990 and published. This comment was based on findings which were established during a working seminar (25 and 26 January 1990) held by the SAAS in the Greifswald nuclear power station, in technical discussions held following that seminar and from documents which were immediately available. The comment was principally concerned with an assessment of the pressurized components of the primary loop, and especially the available knowledge on the embrittlement of the reactor pressure vessels of the units caused by neutron irradiation. In addition, first estimates were made regarding individual aspects of the safety-related design of the installations.

Based on the assumed values for the nil ductility transition temperature of the reactor pressure vessel weld seams close to the core in units 2 and 3, and on the uncertainties regarding materials data and prior test results and analyses, a precautionary shutdown of units 2 and 3 was recommended in the first interim report.

The second interim report contains an overview of the status of the ongoing investigations. The report is principally concerned with the safety-related design of the installation and the evaluation of operational experience.

To assist in understanding the statements contained in the technical chapters (chapters 4-9), chapter 2 gives a brief description of the basic concept and the safety-related design of units 1-4, of the pressurized water reactors WWER-440/W-230 (first-generation WWER reactors).

Chapter 3 presents the principles, agreed by the State supervisory authorities of the COMECON countries in which reactors of the WWER-440/W-230 type are operated (Bulgaria, GDR, USSR and Czechoslovakia), relevant to backfitting and safe plant operation. Moreover, chapter 3 explains the measures which were additionally required by the SAAS for units 1-4 and by which it is intended to ensure the safety of the installation until implementation of the planned backfitting measures.

To carry out the investigations, the GRS and the SAAS, with the additional participation of further institutions, formed working groups on the following subjects:

- Pressurized components
- Accident analysis
- Confinement (pressure resistant compartment system)
- Systems engineering (process engineering, electrical engineering, instrumentation and control)
- Redundancy overlapping events
- Evaluation of operating experience

The working groups carried out detailed consultations and investigations of these problems. In the assessment of the pressurized components and of the corresponding materials problems, detailed discussions were held, also with Soviet and French experts.

Joint meetings of all working groups were held on 7 and 8 March and on 3 and 4 May 1990 at the SAAS in Berlin. These were attended by Soviet experts and, on 3 and 4 May, by French technical specialists as well. In addition, a discussion of the results of the investigations was held on 22 and 23 May 1990 with Soviet experts in Moscow.

At the present stage of this work, it is not possible to conclusively assess all the safety questions relating to units 1-4 of the Greifswald nuclear power plant. For a number of special questions, it is necessary to examine further documents. Because of the time available, it has not yet been possible to deal with a number of problems and topics; this concerns for example investigations aimed at assessing the strength of safety-related building structures.

To make a conclusive and complete safety assessment, it is necessary to carry out further work. This will include, for example, additional investigations on behavior of materials, system-dynamic analyses of the effectiveness of safety systems and more extensive evaluation of operational experience. The longer-term investigations necessary here are specified in the report. The intention is to carry out the investigations within a joint GRS-SAAS programme, with the participation of Soviet and French experts.

Based on the findings of the working groups, the recommendations for necessary safety-related backfitting measures are summarized at the end of each technical chapter. Generally, a distinction is drawn between safety-related measures which are required in the short term or in the long term. Chapter 10 contains an assessment of the recommendations listed in the technical chapters, broken down into three categories of measures and assessed in summary form.

2. DESCRIPTION OF THE INSTALLATION (UNITS 1-4)

Units 1-4 of the Greifswald nuclear power plant are equipped with pressurized water reactors of Soviet construction, namely WWER-440/W-230 reactors.

The text which follows is a description of the major features concerned with system technology and safety-related systems.

2.1 Basic diagram of the power plant

Figure 2-1 shows, in cross-section, a building to house the WWER-440/W-230 installations. Each unit comprises two loops, the primary system (primary loop, see Fig. 2-2) and the feedwater-steam system (secondary loop, see Fig. 2-3). The systems of the primary loop are enclosed by a pressure resistant compartment system (confinement), and the systems of the secondary loop are located within the machine hall.

Table 2-1 contains data concerning the design of the WWER-440/W-230 reactors. WWER nuclear power plants of class 440 are erected in the form of twin blocks. A characteristic feature of this type of construction is the structural arrangement of both reactors within a common reactor hall. Both reactors are provided as well with independent and separate operating systems and safety systems as with joint ones.

The primary loop consists of a water-cooled and water-moderated power reactor with 6 main coolant loops (MCL). Each MCL comprises a main coolant pump (MCP), a steam generator (SG), two gate valves (GV) with electrical drive to isolate the MCL of nominal diameter 500 at the reactor pressure vessel (PPV).

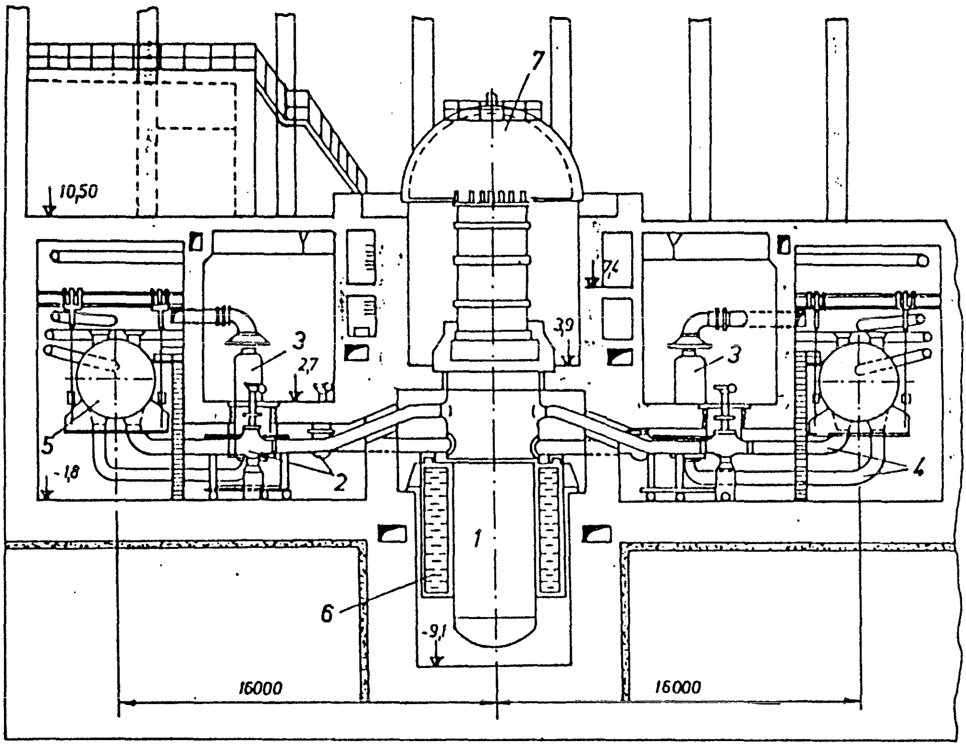
To compensate pressure and volume fluctuations, the primary loop includes a pressurizer (P), which is connected via two pipelines of nominal diameter 200 to a MCL and is equipped with safety valves (2 safety valves supplied by the company Sempell). These blow off into a pressurizer relief tank.

The secondary loop comprises the secondary part of the steam generator, the main steam system, the turbine generators, the condensate and feedwater systems and the auxiliary equipment of the machines. Each unit has two associated turbine generators, each having a power of 220 MW, which operate using 4.4 MPa saturated steam.

Table 2-1:

Main characteristics concerning the 440-MW unit WWER-440/W-230 (Units 1-4)

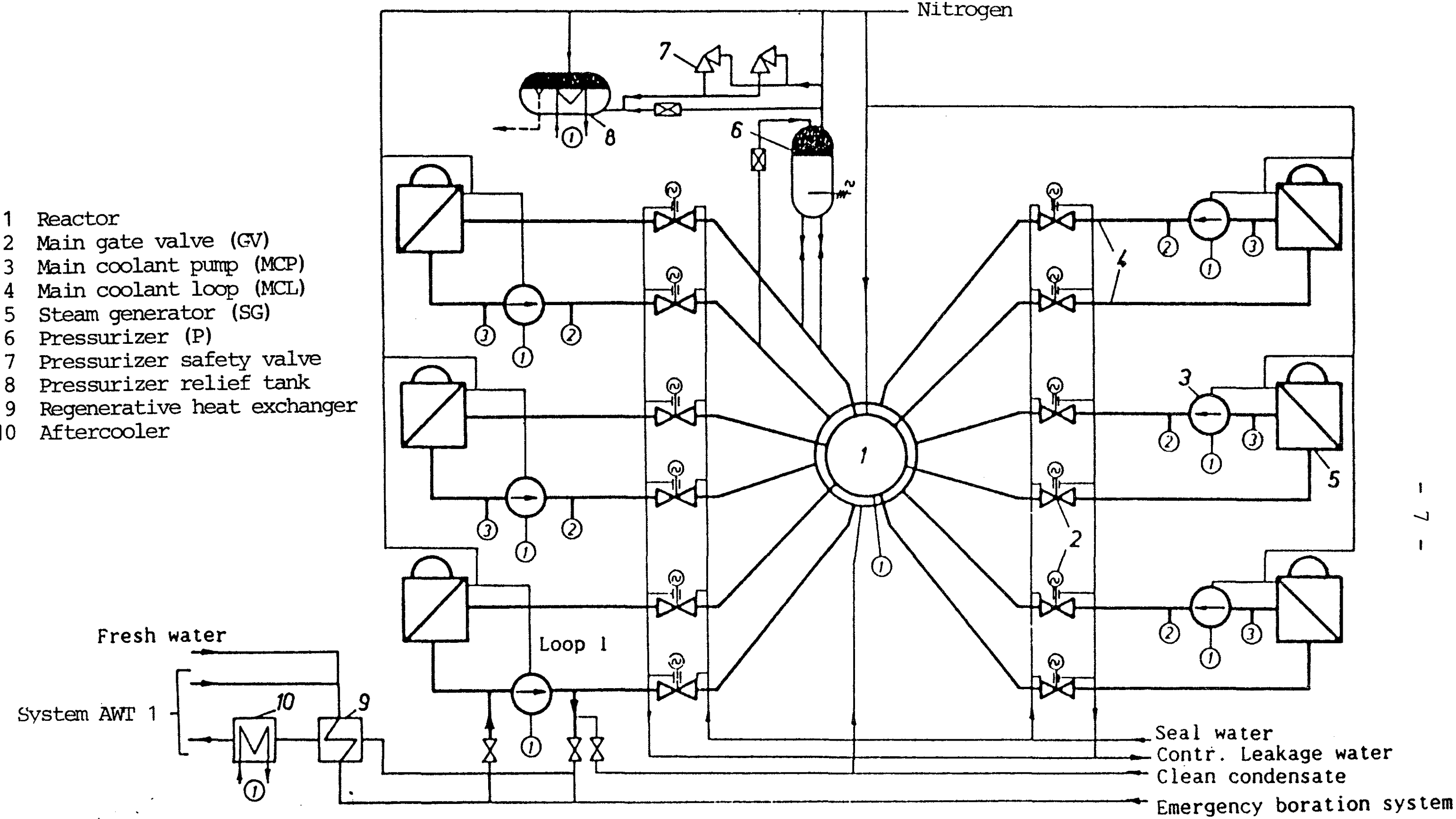
Characteristics/specifications	Unit	
Reactor type		WWER-440
Reactor power		
- thermal	MW	1375
- electrical	MW	440
Pressure in the primary loop	MPa	12.3
Water temperature at the reactor outlet	°C	293
Average water temperature warmed up within the core	K	28
Number of loops		6
Main steam parameters upstream of the turbine		
- Pressure	MPa	4.3
- Temperature	°C	254
Feedwater inlet temperature	°C	222
Turbine type		K-220-44
Number of turbines		2
Power of each turbine	MW	220
Generator type		TWW-200-2
Generator voltage	kV	15.75
Voltage fed into the power grid network	kV kV	220 (TS 1-5) 380 (TS 6-8)
Efficiency		
- gross	%	32
- net	%	30



1 Reactor; 2 Main isolating slide valve; 3 Reactor coolant pump; 4 Reactor coolant loop; 5 Steam generator; 6 Annular water container; 7 Protective hood

Fig 2-1: Spatial arrangement (side elevation)

1 Reactor
2 Main gate valve (GV)
3 Main coolant pump (MCP)
4 Main coolant loop (MCL)
5 Steam generator (SG)
6 Pressurizer (P)
7 Pressurizer safety valve
8 Pressurizer relief tank
9 Regenerative heat exchanger
10 Aftercooler



(1) From and to the intermediate coolant circuit (ICC) MCP or control and protection system

(2) Desalination of primary loop for the active water treatment system 1 similar to circulation loop 1

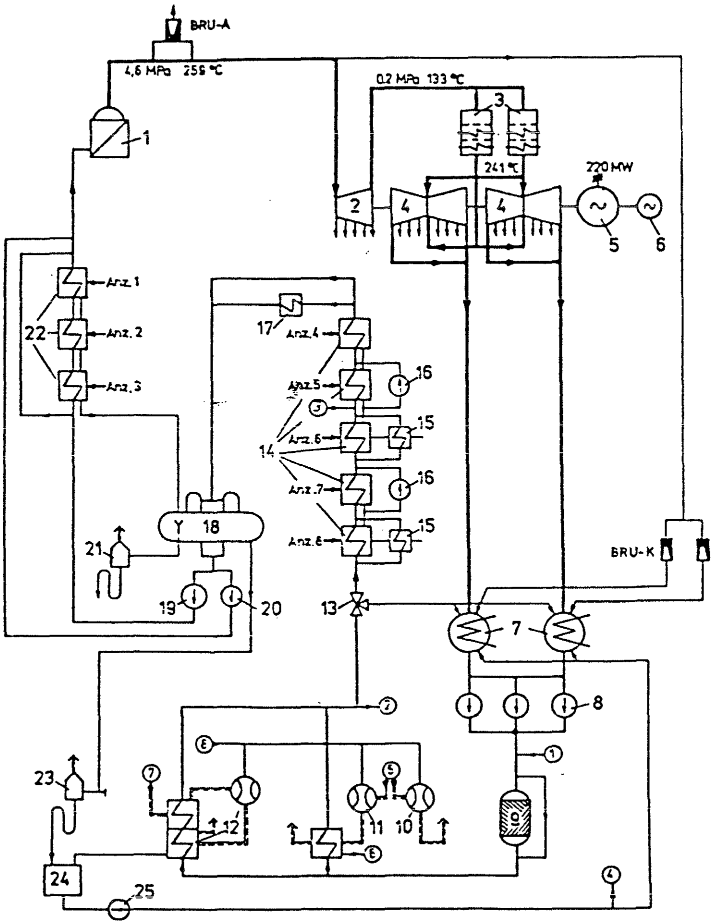
(3) Injection, primary loop, from active water treatment system 1, similar to circulation loop 1

<u>Fig. 2-2</u>:

Basic diagram of the primary loop

Fig. 2-3:

Basic diagram of the secondary loop



1 Steam generator. 2 High-pressure section turbine. 3 Separator-preheater. 4 Low-pressure section turbine. 5 Generator. 6 Power station service generator. 7 Turbine condenser. 8 Turbine condensate pump. 9 Condensate purification system (mixed bed ion exchanger). 10 Priming ejector. 11 Main ejector. 12 Gland ejector. 13 Three-way valve. 14 Low-pressure preheater (LPH 1-5). 15 condensate cooler. LPH 1 and 3. 16 Heating condensate pumps. LPH 2 and 4. 17 Vapour condenser. 18 Feedwater tank (FWT) with deaerator. 19 Feedwater pump. 20 Emergency feedwater pump. 21 Overflow relief valve. 22 High-pressure preheater (HPH 1 - 3). 23 Condensate drain tank relief valve. 24 Condensate drain tank. 25 Condensate drain tank pump.

(1) From the remote heat delivery (passive exchanger) (2). To the regenerative desalination cooler. AWT5. (3) From the regenerative desalination cooler. AWT5.
(4) To the FWT. (5) From the condenser. (6) To the condenser. (7) From the gland steam header.

The power takeoff to the power network takes place at 220 kV or 380 kV, and the station service supply is accomplished at voltages of 6 kV, 380/220 kV and 220 V DC. There is also an energy release of 200 MW thermal to a heat distribution net.

For the supply of coolant water and service water (secondary cooling water), a cooling system is provided, using sea water from the Greifswalder Bodden supplied through an intake structure.

2.2 Safety-related design

The safety-related design of the Soviet pressurized water reactors of the WWER-440/W-230 type was based on requirements which applied at the time in the USSR regarding the planning and construction of these reactor installations. These requirements include the following principles which are set forth in the USSR Technical Plan:

- In view of the measures adopted to secure the reliable construction of the primary loop and as a result the experience gained during the construction of reactor installations in the USSR, there is a low probability of damage such as the breach of large pipes (nominal diameter 200, nominal diameter 500) and corresponding large-scale equipment (reactor pressure vessel, pump housing and slide valve housing). Accordingly, in the plan no consideration is given to accidents initiated by such events.

- In dealing with accident situations, cases in which redundant systems and passive equipment fail are not investigated.

In comparison with current safety requirements applicable to pressurized water reactors, this results in the following deficiencies:

- insufficient confinement
- low capacity of the safety systems
- low redundancy of the safety systems
- interconnection of the safety systems
- interconnection of safety and operating systems
- no physical separation of redundant systems
- no independent long-term emergency cooling.

On the other hand, the installations have properties which from the point of view of safety show positive features. These include:

- the possibility to isolate the individual circulation loops of the primary loop
- the relatively low power density of the core
- the relatively short core
- greatly damped oscillatory behaviour of the power within the core
- the large volume of feedwater in the steam generators.

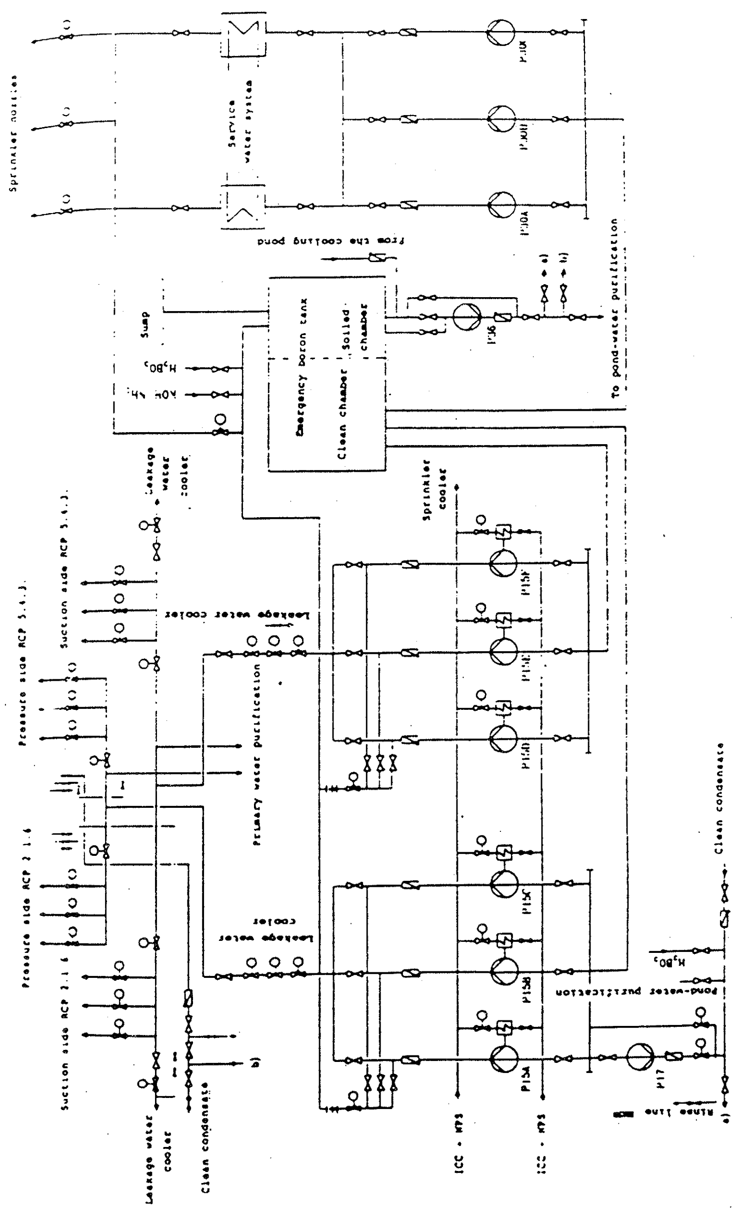
2.3 Safety-related systems

- Emergency cooling system (emergency boron injection system)

For feed into the primary loop, the emergency cooling system (Fig. 2-4) includes 6 pumps which are set up jointly alongside one another. Each set of 3 pumps forms one unit and applies suction via a line from a borated water tank which serves as a reservoir for all pumps, and feeds into the primary loop via two headers. In addition to the boron injection pumps, there are 3 spray water pumps each with 2 coolers, which also apply suction from the abovementioned borated water tank and, in the event of a leak, spray borated water into the pressure resistant compartment system of the reactor building, in order to limit any pressure build-up.

Fig. 2-4:

Basic diagram of the emergency cooling system and sprinkler installation

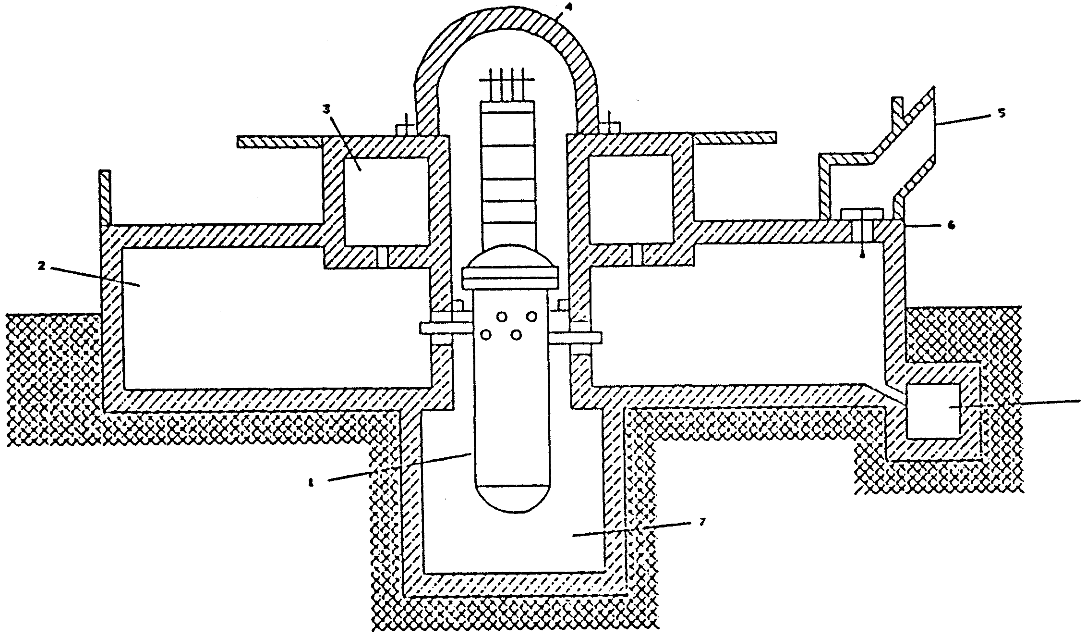


- Confinement (pressure resistant compartment system)

The pressure resistant compartment system (Fig. 2-5) is a closed compartment system, which encloses as a confinement the principal systems of the primary loop (reactor pressure vessel, steam generators,main reactor coolant pumps, main gate valves, pressurizer, emergency boron tank, primary water injection system). The compartments have a total net volume of approximately 14,000 m³ and are designed to withstand an excess pressure of 1 bar. A gradually reduced pressure is maintained between the inaccessible compartments, the accessible compartments and the atmosphere by means of a ventilation system. Heat lost is removed by means of air recirculation systems with a cooler. Three lines of nozzles of a spray system are arranged within the box of the steam generators and main coolant pumps (pressure reduction, iodine bonding). The pressure resistant compartment system is connected to the environment via nine dampers. In the event of the highest-rated design basis accident, i.e. "fracture of a connecting line to the primary loop of nominal diameter 100 with an outlet limiter of nominal diameter 32" and assuming that the spray system remains operational, the dampers do not open. If the spray system fails in the course of the design basis accident, one damper opens. The opening of all dampers protects the pressure resistant compartments in the event of rupture of the largest connecting line, of nominal diameter 200, at the main coolant loop in the region which cannot be isolated.

Fig. 2-5:

Pressure resistant compartment system of the W-230 Greifswald nuclear power station Units 1-4



1 Reactor

2 Steam generators

3 Compartment for drive systems

4 Protective hood

5 Blowout shaft

6 Overpressure dampers

7 Reactor cavity

8 Storage reservoir of the emergency reactor cooling system/sump

- Emergency feedwater supply (emergency feedwater system)

The emergency feedwater system (Fig. 2-6) consists of two pumps which are set up alongside the main feedwater pumps in the machine hall. The pipelines are interconnected on the suction and pressure sides.

- Cooling water supply (service water system)

The service water system (Fig. 2-7) includes 5 pumps for one double unit which are set up in a common compartment and which feed on the pressure side to a header and from there supply all consumers of a unit (operating and safety systems) with cooling water via a train.

- Electrical power supply (Fig. 2-8)

1. Connection to the grid: a start-up grid connection is provided, and generator switches have been backfitted for the main grid connections. This guarantees isolation of the generator from the grid, and makes it possible to provide the power station service system of the unit from the grid. The power output of the units is passed into the 220 kV and the 380 kV grid.

2. Power station service system: facilities are provided for connection to the adjacent unit. In the event of an emergency power supply being required, the run-down energy of the turbine generators is utilized for the continued operation of the main coolant pumps during a limited period.

Fig. 2-6:

Basic diagram of the emergency injection pumps

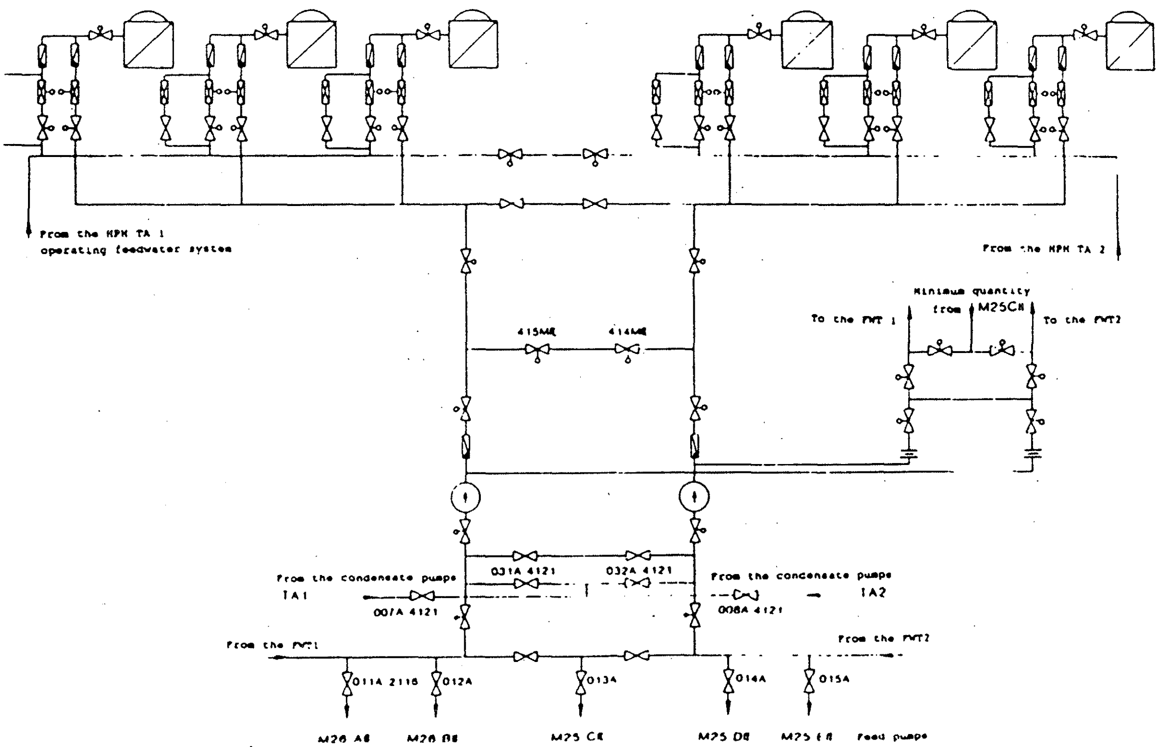


Fig. 2-7:
Basic diagram: service water

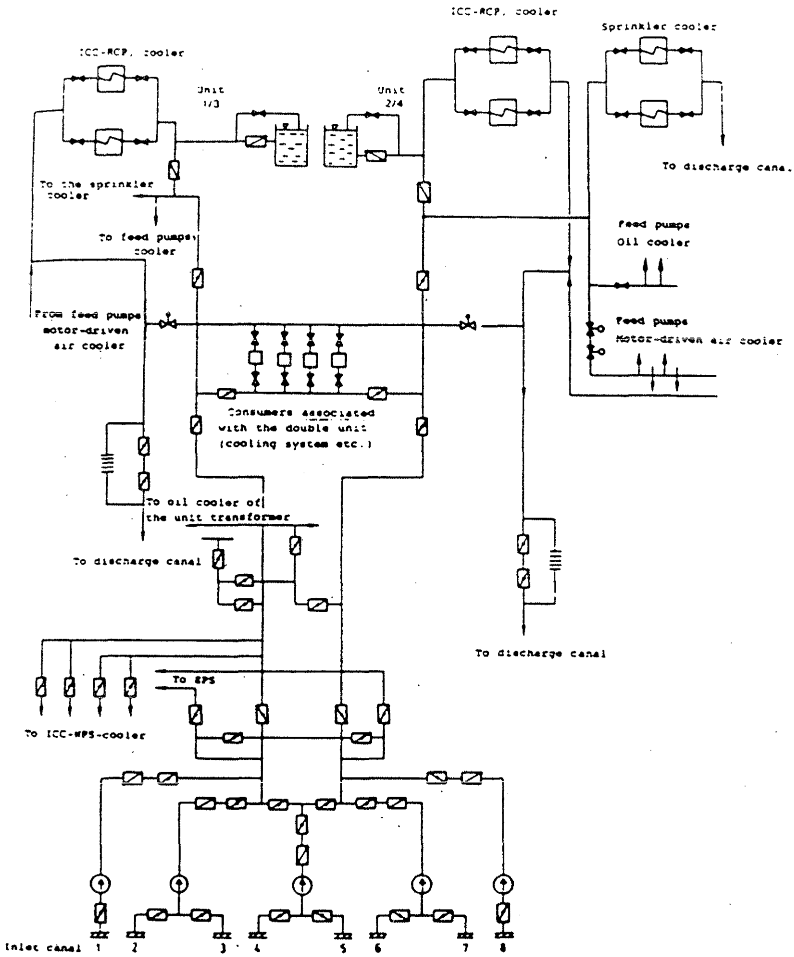
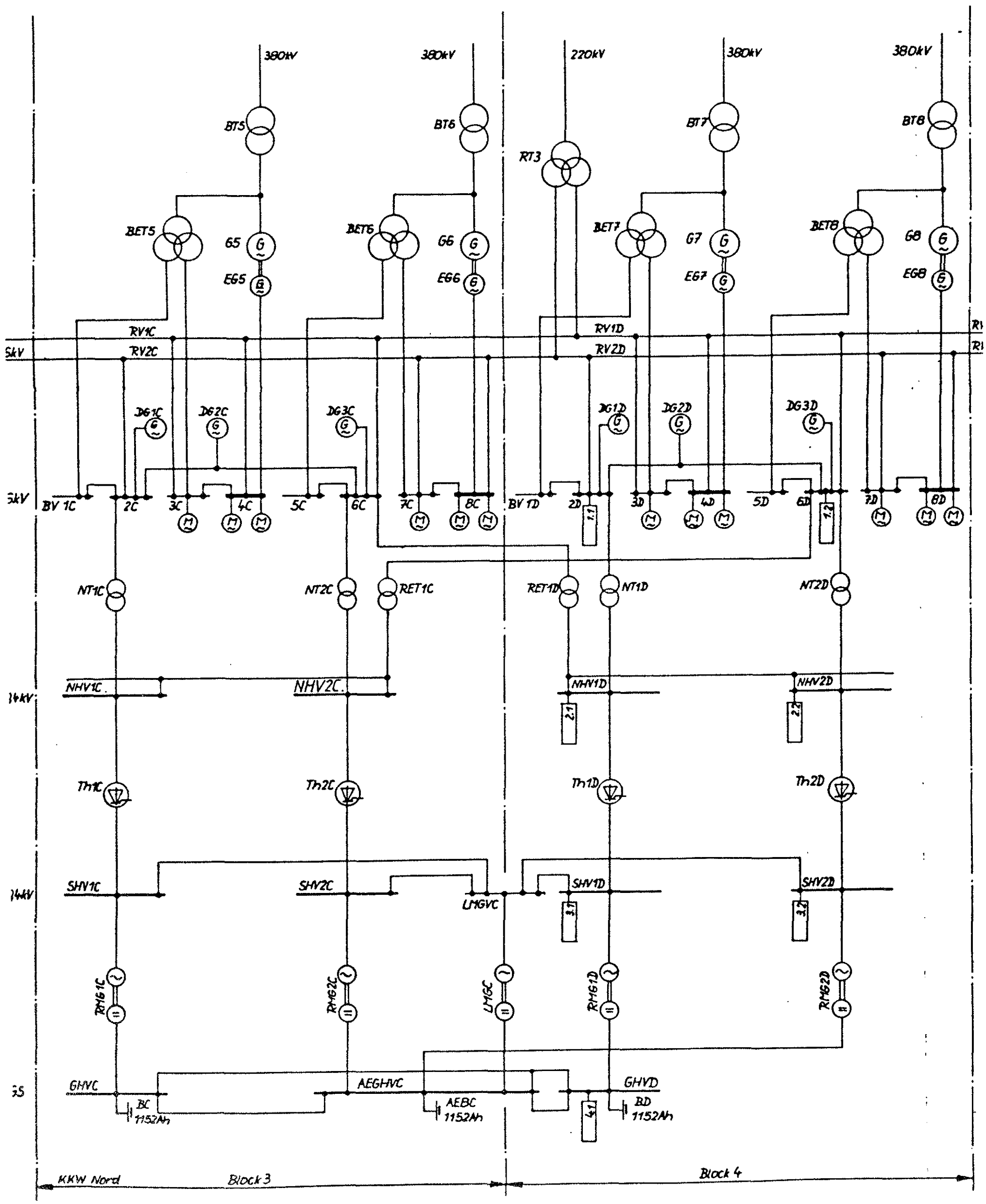


Fig. 2-8:

Electrical power supply of units 3 and 4



3. Emergency power system: the emergency power system is to a large extent, but not entirely, constructed in two lines. Interconnections exist at various levels (diesel generators, switching systems, DC systems).

3. SAAS REQUIREMENTS CONCERNING BACKFITTING AND SAFE PLANT MANAGEMENT

In the mid-eighties, the analysis of existing safety deficiencies in comparison with the current status of the safety-related design of pressurized water reactors gave rise to the view that long-term continued operation was not possible without comprehensive safety-related backfitting to match the safety level of modern pressurized water reactors.

Because of the inadequate progress of work towards this backfitting, the SAAS laid down in March 1989 minimum requirements applicable to safety-related backfitting. The requirements of the SAAS were discussed with the supervisory authorities in the USSR, Czechoslovakia and the Republic of Bulgaria, which operate nuclear power plant units of the same type; they were developed into a common view in the form of 16 principles for safety-related backfitting.

This common view, which requires the completion of the backfitting by the end of 1992, was transmitted to the Greifswald nuclear power plant as a specific statement of the minimum requirements, and the introduction of a specific operating regime was requested until backfitting was complete.

The common view of position of the regulatory authorities in the countries concerned assumes that complete conformity with the current international safety level of modern nuclear power plants operating pressurized water reactors cannot be achieved, but a close approximation must be ensured. To this end, the following principles were formulated:

- The spectrum of design basis accidents is to be extended as far as possible and accident management measures are to be introduced for beyond design basis accidents.

- It shall be demonstrated that leaks > nominal diameter 100 mm occur with a probability < 10^{-5} per unit per year and that timely execution of the required measures is guaranteed by "leak before breach" and by leak detection systems as well as by tests.

- The emergency cooling system must consist of at least two independent lines and cope with leaks < nominal diameter 100 according to plan. In the case of leaks > nominal diameter 100, measures to reduce damage to the core must be taken. The need for core flooding tanks is to be analysed.
- Measures to increase the effectiveness of pressure reduction within the pressure resistant compartment system to reduce the opening of the over-pressure dampers in the event of accidents with leaks < nominal diameter 100.

- Measures to enhance the tightness of the confinement in the case of leaks ≤ nominal diameter 200 are to be provided.

- The emergency power supply must consist of at least two independent lines; further, its level of reliability must not be below that of the safety systems to be supplied.

- The reactor protection system must have physically separate lines with three channels in each instance. Establishment of a reserve control room to shut down and to cool the reactor.

- Assurance of aftercooling in the event of any transients and accidents not associated with leaks in the primary loop.

- Assurance of radiation monitoring in the event of accidents within and outside the installation.

- Assurance of accident protection for personnel and population, in accordance with international requirements.

- Introduction of measures to prevent brittle failure of the reactor pressure vessels.

- Implementation of measures for improved fire protection, such as extensive reduction of the fire load and physical separation of redundant cables.

- Reduction of the probability of core meltdowns in the event of beyond design base accidents by accident management measures.

- Improvement of training facilities for the personnel, e.g. full-scale simulators on which also beyond design basis accidents can be simulated.

- Analysis of accidents caused by external effects and procurement of technical and organizational measures for prevention and mitigation.

- Analysis of materials, of the state of equipment and structures and applied loadings, to determine prerequisites for backfitting measures.

In addition, in February of this year the SAAS established conditions for ensuring safe operation until shutdown of units 1-4 for the implementation of backfitting (35-point programme). In addition to measures for the completion and precise formulation of the operating conditions and specifications as well as the systematic recording and assessment of applied loadings, essential points in this programme include

- 100 % testing of all steam generator tubes
- Stabilization of the cooling water supply of the emergency diesel generators
- Backfitting of systems for power density monitoring and plant diagnostics
- Installation of leak detection systems
- Provision of fire alarm and extinguishing systems in the feedwater areas in the machine hall
- Implementation of measures for accident management within the plant concerning the feedwater supply.

4. ASSESSMENT OF THE PRESSURIZED COMPONENTS OF THE PRIMARY AND SECONDARY LOOPS

The safety assessment of the pressurized equipment (vessels, pumps and block valves) and pipes requires the demonstration of their integrity under conditions of normal operating loads, operating transients and accidents. To this end, it is necessary to analyse

- technical details of the as-built structure,
- the suitability of the structural materials employed,
- mechanical and thermal loads,
- defects detected in the course of periodic tests, and
- possible interactions of the structural materials with the medium.

It should be borne in mind here that the reactor installations were constructed, designed, produced and erected in accordance with technical regulations and standards which are in accord with the state of technology in the 60s in the USSR.

4.1 Primary loop and parts of the main steam and feedwater system

4.1.1 Layout of the loops

- Primary loop

The pressurized walls of the vessels of the primary loop (reactor pressure vessel, steam generators, pressurizer) consist of ferritic steel. A low-alloy chromium-molybdenum-vanadium steel is used for the reactor pressure vessels, and a carbon steel for the steam generator jackets and the pressurizers. The connecting pipes between the vessels and the housings of the slide valves and pumps consist of a titanium-stabilized austenitic chromium-nickel steel (similar to CrNiTi 1810). This is also true for the steam generator tubes. The basic materials used for the various components and pipes are presented in Table 4-1.

Table 4-1:

Table of steels of the WWER 440 system and examples of practical application

Brand of steel	Chemical Composition (%)									Example of practical application
	C	Si	Mn	Cr	Ni	Ti	S	P	Elements	
08Ch18N12T	<0.12	<0.80	<2.00	17.0-19.0	11.0-13.0	5S[%C]<0.7	<0.020	<0.035		MCL
08Ch18N10T	<0.08	<0.80	<2.0/-<1.5a	17.0-19.0	9.00-11.0	5S[%C]<0.7	<0.020	<0.035		Needle tubes SG, reactor internals Protection system, MCL-part Primary circuit, MCP
10CH18N10T	<0.12	<1.00	<0.20	17.0-20.0	8.00-11.0	5S[%C]<0.6	<0.030	<0.035		Block valve housing
15Ch2MFA	0.13-0.18	0.17-0.37	0.30-0.60	2.50-3.00	<0.40	-	<0.025	<0.025	Mo:0.60-0.80 V :0.25-0.35	RPV
25Ch1MF	0.22-0.27	0.17-0.37	0.40-0.70	1.50-1.80	-	-	<0.030	<0.025	Mo:0.25-0.35 V :0.15-0.30	Studbolts : RPV, SG sec.
25ChMF KP70										Studbolts :NPP Rheinsberg
ChN35WT	<0.12	<0.60	1.00-2.00	14.0-16.0	34.0-38.0	1.1-1.5	<0.020	<0.030	W :2.80-3.50	Studbolts: main block valve SG prim.
38ChNKA (38CHN3MA)										Studbolts: MCP
2Ch13	0.16-0.24	<0.60	<0.60	12.0-14.0			<0.025	<0.030		Screws upper block
09Ch17N										Control and protection system
Ch18N22W2T2										Protection system
ZrNb1									Nb:1.0	Cladding tube
22K	0.19-0.26	0.20-0.40	0.75-1.00	<0.40	<0.30	-	<0.025	<0.025		Pressuriser, SG housing
20K (GOST 5520-69)	0.16-0.24	0.15-0.30	0.35-0.65	<0.30	<0.30		<0.040	<0.040	Cu: <0.30 Cr+Ni+Cu<0.6	Feed water tank
Stahl 20	0.17-0.24	0.17-0.37	0.35-0.65	<0.25	<0.25		<0.040	<0.035	Cu: <0.25	Fresh steam line, high pressure heater tubes

Table 4-1 (continued)

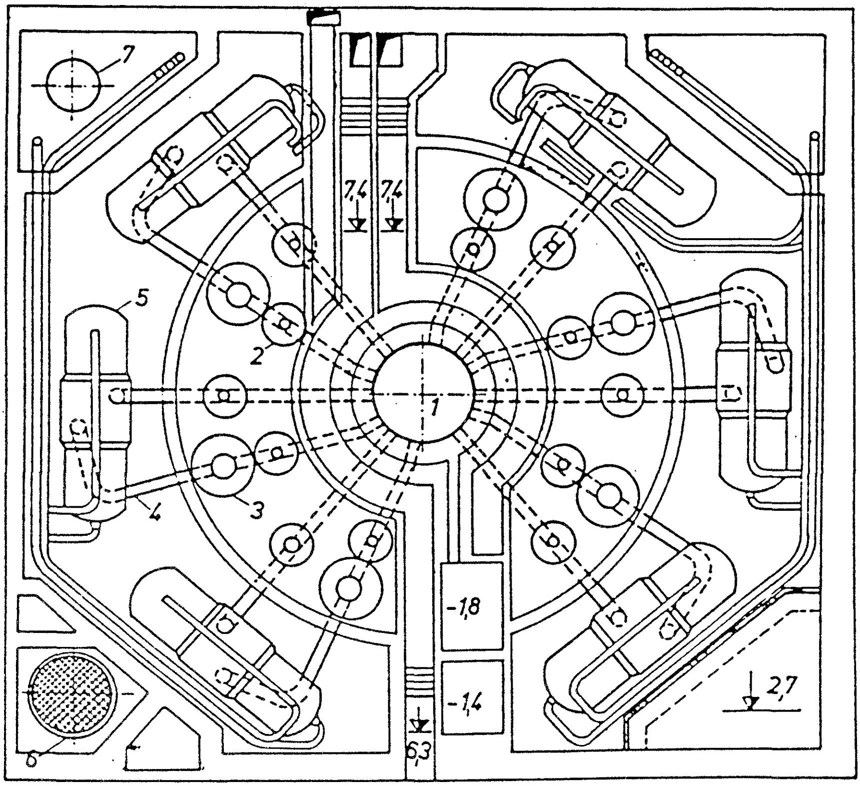
1Ch13 (08Ch13)	0.09	<0.60	<0.60	12.0-14.0	-	-	<0.025	<0.030		Turbine blades, stage 1-3
15Ch11MF										Turbine blades, stage 4, 5
34 ChN1	0.30-0.40			1.30-1.70	1.30-1.70					Axle of low pressure rotor
34 ChN3M	0.30-0.40			0.70-1.10	2.75-3.75				Mo:0.25-0.40	Axle of high pressure rotor
25L										Housing of high pressure part (cast steel GS-C25)
15Mo3 (TGL 7961)	0.21-0.20	0.17-0.37	0.50-0.80	-	-		<0.040	<0.040	Mo:0.25-0.35	T-piece fresh steam lines, pipings secondary circuit
K 15Mo3 y	0.21-0.20	0.15-0.37	0.50-0.80	<0.30	<0.30		<0.025	<0.030	Mo:0.25-0.35	Valid for orders from 1988 pipings sec. circ.
St38b-2 (TGL 7960)	0.12-0.20	0.15-0.35	0.40-0.65				<0.050	<0.045		Feedwater tank
St45.8 (DIN 17175)	<0.22	0.10-0.35	<0.45				<0.050	<0.050		Heat-condensate pipings fresh steam line
CuNiFe5										Condensat tubes
CuNiFe10										Condenser tubes of TS1 after backfitting
CuNiFe30										Condenser tubes from unit 5
CiZu30 (L 68)										Oil cooler tubes low pressure heater tubes before backfitting
X8CrNi18.10										Low pressure heater tubes after backfitting
13CrMo4.4	0.10-0.18	0.15-0.35	0.40-0.70	<0.30			<0.035	<0.035	Mo:0.40-0.65	High pressure heater tubes after backfitting
10CrMo9.10	0.08-0.15	0.20-0.50	0.40-0.70	2.00-2.50	-	-	<0.035	<0.035	Mo:0.90-1.10	Heat-condensate pipe

a - applicable to pipe   y - KKAB Reports 54/2   MSL - Main steam line

Figures 4-1 to 4-3 give an overview of the general arrangement. The reactor pressure vessel is the anchor point. Sliding bearings are provided in each instance in the region of the main block valves and the main coolant pumps. the horizontal SG is displaceably mounted by means of hinge links. The main coolant pump is situated in the cold line. The main coolant loops of all six circulation loops are to a large extent of similar construction. By means of a pipe fitting on the hot side, three lines of nominal bore 200 branch off from one circulation loop, two lines being connected to the lower part of the pressurizer and one line opening into the upper pressurizer connection. In operation, this line is isolated by a non-return flap valve. A connecting line of nominal bore 100 leads from the upper pressurizer connection to the cold line. A control valve is provided in this connecting line. Upstream and downstream of the pump, nozzles are fitted on the main lines of nominal bore 500, and the pipes of nominal bore 50 of the blowdown system are welded onto these connections. Additional structural details and the data concerning the materials, types of weld, lists of nozzles, tests etc. are summarized in /1/.

The reactor pressure vessel is a vertical cylindrical welded vessel with an arched head and base. The internal diameter is 3560 mm, and the design is shown in Fig. 4-4. The cylindrical part consists of 3 seamless forged cylindrical strakes having a wall thickness of 140 mm, 2 seamless forged cylindrical strakes having a wall thickness of 200 mm with the connections opened out (Fig. 4-5) and one seamless forged conical strake with 60 blind bores for the studs. The individual strakes are connected to one another by girth welds (2/3X, SAW). Details of the structure of the welds are given in /1/. The hot pressed arched base consists of two sections having a wall thickness of 160 mm, which were joined together prior to heat treatment by means of electroslag welding. The head consists of the head flange having a wall thickness of 300 mm and 2 hot pressed head sections, which were likewise joined together prior to heat treatment by means of electroslag welding. The wall thickness is 246 mm. Below the level of the main coolant connections, there are no penetrations in the vessel. The connections of the control and protection systems as well as of the system for monitoring the fuel assembly outlet temperatures and other connections are situated in the head.

Fig. 4-1: Spatial diagram of the primary circuit, plan view



1 Reactor; 2 Main block valve; 3 Main coolant pump; 4 Main coolant line; 5 Steam generator; 6 Inspection well; 7 Pressurizer

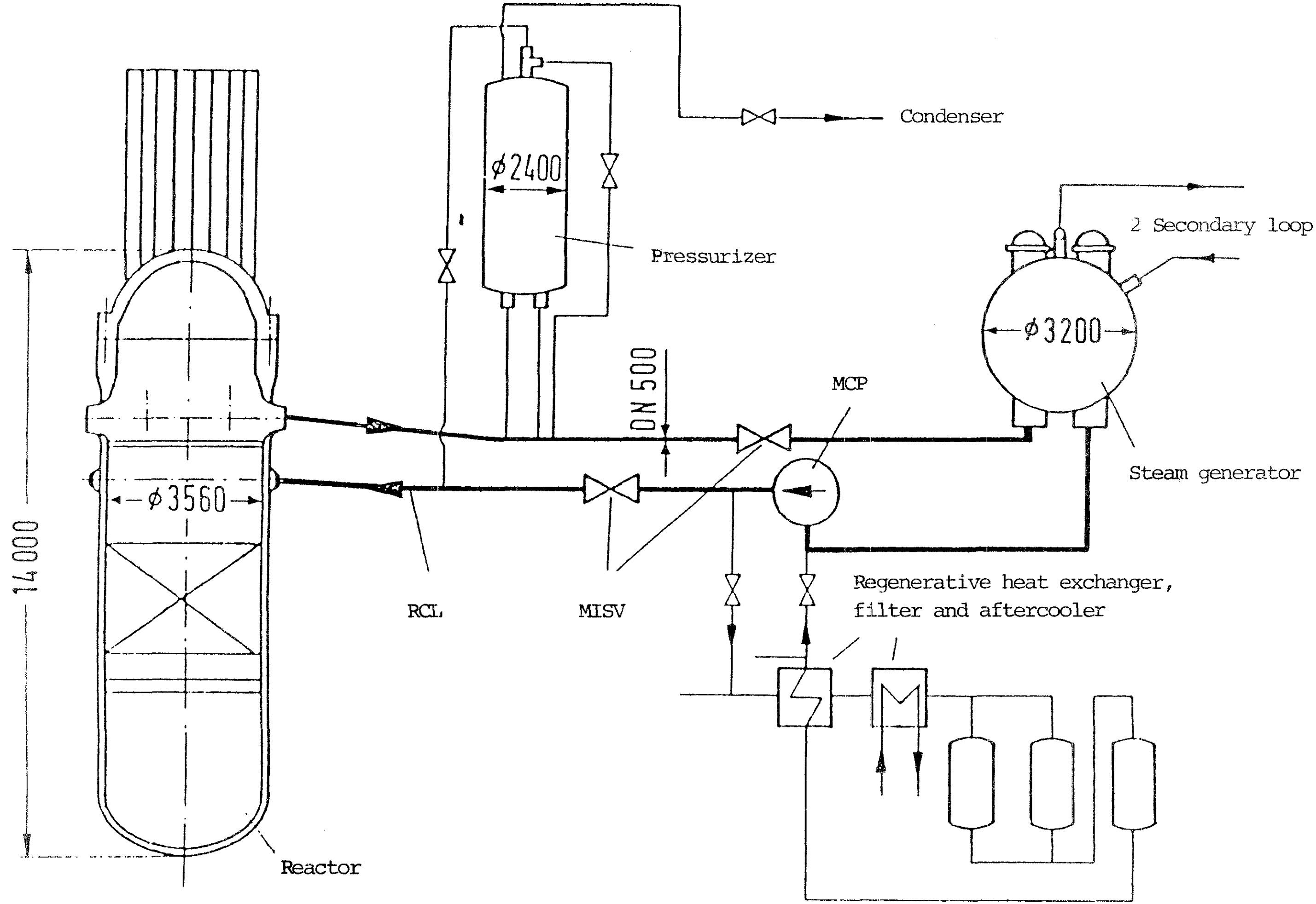
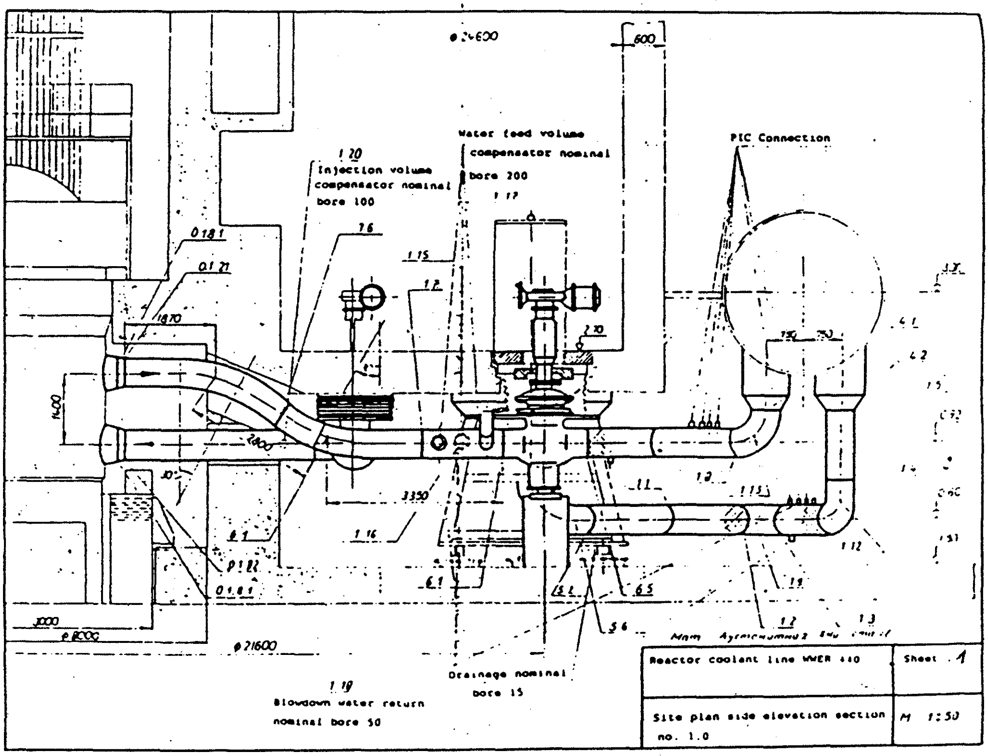


Fig. 4-2:

Reactor installation WWER-440

Fig. 4-3:
Spatial arrangement of the primary loop (longitudinal cross section)



<u>Fig. 4-4</u>:
Lower part of reactor pressure vessel, WWER-40/W-230

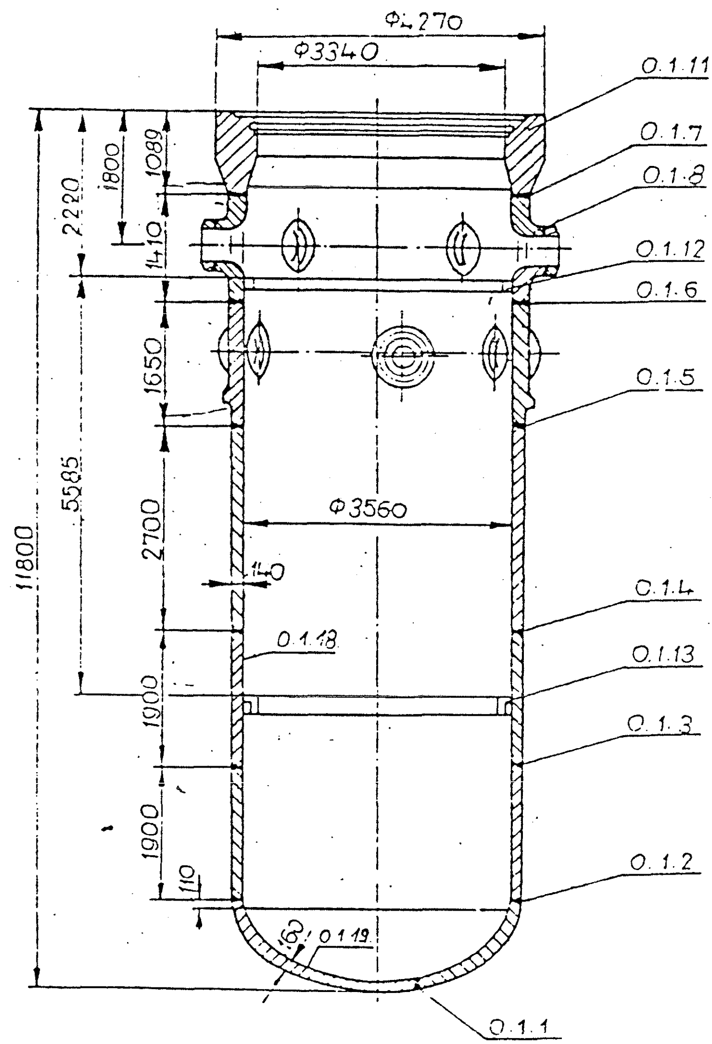
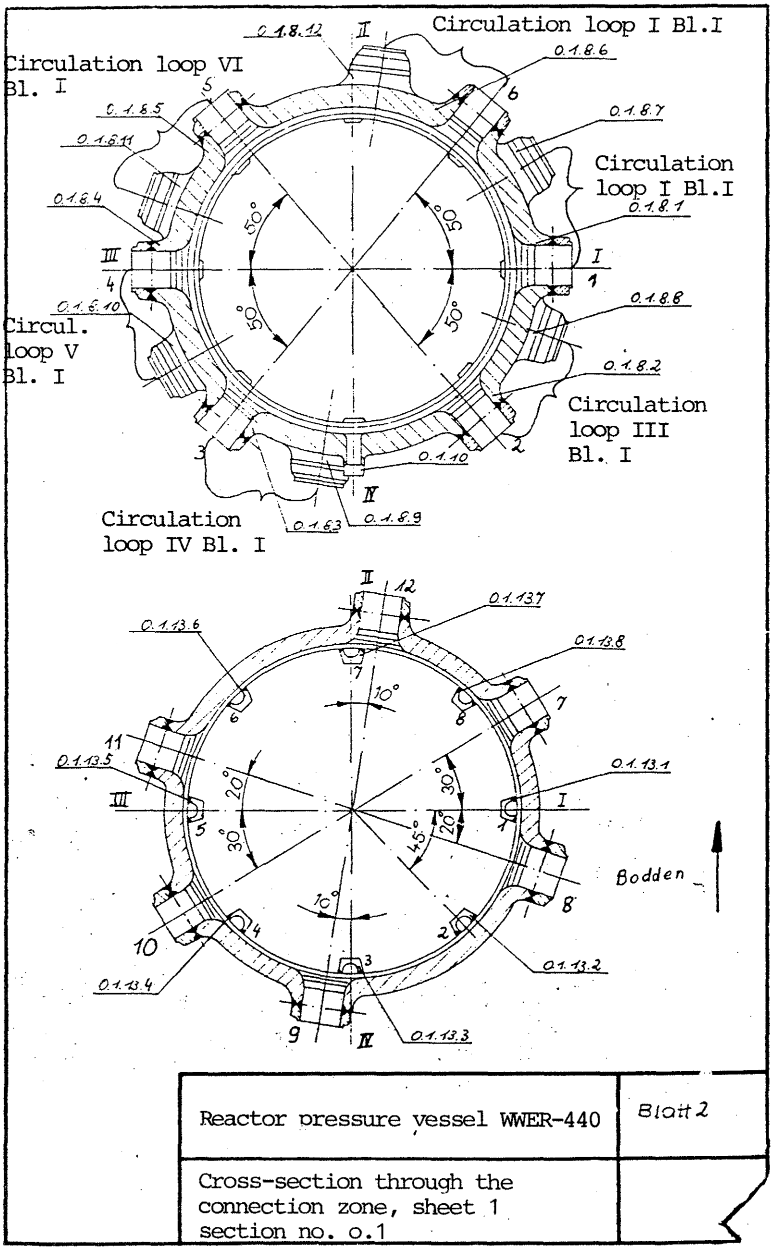


Fig. 4-5: Reactor pressure vessel, nozzle rings, WWER-440/W-230



The reactor pressure vessels of units 1 and 2 are unplated, while those of units 3 and 4 are double-layer plated. The plating is provided in the form of strip plating having a width of 60 mm and a thickness of 9 + 2 mm. In the case of all reactor pressure vessels, the head is plated on the entire internal surface and part of the external surface in the region of the connections, /1/; moreover, all sealing surfaces of the lower and upper parts of the flanges are plated.

At the vertical level of the lower third of the core there is a girth weld (0.1.4), which, in comparison with the other weld seams, is exposed to a considerably greater neutron flux. /1/ contains further information concerning the constructional details of the pressure vessels, materials, types of connection etc.

The detailed design of the pressurizers, steam generators, pumps and slide valves, the material-related transitions between ferritic and austenitic parts as well as further information on the reactor coolant lines and the branch lines can be found in /1/.

- Main steam and feedwater system

Only since 1984, when Technical Quality Standard 43272 was introduced, have the components of the secondary loop been classified as relevant with regard to safety in the sense of ensuring nuclear safety. Data on the materials can be found in /1/. Before this date the design, erection and testing of the system and its operation were carried out in accordance with the applicable regulations concerning steam and pressure engineering. In addition, the testing requirements were established in Technical Standard 30316 in 1987.

In the main steam system, the sections from the SG to the turbine trip valve control system and in the feedwater system the sections from the feedwater tank to the SG are to a limited extent also considered in the present report. Because of the time available, it has not yet been possible to undertake an in-depth analysis, especially of the quality of the feedwater tanks.

4.1.2 Suitability of the structural materials employed

To construct the safety-related systems and pipes of the WWER-440/W-230 reactor installations, use was made of structural materials as shown in Table 4-1, which were subsequently accepted in the standards relevant for the purpose of computing the strength of nuclear power station components (Moscow, Metallurgia 1973). The characteristic data for basic materials and weld metal which are concerned with mechano-technological aspects and which are binding for design purposes, as well as the chemical analysis and heat treatment data, are specified in this standard or in State standards.

The items of documentation (so-called permits) supplied by the manufacturer and concerned with systems and pipes contain, in some cases, data which relate to the individual semi-finished products and which are concerned with chemical analysis, heat treatment and mechano-technological characteristic data. Data concerning repairs are also included. An in-depth evaluation of these items of documentation still remains to be undertaken. To a limited extent, evaluations derived from these permits are included in the data given in /1/ which relate to specific components.

In the assessment of the austenitic materials employed, it is possible to obtain meaningful information not only from the results of the production tests and the inservice tests, but also from the extensive investigations within the terms of the so-called 100,000 h programmes with respect to components which have been removed. At the present time, an extensive programme of investigations is being processed, in which specimens of materials and equipment are recovered from units 1 and 2 after a period of approximately 100,000 h under operational loading, and mechano-technological properties are to be experimentally determined and the microstructure analysed. In addition, comparisons with results derived from follow-up investigations on components which have not been used provide information on changes to materials caused by the operational conditions. These investigations will be completed in 1991 and will be able to provide a more precise statement of the current state of knowledge. With regard to austenitic parts of the reactor coolant line which were used in the Rheinsberg installation, follow-up investigations have shown that, in spite of the inhomogeneities and instances of non- uniformity of the microstructure which were found in the materials, as a rule the minimum requirements of the standard were met and were in some cases considerably exceeded. The cases of precipitation, non-metallic inclusions and coarse grain detected in the specimens of material were classified as being to a large extent caused by the production process. An overview of the results is given in /1/. It was also possible to utilize results derived from tests on specimens of material from Soviet nuclear power stations of the same type for the purposes of an assessment, after these results had been evaluated.

A matter of considerable significance from the point of view of safety is the deviation from the planned condition with regard to the neutron embrittlement of the weld seam 0.1.4 of the reactor pressure vessel. The brittle fracture transition temperature of the material of the weld seam 0.1.4 increases, as compared with the planned condition, more rapidly than was forecast, by a factor of approximately 3. This information became evident from results of tests on the supporting systems specimens taken from the Loviisa installation, from the second unit of the Armenian nuclear power station, from units 3 and 4 of the Kola nuclear power station and from units 1 and 2 of the Rovensk nuclear power station. According to actual knowledge, the cause of this is the inadequate specification - at the date of production of the reactor pressure vessel - of the limit on the content of copper and phosphorus in the weld seam. The assessment of the embrittlement is made more difficult by the fact that in the case of units 1 to 4 no surveillance specimens comcomitant with operation were provided and at the date of production of the reactor pressure vessels the chemical composition of the weld seam, and especially the contents of phosphorus and copper as well as the brittle fracture transition temperature in the starting condition were not determined experimentally. According to Soviet experts, the chemical composition of the weld seam was determined using a model which takes account of the transfer of material between welding electrode and welding flux. In the case of individual vessels, the computed values show good agreement with values determined experimentally, having regard to the uncertainties which are to be expected. This statement is also supported by the investigation of metal shavings taken from the reactor pressure vessel in unit 1.

The effect of the chemical elements Cu and P on the embrittlement of the material, expressed by the shift in the brittle fracture transition temperature, is described by a coefficient of embrittlement. The empirical statement derived for this quantity is set forth in the standards and is based on a large number of results of investigations which were obtained using specimens from research reactors and surveillance specimens. The values determined for the brittle fracture transition temperature of the regions close to the core (weld seam and base material) at start-up and at the present time as well as the data concerning chemical composition are shown in Tables 4-2 and 4-3. The procedure adopted for the determination of the shift in brittle fracture transition temperature is explained in /1/. When assessing the abovementioned shift in the brittle fracture transition temperature, on the basis of the current state of scientific knowledge it is necessary to carry out a supplementary assessment with regard to possible scatter bands. Investigations by Soviet experts provide indications that the embrittlement found in the wall of the reactor pressure vessel is greater than the values determined from surveillance specimens; see Figs. 4-6 and Fig. 4-7 reproduces curve 4, which arises from the application of the computation methods in accordance with the Soviet standards. In this practical application, the experimental results are virtually identical. The question whether this is the case for all combinations of materials remains to be answered by the follow-up investigations both on the Novo Voronezh installation and also on the specimens derived from unit 2 of the Greifswald NPP and other investigations. To establish safety, it is sufficient to verify a commensurate safety margin.

For the purpose of restoring the properties of the material which have been altered by the neutron irradiation, heat treatment processes have been developed all over the world. The data, established on the basis of Soviet investigations, relating to the performance of the heat treatment in the form in which it was implemented in the case of unit 1, are summarized in Table 4-4. According to a current estimate, the reactor pressure vessels of units 2 and 3 have also reached a state which makes a heat treatment necessary. This is to be carried out during the current shutdown.

According to the present state of scientific and technological knowledge, the heat treatment process is in principle suitable to anneal or to reverse to a large extent any material embrittlement which has occurred over the period of operation. In the case of the reactor pressure vessel of unit 1, the extent of recovery of ductility of the material as a result of the heat treatment has not yet been adequately quantified.

Table 4-2 : Data concerning operation time since the first start-up, prognosis of fluences on the RPV wall (weld close to core-SG central core region - GW max) and calculated brittle facture transition temperatures at the beginning of operation (T_{ko}), in 1989 (T_k) and maximum value (T_kEOL).
SG = weld material
GW = basis material

Unit No. / Hours of operation	Fluence in cm^{-2}, Neutron energy > 0.5 MeV: 1989 WM	1989 BM_{max}	EOL[b)] WM	EOL[b)] BM_{max}	Calculated brittle facture transition temperature: T_{KO} WM	T_{KO} BM_{max}	T_K 1989 WM	T_K 1989 BM_{max}	T_KEOL WM	T_KEOL BM_{max}
Unit 1 105 700 h	$4.5x10^{19}$	$7x10^{19}$	$5.7x10^{19}$	a	(+46°C)	(0°C)	the value after annealing remains to be determined	c	(150) d	c
Unit 2 101 700 h	$6.4x10^{19}$	$8.2x10^{19}$	$6.4x10^{19}$	a	(+ 4°C)	(0°C)	(>147°C)	c	(147)	c
Unit 3 90 300 h	$5.5x10^{19}$	$7x10^{19}$	$6.5x10^{19}$	a	(+23°C)	(0°C)	(152°C)	c	(163)	c
Unit 4 73 300 h	$5x10^{19}$	$5.8x10^{19}$	$10.8x10^{19}$	a	(-13°C)	(0°C)	(124°C)	c	(163)	c

a) Values for basis material in the central core region (GW max) depend on the future loading of the core.
b) End of life-value ≙ maximum admissible value for design base; EOL-value is fixed again after annealing above T_k.
c) Values are rated, according to project data, at T_k < 70°C after 30 years; supplementary assessment is necessary.
d) Value determined prior to annealing: 186 °C.
() Values determined for empirical relationships.

Table 4-3 : Data concerning the fractions of Cu and P in the chemical analysis and derived coefficients for the estimation of the shift of the brittle fracture transition temperature during irradiation.

		Values for Cu, P Weld close to the core Basis material Weld material BM1 %	WM %	BM2 %	Coefficient (A)[23] for the estimation of the shift of brittle fracture transition temperature BM2	WM
Unit 1	Cu	(0.17)	0.086 - 0.104[1] 0.109 - 0,123[1] (0.12)	(0.13)	(11.2)	33 (35.5)
	P	(0.010)	0.032 - 0.034[1] 0.035 - 0.036[1] (0.036)	(0.012)		
Unit 2	Cu P		(0.18) (0.032)		(11.2)	(37.7)
Unit 3	Cu P		(0.12) (0.035)		(11.2)	(34.8)
Unit 4	Cu P		(0.16) (0.035)		(15.6)	(37)

() Values in brackets are estimated, referring to heaths manufactured during period of fabrication.
1 Couples of values from 2 different measurements.
GW1 Basis material below weld close to core.
GW2 Basis material above weld close to core.
2 Data from manufacturer.
3 For the calculation, values from litterature or standards were used as
AF (270°C) = 800 (P + 0,07 Cu) for basis material and weld material
or AF (270°C) = 1100 (P + 0,07 Cu) for basis material
AF (270°C) = 800 (P + 0,07 Cu) for weld material.

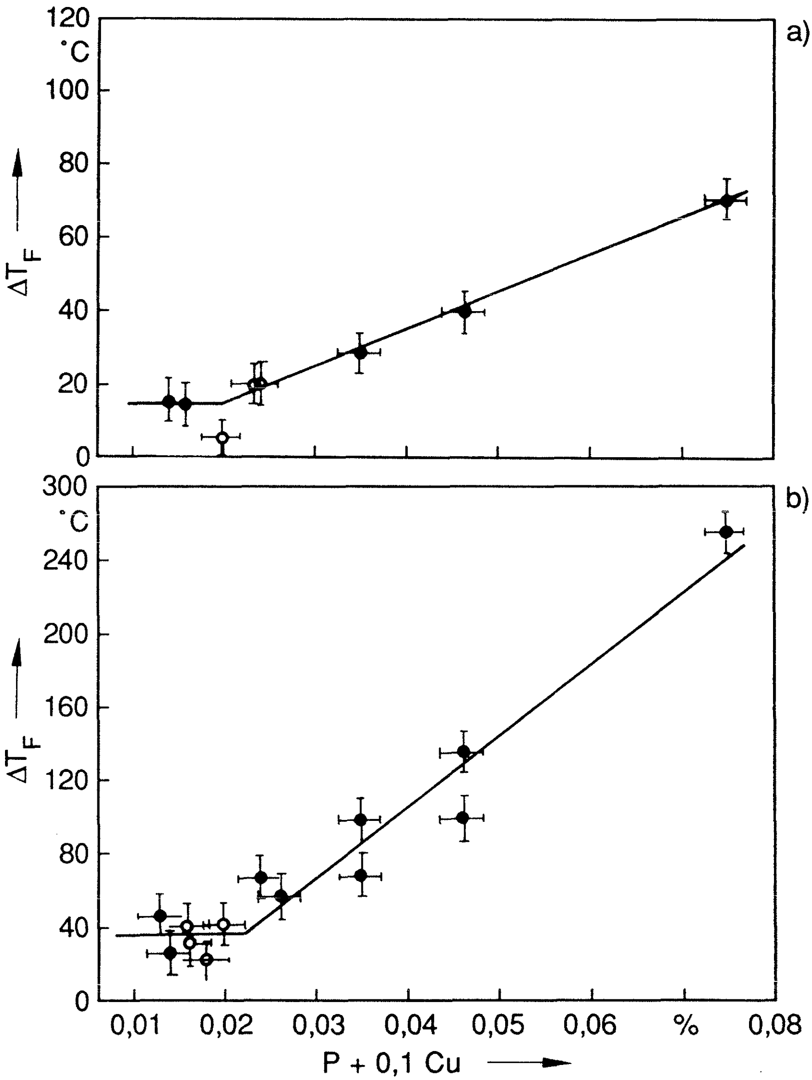


Fig. 4-6 :

Shift in the brittle fracture transition temperature[1]

ΔT_F shift in the brittle fracture transition temperature[1]

∘ weld seams

• 15Cr2MFA steel with differing contents of P and Cu

a) Specimens from Rovensk NPS ($F_{E > 0.5 MeV} = 1.1 \cdot 10^{19}$ cm^{-2})

b) Specimens from the Armenian NPS ($F_{E > 0.5 MeV} = 1 \cdot 10^{20}$ cm^{-2})

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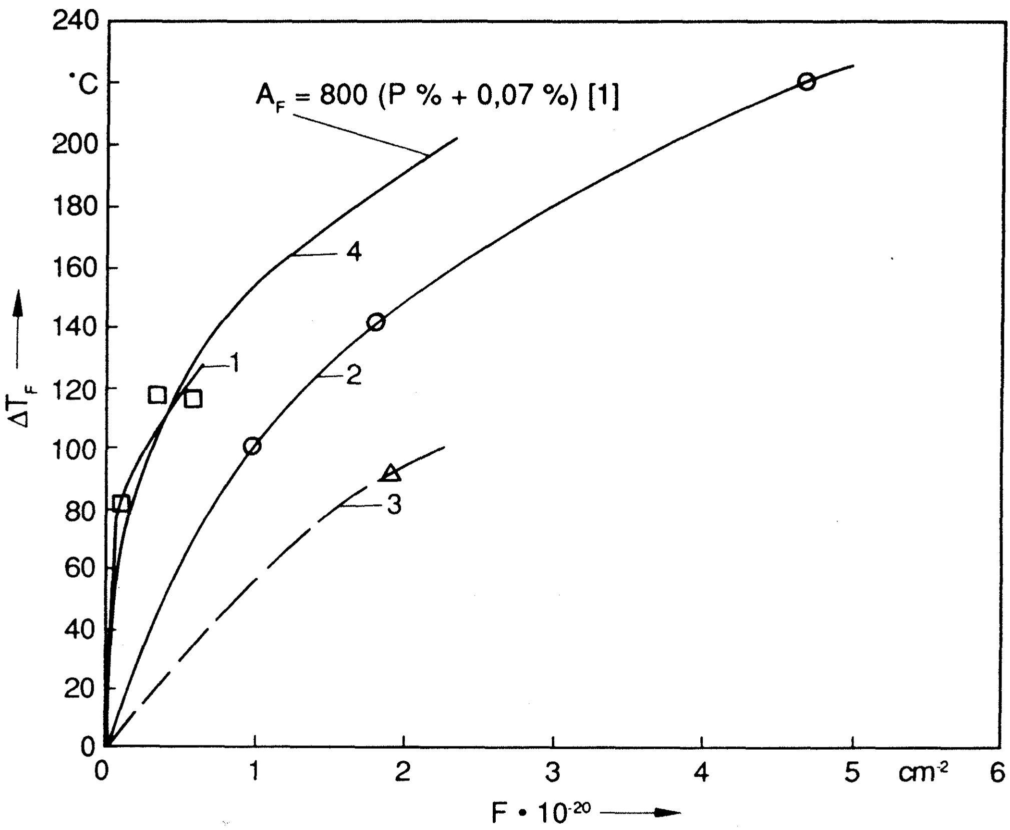
1 Source:
Amajev, A.D., A.M. Krjukov, M.A. Sokolov: Neutron embrittlement of the materials of the pressure vessel of the WWER-440 on the basis of experimental results from monitoring specimens

Fig. 4-7:
Radiation embrittlement of weld metal material at an ambient temperature of 270°C' and the calculated curve



1, □ $\phi \sim 4 \cdot 10^{11}$ cm^{-2} · s^{-1}
2, o $\phi \sim 4 \cdot 10^{12}$ cm^{-2} · s^{-1}
3, Δ $\phi \sim 7 \cdot 10^{12}$ cm^{-2} · s^{-1}

4 - mit Formel [1] errechnete Kurve

Amajev, A.D., V.I. Vichrov, A.M. Krjukov, M.A. Sokolov
Radiation embrittlement of materials of the WWER-440-pressure vessel

Table 4-4:

Data on the heat treatment of the reactor pressure vessel of unit 1

Rate of heating	22 K/h,
Annealing temperature	475 °C ± 10,
Stabilisation time	152 h,
Rate of cooling	30 K/h.

Annealing of the girth weld close to the core took place with basic material at the top and bottom, amounting to an overall height of 70 cm.

Shavings which were removed were analysed for Cu and P (see Table 4-3)

Hardness measurements were carried out on the internal surface of the reactor pressure vessel

Note:

For a supplementary check, 4 specimen plates were removed by spark erosion from the internal surface of the annealed region. The size of the specimen plates was 30 x 70 x 5 mm; 3 specimens were taken from the weld seam, and one specimen from the basic material. The investigation of these specimen plates still remains to be carried out; the intention is to perform notched impact bending tests and tensile tests, as well as supplementary investigations within the field of materials science.

The intended supplementary investigations on the specimens of material taken have not yet been completed. To improve the quantification of the result of the heat treatment, it has been specified that specimens both from the reactor pressure vessel weld seam 0.1.4 and also from the basic material will be taken from unit 2 before and after annealing. In the case of unit 3 this matter has not yet been fully considered, since the removal of specimens would require repair welds to be undertaken on the plating. An overview of the situation regarding the specimens, the intended scope of the investigation and the methodology employed is given in /1/.

Impairment of the material in interaction with the operating medium is discussed in section 4.1.5.

4.1.3 Loadings

The design of the equipment and pipes is based on the Soviet "Standards for computing the strength of NPS components". These specifications prescribe, for loadings under normal operational conditions, permissible stresses comparable with the corresponding FRG regulations. The limitation of the stresses in respect of operational transients and accidents is more stringent. However, in this connection it is necessary to give attention to the spectrum of accidents which is under consideration in each instance. The pressure/temperature conditions which are to be observed on start-up and shutdown are shown in Fig. 4-8.

In units 1-4, impermissibly pressure transients in the primary loop at low temperatures of the loop are not automatically prevented. Accordingly, chapter 7 requires that the administrative regulations regarding start-up and shutdown procedures as well as those regarding tests due to automatic restrictions should be supplemented, in order to prevent impermissible pressures in the cold condition (T < britte fracture transition temperature). An analysis is to be carried out regarding the possibility of consequential damage, to adjacent units as well, on failure of the reactor pressure vessel in the course of pressure testing.

The accidents considered in the design, including the relevant conditions for recording the most unfavourable loadings of the reactor pressure vessel, are shown in Table 4-5.

Loadings caused by internal events (e.g. fire and external events e.g. earthquake are discussed in chapter 8.

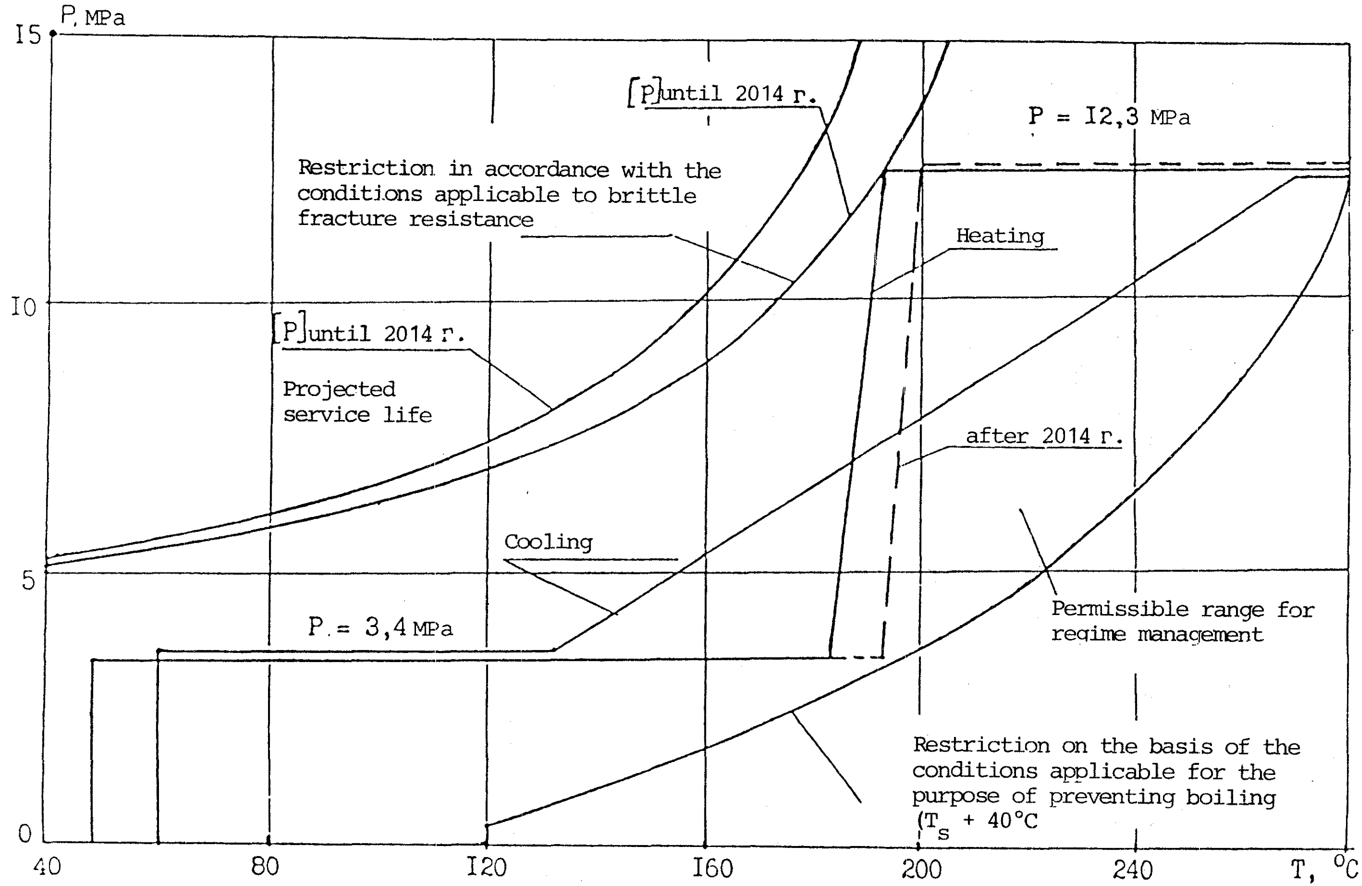


Fig. 4-8:

Permissible range for executing planned heating and cooling regimes

Table 4-5:

Basic loss of coolant accidents investigated and conditions applicable to the demonstration of safety with regard to brittle fracture of the reactor pressure vessel.

Condition of the installation:

- Emergency power required, main coolant pumps inoperative
- 4 emergency cooling pumps with a capacity of 50 t/h each

In the case of the unplated and plated construction of the reactor pressure vessels, the following accidents were considered in each instance:

1. Loss of coolant accidents within the primary loop

   - Leak equivalent to nominal bore 32, complete mixing
   - " " " " development of 6 cold entrainment paths
   - " " " " development of 1 cold entrainment paths
   - Leak equivalent nominal bore 20, development of 6 cold entrainment paths
   - " " " " development of 1 cold entrainment paths
   - Opening of pressurizer safety isolation valve
   - Non-closure of the pressurizer safety isolation valve (30 t/h)
   - " " " " valve (110 t/h)

2. Loss of coolant accidents within the secondary loop

   - fracture of the main steam line
   - non-closure of the steam generator safety isolation valve
   - fracture of the main steam line, fast-flowing medium

- Reactor pressure vessel

The computed stresses for normal operation, start-up and shutdown and hydro tests are listed in /1/. The stresses are below the permissible values.

A matter of particular importance is operation under load with supplies of cold water at operating pressure or pressure during an accident. In these circumstances additional high levels of tension are as a rule generated at the internal surface of thick-walled components at operational temperature (thermal shock), so that where frequent changes take place cracks may form or at a single loading latent cracks may be stimulated to initiation or growth in an unstable mode.

In the case of the reactor pressure vessel, safety analyses were performed for the weld seam in the region close to the core, with hypothetical latent cracks, in accordance with the regulations. Where cooling is uniform in the circumferential direction (rotational symmetry), it is mainly longitudinal cracks which are significant, while where cooling takes place in marked bands entrainment paths circumferential cracks are as a rule important.For accidents during which emergency cooling is required, stress conditions occur locally due to unsteady band cooling of the wall of the reactor pressure vessel, and these stress conditions may jeopardise the integrity of the reactor pressure vessel. These additional stresses were computed for leak-initiated events within the primary and secondary loops as far as the case of a design basis accident involving leakage (leakage area 10 $cm^2 \triangleq$ leakage of nominal bore 32)[1] with due consideration being given to the technological conditions of the reactor installation, and their effect on the stability of assumed cracks as far as a depth of 1/4 of the wall thickness was analysed. On the basis of these results, the permissible brittle facture transition temperature for the weld metal of the girth weld 0.1.4 of the reactor pressure vessel is in each instance limited (see also section 4.1.2).

[1] see also the definition in section 2.3

To make a conclusive assessment of the results available to date, it is necessary to carry out supplementary investigations of the mixing conditions prevailing in the main coolant loop and in the reactor cavity, of the temperature development within the cold band, of the heat transfer conditions prevailing at the reactor wall and of the influence of material properties which are correlated with the temperature dependence. Additionally, to assess the safety reserves of the reactor pressure vessel it is necessary to analyse the aspects concerned with increasing the injected quantity of the emergency cooling system and loadings due to increased asymmetry (3 adjacent cooling bands are possible in the event of switching operations to localize a leak).

Given the results of these supplementary investigations, it may be necessary to decide upon modifications to the installation (e.g. hot-leg injection of the emergency cooling).

As regards the loading and stress conditions of the reactor pressure vessel head together with its welded-on nozzles, it will be necessary to carry out in-depth analyses, including examinations of crack growth, with due consideration being given to the test restrictions.

- Steam generator

Steam generator tubes

The steam generator tubes, of dimensions 16 x 1.4 mm (material 08Ch18N10 T) are considerably oversized. This advantage is greatly reduced by advanced pitting corrosion in conjunction with stress corrosion cracking, especially in the steam generators of units 1 and 2. Within the steam generators of these units, approximately 20% of the tubes have locally developed a wall thickness reduction of between 60 and almost 100%, especially in the region of the supports; further details are given in /1/.

For the existing degree of damage, the hydro test on the primary loop does not provide any certainty in comparison with the loading due to fracture of a main steam line (MS fracture). The required minimum wall thicknesses to reduce the loading arising from the compressive test and for the maximum possible pressure difference in the event of MS fracture are different by only tenths of a millimetre, which in the event of active pitting corrosion can quickly be consumed.

At the present time, steam generator tubes are sealed in circumstances in which in the sealing test they are found to be leaking or the eddy current test indicates a wall thickness reduction >90%. A defect length criterion is not applied. To date, the eddy current test has covered approximately 20% of the tubes.

Approximately 60 tubes removed from the steam generators display individual corrosion defects having a length of up to 20 mm, predominantly in the axial direction. With the existing degree of damage, the rupture of one or more tubes cannot be ruled out.

To determine the sizes of leaks, it will be necessary to perform more extensive analyses having regard to the strength values actually applicable, the defect conditions for 100% tested steam generator tubes and the loadings in the event of incidents and accidents.

Collectors

As far as the collectors are concerned, it is necessary to consider not only the design calculations but also, in particular, loadings in the event of incidents and accidents occurring within the secondary loop (e.g. leak in steam lines, steam-out of a steam generator, cold-leg injection on failure of the high-pressure preheaters, thermal shock loading of the collectors in the event of leakage of the inner feedwater line). At the present time, only estimates are available regarding the consequences of some instances of loading. The calculations which are not yet available are to be completed within a short period. Accordingly, only the nature, scope and frequency of in-service tests for assessing the condition can currently be relied upon for the purpose of demonstrating safety.

· Pipes of the primary loop

A design calculation is not available. A follow-up calculation, presented by the operator, performed on the main coolant loop (MCL) was made for normal operating conditions (operating pressure 12.5 MPa. operating temperature of the hot leg 300 °C, cold leg 270 °C). The MCL nozzles of the reactor pressure vessel were adopted here as anchor points and all bearing surface (see Fig. 4-3) were assumed to be fully operational. On these assumptions, the longitudinal stresses of critical importance for a guillotine breach in the MCL were calculated as having, for example, the following values:

Axial membrane stresses ~ 50 MPa; as maximum value of the longitudinal bending stress ~ 20 MPa at the suction bend of the main coolant pump; see also /1/.

Operational experience gained to date shows that in certain regions the effect of temperature changes or of oscillations was underestimated. The damage to the spray line of nominal bore 100 of the pressurizer injection unit or at a T-piece of the emergency cooling system represents examples of inadequate consideration being given to fatigue stresses. The loading cycles for fatigue analyses still to be performed are to be determined by test programmes which also include operation in the start-up and shutdown phases.

In safety verification, the designer assumes that the fracture of the MCL of nominal bore 500 and adjoining pipes of nominal bore 200 is ruled out. These statements are to a large extent based on the specified production quality, the high ductility of the material and the low stress level applicable within the system. The Soviet Union has set up studies which use current methods to determine very low fracture probabilities. As far as the pipes in units 1 to 4 are concerned, no special analyses have to date been undertaken.

An examination, specific to the particular installation, concerned with ruling out the fracture of pipes requires a detailed assessment of the stresses which exist in each instance, the consideration of the additional thermal loadings determined from the test programmes, the estimation of the effects derived from geometric discontinuities (e.g. local inadequacies in grinding, wall thickness variations etc.) and a confirmation of the results of the earlier tests, which were conducted from the external surface, by supplementary internal tests based on particular areas. Some of this information may be obtained within the context of the impending 100,000 h programmes.

By way of redundancy for these verifications established by operational experience and calculation, it is necessary to set up a leak monitoring mechanism, which detects at an early stage any through-wall cracks which may be forming. Furthermore, it is necessary to establish the loading conditions for which the stability of postulated cracks is to be verified.

Pressurizer

The computed stresses for normal operation and hydro tests are shown in /1/. The stresses are below the permissible values. The available calculations are to be supplemented by more precise loading assumptions which take due account, for example, of temperature changes due to discontinuous operation at the injection unit as well as media thrust (piston flow) and, where appropriate, temperature stratifications within the MCL connecting lines at the bottom of the pressurizer.

· Main steam and feedwater system

When the report was drafted, the design computations for these systems were not available at the SAAS and could not be taken into consideration.

In respect of these systems, calculations are necessary which take account of the actual condition of the building and the reference quantities to be obtained from displacement measurement programmes.

4.1.4 Condition of the equipment and pipework

The condition of the systems and pipes is monitored, during operation, by means of

- inspection,
- pressure tests of tightness,
- hydro tests,
- periodic non-destructive material tests and
- analyses of the operational behaviour.

Success in the pressure test of tightness at 4 MPa and 14 MPa respectively is a prerequisite for start-up after an important maintenance program of the reactor installations. The periods of time allowed for hydro tests (4 years) are set out in standards (Technical Standard 43272 and Technical Standard 30316).

For the periodic non-destructive material tests, the following items are specified in the basic test programme, see /1/,

- test locations,
- test procedures,
- scope of test and
- test interval

The tests have been staggered in terms of time within unit-related annual plans to remain within the prescribed test intervals. The results are documented.

In comparison with the scope of the tests applied to installations within the Federal Republic of Ge many, more visual inspections and surface crack tests on the internal surfaces take place. As a result of the austenitic pipe system and the bimetallic welds on the main components as well as the special nozzles designs on the head of the reactor pressure vessel, there are different test conditions relevant to the application of ultrasonic techniques as compared with installations within the Federal Republic of Germany. It is also necessary to accounter for the crowns which are left on the weld seams on the pipes and parts of the equipment. Special restrictions on volume testing are given, for example, to

- the meridian weld in the reactor pressure vessel top cover (ES weld seam heat-treated as well),
- webs between the nozzles,
- welds of the nozzles of the fuel assembly outlet temperature system (welded onto the outer plating),
- nozzles of the protection and control rod system (shrunk, with a sealing weld on the internal surface),
- connecting welds of the bends of the MCL to the straigh sections.

It should also be noted that the weld seams of the 3 pressurizer connecting lines of nominal bore 200 are currently tested only externally by surface crack test procedures. A radiographic test is provided for the welds to the pipe fitting.

The process-related requirements and specifications concerning register tolerance limits are essentially in conformity with Nuclear Technology Commission [KTA] Rule 3201.4. In the ultrasonic testing of austenitic weld seams, assessment is carried out only of those data which have a signal-to-noise ratio exceeding 10 db. Further details of the technical test conditions and error assessments are given in /1/. Data concerning the qualification of personnel, supervision of the performance of the tests and documentation are also contained therein.

On account of the various executions of the weld seam (use of spacer rings, root sag or crown only smoothed), there are a multiplicity of signals which are assessed by means of a particular evaluation algorithm. The data which are compiled in /1/ and which are subject to a registration obligation require a further, more detailed assessment or supplementary test. A comparison with the current tests on the internal surfaces in unit 2 permits a supplementary assessment of the reliability of previous tests. Accordingly, a conclusive assessment of the test results cannot yet be made. As concerns the completeness of the tests, the evidence will be presented on a basis specific to individual components.

Within the operating period hitherto, the tests the primary loop were essentially performed within the set terms. The findings are summarized in /1/. The majority of the findings, especially at the surfaces, were ground out or repaired by welding during the shut down periods of the reactor installations.

In the secondary loop, there are in some cases considerable deficiencies in the implentation of the newly introduced standards (see section 4.1.1) for the hydro tests, weld seam tests and wall thickness measurements. Mention may be made, for example, of the absence of hydro tests, which are due according to the prescribed cycle, on the main steam system for every semi-system of units 1 and 3, as well as on the feedwater system for every semi-system of units 1 to 4.

Outside the pressure resistant compartment system, only 30% of the ultrasonic weld seam tests in the main steam system and feedwater system have been completed within the set term. It is also necessary to record the absence, in some cases, of ultrasonic wall thickness measurements on pipe fittings, where recalculations have indicated cases of undersizing.

At the present time, the following findings have been made in the pressurized walls of the equipments and pipes:

- Reactor pressure vessel
  - Corrosion pits (up to 5 mm deep, the size of a coin) on the unplated internal surface of the reactor pressure vessel in unit 2.
  - Numerous ultrasonic indications in the region of the fusion line base material/plating on the reactor pressure vessels in units 3 and 4. The indications are interpreted as adhesion defects, slag inclusions and underplate cracks.
  - Stress corrosion in the plating of the reactor pressure vessel flanges of unit 1 and 2 in the region between the operational seal and the emergency seal. The locations found were only in part ground out. There are no indications that the plating has been ruptured. The leakage monitoring lines penetrating between the seal grooves were exchanged due to stress corrosion damage.
  - Ultrasonic tests carried out on the girth welds in the cylindrical part of the reactor pressure vessels in units 1 to 4 revealed sporadic indications having longitudinal dimensions of a maximum of 60 mm. These indications are mainly situated within the volume of the weld seams, relatively close to the external surface.

- Pressurizer
  - Pits on account of earlier pitting corrosion of the unplated internal surface of the pressurizers in units 1 to 4. These locations were retained as reference surfaces and will be monitored annually. Progression of the corrosion pits was not found.
  - An ultrasonic indication within the volume of the weld seam

3.15 (upper bottom) in the pressurizer of unit 1. In the course of repeated examinations, no change of the indication (5 x 40 mm) was found. The targeted acoustic emission analysis during a compressive strength test revealed no signals which could be correlated with the indication.

- Steam generators

  - Pitting corrosion in conjunction with stress corrosion on steam generator tubes, especially in the regions of the supports. The damage to the steam generators of units 1 and 2 is especially advanced; in these cases, approximately 20% of the generator tubes exhibited eddy current indications, which correspond to wall thickness reductions of between 60 and 100%. After alteration of the water treatment in the secondary loop, the progress of the damage declined to a great extent.

- Main coolant lines

  - Numerous ultrasonic indications subject to a registration obligation, which can be conclusively assessed only after the results of the inner surface cracks tests have been made available.

  - Slag inclusions were detected in the radiographic tests on the peripheral welds of the main block valves. The size of the indication exceeds the values permissible in accordance with Technical Standard 43274 (production test). The former assessment of the strength of the components with possible defects is not conclusive and must be examined further.

In the independent check concerning the execution of tests and assessment of the results, there are considerable deviations from the requirements in the FRG. In respect of the pending tests, improvements to the check are necessary.

4.1.5 Interaction of the materials with the medium

- Primary loop

Power operation

The water treatment corresponds to the design; mixed alkalination using ammonia and caustic potash at $pH_{25} \geq 6$ corresponding with $pH_{265} = 7.1 \div 7.3$. With the very low chloride and oxygen levels, no selective or local onset of corrosion induced by the chemistry of the water was detected except in the regions of the seals.

Outage

Undefined water conditions during outage led to the local activation of the surfaces on the unplated reactor pressure vessels in units 1 and 2 and the pressurizer in units 1 to 4 (pitting corrosion). Since 1979, reducing conditions have been set in the stagnant water during outage, by means of hydrazine addition. After this, no further damage due to corrosion has occured.

- Secondary loop

Until the end of 1982, the water treatment was in accordance with the specifications of the design: an unconditioned operation in the presence of oxygen. In doing so leaks of sea water on the secondary side of the steam generator tubes have intensified the occurrence of pitting corrosion together with stress corrosion, especially in units 1 and 2. By hydrazine addition by condensate purivication at 100%, by exchange of the Cu-containing materials in the low-pressure preheaters, by wet preservation of the steam generators during outage and by removal of the deposits on the steam generator by chemical treatment during outage as well as by improving the monitoring of the steam generator water, it was possible to achieve a considerable reduction in the level of damage to the steam generator tubes. A check is to be made as to whether the corrosion problems on SG tubes, pipes and vessels can be solved by tight condensers and adjusted water chemistry. On account of the procedure using a reduced O_2 content at a low pH value, corrosion by erosion was detected in regions with 2-phase flow and in the high-pressure preheaters.

To date, no clearly verifiable reduction of the erosion corrosion has been achieved by tentative addition of octadecylamine. No other types of damage were detected.

- Deviations from standard

The conditioning of the coolant in the primary loop is monitored on a discontinuous basis. For each refuelling cycle and each unit, on average 5 deviations from the prescribed chemical standard values were detected in power operation. As a rule, these deviations were of brief duration (a few hours) and slight (< 10% from the value according to standard). These deviations were caused by incorrect feed quantities of metered chemicals, as well as by the use of contaminated chemicals.

Ion exchange resins arrived in the primary loop during a regeneration of filters. However, experience gained during operation does not give any indications that crack-initiating chemical conditions have arisen.

Twice, on average, per refuelling cycle and per unit, impurities passed into the secondary loop together with the cooling water via leaks in the turbine condenser.

The deposition - caused by the nature of the process - of impurities on the outer surfaces of the SG tubes, may create conditions promoting pitting corrosion. These deposits are removed by chemical treatment at 2-3 year intervals, using a 2-stage process. The quantities and composition of the residues are summarized in /1/.

The possibilities of exceeding the permissible values for impurities, and especially of the chloride content in the primary loop, are to be analysed within the framework of the system-related investigation and evaluation of operational experience.

4.2 Supplementary investigations

The statements made hitherto concerning the brittle fracture transition temperature of the weld seam 0.1.4 of the reactor pressure vessels are to be made more specific by the following investigations:

- The results of the tests on the specimens obtained from unit 1, including correlations with ISO-V specimens, are to be assessed on a conclusive basis before the start-up following general maintenance.

- Prior to and after the thermal annealing of the reactor pressure vessel in unit 2, it is necessary to take from the weld seam 0.1.4 specimens for destructive testing to measure the notched impact toughness and characteristic strength values, as well as metal shavings for chemical analyses from graded depths. The conditions for return to service are to be established with consideration also being given to the results of the investigations of the specimens from unit 1.

- In order to broaden the data basis, it is intended to take up the offer made by Soviet experts to carry out joint tests on further specimens (irradiated and non-irradiated). An associated working programme is being formulated, by MPA Stuttgart involving all participants.

The results available to date concerning the asymmetric temperature distribution in the reactor pressure vessel wall in the course of emergency coolant injection are to be checked by comparative calculations:

- It is necessary here to take note of the temperature dependence of the properties of the materials as well as the heat transfer conditions at the reactor wall, and to check the applicability of approximations by comparison with 3D FEM calculations. The data sets should be agreed by all participants.

- In the case of backfitting of a more powerful emergency cooling system, it is necessary to analyse afresh the accident spectrum considered, together with the altered performance parameters.

- When using switching operations to locate a leak in the primary loop, it is possible for the emergency coolant water to be injected only into 3 adjacent circulation loops. The calculation of this loading condition for the reactor pressure vessel is to be added.

The currently applied sealing criterion for the steam generator tubes is to be reformulated. It is necessary to formulate a criterion which, where appropriate, takes into account not only the defect depth but also the defect length. The test-related implementation of the requirements of the modified sealing criterion is to be demonstrated. The results of the SG tube test which is currently in progress are also to be included.

The ultrasonic testing of the reactor pressure vessels in units 3 and 4 has shown numerous indications from the region of the plating/base material fusion line. An improved assessment of the indications is required, possibly involving the use of special analysis procedures. It is necessary to analyse the behaviour of model defects for cracks underneath the plating under operating load and under accident-initiated load, in order to determine critical defect configurations.

The mixing behaviour on injection of emergency cooling water into the cold legs of the MCL under various conditions determined by the flow pattern (from natural circulation to the no-flow condition) is to be analysed. The results are to be used to draw conclusions regarding the intensity of the cooling effect for the nozzles and the wall of the reactor pressure vessel.

Appropriate measurement programmes are to be implemented in order to provide a precise statement of the unsteady cyclic temperature loading

- at the spray line of the pressurizer injection unit and
- at the connecting lines, of nominal bore 200, from the pressurizer to the MCL (media thrust and, where appropriate, temperature stratification).

To supplement the design calculations, it is necessary to carry out displacement measurements

- at the MCL
- at the main steam line and
- at the feedwater line.

4.3 Safety-related assessment and measures required

In their structural design, the components essentially also correspond to the standards and technical regulations currently applicable in the GDR. In comparison with the technical regulations and safety-related requirements in the Federal Republic of Germany, as far as the primary loop is concerned there are substantial deviations in two respects:

- the high neutron fluence within the cylindrical region of the reactor pressure vessel,
- testing restrictions in partial areas, e.g. nozzle welds, pipe welds without supplementary evidence.

In terms of long-term operation, the materials employed for the various components have essentially proved satisfactory, with the exception of the weld seam 0.1.4 of the reactor pressure vessel close to the core with its high sensitivity to neutron irradiation. The extensive damage to the steam generator tubes is not caused by the materials, but is to be attributed to inadequacies in the water chemistry of the secondary side during the first years of operation.

The high fluence, in conjunction with the sensitivity of the weld metal of the seam 0.1.4 close to the core has led, in the case of the reactor pressure vessels of units 1, 2 and 3, to an unacceptable impairment of the condition of the material. By applying a controlled heat treatment (thermal annealing), the material is to a large extent restored to its initial condition. In the case of several reactor pressure vessels of this type, the annealing has meanwhile taken place, inter alia in 1988 in the case of unit 1. In the case of units 2 and 3, this heat treatment is will take place in 1990. There are no objections against this procedure. With regard to the quantification of the recovery effect, reference is made to the statements in section 4.1 and to the supplementary material investigations in section 4.2.

Having regard to the statements contained in section 4.1 concerning the safety evidence, the use of shielding assemblies is recommended in principle for the further operation of all units. This also applies to return to service after annealing, in order to maintain higher safety margins on a longer-term basis.

The loadings which have hitherto been taken as a basis for the demonstration of safety are inadequate according to the current state of knowledge, and must be supplemented for various component areas. Such requirements are included in sections 4.1 and 4.2. In particular, it is necessary to implement measures which reliably prevent the exceeding of the permissible pressures below the brittle fracture transition temperature in the course of start-up and shutdown. The procedure for the tightness test at a high pressure level and $T > 100°C$ is to be reviewed. In this connection, supplementary analyses are required for accidents in connection with the pressure test, including possible consequential damage to adjacent installations.

Operational experience has shown that unexpected thermocyclic loadings have occurred, especially in the region of the pressurizer injection system, just as they have occurred in reactor types from other reactor manufacturers. Expanded measurement programmes are demanded in section 4.2, and an examination of the assumed loads for the pipe calculations of the primary and secondary loops is also required.

On the basis of all the reported results of the in-service tests, there are to date no indications of stress corrosion in the medium-wetted vessels and pipe areas of the primary loop. The defects found were restricted to the gasket areas, which normally are not in contact with the medium.

Essential findings on tested parts of the secondary loop are pitting corrosion with subsequent stress corrosion on the external surface of the steam generator tubes, centred on the support areas. In addition, there were considerable findings of cracks due to stress corrosion on the austenitic collectors of the steam generators on the external surface in the region of a covered weld seam. The instances of damage to the collectors have been remedied, and the required measures taken. In the course of inservice tests, no new indications have emerged to date.

Section 4.1 contains further statements concerning the corrosion damage during shutdown on the primary side.

The condition of the components will be checked periodically by using non-destructive test, methods and pressure tests at specified short time intervals. As far as the safety assessment is concerned, the internal surface crack examinations have particular importance. Relevant statements are made in section 4.1.4 concerning the test restrictions in the case of individual components as well as concerning the reliability of the testing of the austenitic components. Particular importance is ascribed to the supplementary surface crack examinations from the inside as well as to the destructive tests within the framework of the 100,000 h programmes, for the purpose of clarifying the open questions. A conclusive assessment of the results of the tests on the main coolant line and of the indications on the plating and in the region of transition to the base material of the reactor pressure vessels of units 3 and 4 cannot be made at the present time.

For the purposes of the overall safety assessment, the possibility of leak events is to be investigated. On a worldwide basis, no large fractures have occurred where austenitic pipes are concerned. Required measurements for checking the loadings and in-depth investigations for the interpretation and assessment of the ultrasonic tests on the austenitic components are discussed in sections 4.1 and 4.2. In addition, indications are to be given of the loadings to demonstrate the leak-before-break behaviour. It is necessary to improve the leakage monitoring for the early detection of leaks on the primary and also on the secondary side. These measures should be introduced immediately for the component areas which are exposed to increased thermocyclic stresses. The in-service tests, described in the basic test plan, on the pipes of nominal bore 100 and nominal bore 200 (essentially, the external surface crack test) do not guarantee freedom from cracks to an adequate extent. In the course of the inspections, in the case of all units the pipes of nominal bore 200 and nominal bore 100 in the primary loop are also to be tested to be free of cracks on the inner surface. An appropriate test programme is to be presented.

The currently applied sealing criterion for the steam generator tubes is to be reformulated. It is necessary to formulate a criterion which takes account not only of the defect depth but also of the defect length. Specifications regarding the leakage rate are to be supplemented.

No statement can yet be made regarding possible leaks in the main steam and feedwater systems.

It is expected that the current investigations on reactor pressure vessel materials and concerning the assessment of the indications of the ultrasonic test on the austenitic pipes as well as the extensive steam generator tube tests will in the coming months permit an improved assessment of the condition of parts of the equipment and pipes. Furthermore, requirements have been set regarding expanded demonstrations and backfitting measures. As far as units 1 and 4 are concerned, there is no reason for premature shutdown on the basis of the assessment of the condition of the components.

5. ACCIDENT ANALYSES

5.1 Assessment of existing accident analyses in terms of the actual state

The reference plant for all accident analyses presented is unit 1. The essential difference as compared with units 3 and 4 is the smaller capacity of the emergency core cooling pumps in units 1 and 2. The scope of the evaluation is based on the "Guidelines for the assessment of the design of nuclear power plants with pressurized water reactors to counter accidents within the meaning of § 28 para. 3 of the Radiation Protection Order - Accident Guidelines -" of the Federal Minister of the Interior in the version dated 18th October 1983, as well as on the draft of the Standard Technical Document (NTD) of the COMECON member states "Standardized Content of the Safety Report" in the version of March 1987.

The analyses under consideration were essentially performed using the programs TRAVO, RAMPA and RELAP4/MOD6, which are used by the utility Kombinat Kernkraftwerke, and using the program KONTUR which is used in the Soviet Union.

The code TRAVO was developed by the utility Kombinat Kernkraftwerke. TRAVO is a computer code for the simulation of transients close to operational conditions without boiling of the coolant in the primary loop.

Particular emphasis is laid upon the modelling of control systems. Except in the latest version 3.3, the mass flow in the primary circuit is modelled as a time function.

The codes RAMPA-M and RAMPA-G for accidents with an intermediate and a large leak respectively are advanced developments of the similarly named Soviet computer codes, (direct cont. without new paragraph).

The three conservation equations for the mixture are solved. Mechanical nonequilibrium is taken into account by a drift flux model or by a bubble rise model. A relatively coarse nodalization of the primary loop is used due to limited computer capacity. In more recent analyses, the code RELAP4/MOD6 is used instead of RAMPA.

The computer code KONTUR was developed by the Thermotechnical Institute in Moscow. Regarding unit 1 of the nuclear power plant, the program was applied for secondary-side leaks only. A description of the models of the program is not available.

Reliable verification calculations referring to tests specific to the WWER are not available for the codes TRAVO, RAMPA and KONTUR.

In order to check the presented results, plausibility analyses and simplified calculations without using computer codes were carried out.

5.1.1 Loss of coolant from the primary coolant system within the confinement system

- Design basis accident: equivalent leak with nominal diameter 32 mm

This case is the design basis accident for the emergency core cooling system and corresponds to the break of the spray line to the pressurizer having a nominal bore of 100 and flow limiter. This connection is located in part of the primary loop which cannot be isolated. The break of the connection lines of nominal diameter 50 in that part of the primary loop which can be isolated is not covered by this. In the latter case, manual measures are required.

The analyses show that the accident is reliably controlled by two emergency cooling pumps of the type EP 50; in this case, the secondary side is initially required for heat removal. On this basis, a single failure in the event of total loss of power is not covered. The control of the situation using only one pump must be demonstrated by further analyses; a potential for this is apparent.

In units 3 and 4, on account of the higher injection rate of the pumps of type ZN 65 it can be assumed that the situation will be controlled with only one pump.

Leaks of diameter < nominal diameter 32 are covered by the design basis accident, as far as the effectiveness of emergency core cooling is concerned.

RALOC analyses with realistic boundary conditions showed that the signal P_{DE-BOX} > 1.3 bar is only reached after approximately 5 mins. (see chapter 6.4). Without spray from the sprinkler system, the signal would appear only after approximately 3 mins. The reason for the later response as compared with the results of earlier analyses is the effective condensation on structures within the confinement system. It emerges from the analyses that in the case of leaks of nominal diameter < 32 it is necessary to expect a very late response or indeed a response failure on the part of the above-mentioned signal. This leads to the requirement that the reactor protection signal should be friggered at the pressure p = 1.2 bar, at which the activation of the sprinkler pumps is also triggered.

- Pressurizer steam-side leak of nominal diameter 90

The existing accident analyses regarding the rupture of the connecting line between the pressurizer and the safety valves show that by injection of two emergency core cooling pumps of the type EP 50 the leakage mass flow is compensated by the injection mass flow after approximately 35 mins. Within the analyzed period, the core is constantly covered by a two-phase mixture.

- Leaks > nominal diameter 100

The analysis of the equivalent leak of nominal diameter 100 in the cold leg with injection by two pumps of type EP 50 shows that after approximately 30 mins. the leak mass flow rate is compensated. Fuel rod damages are to be expected, while core meltdown can be excluded.

The equivalent leak of nominal diameter 200 was also analyzed with safety injection by two pumps. At the end of the blowdown phase after approximately 500 secs., the leakage rate cannot be compensated by two EP 50 pumps. If four pumps are available, i.e. without total loss of offsite power and single failure, there is on the other hand the chance of preventing an extended core meltdown by timely refilling.

For the case of double-ended break of the MCL, no reliable thermohydraulic analysis is available. Estimates show that, even with four pumps of type EP 50, it is not possible to prevent core meltdown.

5.1.2 Damage to steam generator tubes

Using the computer code TRAVO, analyses of steam generator tube leaks were carried out for the case of the double-ended rupture of 1 to 2 tubes.

The analysis of the double-ended rupture of one tube shows that in certain circumstances no automatic reactor shutdown occurs, since this can be prevented by the activation of the emergency core cooling pumps before the reactor protection system responds. For this reason, a check should be made as to the efficacy of different excitation criteria for the emergency core cooling and the reactor protection system in consequence of a low pressurizer water level. From the point of view of the emergency core cooling system, this break can be compensated by means of one emergency cooling pump.

With larger leakage cross-secs.tions, the excitation criteria for the reactor protection system are reached. At least two emergency cooling pumps of type EP 50 are required for the compensation of the double-ended rupture of two steam generator tubes.

A series of manual measures are, in principle, required for the long-term control of steam generator tube leaks. The time available for this is not limited by the water stocks in the emergency boration vessel, but by the need, on account of the risk of condensation shocks within the main steam system, to isolate the leak before the steam affected generator is filled. For a 2F break of one SG tube, approximately half an hour is available for this. The required manual measures are specified in the emergency operation regulations. The essential steps are: checking of the reliable shutdown of the reactor, isolation of the primary water purification system and location and isolation of the leak by alternate isolation of three respective primary circulation loops by means of the main gate valves.

5.1.3 Loss of coolant from the secondary circuit

Until the installation of quick-closing valves in the connection lines between the main steam lines and the common steam collector (MSC), large leaks within the secondary loop were not considered as design basis accidents. Within the context of the present assessment, the thermohydraulic processes in the event of loss of coolant from the secondary loop were considered without taking into account operational leakages and breakes of steam generator tubes. It emerges that the accidents with simultaneous loss of onsite and offsite power in general proceed less problematically, on account of the reliable shutdown of the reactor associated therewith. As a result of the more intense over cooling of the primary system in the event of total loss of power with subsequent injection by the emergency core cooling pumps, the further development of the accident may involve the response of the pressurizer safety valves.

For the accidents with leaks on the secondary side which are considered in the text which follows, a plant condition after the backfitting of quick-closing valves (QCV) in the main steam system was taken as a basis.

- Break of the main steam line in the steam generator box

As a result of an increase in pressure in the SG box, the reactor is shut down within 1 to 2 secs. The next automatic action is the response of the interlok 6.4.19 (P_{SG} < 35 bar AND $\nabla p_{MSC,SG}$ > 5 bar). This brings about the closure of the QCV (2-5 secs. closing time), the immediate shutdown of the MCP as well as the water-side isolation of the steam generator of the affected reactor coolant loop. The flashing in this steam generator leads to subcooling of the primary system with subsequent core cooling injection; in this case, the coolant temperature does not fall below 180 °C. If the MCP does not start up again after running down, the result is the opening of the pressurizer safety valves. This applies to the case of loss of preferred power with failure to switching over to the electrical reserve supply.

- Break of the main steam line upstream of the QCV

At the start of the accident, the rapid drop in pressure in the main steam manifold causes the trip of at least one turbine. Although both turbines should be shut down by the same pressure in the header, it is possible as a result of random deviations of the set values in combination with pressure differences in the header that one turbo generator set remains in operation and thus the reactor is not shut down. According to the existing calculation, the interlock 6.4.19 will respond after approximately 4 secs., after which time the reactor has to be shut down manually in accordance with the operating manual.

- Break of the main steam line upstream of the turbine

Because of the smaller leak size, the pressure decrease within the main steam header proceeds more slowly than in the cases previously considered. In certain circumstances, there is no response by the interlock 6.4.19, nor is the reactor shut down by the shutdown of both turbines. In this case, manual measures are indispensable. Reliable "best estimate" computations are required.

- Rupture of the main steam header

As a result of the rapid pressure drop in the header, the fast-acting main turbine valves of both turbines close shortly after the initiation of the accident, with subsequent shutdown of the reactor. A short time later, at a pressure of P_{MSS} < 35 bar, the lock 6.4.6 responds, and this brings about the closure of all QCVs. As a result of this, the main steam header is separated from the main steam system. The decay heat removal from the primary system occurs via the SG safety valves. On the primary side, neither the response criteria for the emergency cooling system nor the response pressure for the opening of the safety valves at the pressurizer become operative in the calculation presented. In the short term, no manual measure is required. In the long term, the objective is to terminate the loss of inventory loss via the steam generator safety valves.

- Unintentional opening of a turbine bypasss valve, SG relief valve or a SG safety valve

In the event of the unintentional opening of a turbine bypass valve or a relief valve at the steam exhaust station (atmospheric exhaust system), or of a steam generator safety valve (SG-SV), the capacities of which amount to 440 t/h (turbine bypass valve or atmospheric exhaust system) or 250 t/h (SG-SV), no criteria for reactor protection or criteria for closing one or more QCVs are reached, because the steam loss is compensated by the control system. Using manual measures, the unit is to be taken out of service and the leak isolated.

- Break of the feedwater header or failure of all feed water pumps

These two cases proceed on an identical basis, assuming that the nonreturn valves in the feedwater lines operate in accordance with the design specifications. The shutdown of the reactor takes place after secondary-side pressure drop or primary-side pressure increase after approximately 4 mins. when the secondary side has almost lost the complete inventory. Even considering the further decay heat removal by injection and evaporation of emergency feedwater into the almost empty steam generators, this scenario of the accident must be regarded as unacceptable. Accordingly, attention is emphatically drawn to the installation (required in the 35 point programme) of a reactor shutdown signal initiated by a low steam generator water level ($L_{SG,mins.} = -0.5$ m). By means of this signal, the reactor would be shut down early enough that the inventory in the steam generators would be sufficient for decay heat removal over approximately 2h, even without feedwater supply.

- Break of a feedwater line between SG and check valve

As a result of the great pressure drop in the feedwater header, the feedwater pumps are automatically shut down. It must be assumed that the emergency feedwater pumps feed towards the leak. In the existing KONTUR analysis, no consideration was given to the possibility of an early shutdown of the reactor as a result of the pressure increase in the SG box. Reactor shutdown occurs only as a result of shutdown of the second turbo generator set after approximately 4 mins. as a result of low main steam pressure in the header. The residual water content in the unaffected steam generators is sufficient only for decay heat removal for a period of approximately half an hour. Accordingly, especially for such an accident, an automatic reactor shutdown should be provided in the initial phase, e.g:

- at low level in one steam generator,
- at lowin level in at least two steam generators or by failure of the feedwater pumps.

In this context, it is necessary to check the efficacy of the staggered triggering of the spray system and of the reactor shutdown system as a result of the rise in pressure in the SG box, since in certain circumstances in the current state of the plant this may lead to the prevention of the reactor shutdown.

- Break of a drain line (nominal diameter 80 or 50)

To date, such an accident has not been investigated. In the event of a break within the SG box, it has to be investigated whether the shutdown criterion is satisfied in consequence of pressure build-up in the box. When the criterion is not reached or the leak is situated outside the box, compensation of the water loss by main and emergency feedwater supply can be assumed. An automatic shutdown of the unit would not occur. Detailed accident analyses are necessary.

5.1.4 Loss of primary coolant outside the confinement system

No analyses are available with regard to these accidents. The lines of the emergency core cooling system are protected within the confinement by two valves and a non-return flap valve. In the event of a break between the feed pump and the non-return valve in the feed line to the primary water purification system, a leak of nominal diameter 50 outside of the confinement can occur if a single failure (failure of the non-return valve is assumed. It is to be expected that, in consequence of the long pipes and the flow resistances in the heat exchanger, no larger leaks will occur than in the case of the leak of nominal diameter 32. The leak must be isolated manually after being located, and the respective emergency core cooling pumps must be shut down. The emergency core cooling must be assured by the non-affected redundancy. If the reactor shutdown has not occurred automatically, manual initiation is requested.

Instrumentation lines do not lead out from the confinement system. Leaks in sampling lines are coped by the emergency core cooling system. They must be isolated manually.

5.1.5 Reactivity initiated accidents

The documented basis of reactivity-induced accidents is inadequate for units 1 to 4.

Previously, only computer codes for one-dimensional coolant channel analysis with point kinetics or one-dimensional neutron kinetics were available for the investigation of dynamic processes in the core.

Only in 1989 the state of development of the three-dim mensional dynamic programs reached a level which is suitable for accident analyses for WWER-440.

For units 1 to 4, analyses are available for the following accidents at full power:

a) ejection of the most effective control rod
b) break of the main steam line in the secondary circuit and loss of onsite and offsite power
c) spurious opening and stuck-open condition of 2 steam exhaust stations and loss of onsite and offsite power

The analyses which have been performed show that in the case of the accident mentioned under c), recriticality is expected, which however does not lead to serious effects.

Since the construction of the core and of the internals of the pressure vessel of types W-230 and W-213 is identical, the accident analyses performed for W-213 (unit 5 in the Greifswald nuclear power station) can be evaluated for units 1 to 4.

The results of analyses for short-term processes in the core are valid for units 1 to 4 too. Other processes, in which the design of the primary coolant system plays a significant part, require a differentiated consideration.

On the above mentioned assumptions, the analyses of the following reactivity-induced accidents which were performed for the W-213 can be included in the consideration for units 1 to 4:

d) Uncontrolled withdrawal of control rods (individually and in groups at zero power and full power conditions)
e) Ejection of the most effective control rod (under at zero power and full power conditions)
f) Drop of the most effective control rod
g) Opening of main gate valves in cold circulation loop and switching on the respective main coolant pump in the case of operation with 3 and 5 circulation loops

Incidents arising from power conditions are of minor importance than incidents arising from zero power condition, as far as safety is concerned.

In the event of the ejection of control rods at zero power condition (case e), damage to 3% of the fuel rods is to be expected. In the event of case g damage to fuel elements is likewise to be expected.

5.1.6 Loss of preferred power/loss of onsite and offsite power

At loss of onsite and offsite power reactor scram is initiated after approximately 2 secs. The pressure in the primary loop rises by a maximum of 3 bar. The setpoints of the pressurizer safety valves are not reached. The SG-SV can open for a short period of time in the initial phase. Main steam is passed out in a controlled manner via the fast-acting atmospheric exhaust system. In the longer-term phase, the feed to the steam generators is guaranted by emergency feedwater pumps operated by the emergency power supply.

5.1.7 Failure of several main coolant pumps (MCP)

Due to the low moment of inertia of the MCP, in the event of failure of the power supply it is necessary to ensure electromechanical rundown using the rundown energy of the turbo generator sets for a minimum number of MCPs. The simultaneous rundown of more than two MCPs leads to reactor scram.

The supplement to the Technical Project contains an analysis of the simultaneous failure of four six MCPs with immediate outage. This analysis involves a pressure build-up in the primary system leading to the response of the pressurizer safety valves. At hot fuel rods short-term boiling may occur. In this case, it cannot be excluded that the tightness of some claddings get lost.

For the simultaneous failure of all six MCPs, analyses are available from the Soviet main constructor (GIDRO-PRESS) and from Bulgarian analyses applying a modified version of the COBRA IIIC computer code.

According to the analyses, the pressure within the primary coolant system remains below 16 MPa. Since the existing calculations are not comprehensable in detail, it is necessary to perform specific analyses using "best estimate" computer codes.

5.1.8 ATWS accidents

ATWS accidents have to date not been investigated for the WWER-440 plants. No statements regarding the maximum pressure and the minimum DNB ratio can be made without such analyses.

5.2 Accident analyses to be performed

It is necessary to perform supplementary accident analyses with respect to regard the actual condition of the WWER-440/W-230 using the computer program ATHLET, in order to verify the results of the analyses already available and to investigate accident sequences which have to date not been considered. It is intended that these analyses should be undertaken in 1990. With regard to the status arising after reconstruction, additional analyses may become necessary at a later date.

As a result of the tight deadline, the number of analyses in the year 1990 has to be limited to four, or a maximum of five. Having regard to the significance and the urgency of this matter, the following cases have been selected:

1. Leak in the cold leg with an equivalent diameter of 32 mm, emergency core cooling by one pump of type EP 50 after no less than 115 secs. (with consideration being given to total loss of power and single failure)

2. Double-ended break of a connecting line of nominal diameter 200 mm between pressurizer and primary loop, emergency core cooling by four EP 50 emergency pumps (rupture of the largest connecting line under conditions of maximum emergency cooling capacity), depending on the result, an additional calculation with 50 % power may be required

3. Failure of all six MCPs with consideration being given to their moment of inertia under conditions of loss of onsite and offsite power (determination of the maximum pressure load to the primary system and of the DNB ratio); additional sequences still

to be determined after performing the main calculation, e.g. temporally staggered failure, or failure of 4 out of 6 MCPs

4. Loss of onsite and offsite power without reactor shutdown (ATWS) with the core condition at begin of cycle (first investigation of the plant behaviour in the event of ATWS accidents)

5. As regards the last case of analyses, there are several conditions of which one will be selected finally

- double-ended break of the emergency boration line of nominal diameter 50 mm with flow limiters and reverse flow via the header (rupture of the largest connecting line ≦ nominal diameter 100)

- rupture of the main steam line with steam generator tube rupture as consecutive damage (determination of the primary inventory discharge to estimate the radiological relevance)

- rupture of the main steam line upstream of the turbine (verification of existing calculations with respect to response of the interlock 6.4.19 to close the quick-closing valves)

In addition to these proposed accident analyses more detailed investigation of existing analyses or continuation thereof is required, as outlined in greater detail in chapter 5.1.

5.3 Comments on safety-promoting measures from the 35-point SAAS programme

Concerning item 6: Operating manual concerning the behaviour in the case of small leaks

The requirement to review the appendix 15 of the operating manual NO1 is supported.

Concerning item 9: Eddy current testing of the SG tubes

The probability of breaks and possible consecutive failures has to be quantified on the basis of the eddy current tests. Proceeding thus, the need for and scope of new accident analyses have to be specified.

Concerning item 11: Heating system for the emergency core cooling water

The proposed measure increases the reliability of the emergency core cooling system.

Concerning item 16: Installation of the quick-closing valves.

The installation of the main steam quickclosing valves has a decisive effect on the accident behaviour in the event of leaks in the main steam system. In the preceding evaluations, such installation was already assumed.

Concerning item 17: Reactor protection system actuation in the event of a low SG water level

Accident analyses show that it is absolutely necessary to shut down the reactor on falling below the permissible level in one steam generator. Where possible, two diverse should be used, e.g. coarse and fine level measurements including a time delay element (approximately 20-30 secs.), to avoid erroneous initiation.

Concerning item 19: Supply of cooling water to the diesel emergency power supply system

From the point of view of accident analysis, an increase in the reliability of the diesel emergency power supply system is of essential significance.

Concerning item 20: Power density monitoring

In conjunction with the later introduction of a DNB signal, the planned measures for power density monitoring are considered to be necessary.

Concerning item 21: Leak detection system

In order to cope with the accident by operator actions and to use the "leak before break" criterion, leak location which is as precise as possible is desirable, especially in view of the possible isolation of individual circulation loops by means of the main gate valves.

Concerning item 31: Accident Management Measures

The main emphasis of the discussed accident management measures lies in the maintenance of the secondary side heat sink. In this point the views of the SAAS and of utility (KKKW) with the result of the German Risk Study, Phase B, Possibilities for implementation in this regard are, for example:

- interconnection of the emergency feedwater systems of all units
- diverse feed into the drain lines of all units
- mobile pump with independant drive injection from the feedwater tank by pressure difference

Primary-side measures for leak compensation are proposed by utilizing of adjacent units.

- Supplement to the 35-point programme

- A recommendation is made to check the feasibility of short-term replacement of the EP 50 pumps by pumps of type ZN 65 in units 1 and 2. The advantage would be a higher degree of reliability and an increase in the effectiveness of the emergency core cooling, especially at low pressure.

- Checking the possibility of setting up an additional LP emergency cooling pump or utilization of an available LP pump for injection into the primary system as an accident management measure.

There are serious reservations against further operation of the plant while maintaining the emergency core cooling concept according to the project, as regards a design basis leak of an equivalent diameter of 32 mm. If it cannot be demonstrated that under "best estimate" conditions the second design limit value according to OPB 82 will not be exceeded in case of the break of the largest connecting line (cf. paragraph 5.2), the following alternatives were discussed in the "accident analyses" working group:

- With regard to the development of serious core damage in case of a break of primary system connecting lines up to the double-ended fracture of the largest connecting line, the surge line, estimates sation line, estimates utilizing assumptions which under assumptions which are in certain cases conservative show that, on operation of the plant with reactor power reduced to approximately 50%, core meltdowns can probably be prevented by means of the available emergency core cooling system (without the emergency core cooling system (without the assumption of a station black out and the assumption of a single failure). In order to verify the results of the estimates which are based on simplified mass and energy balances, it is necessary to perform analyses with advanced thermohydraulic codes.

- A further possibility for increasing the effectiveness of the emergency core cooling is the shutdown of one unit out of the double unit with continued use of safety systems for the unit still being operated. By allocating a fourth diesel to the operating unit, it is also possible to employ four emergency core coolant pumps for injection in the event of the loss of onsite and offsite power. The activation of further pumps (a maximum of 4) of the adjacent unit by hand is considered to represent a potential for core damage reduction in the case of relatively large leaks, especially in the low pressure phase. The availability of the secondary side heat sink is improved by using the emergency feedwater system of the unit shut down. On account of the relatively large water stocks in the SGs, there is adequate time available for manual activation.

In the discussion of these variants, the Soviet experts expressed fundamental reservations on account of possible restrictions of stability and reliability of operation. The safety-related advantages and disadvantages of possible solutions must be weighed up one against the other.

5.4 Thermohydraulics concerning the formation of cold water plumes (CWP)

Cold water plumes are of essential significance for the assessment of the brittle fracture resistance of the reactor pressure vessel, especially where they occur at high pressure of the primary system and distributed asymmetrically over the periphery of the downcomer. A matter of particular interest concerns cold water plumes at the level of the core region where the brittle fracture transition temperature of base material and weld seam is increased by the action of the neutron radiation.

Depending upon their origin, cold water plumes can be classified in two groups: on the one hand secondary side subcooling transients, and on the other hand primary side leaks which are partially compensated by the emergency core cooling system. Moreover, the overlap of the two groups requires consideration.

Analyses on this topic were carried out by the Soviet manufacturer "GIDROPRESS" in 1984 and 1987; the calculations performed in 1987 are applicable only to a limited extent to the units 1-4, since they presuppose the modification of the emergency core cooling injection system to the hot legs. Investigations regarding WWER-440/W-213 reactors are also to be taken into account for the purposes of the assessment, where, as in the case of units 1-4, cold side injection of the high-pressure emergency cooling pumps is provided. These investigations include measurements in the Soviet KOLA-IV power plant and the probabilistic PTS ("Pressurized Thermal Shock") study regarding the Finnish LOVIISA plant. However, it should be stated that these plants of the type WWER-440/W-213 are equipped with an emergency core cooling system of larger capacity and an injection system distributed unequally between the loops. The Zittau Technical University has undertaken further experimental and analytical investigations concerning the flow distribution of large volume cold water plums within the annulus and lower plenum, as they may occur in the case of secondary side subcooling transients.

For the occurrence of large volume cold water plumes arising from individual cold plumes following on secondary-side subcooling transients, a coolant circulation in the primary system is required. On the other hand, cold water plumes may occur in consequence of the high-pressure emergency core coolant injection only at very low flow speeds or under stagnation conditions, e.g. at very low residual power levels at the beginning of a cycle or at conditions of a temporary loopseal.

Also important in the case of asymmetric cold water plumes are manual measures, which may be performed in case of primary-side leaks for leak detection and location. As a result of the isolation of a group of three respective loops by means of the main gate valves and the associated valves in the headers of the high-pressure injection system, asymmetric injection phenomena will occur with locally doubled rates of injection. This receives attention in recent analyses, by KWU (Kraftwerks-Union), as does the fact that the injection rates of the high-pressure emergency core cooling pumps of units 3 and 4 considerably differ from the injection rates of the respective pumps of units 1 and 2 which were previously taken as a basis in the analyses.

With regard to the cold water plumes due to high-pressure emergency core cooling, it should be stated that considerable conservative features were taken into consideration in the analyses available so far. In some cases, the Soviet analyses employed empirical equations based on model tests (scale 1:7), the results of which have proved in retrospect to be absolete by comparison with the above mentioned measurements at the KOLA-IV plant. In other analyses, a model of a "complete mixing" of the high-pressure emergency cooling water with stagnating water in a defined control volume of the downcomer and a part of the lower plenum was used, which takes inadequate account of the physical processes in the formation of cold water plumes. According to the KOLA tests, it is to be expected that the cold water plumes originating from the high-pressure injection by the EP 50 emergency cooling pumps (units 1-2) have been almost entirely mixed, when they reach the weld seam in the core region.

The KWU has supplied preliminary information on the fluid/fluid mixing analyses, with the result that in case of high-pressure coolant injection no temperature differences between cold water plumes and the surrounding fluid greater than approximately 30 K act on the weld seam in the region of the reactor core. The analyses are based on HDR (superheated steam reactor) experimental results which were evaluated by means of the theory of planar plumes according to Chen/Chen. The intensity of mixing increases with increasing injection mass flow so that the above indicated result (30 k) is only slightly dependent upon the injection mass flow.

With regard to pressure load, leaks which can be compensated are to be assessed as more unfavourable than the design basis case with a leak of nominal diameter 32, since the pressure re-increases in the course of the transient.

With regard to the secondary side subcooling transients the complete break of the main steam line was considered comprehensively in the Soviet analyses. This case is characterized by the response of the interlock 6.4.19, which leads to the closure of the quick-closing valve and the feedwater supply as well as to the shutdown of the MCP in the affected loop. Since at least three of the remaining MCPs run down normally within approximately 3 mins., the primary side loop, which is affected by the rupture of the main steam line, experiences a coolant flow in the reverse direction. Within approximately the same time, the water in the steam generator has already been evaporated to a large extent. Consequently, between the termination of the pump rundown and the evacuation of the steam generator only a relatively short period of time exists for the entry of a large volume cold water plume into the downcomer. It is doubtful whether smaller main steam line leaks are covered by the total break of the main steam line, as far as the danger of brittle fracture is concerned.

Finally, it should be stated that the analyses available for units 1-4 are inadequate and, where they are available, are not reliable enough to permit an unambiguous assessment. It is recommended to perform reliable and appropriate analyses regarding the formation of cold water plumes, with reference to the PTS study set up in Finland for LOVIISA emphasizing asymmetric subcooling transients.

5.5 Supplementary work to by done for the verification of the used computer codes

Especially in the area of the primary side steam generator flow, the existing computer programs are inadequately verified. In connection with this problem area, experiments have been conducted at the test facilities PMK, REWET III, at an experimental facility operated by GIDROPRESS (USSR) and in Vitkoviz (Czechoslovakia). These experiments, as well as the experiments planned at the new test facility PACTEL in Finland and the steam generator test facility in Vitkoviz (Czechoslovakia), must get detailed evaluation and assessment.

It is recommended that the responsible authorities in the German Democratic Republic and in the Federal Republic of Germany should arrange as soon as possible for a detailed assessment of the experimental results and their utilization for code verification by their respective experts.

5.6 Safety-related requirements

On the basis of the preceding investigations, the following recommendations are made for necessary safety-related backfitting measures :

- RPS 2 initiation by "pressure in steam generator box high" to be reduced from 1.3 to 1.2 bar

- RPS 1 initiation by "pressurizer water level low" to be increased from 2.4 m to 2.56 m (logic "and" with low primary pressure)

- RPS initiation by "water level low in steam generators" to be backfitted

- Installation of the quick-closing valves into the main steam lines of units 2 and 4

- Replacement of the EP 50 emergency core cooling pumps by ZN 65 or installation of additional low-pressure emergency core cooling pumps respectively utilization of existing ones

- Performance of analyses concerning the development of cold water plumes in reliance upon the study carried out in Finland for the LOVIISA plant, concerning mainly asymmetric subcooling transients, and submission of these analyses for assessment

- Extension of the design basis loss of coolant accident to the break of the largest connecting line taking into account total loss of onsite and offsite power and single failure: for short-term continued operation, it must be shown that under "best estimate" conditions the second design limit value according to OPB 82 will not be exceeded in the event of leaks up to break of the largest connecting line

- Installation of a separate heating for the emergency core cooling water; for short-term continued operation, it has to be ensured by operational measures that during the use of the emergency core coolant pumps for this purpose there is no substantial restriction of the availability of these pumps

- Interconnection of the emergency feedwater systems of all units

- Interconnection of emergency core cooling injection systems between adjacent units

- Diverse injection into the drain lines of the steam generators

- Provision of a mobile pump with independant drive for secondary side emergency feedwater

- Utilization of the possible injection from the feedwater tank by pressure difference

- Within the terms of comprehensive backfitting, evidence should be provided of adequate emergency core cooling in case of a doubleended break of the MCL (the objective is to remain well below the radiological intervention level for evacuation of the public).

6. THE PRESSURE RESISTANT COMPARTMENT SYSTEM AS CONFINEMENT

The confinement of units 1-4 of the Greifswald nuclear power station does not satisfy the requirements set forth in the safety criteria for nuclear power stations /2/.

6.1 Current layout of the pressure resistant compartment system

6.1.1 Basic operation

The components of the primary loop of the WWER-440/W-230 reactors are installed within a pressure resistant compartment system which is intended to fulfil the function of a containment. In the design basis case (see chapter 6.1.3), the pressure and temperature in the pressure resistant compartment system are limited or reduced over a long period by a spray system. Radioactive substances released from the primary system into the pressure resistant compartment system are washed out from the atmosphere using chemical additives to the spray water. In the case of leak-initiated accidents exceeding the design specification, there is an excess pressure security system operating by simple-design dampers, which open before reaching the design pressure of the pressure resistant compartment system. In this manner, in the first instance large portions of non-condensable gases (air) are expelled from the pressure resistant compartment system. After the dampers have been closed, the condensable gases (steam) are condensed by means of the spray system.
In this case it is assumed that damage to fuel elements with an increased release of radioactivity occurs only <u>after</u> closure of the dampers, and the long-term leakage from the pressure resistant compartment system can be kept at a low level.

6.1.2 Structure

The pressure resistant compartment system consists of a total of 23 compartments. These compartments accommodate all main and auxiliary systems of the reactor installation, submitted to pressure and temperature of the primary loop during power operation.

The total volume of the compartments comprises approximately 14,000 m^3. Most of the compartments are connected to one another by large, non-closable openings. Some compartments are connected to the remainder of the pressure resistant compartment system by overpressure dampers. In the event of pressure build-up, the dampers of these compartments open only in the direction of the pressure resistant compartment system. The pressure resistant compartment system is designed for a maximum internal pressure of 2 bar. The compartment walls and ceilings perform simultaneously the function of radiation shielding towards the adjoining passable compartments. The compartments are sealed by a steel liner: The upper region of the reactor well is sealed by a dismountable protective hood.

To limit the pressure and to provide pressure relief, the pressure resistant compartment system is equipped with 8 overpressure dampers of nominal diameter 1130 and 1 overpressure damper of nominal diameter 520. The small dampers open at a pressure within the pressure resistant compartment system of 1.6 bar; the large dampers open at 1.8 bar. When the large dampers are opened, the atmosphere of the pressure resistant compartments is discharged into the environment via two blow-off shafts. A pipe leads directly from the small damper into the free atmosphere.

6.1.3 Design basis accidents

In conformity with the safety concept of the entire plant, the design of the pressure resistant compartment system was based on the breach of a connecting line of nominal diameter 100 to the primary loop with an flow limiter (nominal diameter 32).

With regard to the action of the spray system, it is intended that in this case the pressure in the pressure resistant compartment system should not reach the opening pressure of the dampers. The spray system should ensure that the overpressure within the pressure resistant compartment system is reduced in a maximum of 30 min. For the purposes of such an accident, it is assumed that no damage takes place to the fuel elements of the core. Only the radioactive substances contained in the cooling water of the primary coolant loop under normal operating conditions are released into the pressure resistant compartment system.

In addition, the pressure resistant compartment system is designed in accordance with the data supplied by the Soviet general designer (Technical Design 1968) for a leak having a magnitude of nominal diameter 200 ,1F at the primary coolant loop. The pressure arising in this case within the pressure resistant compartment system exceeds the opening pressure of the dampers. The steam/air mixture is blown off into the environment. After the first expulsion, the dampers shall close tightly again. In this connection, the repeated opening and closing of the dampers is to be assumed. As regards the release of radioactivity, this means that on the first expulsion only the radioactivity contained in the primary coolant loop under conditions of normal operation passes into the atmosphere. Until the possible occurrence of fuel rod damage due to overheating, the dampers shall be closed and the pressure resistant compartment system tightened. The spray system should ensure a pressure reduction in the long term. No results of calculations concerning such an accident are given in the design documents.

6.1.4 Tightness requirement for the pressure resistant compartment system

The required tightness of the pressure resistant compartment system was determined in the Soviet Technical Design for the "leak nominal diameter 32" design basis accident, under the following conditions:

- The total body exposure of the personnel remaining within the adjoining compartments during the accident must not exceed 5 rem (50 mSv).

- The ventilation systems in passable compartments continue to operate normally during the accident.

A permissible leakage rate from the pressure resistant compartment system into the adjoining compartments of approximately 3600 m^3/h was calculated. The test pressure for determining the leakage was established in conformity with the maximum compartment pressure calculated for the leak of nominal diameter 32, at a level of 1.25 bar.

6.1.5 Bearing capacity of the building

The maximum permissible loading of the pressure resistant compartment system with internal pressure was indicated at 2.0 bar in the design plan. No stress analysis of the building is available from the Soviet designer. A pressure test using the design pressure or a value above this has not to date been carried out. Subsequent calculations by the Academy of Civil Engineering of the GDR have established that the pressure resistant compartment system has no strength margins at design pressure.

6.1.6 Sealing of the pressure resistant compartment system/tightness test

For sealing purposes, the pressure resistant compartment system is provided with a welded steel plate lining. For penetrations through the walls and ailings, such as dampers, hatches, pipes, cables and shafts, constructions were developed which satisfy the above tightness requirements, but no longer comply with the current state of the art. There are various weak points such as the shaft penetrations of ventilating fans. A leak test on the pressure resistant compartment system will be performed annually as a pressure drop measurement (compressed air at ambient temperature), using the aforementioned test pressure.

In the case of all 4 units, the measured leakrate was steadily below the permissible leak rate. The measured leak rates are within a range from 250 to 2300 m^3/h.

6.1.7 Isolation of the pressure resistant compartment system

During operation, individual compartments of the pressure resistant compartment system must be visited in turn for the purpose of monitoring and maintenance work. Access to the compartments of the pressure resistant compartment system is achieved via airlocks or double dampers, so that on obtaining access to the pressure resistant compartment system during operation of the installation the tightness is steadily guaranteed. The ventilation and air exhaust systems, the lines of which pass out from the pressure resistant compartment system, have either weight loaded overpressure dampers or motor-driven valves with electrical power for the purpose of isolation in the event of pressure build-up within the pressure resistant compartment system. There is only one of such dampers provided in each line. The closing times of the electrically driven valves are between 30 sec (nominal diameter 250) and approcimately 60 sec (nominal diameter 1000). The electric drives are connected to the emergency power supply. The control of electric drives takes place by means of a measurement of the compartment pressure (closing impulse at 25 mm water column excess pressure). The systems for dosimetric compartment monitoring (gas activity and aerosol measurement) also have motor-driven valves and are closed at the same time as the ventilation lines. All other pipes - including those which are connected to the primary coolant loop - have no isolating valves which are automatically closed on a rise in pressure within the pressure resistant compartment system.

6.1.8 Operation of the pressure resistant compartment system

The removal of the heat given off from the technological systems to the compartment air takes place mainly by means of a redundant air recirculation cooling system (R1). The heat is removed in air coolers to a service cooling water system (ICC-NPS). The ventilation of the compartment (A 102) for the drives of the main coolant pumps and the main gate valves takes place by means of ventilation and exhaust system (P 4 and W 4). Sience the compartment must be visited periodically when the units is in operation. Within the entire pressure resistant compartment system, for reasons of radiation protection, a sub-pressure of 20 to 30 mm water column is maintained at all times by an exhaust system (W 2). The exhaust air from the pressure resistant compartment system is purified in air filters to remove radioactive substances (W 2 aerosol plus iodine filter, W 4 aerosol filter), before it is passed into the environment via the exhaust gas chimney.

For the plant and equipment of the pressure resistant compartment system, the following automatic locking devices are introduced as a function of the compartment pressure:

+ 25 mm water column:	- Closure of the motor-driven valves in the exhaust air lines - Shutdown of the air supply and exhaust systems - Closure of the valves in the dosimetry lines
+ 0.2 kp/cm²: excess pressure (120 kPa)	- Activation of the spray system
+ 0.3 kp/cm²: excess pressure (130 kPa)	- Shutdown of the reactor by the emergency protection system RPS 2

6.1.9 Spray system

The spray system consists of 3 pumps (capacity 400 m³/h each), 2 coolers and 3 nozzle trains within the pressure resistant compartment system. When required, two pumps, two coolers and two nozzel trains are in operation. The pumps take water from the emergency borating tank. The water reservoir amounts to 800 m^3.

Since the water in the emergency borating tank is heated to approximately 60°C, it must be cooled to approximately 40 °C to improve the spray effect, before it is sprayed into the pressure resistant compartment system. The spray water including the leakages from the primary coolant loop runs back into the emergency borating tank. The spray coolers are thus at the same time used to remove the decayheat from the pressure resistant compartment system. The heat is given off, in the coolers, to the service cooling water.

6.1.10 Periodical inspections

- Leak test on compartments: see chapter 6.1.6
- Ventilation valves: the closing function of the motor-driven flaps in the air exhaust lines is tested monthly. The tightness of the flaps is tested jointly with that of the pressure resistant compartment system (subatmospherie pressure tests). The overpressure dampers of the air supply system P4 (A 102) are tested only for their wear condition (sealing surface, mechanical mobility).

- Dampers of the pressure resistant compartment system: each year, 50% of the dampers are checked. The lifting power required to open the dampers is measured in accordance with the test specification, and the tightness is tested by compressed air (incorporation of an intermediate floor).

- Dampers, hatches, shaft penetrations, cable penetrations: the testing of the tightness of these items is carried out jointly with the integral testing of the pressure resistant compartment system.

6.2 Safety-related assessment

6.2.1 Leakage of the pressure resistant compartment system in case of an accident

According to /3/, the maximum permissible leak rate for the pressure resistant compartment system is 3600 m^3/h at a pressure of

1.25 bar. This maximum value leads to an air change cycle of 6/d (= 600 vol.%/d) for the compartment system. Substantial contributor to this high value are the shaft penetrations of the air recirculation system cooling the pressure resistant compartment value of 3600 well as pervious ventilation valves. The permissible value of 3600 m^3/h is exhausted, especially for units 1 and 2, to the extent of 2/3. According to the power /3/, the maximum leak rate should be limited, by appropriate backfitting measures, to 300 m^3/h. The number of air changes which can be achieved thereby, amounting to 0.5/d, is, compared with the approved values in the FRG for large, dry containments of 0.25 vol.%/d at design pressure, still very high. If, on account of the differing sizes of the compartment areas surrounding the primary loop, a direct comparison is made between the values in m^3/h, the result is still a ratio of approximately 40. The question of whether this value is also applicable to the pressure resistant compartment system at design pressure (approximately 2 bar) and whether the radiological consequences in the environment of the plant on account of such high leakages in the course of accidents are acceptable is to be examined (see chapter 6.4).

6.2.2 Isolation of penetrations of the pressure resistant compartment system

The basis for the operation of the pressure resistant compartment system as containment is a reliable isolation of the penetrations of the system at the occurrance of an accident. For example, the breach of lines outside the system which are connected to the primary circuit may cause the pressure resistant compartment system to be bypassed; accordingly, such lines should in principle be provided with two isolation valves, one on the inside and one on the outside. This principle has not been consistently followed in units 1-4 (see chapter 7.3.6). Furthermore, several ventilation valves are in the open condition during power operation. These valves are provided only singly; they lead, if they fail when required, to high release rates from the pressure resistant compartment system into adjoining regions of the building. This may result in impairment of control and operation actions. Isolation valves and pipe and cable penetrations must also be protected in the region of the penetrations against subsequent damages caused by outflowing media, reaction forces and fragments. To what extent such a requirement is fulfilled for the installations considered here should be examined.

6.2.3 Excess pressure protection for the pressure resistant compartment system

To protect the pressure resistant compartment system against an overpressure failure caused by a release of mass and energy from the cooling loops into the compartment system exceeding the design basis (break of nominal diameter 32), nine so-called dampers are installed for each unit. The media escape from the system via these dampers and two adjoining shafts directly into the environment. The principle of the confinement is infringed when the dampers respond. However, it is essential that the dampers respond during loss of a coolant accident (LOCA) in the phase in which the occurrence of fuel damage due to overheating of the core region is not expected yet. This should be secured by the design of the emergency cooling systems. If this condition is satisfied, in the event of beyond design basis accidents in the early phase, only a part of the activity present in the primary coolant loop at the beginning of the accident can be released into the environment.

If the pressure resistant compartment system is intended to fulfil the function of a long-term containment, then the reliable closure and sealing of the dampers after a pressure drop in the compartment system and an operationally effective spray system acquire great importance as safety features. Thus, the reliability of the dampers is a decisive factor in the assessment of the pressure resistant compartment system as a containment.

6.2.4 Cable penetrations through the pressure resistant compartment system

Cables leading out of the pressure resistant compartment system are permanently connected to the building structure. The cables are laid in pipes of nominal diameter 50 and sealed by means of a sealing compound or PUR foam. Cable ducts must be designed so that the seal is not threatened by pressure and temperature effects. The behaviour of PUR foam or of the sealing compound under accident conditions should be tested. It should also be possible to control consequential damage due to escaping media (e.g. direct radiation).

6.2.5 Stresses on the structure

The loading capacity of the building structure of the pressure resistant compartment system by a quasi-steady pressure build-up in the compartment system is indicated in various documents (e.g. /4/, /5/) as being 2.0 bar. No relevant demonstrative statement in support of this value is available. In the case of accidents exceeding the design basis, such as for example a breach in the primary coolant loop with a leak equivalent to nominal diameter 200, the occurence of relatively large pressure differences between the individual compartments of the pressure resistant compartment system is a factor to be reckoned with.

Dynamic stresses of the structures of the pressure resistant compartment system due to media escaping under pressure, the impact of loose or broken pipes, reaction forces on the anchorages of individual components and fragments which may act as projectiles, must be estimated, in order to be able to assess the integrity of the pressure resistant compartment system during loss of coolant accidents exceeding the design basis.

6.2.6. Spray system and cooler

To limit the partial pressure build-up due to condensable gases which are released during an accident from the coolant loops into the pressure resistant compartment system (especially steam), a spray system is installed in the steam generator box.

The effectiveness of this system determines whether, in the course of loss of coolant accidents, the dampers respond and close again after a response. The water sprayed into the SG box forms alongside the structures of the walls and internals the only heat sink for the long-term removal of decay heat from the pressure resistant compartment system. Additionally, the spray system serves, as a result of appropriate additives to the spray water, to bind fission products such as elementary iodine. Its effectiveness is influenced by the fact that the spray devices reach only a fraction of the atmosphere in the SG box. The region above the spray nozzles as well as the regions below the drive compartment are inadequately served with spray water. The spray system is of great importance from the safety point of view. In the event of failure of the not entirely redundant system, a part of the possible accident spectrum cannot be controlled.

6.2.7 Sump drain

As a consequence of a loss of coolant accident, the sump region in the pressure resistant compartment system fills with water. The water flows over a weir and through a pipe into the emergency boration tank. There is only one inlet to the emergency boration tank. This inlet is protected by coarse and fine mesh filters. Blockages, for example due to insulating materials which have been torn off may impair the removal of decay heat. Accordingly, it is recommended to install a second outlet.

6.2.8 Broadened spectrum of possible accidents

With regard to possible effects on the pressure resistant compartment system, the design case with a breach size of nominal bore 32 and the beyond design basis case of a leak of nominal diameter 200 in the primary coolant loop were investigated. Furthermore, the following scenarios should be investigated in particular:

- break (2 x pipe cross-section) of the largest connecting line to the primary coolant loop (nominal diameter 200),

- break of a main steam line within the pressure resistant compartment system, and

- leaks at the control rod guide tubes in the region of the reactor top cover.

6.2.9 Formation of combustible gas mixtures in the pressure resistant compartment system

During a loss of coolant accident, hydrogen (H_2) is formed due to the radiolysis in the core region and in the sump, due to the oxidation of zirconium in the fuel rod claddings at temperatures exceeding 1150 K, due to chemical processes involving protective coatings in the pressure resistant compartment system in a superheated steam atmosphere etc. In the case of a reliably controlled loss of coolant accident, the long-term radiolysis process forms the dominant H2 source. It lasts however days or weeks before the gas mixture in the pressure resistant compartment system becomes burnable.

In the event of beyond design basis accidents, involving high degrees of core damage, the exothermic reaction of zirconium with superheated steam is the dominant H_2 source.

If the dampers respond, in the first instance no combustible gas mixtures or gas mixtures which burn only to a limited extent can form on account of the low-oxygen atmosphere in the pressure resistant compartment system (expulsion of part of the air through the dampers). However, in the long term it must be expected that air will pass into the pressure resistant compartment system via inwardly directed leaks (reduced pressure in the pressure resistant compartment system due to steam condensation) and combustible gas mixtures may be formed. In units 1-4 at the present time no measures are provided for the purpose of preventing the formation of gas burns in the pressure resistant compartment system.

6.3 Thermohydraulic analyses

6.3.1 Investigations already performed

The following leak-induced accidents have been analysed by the operator of the installation:

Leak 1 F, nominal diameter 32 with total loss of power (1 F = 1 x pipe cross-section)

Leak 2 F, nominal diameter 32 with total loss of power (2 F = circumferential break = 2 x pipe cross-section)

Leak 2 F, nominal diameter 90 with total loss of power

Leak 1 F, nominal diameter 100 with total loss of power

Leak 1 F, nominal diameter 200 with total loss of power

Leak 2 F, nominal diameter 200 without total loss of power.

The calculations were essentially carried out using the program BRACO-1 (see annex 2).

The accidents 2 F, nominal diameter 32 and 1 F, nominal diameter 200 are described in greater detail below.

- Design basis accident 2F, nominal diameter 32 with total loss of power

The break of a pipe of nominal diameter 100 with flow limiter (nominal diameter 32) does not lead to the opening of the dampers under the following conditions:

- Operation of 2 spray pumps at 100 % nominal flow rate (independent of counterpressure)
- Temperature of the spray water: 30°C

The calculated maximum pressure in the steam generator box is approximately 142 kPa after 425 sec.

If only one spray pump is available, the small damper opens after 350 sec. The mass and energy removal from the primary coolant loop into the pressure resistant compartment system (leakage function) was determined using the program RAMPA.

- Leak 1 F, nominal diameter 200 with total loss of power

Approximately 6 sec after beginning of the accident, the opening pressure of the large dampers is reached. In the interval from 6 to 344 sec, the pressure time history in the steam generator box shows large oscillations. 344 sec after onset of the accident, the large dampers close. As a result of the expulsion of the steam-air mixture into the environment, the peak pressure is limited to values of around 1.80 bar. The maximum temperature is 117°C. Under these loadings, destruction of the pressure resistant compartments is not expected. The blowdown rates from the primary coolant loop were computed using the program RELAP4/MOD6.

6.3.2 Confirmatory analyses

The analyses, carried out by GRS, regarding the behaviour of the pressure resistant compartment system in the event of loss of coolant accidents had as their objective the checking of the former results from the operator and verification of those results. GRS used computer programs which are used in the licensing procedure for light water reactors and which were verified by experiments (Marviken, Battelle Institute Frankfurt, HDR) (see annex 2).

Besides the general objective of verifying load data for the pressure resistant compartment system, the intention was also to investigate the effectiveness of the spray system and the mode of operation of the dampers (as weight-loaded flap).

6.3.2.1 Comparative analyses performed

For the purposes of the comparative analyses, in the first instance the design basis accident involving breach of a connecting line of nominal diameter 100 with flow limitation equivalent to nominal diameter 32 (see chapters 6.1.3 and 6.1.8) was selected. In order to determine the pressure and temperature loadings in the pressure resistant compartment system of unit 1, the GRS computer programs COFLOW and RALOC (see annex 2) were utilized. As input data regarding the compartment volumes, the connecting openings, heat conducting structures, spray system, ventilation data etc, the basic data set /6/ established by the operator was used. The blowdown rates were taken from more recent computations using the computer code RELAP4/MOD6, which were carried out in 1989 by the operator. These correspond to a break area equivalent to nominal diameter 32.

A check on the flow rates by GRS computer programs (e.g. DRUFAN, ATHLET) is not yet available at the present time (see chapter 5). However, it is expected that no substantial effects on the current GRS statements concerning the loading of the pressure resistant compartment system will emerge from this check. This statement is based on earlier comparisons between the computer codes RELAP and DRUFAN.

An initial computation using the total number of compartments (23 compartments + environment zone) of the pressure resistant compartment system proved to be inappropriate, since a considerable increase in computer time is caused by relatively small compartments connected to the SG box by relatively large cross-sectional connections. This sham accuracy does not help to increase the cogency of the evidence. Accordingly, the 23 compartments were combined into five model compartments, as had already been proposed in /6/.

By comparing computations in which all compartments of the pressure resistant compartment system are individually modelled with a five compartment model it emerged that the five compartment model meets the requirements. The further investigations were therefore carried out using this model. In addition to the investigations relating to the design basis accident (chapter 6.1.3), the breach of a line of nominal diameter 200/1 F was also included in the GRS analyses. This case serves mainly to simulate the operation of the dampers.

6.3.2.2 Results of the comparative GRS analyses

To examine the effectiveness of the spray system, orienting computations were initially performed, using the pressure difference program COFLOW. For these initial qualitative investigations, simplifying assumptions were made in the data set. First, the effect of the initially open ventilation was disregarded (i.e. the ventilation was assumed to be closed). Likewise, the steel and concrete internals were not considered as a heat sink in these calculations.

The results of these calculations are shown in Fig. 6-1 for the influence of the spray efficiency and in Fig. 6-2 for the effect of the spray temperature on the pressure within the pressure resistant compartment system.

Further, more precise investigations were carried out using the GRS program RALOC on the following assumptions:

- ventilation of the pressure resistant compartment system closes after 30 sec

- spray system becomes effective after 125 sec in case of total loss of power

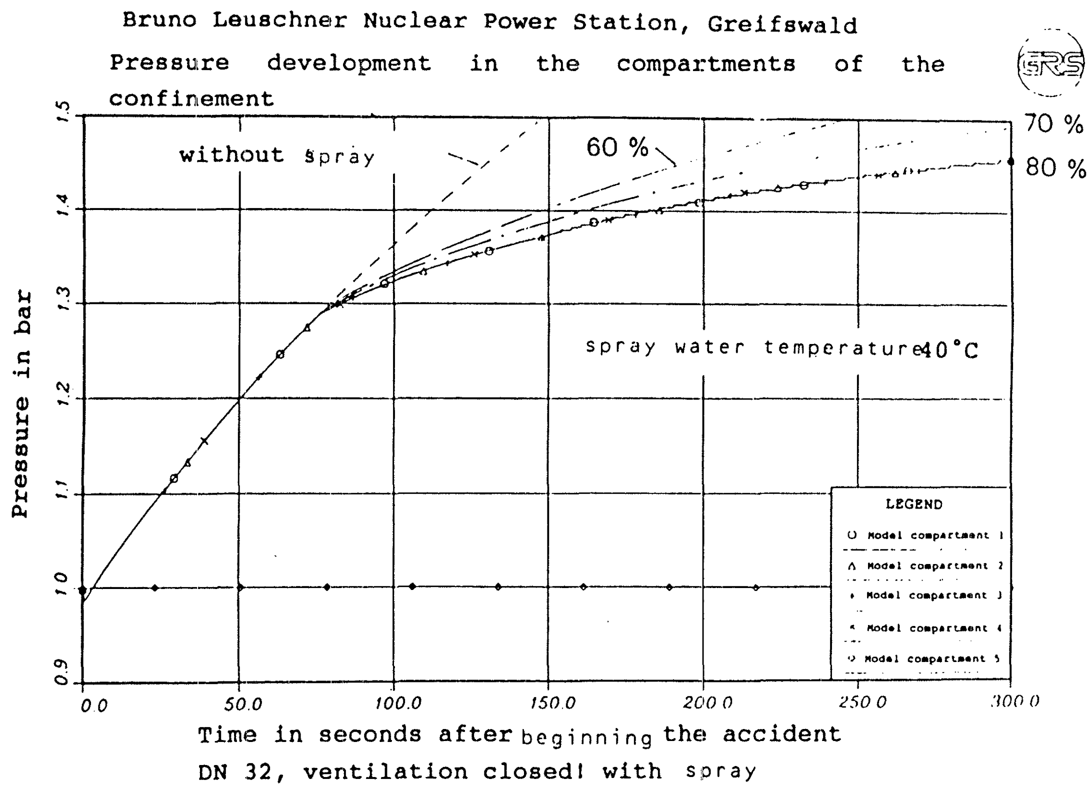
- 70 % efficiency of the spray system (from a simple estimate based on nozzle construction and arrangement in the pressure resistant compartment system)

- spray water at 40°C with two spray water pumps and coolers in operation.

Fig. 6-3 shows the pressure time history within the pressure resistant compartment system under the aforementioned conditions; the upper curve shows the calculation without and the lower curve the pressure development with consideration of the heat removal into the steel and concrete structures of the pressure resistant compartment system. This provides confirmation of the statement in chapter 6.1.3 according to which in the design basis accident

Fig. 6-1

Effect of the spray efficiency



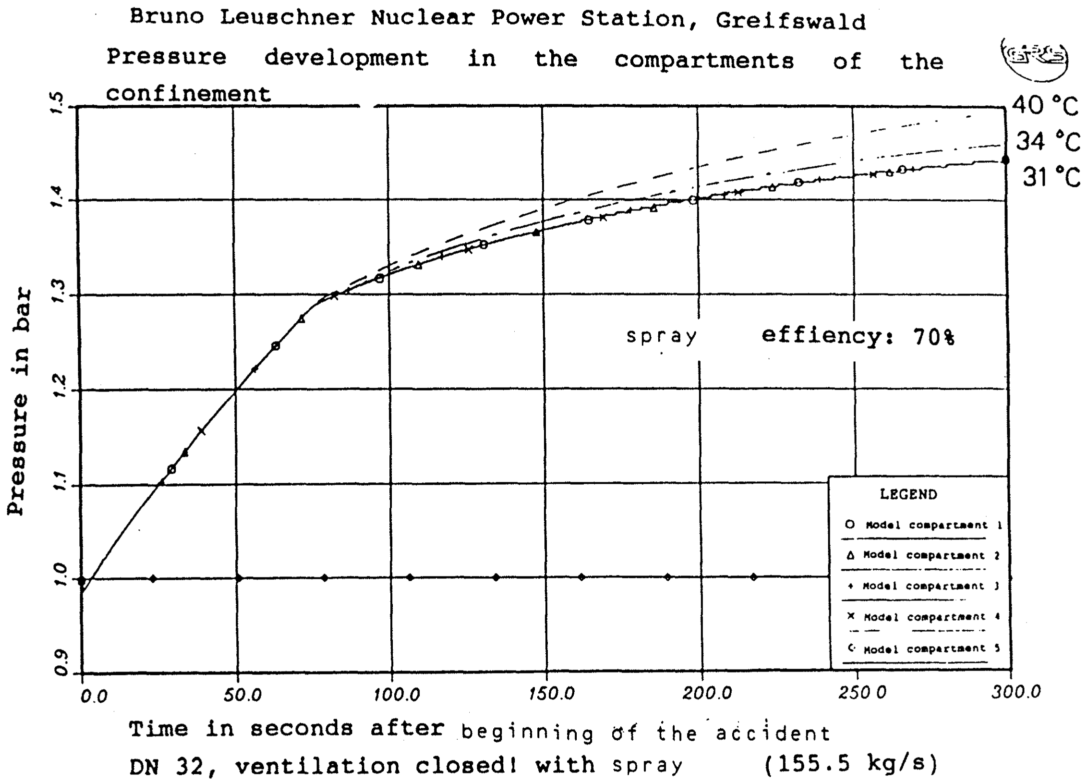


Fig. 6-2: Effect of the spray temperature

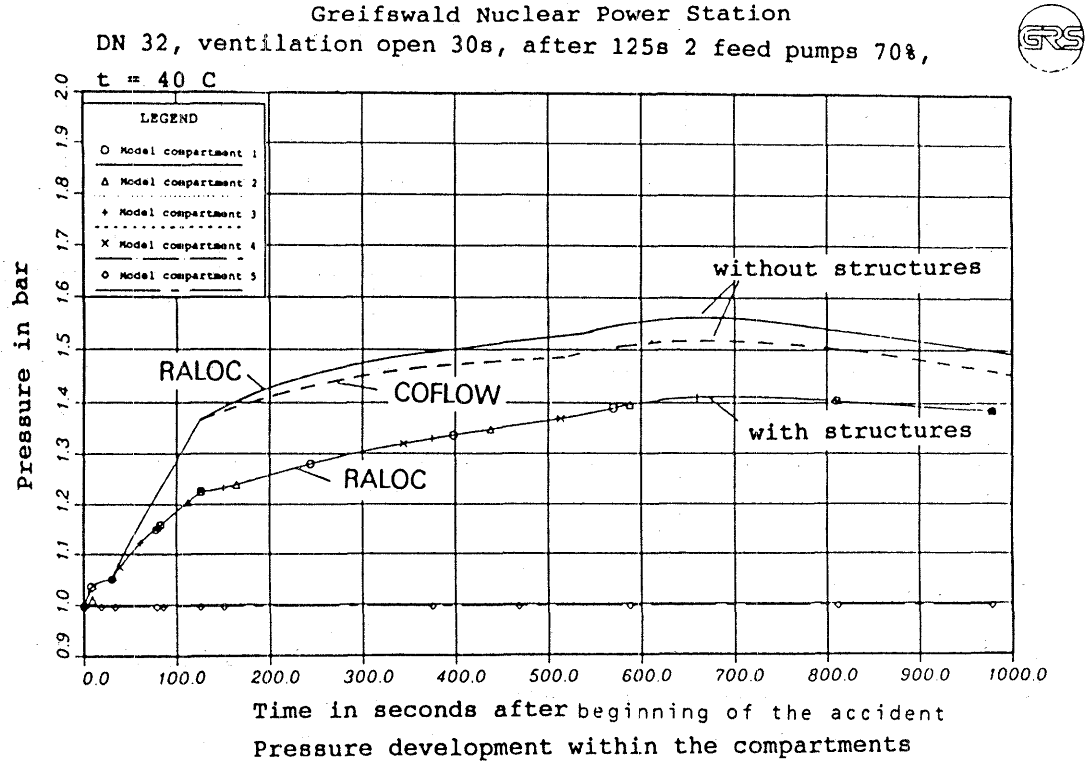


Fig. 6-3: Calculation for the design basis accident

(leak equivalent to nominal diameter 32) no response of the dampers takes place. The calculation without heat conductive structures, which is conservative with respect to the pressure development, is confirmed by comparison with the programme COFLOW (see Fig. 6-3).

If only one spray water cooler is available, the spray water flow rate is reduced from 155.5 to 145.6 kg/sec, and the spray water is increased from 40°C to 50°C. In this case (Fig. 6-4) the pressure within the pressure resistant compartment system remains below the operating pressure of the dampers. The removal of heat into the structures was taken into consideration here (see the lower curve in Fig. 6-3).

/6/ shows for the design basis accident a pressure development which was determined by the program BRACO (see annex 2). Comparative calculations using the programs RALOC and COFLOW show good agreement with the BRACO result under identical boundary conditions (see Fig. 6-5). In the calculations, different times for the spray system to become effective (without total loss of power) were assumed as a function of reaching the initiation criterion for the spray system (1.2 bar and 25 sec running time for filling the pipes). The analyses show that under realistic conditions in the case of the design basis accident (leakage equivalent to nominal diameter 32) the pressure relief dampers of the pressure resistant compartment system do not respond.

Accordingly, the functioning of the pressure relief dampers of the pressure resistant compartment system was also investigated in analyses for the pressure of a connecting line of nominal diameter 200 with unilateral flow (leak of nominal diameter 200/1 F). These analyses

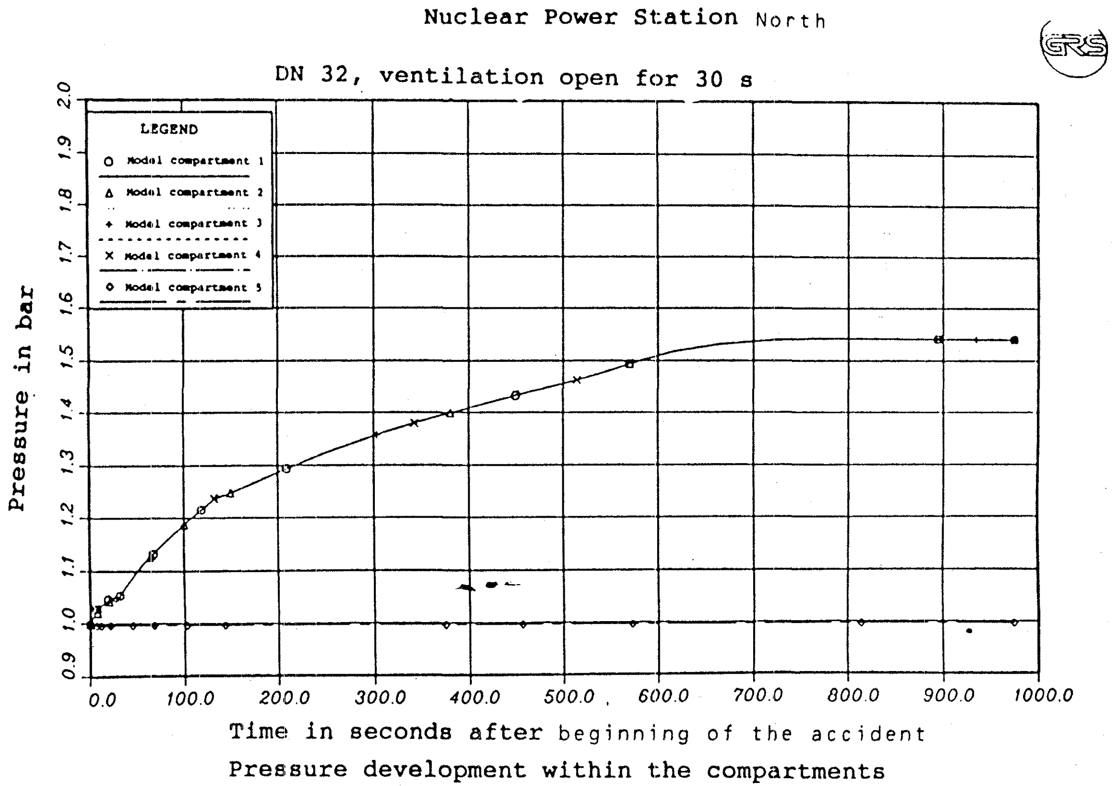


Fig. 6-4: RALOC computation with only one spray cooler

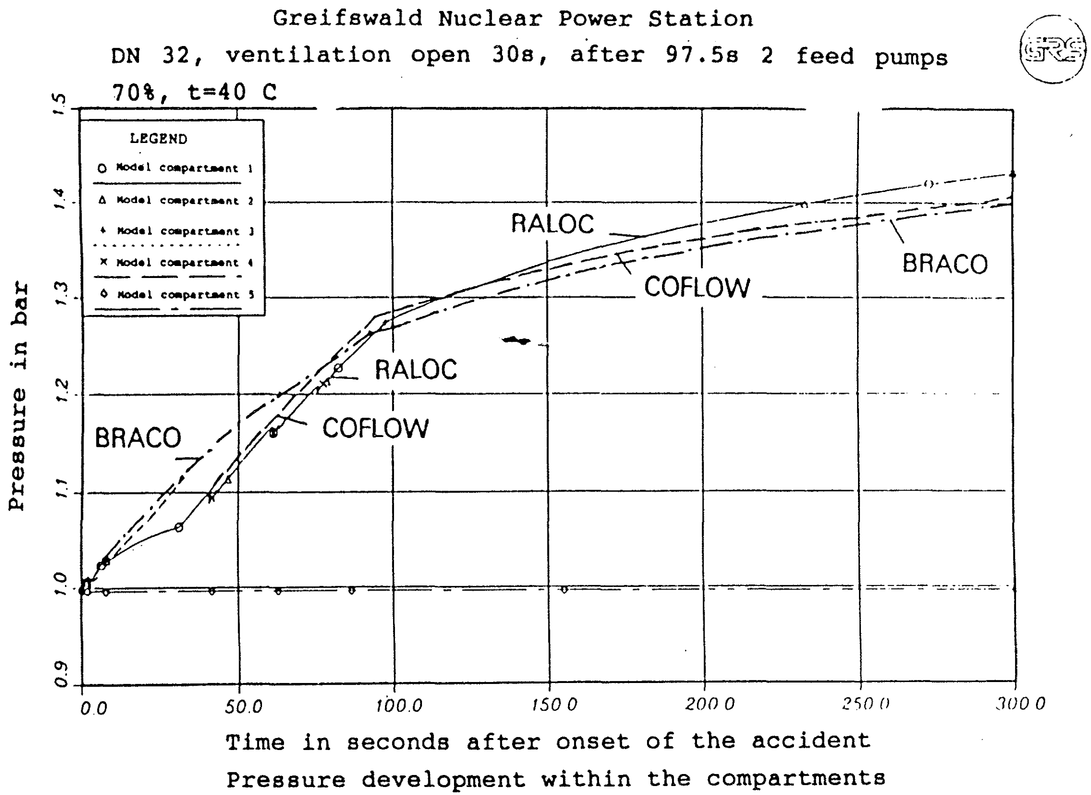
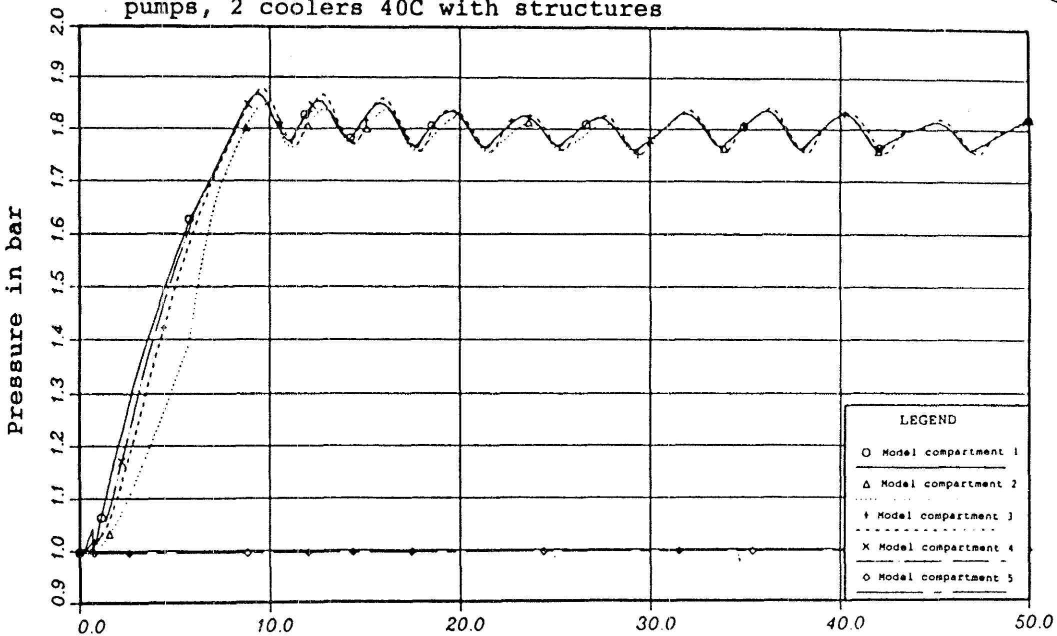


Fig. 6-5:
Program comparison between RALOC, COFLOW and BRACO

Fig. 6-6: Investigations concerning the damper function using RALOC

Nuclear Power Station North

1-F DN 200, without total loss of power, 70% 2 feed pumps, 2 coolers 40C with structures



Time in seconds after beginning of the accident

Pressure development within the compartments were carried out using the program RALOC, in which the dead weight of the dampers is taken into account. By this means, the opening function represented in /6/ (open - closed without an inertia term) is more accurately covered. The data of the blowdown (mass, enthalpy) were taken from documents of the operator, which had been determined using the program RELAP4/ MOD6. Figure 6-6 shows the pressure development of the RALOC calculation sequence with boundary conditions comparable to Fig. 6-5: ventilation closed after 30 sec, 25 sec after reaching a pressure of 1.2 bar in the pressure resistant compartment system, the spray system injects 155.5 kg/sec (70% of the nominal flow rate) of water at 40°C into the compartment system (without total loss of power). The removal of heat into the structures was taken into account in the calculation.

The comparison with the pressure development determined using BRACO (from /6/, Fig. 6-7) shows the clear reduction in the frequency of opening of the dampers. As time passes, the amplitude and frequency of oscillation decrease, but in the case of the RALOC results, due to the consideration of mass (inertia) of the damper, an overswing beyond the opening pressure of the large damper of 1.8 bar can nevertheless be observed. The markedly lower opening frequency, which becomes evident in a realistic simulation, should mean a higher degree of reliability in the operation, especially the closing function, of the dampers.

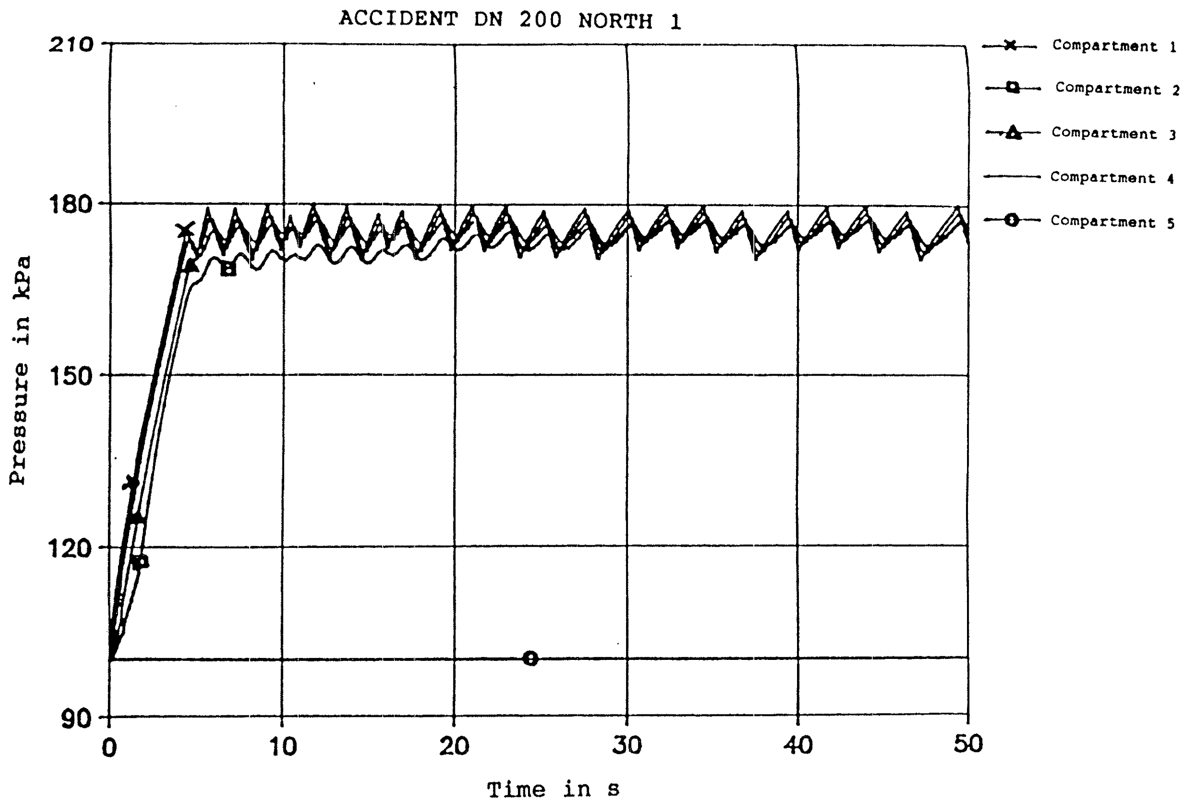


Fig 6-7: Investigations concerning the damper function using BRACO.

6.3.3 Outlook to further analyses

For the purposes of an assessment of the pressure resistant compartment system including an expanded spectrum of design basis accidents, the following analyses are proposed:

a) Design basis accident (leak of nominal diameter 32), determination of the leakage rates from the pressure resistant compartment system to assess the radiological charge of the environment of the system (ventilation, leaks at the fan etc.),

b) LOCA (nominal diameter 200, 1 F) using a revised model for the damper function for the determination of the release of radioactivity,

c) Analysis of the controllability of the break of the largest connecting line (2 F, nominal diameter 200) to the primary coolant loop in the compartment system (pressure build-up, pressure differences between individual compartments, temperature development) and determination of input data for a radiological source term calculation,

d) Break of a main steam line (as in c),

e) Investigation of various conditions relating to the effectiveness of the spray system and dampers.

f) Effect of the closing times for ventilation valves and of the failure of individual ventilation flaps on the accident sequence.

6.4 Safety-related requirements

6.4.1 Measures to be implemented on a short-term basis

a) The number of isolation valves (air exhaust valves) of the ventilation system W 2 which are open in operation is to be reduced to the minimum number required for depressurization within the compartments of the pressure resistant compartment system. The remaining air exhaust flaps are to be closed and secured before operation, and their tightness checked. The open isolating valves are to be protected against penetration by foreign bodies, which may prevent closure.

b) By an intensification of the maintenance work or additional sealing operations, the actual leakage rate of the pressure resistant compartment system on operation of the units is to be reduced to 300 m3/h (test pressure 1.25 bar). Unit 3 is already close to this value. Moreover, measures are to be adopted to reduce the closing times of the ventilation armatures, especially the air exhaust flap of the system W 4, of nominal diameter 1000.

c) The tightness tests on the pressure resistant compartment system have to date been carried out only with a test pressure of 1.25 bar. In order to clarify the pressure dependence of the leakage rate, tightness tests at the maximum permissible test pressure should be carried out on at least one unit.

d) The operative capability and tightness of the dampers of the pressure resistant compartment system is of prime importance to the operation of the compartment system as confinement. Accordingly, it is necessary to demonstrate adequate reliability of the dampers.

If adequate reliability cannot be demonstrated, backfitting measures are required.

e) The tightness and loadability of the penetrations through the walls of the pressure resistant compartment system (e.g. of the cable penetrations) are to be checked at design pressure and temperature (type testing).

f) It must be shown that long-term decay heat removal from the pressure resistant compartment system is assured by one spray cooler. Otherwise, the redundancy of the coolers is to be increased.

g) The formation of combustible gas mixtures within the pressure resistant compartment system (hydrogen) must be prevented following accidents involving loss of coolant. Investigations concerning the limitation of local hydrogen build-up should be carried out and the required backfitting measures formulated.

h) On inclusion of an expanded breach spectrum related to the primary coolant loop, proceeding as far as the break of the largest connecting line (nominal diameter 200, 2 F) in the backfitting of the pressure resistant compartment system, it is necessary to include possible dynamic stresses of the structure and of the safety systems (shock waves, jet forces, missiles etc.) and associated consequential damage in the preparation of relatively long-term backfitting measures.

6.4.2 Measures to be implemented on a longer-term basis (backfitting measures)

For the purposes of relatively long-term continued operation of the nuclear power station units (beyond 1992), the implementation of the following backfitting measures is considered necessary:

a) The isolating valves for pipes which are connected to the primary coolant loop and penetrate the pressure resistant compartment system are to be designed on a redundant basis. This process should be in accordance with criterion 8.4 of the safety criteria for nuclear power stations in the FRG /2/.

b) In order to ensure a long-term removal of decay heat from the pressure resistant compartment system, at least a second sump drain is to be provided from the compartment system into the emergency boration tank(s).

c) The required tightness of the pressure resistant compartment system is to be determined for an expanded spectrum of accidents proceeding as far as the break of a line of nominal diameter 200 (2 F). Any measures required to increase the tightness are to be implemented.

d) In order to control an expanded spectrum of accidents (leak > nominal diameter 32), the function of the pressure resistant compartment system - rapid pressure relief associated with reliable, early closure of relief flaps - is to be guaranteed. Measures required within the region of the dampers are to be prepared and implemented.

e) The measures derived from the investigations concerning dynamic stresses of the structure and the safety systems (see item 6.4.1) are to be implemented.

f) Measures are to be implemented to prevent the formation of combustible gas mixtures within the pressure resistant compartment system.

7. SYSTEMS ENGINEERING

7.1 Introduction

The assessment of the systems engineering is undertaken with reference to a list of events which may initiate accidents (initiating events). In selecting these initiating events, consideration is given to operational experience and to plant specific conditions. For each initiating event, event sequences are discussed which may arise depending on the functioning or failure of the systems actuated. A system fails when it cannot meet the minimum requirements imposed upon it. The minimum requirements are based on the reactor-dynamic and thermohydraulic accident analyses (chapter 5).

In assessing the systems engineering, consideration is given to the following aspects:

- Actuating criteria
- Required auxiliary systems (power supply, cooling, lubricant supply)
- Degree of redundancy
- Reduction of the degree of redundancy in emergency power supply mode
- Consequential failures and functional dependencies
- Human interventions and time available
- Alternative measures

7.2 Initiating events

The investigation is restricted to events which relate to the reactor core. Loss-of-coolant accidents are considered as initiating events.

Loss of coolant accidents

- Leak ≦ nominal diameter (ND) 200 (314 cm^2)
- Leak ≦ ND 32 (8 cm^2) with one-sided escape
- Leak at the pressurizer ≦ ND 90 (64 cm^2)
- Leak at a steam generator heating tube ≦ ND 13 (1.3 cm^2)
- Leak at several steam generator heating tubes
- Leaks via connecting lines of the primary loop which leave the confinement.

Only those leaks are considered which are not overfed by the operational feed system and which result in an actuation of safety systems.

A rupture of the pressurizer surge line upstream of the check valve installed in this line leads to a leak of ND 200 (314 cm^2) with one-sided discharge. A rupture of one of the pressurizer connecting lines or of the pressurizer surge line downstream of the check valve leads to a leak of ND 200 with two-sided discharge.

The emergency cooling system (emergency boration system) in units 1 and 2 (six emergency cooling pumps of type EP 50) was designed to cope with leaks equivalent to ND 32 (8 cm^2) with one-sided discharge; this is the diameter of the discharge limiter in the pressurizer spray line of ND 90 (64 cm^2)). The six emergency cooling pumps in units 3 and 4 (type ZN 65) have a substantially higher capacity. The 12 emergency cooling injection lines of ND 50 (20 cm^2) have discharge limiters of ND 26 (5 cm^2) at the positions of bonding into the main coolant lines. Accordingly, a rupture of one of these lines gives, by summation of the two discharge cross-sections, a leak cross-section of 25 cm^2. Even more unfavourable is a rupture of an emergency cooling injection header in the region of the separation valves. In this case, the discharge cross-section amounts to 32 cm^2. Such ruptures can be limited by appropriate isolation measures to 5 cm^2 with one-sided discharge or can be completely isolated. However, the leakage location required for this purpose is costly in terms of time. It remains to be analysed whether damage to fuel elements is to be expected in the case of leaks of 32 cm^2 without isolation measures, if only two emergency cooling pumps are available.

The leak of ND 90 (64 cm^2) corresponds to a rupture of the line from the pressurizer to the pressurizer safety valves. It represents the largest pipe leak with discharge of steam. The line to the spray distributor also has a ND of 90 (64 cm^2), while the total cross-section of the spray nozzles is only approximately 28 cm^2. In the reactor pressure vessel head there are 37 apertures for the control rod drives (control and protection system standpipes) at which leaks may also occur, corresponding to a maximum of ND 90 (64 cm^2).

With regard to leaks of ND 200 (314 cm^2) with one-sided or two-sided discharge, it has to be investigated whether core meltdown and failure of the confinement can be prevented using four available emergency cooling pumps.

For leaks large enough to trigger response of overpressure dampers of the confinement (8 dampers of ND 1130, operating pressure 0.8 bar and 1 damper of ND 500, operating pressure 0.6 bar) it has to be investigated whether the long-term effectiveness of the emergency cooling is unacceptably impaired if overpressure dampers incorrectly remain open.

For all primary leaks within the confinement, this question is to be clarified also with regard to a failure of the isolation of the ventilation lines.

Larger leaks, e.g. of ND 500 (main coolant line), can not be coped with by the emergency cooling systems. They lead, moreover, to a failure of the confinement.

The failure of the following primary loop connections is covered by the above mentioned leakage cross-sections:

- Auxiliary spray line of ND 50 (20 cm^2)
- 12 connections on the reactor pressure vessel head for core outlet temperature measurements
- 12 connections on the reactor pressure vessel head for in-core instrumentation
- 104 penetrations for pressurizer heating elements
- one connection on the reactor pressure vessel for pressure and differential pressure measurements
- various connections with small cross-sections (measurement and vent lines, drainage arrangements, sampling lines)

Transients

- Loss of normal a.c.power (emergency power mode)
- Loss of main heat sink
- Loss of main feedwater
- Overfeeding of pressurizer
- Loss of a turbo generator set
- Start-up and shutdown procedures
- Leak of a main steam line (various leak locations)
- Leak of the main steam header
- Leak of a feedwater line (various leak locations)
- Leak of one of the three feedwater headers
- Loss of the service water system
- Loss of several reactor coolant pumps
- Loss of the protected 380 V three-phase main distribution system
- Loss of the 220 V d.c. main distribution system
- Reactivity perturbations
- Anticipated transients without scram (ATWS)

7.3 Event sequences

This section contains details of the systems required to cope with the initiating events, and of system weak points. Each system weak point is discussed only once, i.e. it is not recurred in the discussion of further initiating events.

7.3.1 Leak ≦ nominal diameter 200 (314 cm^2)

Reactor shutdown is initiated either by the criterion "pressure in the primary loop ≦ 12.0 MPa and pressurizer level low" (actuating RP 1, dropping of all control rods within 6 to 13 secs) or by the criteria "pressure in the primary loop ≦ 11.5 MPa" or "pressure in the confinement high" (actuation RP 2, staggered dropping of the control rods). The criteria are diverse to each other.

The emergency cooling systems are activated by the criteria "pressurizer water level < 2.56 m" or "pressure in the primary loop < 10.8 MPa". The emergency cooling pump which, for the purpose of preheating, operates in recirculation mode, is not available for an automatically activated injection.

A serious weak point here is the low capacity of the emergency cooling systems, especially after a loss of normal a.c. power which may be caused by the LOCA.

In order to minimize release and to prevent consequential damages by discharging steam in loss-of-coolant accidents, it is on the one hand necessary that overpressure dampers reclose after response, and on the other hand that the ventilation lines of the confinement are isolated. To this end, 14 air exhaust valves of ND 250 and one air exhaust valve of ND 1000 are to be closed by motor drive.

These valves are existing only once in each ventilation line; their failure probability can be estimated by evaluating the functional tests to about $5 \cdot 10^{-3}$ per demand.

In addition, there are a total of 42 oscillatory underpressure dampers, which must have to close passively. Failure of such dampers causes release of primary coolant steam either into transmitter rooms or into rooms of the ventilation system. The oscillatory dampers appear to be relatively reliable, but are in each instance likewise provided only singly.

The overpressure dampers are basically considered to be capable of satisfying the requirements set. However, they are also provided only singly in each case.

The minimum requirements placed upon these dampers are to be determined. Their operational capability with regard both to opening and to closing must be demonstrated. Any backfitting measures which may be required as a result must be implemented.

In connection with possible releases, it has to be considered that the pressure within the confinement may increase again after initial pressure relief via the overpressure dampers, caused by a failure of the spray system or by this system being switched over from spray operation to direct cooling of the emergency core cooling water supply. Such switchover takes place automatically if the temperature of the emergency core cooling water rises by approximately 5 K as a consequence of the return flow of the sump water. The time at which this value is reached has to be evaluated.

7.3.2 Leak ≦ nominal diameter 32 (8 cm^2) with one-side discharge

Because of the lower leak rate as compared with leaks discussed under section 7.3.1, the reactor shutdown is delayed. The actuation criteria are the same as in section 7.3.1.

At the present time, it is assumed that two emergency core cooling pumps of type EP 50 or one pump of type ZN 65 are required to cope with the accident. It should be investigated whether one pump of type EP 50 is sufficient.

Additional operator actions are required to cope with a rupture of an emergency core cooling injection line, a simultaneous loss of normal a.c. power and a simultaneous single failure in an emergency power distribution system or at a valve in an injection line.

If the leak is located between the last check valve of an emergency core cooling injection line and the separating valves in the emergency core cooling injection header, and if there is an additional single failure, the situation can be managed only by leak isolation; however, the localization of the leak is time-consuming.

Weak points of the emergency core cooling systems are considered to be:

- the three (in units 1 and 2) or two (in units 3 and 4) motor valves in series in the emergency core cooling injection lines, which must open on demand (the EP 50 pump requires an active control of counter-pressure, while the ZN 65 pump requires only an orifice),

- the inadequate emergency power supply of the pumps, especially of the emergency core cooling pumps, of which only two are electrically supplied in case of a loss of normal a.c. power,

- the single-train design of the component cooling water circuit, by which the emergency core cooling pumps are cooled (a common forward and return flow line for all emergency core cooling pumps),

- the additional operational utilization of the emergency core cooling pumps for preheating the emergency borated water, whereby the pump employed for this purpose is not automatically available for emergency cooling if required,

- the absence of an independent long-term low-pressure injection into the primary loop,

- the low redundancy of the spray water system; the cooling of the emergency coolant water requires both spray water coolers, both preselected spray water pumps and the opening of four motor valves in the return lines leading to the service water system. In order to limit the pressure in the confinement two out of two motor valves have to open on demand. The third spray water pump can be started and the third valve can be opened by manual command. However, depending upon the leak rate, this command may take place too late to prevent the response of the overpressure dampers of the confinement. It should be investigated whether one spray water cooler and one spray water pump would be sufficient.

- If certain single failures occur, corrective human interventions are necessary to cope with the accident.

For the leak sizes investigated here, decay heat removal via the secondary side is necessary, since the energy discharge through the leak is unsufficiant. To this end, it is possible to utilize the operational systems. After loss of normal a.c. power, the emergency feedwater system is required for feeding the steam generators, and for main steam release one out of two relief valves are required. A shutdown of the system by means of these valves must be controlled manually. One secondary pump and two steam generators are required for the removal of decay heat.

The availability of the secondary side is expected to be higher than of the primary side. So the systems, for the "main steam discharge" function exhibit a high redundancy. For the system function "steam generator feed" there is sufficient time available for human interventions.

7.3.3 Leak at the pressurizer ≤ nominal diameter 90 (64 cm^2)

The reactor shutdown (RP 2) is initiated by two criteria ("primary loop pressure low" or "pressure in the confinement high").

It is assumed that three emergency core cooling pumps are required in order to cope with leaks of ND 90 (64 cm^2). It should be investigated whether two emergency core cooling pumps and one spray water cooler would be sufficient. Energy from the confinement is removed by means of the spray system via the spray water coolers. It is not possible to cope with a simultaneous loss of normal a.c. power and a single failure in the emergency power supply, or with a single failure in an injection line or in a service water line to the spray water coolers.

In the case of leaks at the pressurizer as a consequence of inadvertent opening of one or both pressurizer safety valves (each ND 32 (8 cm^2) each), one or two emergency core cooling pumps are required.

7.3.4 Leak at a steam generator heating tube ≦ nominal diameter 13 (1.3 cm^2)

The reactor shutdown is delayed because of the low discharge rate; the following actuation criteria are expected:

- "Pressure in the primary loop ≦ 11.5 MPa"
- "Pressure in the primary loop ≦ 12 MPa" and "Pressurizer water level low"

The operating personnel has to recognize the accident, to identify the defective steam generator and to isolate it on the primary and secondary side. Various indications are available for such detection:

- A drop in the pressurizer water level
- Automatic activation of the feeding pumps
- An increase of the activity in the secondary system (sensors at the turbine and at the main steam lines)

Prior to the isolation of the affected main circulation loop (closure of the main block valves and of the forward and return line of the loop demineralisation system) becomes effective, an emergency cooling pump is required for leakage compensation. The coolant emerging from the leak is lost via the secondary side.

If the motor drive of a main block valve fails, the valve can be closed manually by the personnel in situ (time required: approximately 30 minutes). If this is not possible too, further manual measures must be taken, in particular a reduction of the primary pressure to a level below the operating pressure of the secondary-side safety valves and subsequent secondary-side isolation of the steam generator concerned. However, these manual measures have not hitherto been specified in the operating specifications.

The primary pressure can be reduced by pressurizer spraying; in the emergency power mode, spraying is not possible since the reactor coolant pumps are not operating. Instead of this, for example, a pressure reduction could be achieved by repeated filling and emptying of the pressurizer.

The decay heat must be removed via the secondary side.

7.3.5 Leaks at several steam generator heating tubes

In order to cope with ruptures of several steam generator heating tubes, for each broken tube the quantity transported by one emergency cooling pump is required to control the accident, as long as the defective steam generator is or generators are not isolated on the primary side.

Reactor shutdown is initiated by the criteria mentioned under section 7.3.4, but set points are reached more rapidly in this case. It has to be investigated whether a rupture of three or more tubes in a steam generator due to a main steam line break or leaks at several steam generators as a consequence of a main steam header failure are likely.

7.3.6 Leaks via connecting lines of the primary loop leaving the confinement

Such leaks, e.g. via the emergency core cooling injection line, feed line, pressurizer auxiliary spray line or sampling line, have to be prevented, since the coolant emerging from the leak does not flow back into the sump and the emergency boration tank, but is lost from the confinement. This also leads to the direct release of activity.

Possibilities of loss-of-coolant from the primary loop in systems which exclude feedback must be considered. In principle, the high-pressure systems are separated from the low-pressure systems by double isolation with drainage between. The following low-pressure systems are connected to the primary loop:

- Emergency cooling injection lines of the two emergency core cooling systems and of the intermediate drainage connected to the leakage water sump
- Drainage and ventilation lines of the primary loop, passing via the leakage water cooler into the leakage water sump
- The injection line of the primary loop filling pumps
- The leakage water sump inlet and outlet lines
- The connecting line to the N2 network
- Connecting lines of the resin rinsing lines
- Connecting lines for the clean condensate for resin rinsing and for the regeneration of the filters in the water treatment plant 1
- Chemicals connecting lines for the regeneration of the filters of the treatment station 1 in units 3 and 4
- Connecting lines of the special gas treatment plant
- Sampling lines of the primary loop
- Loss of primary coolant via resin filling connections of the water treatment station 1 leading into the reactor hall.

Furthermore, non-returnable losses of primary coolant into low-pressure systems are possible via defective heat exchangers:

- Intermediate coolant circuit (ICC-NPS) connecting lines of the leakage water cooler
- Heat exchangers of the independent coolant circuit of the reactor coolant pumps (primary coolant leak into the ICC-RCP)
- Cooler of the regenerative heat exchanger of the water treatment station 1 (primary coolant leak into the ICC-RCP)
- Heat exchanger of the coolant circuit of the control rod drives (primary coolant leak into the ICC-CPS)

An undesired discharge of primary coolant is monitored or prevented by the following measures:

- Low-pressure systems are connected to the primary coolant loop via double isolations with intermediate drainage; the entire system is designed to withstand the nominal pressure of the primary coolant system. The valve position is specified for all operational cases; the valves are secured by chain and lock. A lead seal is also provided for monitoring purposes.
- The water in the ICC-RCP, ICC-CPS, ICC-NPS and the steam generators is monitored for activity.

It should be noted that there are no pipe whip restrictions. Accordingly, consequential damage in the event of pipe failures cannot be excluded.

The incorrect switching of connecting lines may lead to serious accidents, and even to a primary leak equivalent to ND 32.

Internal leakages of the heat exchangers lead to loss-of-coolant accidents which are within the controllable leakage spectrum, but the leaked fluid is lost from the coolant inventory. Except for a leak in the cooler of the regenerative heat exchanger, isolation is readily possible.

Leak detection is possible from a wide range of information. By isolation of partial systems, the leak can be located and then isolated. Leakage location is, however, time-consuming.

7.3.7 Loss of normal a.c. power (Emergency power mode)

Reactor scram is initiated by the trip of both turbo generator sets (closure of two out of four quick-closing valves of the last operating turbo generator set). There is a redundant initiating signal via "trip of the last turbine and opening of the associated generator circuit breaker" (the switches at the fast-closing valves, from the position of which the two signals are derived, are present in pairs). Furthermore, on turbine rundown after approximately one to two minutes the diverse criterion "failure of more than two reactor coolant pumps" is initiated.

In the event of voltage dip at the emergency busbars the emergency power diesel are started. Two out of the three diesel generator sets are required to cope with the transients. If an emergency diesel generator set fails to start, the third emergency diesel generator set is automatically switched over to the dead busbar. The automatic switching system operates only as long as the third diesel was not switched to the preselected busbar. After starting and connecting the diesels, the consumer reload sequence program proceeds. Once this program has been executed, the switching-in of emergency power consumers is possible only manually. Likewise, in the event of operational failure of a diesel, manual switchover actions are required in order to restore the supply to the emergency power distribution system which may have failed.

The heat from the primary system is removed via the secondary side. The emergency feed water supply to the steam generators is required approximately 3.5 h after loss of normal a.c. power. Until this time, the feedwater inventory of the steam generators is sufficient.

For the steam release, two steam generators as well as one of the two main steam relief valves or one of the twelve main steam safety valves must be available (on the steam side, the steam generators are connected via the main steam header).

The weak points of an emergency power supply system are:

- Low redundancy of the emergency diesel generator sets
- Interconnection of the two emergency power distribution systems by the third diesel, which is not associated with a particular train
- Frequently inadequate cooling of the emergency diesel generator sets as a result of soiled coolers and due to dependence on the service water system
- No diverse actuation criterion for the diesel start
- The diesel loading sequence programme is liable to malfunction
- Realistic power balances are not available
- Low battery capacity (approximately 30 min).

7.3.8 Loss of main heat sink

The reactor protection system is actuated by the same criteria as for loss of normal a.c. power, but partly with some delay.

It is not necessary to deal with this transient separately, since it proceeds in a manner similar to loss of normal a.c. power, but without requiring the emergency power supply system.

7.3.9 Loss of main feedwater

In this case, there are no secondary-side actuation criteria for the reactor trip. Because of increasing dry-out of the steam generator heating tubes, pressure and temperature will increase. The emergency feed water pumps are activated by the criterion "steam generator water level low".

For some time the turbine control system compensates the pressure drop in the main steam system by closing the turbine control valves.

At a temperature of ≧ 310 °C in the primary loop or at a primary loop pressure ≧ 13.7 MPa, RP 3 (control rod insertion) is initiated and after approximately 20 secs passes over to RP 2 (staggered control rod insertion). At this time, the steam generators have completely boiled out. As a result of the pressure drop, turbine trip and thus RP 1 (dropping of all control rods) are actuated. Subsequently the steam generators are refilled by the emergency feedwater system. It has to be investigated whether the injection of emergency feedwater into the partially drained steam generators may result in damage to the steam generators. Response of the pressurizer safety valves cannot be excluded.

Weak points here are

- low redundancy of the emergency feedwater system
- no actuation of reactor scram by the criterion "steam generator level low". Therefore, in case of failing the emergency feedwater system the time available is not sufficient for manual interventions to restore the steam generator feed water supply.

7.3.10 Overfeeding of the pressurizer

The pressurizer water level is monitored by three fine level measurements and one coarse level measurement. No reactor protection actions are derived from the criterion "pressurizer water level high". Therefore, critical plant conditions may arise in the following cases:

- Tightness test at 4 MPa
  This test is performed prior to start-up of the plant after revision. During this procedure, the pressurizer is entirely filled with water. To this end, the 6 m level interlock of the feed pumps must be disabled. In the event of a pressure rise in the primary coolant loop, e.g. due to disturbances of the heat removal or by start-up without prior reduction of the pressurizer water level the temperature may fall below the corresponding NDT (nil ductility transition temperature)

- Overfeed in power operation mode
  A primary pressure exceeding 13.2 MPa initiates RP 4 (prohibition of control rod withdrawal), a pressure

exceeding 13.7 MPa initiates RP 3, and after further 20 secs RP 2. At 14.2 MPa, the first pressurizer safety valve responds.

7.3.11 Failure of a turbo generator set

If one of the two turbo generator sets of a unit fails then the number of rundown-supported main coolant pumps is reduced. Therefore, the reactor power is quickly reduced. Since the turbine monitoring control system cannot follow quickly enough, the second turbine draws more power than the reactor produces. The resulting subcooling transient may cause the emergency cooling system to be actuated.

7.3.12 Start-up and shutdown procedures

Start-up and shutdown procedures are carried out in each unit at least once a year.

The numerous manual measures required are a potential source of human error, while corrective interlocks may in some cases be disabled.

7.3.13 Leak in a main steam line (various locations)

In the event of a leak near to the turbine, the main steam pressure in the header does not necessarily fall below the set point of the criterion 6.4.19 (closure of the affected steam generator in the direction of the main steam header, main and emergency feedwater systems and drain systems, shutdown of the reactor coolant pump). Furthermore, the set point for the turbine trip (pressure in the main steam header < 4 MPa) is not reached. It is necessary to adopt measures to isolate the leak, for which, however, sufficient time is available.

In the case of leaks which are situated closer to the steam generator, the defective steam generator is isolated by the criterion 6.4.19. Boil-out via the leak takes place.

After failing the main steam isolation valve, the header pressure falls below 4 MPa, whereupon turbine trip and reactor scram are initiated. At 3.5 MPa in the header, all main steam isolation valves close. The main steam relief valves (fast acting atmospheric exhaust system) and the main steam by-pass station (fast acting steam dump system) are separated and are no longer available. The main steam safety valves are activated. In the longer term, the main steam header must be separated by a manual command to the separating valves, in order to restore to service a semi-unit (3 out of 6 generators), i.e. in order to make available a relief valve or a by-pass valve. As a result of the subcooling, subsequent feeding into the primary loop takes place. This can lead to the response of the pressurizer safety valves during further operation.

In the case of such an accident, there is a risk of consequential damage to steam generator heating tubes.

The probability of consequential damage to other components of the secondary side is difficult to estimate, especially on the 14.7 m platform, on which all main steam, feedwater and emergency feedwater lines are routed close together. In the event of such consequential damage, the emergency feedwater system may be impaired at the same time.

If the leak is located within the confinement, the overpressure dampers of the system are required to work. The criterion "pressure high" is additionally available in this case to initiate the reactor scram.

7.3.14 Leak at the main steam header

In the event of such a leak, the header is isolated by the main steam quick-closing isolation valves (cf. Section 7.3.13). The problem of consequential damage in the region of the 14.7 m platform exists also in this case. As a result of the rapid pressure drop, a damage to heating tubes of several steam generators may be caused.

If more than one quick-closing valve fails the subcooling transient could cause a risk of brittle fracture of the primary coolant loop and of recriticality.

Transients as a consequence of inadvertent opening of main by-pass or main steam safety valves are similar in terms of their effects to leaks from a main steam line or the main steam header.

7.3.15 Leak of a feedwater line (various locations)

This transient proceeds substantially in the same manner as a main steam line leak, but at a lower degree of sub-cooling of the primary side. The pressure drop in the main steam system is delayed. Due to the criterion "pressure on the pump pressure side low", all main feed pumps are switched off.

Reactor scram is initiated relatively late; the emergency feedwater injection takes place into almost empty steam generators (see section 7.3.9).

The most unfavourable leak location is between a steam generator and the last check valve downstream of the steam generator. In this case, the emergency feedwater line to the corresponding steam generator must be isolated, since otherwise emergency feedwater is supplied to the leak. The location of larger leaks can be detected in the first phase of the transient by the differing steam generator water levels; smaller leaks are overfed and can be localized by testing various isolating arrangements only. In the event of leak locations within the confinement at a steam generator drain line, the reactor protection criterion "confinement pressure high" is reached.

In the event of a leak in a drain line, there is no automatic shutdown of the main feed pumps. The detection of the affected steam generator is time-consuming.

7.3.16 Leak of one of the three feedwater headers

There are one suction-side and two pressure-side feedwater headers. Due to the failure of the main feedwater supply in the case of a large leak at one of these headers, turbine trip and reactor scram are initiated with some delay (cf. section 7.3.9).

Leaks in the pressure header proceed substantially in the same manner as a feedwater line rupture. By a leak in the suction header, aditionally the emergency feedwater is affected.

The content of the feedwater tank flows out through the leak into the turbine hall. This leads to impairment of electrical systems, pumps and valves by water and steam.

Consequential damage to components in the turbine building in the event of failure of tanks with high energy content (feedwater header, feedwater tank, high-pressure preheater) is possible.

7.3.17 Failure of the service water system

There is one service water system for each double unit. It cools consumers both of operating and of safety systems. So, the unit transformers, the reserve transformers and also the diesel coolers are cooled by the service water system. The unit transformers may possibly be cooled by the main cooling water, but the main cooling water pumps again are cooled by the service water system. A maximum of two of the four reserve transformers of units 1 to 4 can fail as a result of the failure of the service water system of a double unit. The power station internal transformers were converted to air cooling, and this is also intended for the remaining reserve transformers.

Electrical supply, e.g. from the other double unit, may be provided by manual measures. If these manual measures are not implemented, then after about 30 min the batteries are depleted and the plant is out of control.

The service water system could fail as a result of

- flooding of the intake structure, e.g. by leak of a line, under circumstances caused by loads dropping from a crane
- blockage of the water intake
- rupture of the flow pipe
- damage to the cable line common to all pumps

Weak points are on the one hand the inadequate degree of redundancy of the service water system (pipes) and on the other hand the fact that both operating and safety systems (emergency diesel generator sets) are cooled by the service water system.

7.3.18 Failure of several main coolant pumps (MCPs)

This transient is of particular importance, since the MCPs have no fly-wheel for extending the coastdown time the after failure of the power supply, but reach standstill within approximately 3 secs. A simultaneous failure of more than two MCPs leads to reactor scram (RP 1). By means of an analysis it has to be checked whether the simultaneous failure and immediate standstill of all MCPs with correspondingly high reactor power leads to overpressure failure of the primary coolant loop. In the event of a single failure in the electrical power supply, the simultaneous failure of four pumps is possible, if two coupled busbars are affected and a particular switching of the MCPs is active. In this case, the result is a pressure increase in the primary coolant loop until the pressurizer safety valves open.

Such a transient is possible if the specifications contained in the table of permissible power levels are infringed. Furthermore, it has to be investigated whether a failure in the control system may lead to the simultaneous shutdown of all MCPs.

7.3.19 Failure of the protected three-phase 380 V main busbar or of the 220 V main d.c. busbar

The essential functions of the system for uninterrupted power supply are the provision of control voltages, measurement and signalling voltages, the supply to valves, which must be available without interruption, and the power supply to the automatic interlock systems.

Within the uninterrupted power supply system, the following deficiencies exist:

- Only one d.c. busbar is provided per single unit. An additional d.c. busbar is used for two units.

- The rotary converters require frequent outages and may be recoupled only upon shutdown of the associated protected main busbars.

- Double earth contacts at the 220 V d.c. busbars may lead to voltage doubling at the doublefed consumers.

- In the emergency power mode, the rotary converters must be switched from rectifier operation to inverter operation and then back to rectifier operation. This leads to a reduction of the reliability of the supply.

- Various d.c. consumers, such as the 6 kV switch control mechanism, are not fed redundantly.

7.3.20 Reactivity disturbances

During the previous operation of the plants, there was in several cases an unacceptable injection of demineralised water into the primary coolant loop. However, the resulting low boron concentrations did not cause safety concern.

Reactivity disturbances can also arise as a result of failures in the reactor control systems. In order to control the reactor, the 7 shutdown rods of the control group are moved uniformly; the other control rods are fully withdrawn during power operation. During start-up, rods of these control groups are also uniformly moved.

Disturbances of the global reactivity initiate a reactor scram (see section 7.3.21). Disturbances of the reactivity of individual core regions may occur because of the above mentioned mode of operation of the control rods due to

- fluctuations of coolant flow through a fuel assembly,
- unintentional withdrawal of a control rod
- sticking of a control rod
- axial disturbances of reactivity due to altered rod positions.

These disturbances lead to symmetrical loads or interfere with the axial power density distribution.

During operation no events with safety concern did happen. Such events can at the present time be detected only by alteration of the fuel assembly outlet temperatures. Instances of exceeding the permissible fuel assembly heat margins are monitored. Information on the local power density distribution is not available.

(The statements regarding operational experience are still to be checked).

7.3.21 Anticipated transients without scram (ATWS)

Reactor dynamic calculations on ATWS are not available. The system for automatic reactor scram has diversity only in the actuation level, but not in the relay section for the initiation of reactor scram.

A diverse shutdown system by rapid boron injection is not available. In order to compensate slower reactivity variations (temperature, poisoning) and for the long-term maintenance of subcriticality, the feed system and the emergency cooling system are available.

7.4 Control system and electrical power supply

7.4.1 Control system

- Brief description

The control system is built up on a hierarchical basis; in addition to functions related to a single unit, it has also to fulfil double-unit related functions. For each unit there are local measurement centres, local consoles, an unmanned control station for the control and protection system, a control room and a relay room. For the double unit there are local measurement centres, local consoles, an equipment control room in the active part of the reactor building and a dosimetry control room. For communication between the control centres, there are intercom and telephone systems. The main control centre is the control room. From the control room, the most important components and systems of the unit are controlled, regulated and monitored. Due to the large number of systems linked with process technology, such as six MCPs and two saturated steam turbines, the control system has a relatively large number of measurement, control and signalling stations.

Since the degree of automation is low, the operating personnel are required to undertake a significant number of operations. For the display of switching conditions and signals, the unit control room includes a conventional visual display. Each unit is equipped with an IW 500 MA computer which monitors significant unit parameters, e.g. fuel assembly outlet temperatures. In addition to individual indications, it is possible to select a series of analogue measurement values by means of multi-scale systems. The control system comprises the control and protection system, the safety system and the safety-related measurement, as well as the control systems.

The control and protection system comprises:

- The reactor protection system (RPS)
- The reactor shutdown system
- The neutron flux measurement system (NFS)
- The control system for the regulating units of the reactor
- The system for reactor power control
- The system for displaying the positions of the regulating units of the reactor
- The auxiliary media and electrical power supply systems

According to their safety-related significance, the following systems are assessed:

- Reactor protection system

  The RPS is working four graduated levels with the designations RP 1, RP 2, RP 3 and RP 4. Initiation of RP 4 prohibits an increase of the reactor power; RP 3 causes a reduction of reactor power with normal speed of the control rods; RP 2 causes a reduction in reactor power with maximum speed by dropping of shutdown rods and RP 1 causes reactor scram by simultaneous dropping of all 6 shutdown and control rods.

- Reactor shutdown system

  The reactor has 37 control and shutdown rods. Each rod consists of a control assembly and a fuel element follower and is moved by an electric motor over a gear by means of a tooth rack. The control rods are connected to the tooth rack by means of an electromagnetic coupling and after actuating drop into the reactor core within 6 to 13 secs. There are five control rod groups each with six control rods in a group and one control rod group with seven control rods.

- Neutron flux measurement system (NFS)

  The detectors of the NFS are situated outside the reactor pressure vessel; they all have devices for adjusting their position. There are tree different measurement ranges: for the start-up or source range a logarithmic pulse measurement channel with six counters; for the intermediate range a logarithmic current measuring channel with six ionisation chambers, and for the power range a linear current measuring channel with six ionisation chambers. These 18 channels form two independent trains, of which each system contains 9 channels. Each train supplies set point values concerning low reactor period and high neutron flux levels to the reactor protection system. In addition to this, the reactor period and neutron flux are displayed and recorded in the control room.

- Auxiliary media and electrical power supply systems

The control rod drives are supplied with electrical power by the house load grid of the unit via two lines from the compensation main busbars 1 and 3 utilizing 220/380 V a.c. All further trains of the control and protection system receive their electrical energy via double feed systems from the emergency power grid of the unit in the following manner: components or systems utilizing 220/380 V a.c. from the main safety busbar 1 and 2; components or systems utilizing 220 V d.c. from the main d.c. busbar 1 and the general power station internal d.c. main busbar associated with the double unit. Within the control and protection system, further transformations are performed for equipment and assemblies, e.g. high tension for counters and ionisation chambers, and low tension for electronic assemblies.

- Structure and principle features of the control and protection system

A two-train configuration of the reactor protection system is realized only for the NFS. For all other signals, the two-train structure commences only behind of the set point transmitters. Some acutation criteria of the reactor protection system are utilized several times.

For the reactor protection system the zero current principle is applied. However for generating certain criteria the working current principle is utilized. In principle, all components of the control and protection system are tested once a year during refuelling.

The a.c. supply is not interrupted during automatic switchover from one feed line to the other. Because of the time-behaviour of the equipments there have not been any instances of uncertainty up to date.

The DC supply input takes place with diode decoupling via double feed lines.

Compared to international practice, various actuation criteria of the reactor protection system are missing such as, for example, decrease of water level in the steam generators and an increasing rate of main steam pressure decrease. There is a large number of interlock switches for reactor protection system actuation criteria which are prone to cause operational errors. In addition, there are no automatic deblocking devices. So it is also necessary to switchover the measurement ranges of the NFS by manual action of the operator, i.e. from the start-up range to the intermediate range and from the intermediate range to the power range and vice versa.

Since there are no internal testing devices, the functional testing during operation is difficult.

There are infringements of the principle of spatial separation, e.g. the concentration of the majority of transmitters for the reactor protection system within a transmitter room, and the concentration of almost all assemblies of the control and protection system (CPS) in one room (the control room). The accident-proof construction of the transmitters and of the electrical components as well as the instrumentation and control components has not been demonstrated. Flooding of transmitters and of measurement and control equipment cannot be ruled out.

There is no emergency control station or reserve control room.

The system for measuring the outlet temperature at the fuel assemblies is designed as a monitoring system. When set points are exceeded, no criteria are activated. There is no in-core neutron flux measurement system. For the backfitting of the core monitoring, as well as in connection with the reactor pressure vessel embrittlement and the use of shielding assemblies, a system for monitoring the power density distribution is to be provided.

Departing from common international practice, the design does not include a reactor power limiting device.

Modifications of the control and protection system did increase the reliability. The reactor scram has never failed on demand up to now.

Safety control system

The following functions required to cope with leaks are considered:

- Activation of the emergency cooling system
- Activation of the spray system
- Isolation of the confinement
- Closure of the quick-closing valves in the main steam lines
- Closure of the quick-closing valves of the turbines
- Isolation of one cirulation loop after rupture of a main steam line
- Activation of the emergency feedwater system.

As a rule, process parameters are measured three-fold, with subsequent two out of three evaluation; and only one actuation criterion is available for accident detection. Transmitters are used for multiple purposes, for example for the control and protection system, for the safety system and for the operational measurement and control systems.

The circuits for the actuation of the safety systems, e.g. of the emergency core cooling system, of the spray system or of the emergency feedwater system, do not satisfy the single failure criterion. The control circuits for driving redundant trains relevant to safety are provided only once for each train and are monitored only in some cases. Protective signals have no priority over manual commands. The setting of the unlocking switches is not recorded. An adequate spatial separation of mutually redundant components of the safety systems is not provided.

The automatic MCP system to control and monitor the main coolant pumps is structured on a two-train basis. It is responsible for the following functions:

- In the event of shutdown of two MCPs, shutdown of other MCPs is prohibited.

- When the power control device is malfunctioning or shut down, the failure of one or two MCPs due to the criteria
  - drop in power of one MCP or
  - differential pressure over one MCP ≦ 2 bar.

  or the closure of two out of four quick-closing valves of a turbo generator set initiates RP 3 (staggered insertion of the control rod assemblies at operational speed)

- On account of the abovementioned criteria, the failure of more than two MCPs leads to the initiation of RP 1 after 0.6 sec.

- The separation of the turbo generators 1 and 2 from the grid leads to RP 1 after 2 secs, on closure of 2 out of 4 quick-closing valves of the last operating turbo generator set.

- The failure of the control voltage in one train of the automatic MCP system leads to an alarm at the control room. On failure of the control voltage in both trains, RP 3 is initiated after 0.5 secs.

The reactor coolant pumps have unlocking switches which are situated in the control desk. The unlocking of individual MCPs can be detected via the switch position; no signalling is provided.

Independently of the regulation by the automatic MCP system, the reactor power is automatically reduced to 60% or 35% in the event of failing one or two MCPs respectively.

Safety-relevant measuring and control systems

These include systems which do not directly influence the nuclear safety of the unit. They include, for example, the turbine protection system and the aggregate protection system, control circuits for feedwater and preheat columns, control systems for slide valves and pumps, measuring systems for monitoring plant components, and signalisation at the control room or control room consoles in the event of exceeding set points.

These plant components are monitored by local displays and remote displays in the control rooms: actuation of interlock is as a rule effected in 2 out of 3 or 2 out of 2 circuits for the turbine protection system or for the preheat columns, and in 1 out of 1 for the aggregate protection system.

By-pass switches are provided for 80% of all criteria. The instrumentation and control devices are, predominantly, equipped with Soviet systems dating back to the 60s. These involve the following:

- the use of large-scale control panel instruments
- the absence of functional group control systems
- the absence of an accident sequence recording system.

The safety-relevant measuring and control systems are for the most part using relay technology, and do not have self-checking and self- monitoring cpabilities.

An independent accident instrumentation system separate from the unit control room is not provided.

Instead of accident sequence recording, reliance may be placed upon the following information:

- Evaluation of recorder strips
- Daily recording using microcomputer systems for the most important unit parameters
- Fast printouts via the processing computer for fuel element outlet temperatures
- Switching protocols for monitoring the control and protection system.

- Safety-relevant assessment

- Protection and Control system:

Compared with pressurized water reactors in the Federal Republic of Germany, the following actuation criteria for RP 1 are not provided:

- DNB ratio
- Pressure high in the primary circuit (RP 3/RP 2 applicable)
- Pressurizer water level high
- Steam generator water level low
- Activity in the main steam lines high
- Pressure gradient in emergency feedwater and main steam lines

There is no temperature/pressure interlock during start-up of the installation.

Some actuation criteria are provided only once in multiply existing components like, for example, main coolant pumps, quick-closing stop valves of the turbo generator; here the logical linkage of the actuation criteria takes place in a 3 out of 6 or 2 out of 4 selection.

Various actuation criteria are realized on the working current principle, e.g. criteria of electrical power supply.

Spatial separation of redundant components is not existing in some cases. This refers to sensors, signal processing and the logical linkage of the actuation criteria.

There are approximately 40 unlocking switches in the reactor protection system. The position of some interlock switches is not signalled. It is monitored and recorded once per shift. The components of the control and protection system can be checked during operation only to a limited extent.

The dropping of any control rod leads automatically to RP 4 (locking of the withdrawal of a control rod). There is no automatic power reduction.

The protected a.c. power supply for the control and protection system is not uninterrupted.

For the NFS, the switchover of measurement ranges is exclusively manual. Accordingly, during cooldown or after shutdown an incorrect measurement range may be selected. Test switches of the NFS are without signalling.

To ensure operation in accordance with the design specification and to control accidents comprehensive administrative measures were established, which are to be supervised and executed by the operating personnel.

In comparison with pressurized water reactors in the Federal Republic of Germany, the plant is equipped with only a few automatic limiting devices. For example, there is no automatic control rod insertion limit to ensure shutdown reactivity by means of control rods; likewise, there is no coolant pressure limitation, and in particular no brittle rupture monitoring system.

Signals from the reactor control system are used both for safety-relevant operating systems and also for operational control and interlock devices.

There is a risk of flooding of transmitters as well as measurement and control (MCR) devices in several compartments, e.g. in the compartment for the emergency cooling and spray pumps.

The accident resistance of the transmitters and of the electrical and the instrumentation and control components has not been demonstrated.

There is no in-core NF instrumentation; limiting values for the fuel element outlet temperatures are displayed in the control room, periodically recorded and monitored by the operating personnel.

- Safety system:

  - Circuitry design in accordance with the working current principle.
  - The actuation of redundant safety systems may fail because of a single failure.
  - The control current circuit for actuation of redundant safety systems is single only, but fully monitored.
  - Priority of protection signals over manual commands is not provided.
  - Multiple use of transmitters and a high degree of interconnection in the control and interlock systems.
  - No spatial separation of redundant components in measurement and control devices.
  - Unsupervised unlocking switches
  - Switchovers of the power supply of main coolant pumps require manual measures to adjust the reactor power.

- Safety-relevant measurement and control systems:

  - The accident instrumentation is inadequate.
  - There is no comprehensive recording system for accident sequences.

7.4.2 Electrical power supply

- Brief description

The nuclear power plant units are operated on the 220 kV/380 kV switchyard Greifswald, which is connected to the national grid of the former GDR via two 380 kV double lines and three 220 kV double lines. Units 1 and 2 feed into the 220 kV grid, and units 3 and 4 into the 380 kV grid. On site, there is no 220 kV/ 380 kV grid coupling.

The generated power is delivered via two turbine generators per unit to the 220 kV or 380 kV grid. Both generators provide the auxiliary power via auxiliary transformers. Backfitted generator circuit breakers permit to supply the auxiliary power also from the switchyard (the protective connection of units 2 to 4 to the system is not yet complete). A fast reduction of reactor power to home load supply is possible only with a low degree of reliability.

The following four 6 kV busbars are associated with each turbo generator set:

- Busbar 1 resp. 5: General auxiliary power supply
- Busbar 2 resp. 6: Emergency diesel power supply
- Busbar 3 resp. 7: General auxiliary power supply and use of the coastdown energy for a main coolant pump
- Busbar 4 resp. 8: Use of the coastdown energy for two main coolant pumps

Busbar 1 (resp. busbar 5) is coupled to busbar 2 (resp. busbar 6) via circuit breakers (normal case). Busbar 4 (resp. busbar 8) is supplied by an auxiliary power supply generator, and coupling with busbar 3 (resp. busbar 7) via a circuit breaker is possible (special case).

Below the 6 kV distributions, there are 380 V main dis tributions and 380 V subdistributions. The two 380 V main distributions for emergency power supply are directly associated with the two 6 kV emergency power busbars, busbar 2 and busbar 6. The 380 V subdistribution boards for emergency power supply can be supplied via automatic switchover systems from the two 380 V main distributions for emergency supply.

If required, the 6 kV distributions (without busbar 1 and busbar 5) can be supplied directly from a twinned reserve grid. The reserve grid is fed for each pair of units via a reserve grid transformer (for units 1 and 2, one respective reserve grid transformer) from the 220 kV grid. The reserve busbars can be switched through via all units, but are normally electrically separated from one another.

When the 6 kV supply via auxiliary power transformers is not available, the system is automatically switched over to the reserve grid. If this also is not available, then the diesels are started.

In the event of a voltage dip on the emergency power distribution, the emergency diesel generator set directly associated with the 6 kV distribution for busbar 2 or busbar 6 is started and switched to the emergency power distribution which has failed. A third emergency diesel generator set, which is not associated with a particular train, is likewise started and switched to one of the two distributions, depending upon failure criteria and preselected setting. Switching into the circuit takes place in the unexcited condition at frequency differences of less than 1 Hz.

The required switchover operations for the three diesel generator sets are implemented by a twinned control system. The consumers are isolated in a controlled manner by the voltage dip monitoring systems of each distribution board, should the voltage drop.

In emergency power mode (activation of the emergency power distribution), the reconnection of the emergency power consumers takes place by means of a loading sequence program, which is common to both trains and which is twinned. This program controls the time dependent connection of selected emergency power consumers and blocks automatic ON commands which are not included in the program.

The program control is terminated either by opening 2 out of 3 diesel generator circuit breakers or by switching in a reserve feed system or an auxiliary coupling.

Safety-relevant emergency power consumers are connected to various busbars. As a rule, redundant consumers are not simultaneously activated. If required, the preselected consumer is activated. Only if the preselected consumer fails the reserve consumer is activated together with the following reload group.

The system for uninterrupted power supply is formed from two 380 V main distribution systems (safety main distribution 1, safety main distribution 2) and from a series of subdistributions which are structured on a non-redundant basis. The subdistribution boards have feed facilities from both safety main distributions. Furthermore, this system includes a unit-related 220 V d.c. main distribution. Additionally, for two nuclear power station units a further 220 V d.c. main distribution is available for meeting the general internal requirement for two nuclear power station units. In the normal switching condition, the safety main distributions are fed via thyristor power switches from the 380 V diesel main distributions. Rotary converters are connected to the safety main distributions, and these produce the direct current for the consumers at the d.c. main distributions and at the same time keep the 220 V batteries charged as required. Among the batteries there are, similar to the distributions, a unit-related battery and a battery common to two nuclear power station units. In the time interval between voltage failure at emergency power distributions and the restoration of voltage, the rotary converters change over to inverter operation and supply the safety main distributions, fed by the batteries.

The most essential functions of the system for uninterrupted power supply are the provision of the control voltages, the measurement voltages and the signalling voltages, the supply of valves, which must be available without interruption and the provision of voltage for automatic interlock mechanisms which are essential to safety.

The plants are equipped with main coolant pumps without flywheel mass. When the emergency power mode is activated, the coastdown energy of the turbo generator sets is converted into electrical drive energy for the main coolant pumps.

For this purpose, each unit is associated to two auxiliary power generators which are driven by the turbines and supply two main coolant pumps each. The remaining two main coolant pumps also participate in the coastdown via the unit auxiliary power transformers. In this case, all consumers at the respective distribution which are not required, are switched-off by a special automatic system.

One pair of main coolant pumps is supplied from busbar 4 and busbar 8 each, and one pump from busbar 3 and busbar 7 each. The connection to busbar 3 and busbar 7 is made in the form of a trouser leg arrangement, i.e. the two pumps can be supplied both by busbar 3 and by busbar 7.

Safety-related assessment

In units 1 and 2, main and reserve grid connections are supplied from the same grid (220 kV voltage level). There is no possibility of grid-side supply from a different junction.

In units 3 and 4, the main grid connections are supplied from the 380 kV voltage level and the reserve grid connection from the 220 kV voltage level. The expected frequency of an emergency power mode is estimated at 0.1 per year. The number of units at the site has only a slight influence on this frequency.

The auxiliary power supply installations are allocated to the turbo generator sets. Functionally, the emergency power systems are to a large extent twinned; there is no complete redundancy and the trains are interconnected. In the region of the reserve grid, the 380 V reserve auxiliary power supply, the 220 V d.c. distribution, which is supplying two units, and alos the cooling water supply for unit and reserve transformers, there are interconnections between the two units of a double unit. Even within the emergency power generation systems, there are interconnections between the units, e.g. in the d.c. supply, the compartment ventilation and the cooling water supply.

Of the three diesel emergency generator sets of each unit, presently at least 2 sets must start on demand. This condition is unsatisfactory. Furthermore, because of the interconnection of the two 6 kV distributions by the diesel emergency power generating set which is not associated with a particular train, it cannot be ruled out that single failures in this region may lead to the failure of both 6 kV distributions.

Spatial separation of cables of redundant systems is not provided. The main distribution boards are spatially separated on the basis of their train allocation.

At the present time, realistic power balance statements are not available. An assessment of the design specification in terms of power and of the degree of redundancy, especially as regards the diesel emergency power supply, is accordingly possible only to a limited extent.

Under specific boundary conditions, the automatic cut-in of consumers which are significant in terms of safety can be blocked by the diesel loading sequence programme. Manual measures are not affected by the loading programme. The loading sequence is blocked, after excitation of the program, until the mentioned criteria are fulfilled. In the event of reconnection, there may be an overload of a diesel emergency power generator set as a result of commands stored in the automatic system. The reconnection must take place via an isolation of the 6 kV main distribution, since there is no synchronisation device. Provision is made for a live examination of the loading sequence programme only on startup of the unit.

The automatic diesel starting devices are excited only by low voltage at the 6 kV emergency power distributions. There is no frequency monitoring. Both automatic diesel starting devices are independent of one another from the sensors to the actuators. On deactivation of the main grid and the turbo generator set, the feed to the associated auxiliary power busbar is deactivated by an interlock mechanism. By this means, operation of the emergency power busbar with low frequency due to the coastdown of the turbo generator set is prevented.

The testing of important automatic devices and systems is not possible, or is possible only with redundancy restrictions, when the plant is in power operation.

In the event of a single failure at busbar 4 or busbar 8 (e.g. busbar short circuit), three main coolant pumps may fail simultaneously. On operating the plant with only one auxiliary power generator, four main coolant pumps may fail simultaneously. It is possible to exclude the failure of five or six main coolant pumps caused by an abnormal condition within the electrical energy supply system.

7.5 Ergonomic aspects

7.5.1 Introduction

As shown by the discussion of the individual technical systems, on account of the existing plant concept, the operators play a substantial part in managing the plant both in normal operation and in the event of malfunctions or accidents.

The ergonomic design of the parameters which are significant in terms of human reliability often determines whether critical situations result in mistakes and errors.

A selection of essential ergonomic parameters is assessed below. More extensive investigations, especially the development of behavioural models and task analyses for manual measures could not be performed up to now.

7.5.2 Aids to diagnosing accidents

From the point of view of ergonomics, the aids made available to diagnose an abnormal condition exhibit a series of short-comings complicating the diagnostic function and the selection of required accident instructions. This may result in incorrect diagnoses or lost time.

The attention of the operating personnel is drawn, both visually and acoustically, to the response of signal limiting values and the onset of an abnormal condition (direction of attention by horn, flashing lights, change in measured value indicators and the like). As far as the interpretation of the signals presented is concerned, the operators to a large extent dependent on their expert knowledge.

If the operators have selected a particular accident operational procedure based on their initial subjective assessment of the situation, then it is necessary, prior to the implementation of this procedure, to monitor the initial symptoms applicable to this procedure (alteration of process quantities or events). The presentation of the symptoms which is employed in this case does not comply with ergonomic design principles in many respects. There are no logical links, the descriptions of symptoms are not unambiguous and instruments to be monitored are not adequately designated. The operational specifications contain references to symptoms which can be observed only with a time delay, or cannot be observed in the control room or cannot be observed in the event of the basic abnormal condition concerned.

7.5.3 Operating specifications for accidents

In many respects, the structure of the operating specifications is not in compliance with ergonomic requirements, so that in accident situations the likelihood for errors of omission and of execution is increased.

The accident instructions are arranged in the form of lists and are divided into the sections "recognition features", "measures implemented automatically" and "measures to be initiated manually". Ergonomic aids such as for example structuring, emphasis or standard presentation, are not used. The procedures merge undetectably in one another. There is no instruction identification pattern, nor is there a marking scheme for instructions which have been implemented. There is no adequate prevention of possible errors in the procedural sequence. Instructions which apply only subject to certain conditions are frequently difficult to identify as such. Warnings are inconveniently located and in some cases even include additional instructions.

The stress to which the operators are exposed after the onset of an accident leads, especially in the area of knowledge-based action, to a marked increase in frequency of error. Likewise, there is distortion of the perception of time and time-dependent quantities. The procedural instructions do not take account of these interrelationships. The degree of detail of the instructions is generally low. Instructions which are to be implemented in the event of failure of individual system functions are not adequately presented. The identification of systems is inadequate and in some cases not standardized. Missing informations on procedures have to be supplemented from memory. This may result in omission or confusion of the sequence of tasks. For monitoring functions, the operators are required to make a subjective estimate of the time behaviour of process parameters (e.g. if the process parameter varies slowly, rapidly, frequently, greatly etc., then ...).

7.5.4 Qualification and training of the operating personnel

Each of the four control rooms is staffed by a unit operator, a reactor operator, an instrumentation and control board operator and an electrical operator. Deviations from this minimum staffing level are permissible only on a short-term basis. The duty engineer responsible for all four units may be for a short period at the control room of the accident unit (according to information from operators, as a rule for a period not exceeding 10 minutes). Furthermore, there is constant staffing of the equipment control room, the electrical control room, the dispatcher station, the dosimetry control room, the intake building and the discharge building.

A large portion of the measures which are required within the first phase after the onset of an accident must be monitored by the group of persons present in the control room, and must be initiated or implemented by that group of persons. In this procedure, a central role falls upon the duty engineer, the unit operator and the reactor operator. They are expected to be able to carry out the required diagnoses and switching operations on the basis of their acquired expert knowledge, even without written aids. A consequence of this concept is that an ergonomically advantageous arrangement of these aids has not been implemented or has been implemented only to an inadequate extent (cf. also 7.5.2 and 7.5.3).

This concept of largely trusting the expert knowledge of the operators is satisfactory for situations with which the operators are fully familiar. The supposition that this might not be the case in accident situations is indicated, inter alia, as follows:

- The measures regarding qualification are not sufficient to deal with the entire range of accidents in sufficiently short time intervals with the entire operating personnel, and thus to keep the level of knowledge high above average.

- The simulator for practicing is not regarded as an accurate representation.

- An extremely high degree of familiarity with the plant and a permanent detailed knowledge even of infrequent processes can be built up only over a period of many years. Because of changes in personnel, it must be assumed that a considerable proportion of the operating personnel does not yet satisfy this requirement.

- The level of training and thus the initial qualification of the operating personnel is not to be classified as above average. The plant management itself carries out the examinations.

7.5.5 Design of the unit control room

The design of the unit control room corresponds to the state of technology of the 60s. Reorganisation to take account of modern instrumentation and control technology having regard to ergonomic aspects has not taken place. In many respects, the design of the control room is not in agreement with current ergonomic requirements. The question of whether a partial modification of the control room is advisable or not should however be carefully examined in relation to each individual problem with respect to the long experience gained by the personnel with the present arrangement. In this case, an increased frequency of error would have to be expected at least over a relatively long relearning phase. An impression of the problems may be gained from the following examples:

- The selected movement codes (rotating the switch to the left corresponds to closing the armature) and colour codes (e.g. red for valves open), for equipment do not take account of the generally known stereotypes.

- A considerable portion of the instruments incorporated in the wall panels can be read only with difficulty or not at all from the workplaces of the operators. For example, the view of important displays is blocked by the consoles, reflected light interferes with reading the emergency alarm fields, continous-line recorders are placed above head height and can be read only under difficult conditions.

- The arrangement of the actuating switches on the control console has already led in the past to cases of confusion. The colour marking of the known "problem cases" or the covering of these switches with easily removable rings is inadequate. In addition to an improved arrangement of the switches, the equipment feedback communications should also be integrated into the console. At the present time these equipment feedback communications are displayed on the wall panels, so that the operators cannot simultaneously observe the feedback communications and the switches.

- Measures which counteract a possible surge of information and its consequences (e.g. hierarchical alarms, visual/acoustic coding, position coding) are provided only to an inadequate extent.

- Safety-related measurement and limiting values within a system can be read only from multiply utilized display devices. Thus, for example, approximately 30 process quantities within the reactor coolant lines can be read using only two instruments. In the case of the digital display unit, the value is displayed only in the form of a sequence of digits. The unit and the position of the decimal point must be added from memory.

- The interlock switches are inadequately protected against erroneous operation or erroneous setting.

- The lettering of the devices on consoles and wall panels is inadequate. German and Russian codings are used in parallel; however, both systems are not used in the same way by the operators. In some cases designations are absent or are illegible or handwritten on adhesive tape or fitted in such a manner that they could also be applicable to the adjacent operating system.

- The telephone and the intercom are available for communication systems. The intercom causes acoustic disturbances in the control room. As a result of this the attention of the control room personnel is distracted, especially in accident situations.

7.6 Recommended backfitting measures

This section refers to system-related improvements, a distinction being drawn between measures to be implemented in the short term and those which are to be implemented as a supplement to the intended backfitting ("16-point programme"). The short-term measures are considered to be necessary as a supplement to the "35-point programme" set up by the SAAS, in order to permit further operation of units 1 to 4 on a for a limited time.

7.6.1 Reactor shutdown system

In the short term:

- Automatic initiation of the reactor protection system by the criterion "steam generator water level low"

- Monitoring and indication of position of all interlock switches

In the longer term:

- Installation of a quick-acting boron poisoning system

7.6.2 Emergency core cooling system

In the short term:

- Replacement of the EP 50 pumps by ZN 65 pumps (depending on the analyses for the leak of ND 32 with one available pump and for the leak of ND 200 with four available pumps; the replacement may be required only in the longer term).

- Emergency power supply and use of the reactor filling pump P 36 and of the pump P 17 of the spent fuel pool for low-pressure long-term feed. To this end, the delivery capacities should be determined by experiments.

- Heating of the emergency boration tank must be independent of the emergency core cooling system.

- Assurance of the environmental conditions according to the design specification for the emergency core cooling pumps and the spray pumps (insulation of pipes carrying heated emergency core cooling water)

- Relinquishment of closed motor valves, which must open in emergency situations. In this connection, it must be ensured that safety with regard to leaks from the primary coolant loop is not impaired.

- Automatic activation of the reserve pump or of the motor valve in the third spray line in the event of failure of one pump or valve.

- Establishment of a cross connection to the emergency core cooling system of the adjacent unit (within one double unit) and of a spatially separate emergency cooling water return pump. The connection should be installed on the pressure side of the pump and should contain a lock valve.

- Provision of a building drainage pump for the emergency core cooling pump chamber, with location of the leak with regard to origin of the leak water (service water or emergency cooling water).

- Prevention of obstruction of the return line from the sump by improving the sump cover.

In the longer term:

- Wrapping of the injection lines to the hot legs (depending upon the analyses concerning loadings of the reactor pressure vessel by "cold strands", possibly even in the short term)

- Improvement of the capacity, of the degree of redundancy and of the spatial separation of the emergency core cooling and spray devices (accumulators, low-pressure and high-pressure injection systems).

7.6.3 Emergency feedwater system/main steam system

In the short term:

- A lockable cross connection between the pressure lines of the emergency feed water systems of units 1 to 4, with additional hose connections

- Possibility of injection into the steam generator drain lines via a line spatially separated from the 14.7 m platform, likewise with additional hose connections

- Pump for demineralised water with independent drive for emergency feed to the steam generators from the demineralised water tanks

- Incorporation of further water reserves such as district heating transport line, drinking water, water for fire-fighting.

In the longer term:

- Improvement of the capacity, of the degree of redundancy and of the spatial separation of the emergency feedwater and steam release system as well as of the required auxiliary systems.

7.6.4 Service water system/component cooling water system

In the short term:

- A lockable cross connection on the pressure side of the service water systems of both double units

- Building drainage pump for the intake system with automatic activation if water is present.

In the longer term:

- Improvement of the degree of redundancy and of the spatial separation of service water system and component cooling water system.

7.6.5 Overpressure valves and isolation of the ventilation lines of the confinement

In the short term:

- Reduction of the number of air exhaust valves which are open during power operation

- Transmitter compartments, in which the accident instrumentation required in 7.6.6 is installed, are to be protected by additional quick-closing valves (double isolation) to prevent ingression of steam.

In the longer term:

- Double isolation of all ventilation lines by quick-closing dampers

- Mounting of protective gratings to prevent foreign substances to enter the closing valves.

7.6.6 Instrumentation and control

In the short term:

- Single-failureproof activation of redundant safety systems

- Separate construction and monitoring of the control current circuits of redundant safety systems

- Monitoring and display of the position of all interlock switches of the safety systems and of selected important interlocks (e.g. injection pumps via pressurizer level)

- Installation of a power density distribution monitoring system with the use of in-core instrumentation (signalling DNB)

- Establishment of an accident sequence recording system

- Establishment of an autonomous accident instrumentation system in the transmitter compartments spatially separated from the unit control room, which records at least primary pressure, primary temperatures, pressurizer level and steam generator level

- Interlock to ensure the maintenance of a safety margin to brittle fracture by pressure relief at temperatures below NDT temperature

- It has to be checked whether the different set points of the criterion "pressurizer level low" for reactor scram and activation of the emergency core cooling pumps are sensible.

- It has to be checked whether the different set points of the criterion "confinement pressure high" for reactor scram and activation of the spray pumps are sensible.

- Controlled reduction of reactor power on failure of one or both turbo generator sets (grid failure) to the appropriate power level, if possible without response by safety systems (reactor shutdown, emergency core cooling)

- It has to be checked whether the administrative measures for ensuring an adequate number of rundown-protected reactor coolant pumps ("table of permissible reactor power") are to be replaced by electrical interlock mechanisms, but at least the occurrence of unacceptable conditions is to be signalled.

- Improvement of leak detection and leak location

- Installation of further water alarms in the emergency core cooling pump compartment and in the intake structure of the service water system

- The section between the first and second lock valve of connecting lines to the primary coolant loop which are equipped with motor valves is to be monitored for leaks of the first lock valve.

- Fire protection measures (cf. chapter 8)

In the longer term:

- Fundamental renewal of the instrumentation and control system, observing current regulations and guidelines.

7.6.7 Electrical systems

In the short term:

- At least two-train design of the uninterrupted emergency power supply and extensive elimination of the interconnection between the units

- Replacement of components which are susceptible to failure or which are costly in terms of maintenance (use of static inverters, charging rectifiers)

- Improvement of the spatial separation of important safety-related redundant cables or replacement measures (cf. chapter 8)

- Elimination of the couplings between busbar 3 and 4 as well as between busbar 7 and busbar 8, and fixed allocation of main coolant pump 3 to busbar 3 and main coolant pump 2 to busbar 7

- Establishment of realistic power balances for the emergency power systems and, where appropriate, derivation of the required measures (e.g. fourth diesel, increase in the battery capacity)

- Checking, and possible improvement, of the loading sequence programme for emergency power consumers with respect to the priority of manual commands, the blocking of automatic measures originating from consumers which are important from the safety point of view during diesel operation, and the conditions regarding termination of diesel operation

- Reduction of the interconnections of the emergency power supply distributions at the 380 V voltage level

- Improvement of diesel cooling by the provision of an independent cooling system

- Conversion of the cooling of the reserve transformers to air cooling.

In the longer term:

- Establishment of a battery charging circuit monitoring system

- Improvement of earth contact detection and earth contact localisation

- Improvement of the battery compartment ventilation to reduce the compartment temperature

- Reduction of the component protection criteria of the emergency power diesel aggregates which lead to shutdown of the aggregate on demand

- Backfitting of synchronisation devices for the diesel generator circuit breakers

- Matching of the auxiliary systems of the diesel generators to the redundancy and circuitry of the main aggregates

- Examination of possibilities for improvement in the grid-side supply with regard to functional and spatial independence (connection with the national grid, main and reserve grid connections, independent grid power supply in emergency power mode, coupling between the 380 kV and 220 kV grids)

- Improvement of the capacity, of the degree of redundancy and of the spatial separation of the emergency power supply system (in accordance with the "16-point programme").

7.6.8 Administrative measures

In the short term:

- Introduction of a 5th shift, exclusively for qualification and training measures

- Conduct of the expert knowledge tests for the control room personnel by experts independent of the utility

- Permanent presence of an experienced highly qualified safety engineer at the site, to assist the shift personnel in planned situations

- Establishment of an emergency call system for experts and provision of appropriate alarm systems

- Revision of the accident instruction manual, having regard to ergonomic design principles, with the objective of rapid availability of clear practical aids to assist the control room and the inclusion or clarification of instructions regarding specified events, e.g. detection of and behaviour on the occurrence of leaks or impending grid breakdowns

- The labelling of wall and console panels in the control room is to be standardized. For this purpose, use is to be made of a standardized systems identification system. Labels should be readily legible and should be capable of unambiguous cor-relation with the associated component symbols. In this connection, it should not be overlooked that all components and lines should be designated in situ on a standardized basis and these designations should be readily legible

- Regulation of the mode of operation of the crane installations in the reactor hall, in the turbine building and over the intake structure

- Formulation of instructions concerning the management of beyond design basis accidents e.g.

    - Steam generator leak with failure of the primary-side isolation of the defective steam generator

- Failure of emergency core cooling system in the case of loss-of-coolant accidents or of their auxiliary systems, such as the service water system and component cooling water system

- Leaks in connecting lines of the primary loop, which are situated outside the confinement

- Failure of the secondary-side heat removal

- The monitoring of the manual block valve settings in connecting lines from low- pressure systems to the primary loop (in accordance with section 7.3.6) is to be designed on an objectively safe basis (e.g. by means of a key system)

In the longer term:

- Consistent application of ergonomic aspects for the backfitting of the instrumentation and control system (cf. section 7.6.5)

- Matching of the existing training simulator to the specific conditions of the W-230 reactor, especially for the purposes of accident management training.

8. OVERLAPPING EVENTS

Overlapping events are understood as those events which may affect large areas of the plant in a redundancy and system overlapping way. Such events lead either to the mechanical or thermal load of structures, components and systems or to flooding of areas of the plant. Furthermore, a distinction is made between system-internal events due to fire, flood, failure of components in the turbine building (pressure vessels, turbines) and external events due to earthquake, external flood, lightnings, aircraft crash and explosion shock waves. For the evaluation of those overlapping events, their occurrence frequency to be expected under specific plant or site conditions is an essential feature; for the present assessment no quantitative data were available in general concerning the occurrence of events.

Since the effects of individual events may differ, the events are differentiated below. Actions by third parties (terrorism, sabotage) are not investigated.

8.1 Assessment of the current layout

8.1.1 Plant-internal overlapping events

8.1.1.1 Fire

The fire protection system for the Greifswald nuclear power station was planned in accordance with standards applicable in the 60s. From the point of view of current technology, there are three essential weak points:

- Absence of separation, effective for fire protection purposes, of redundant safety-related systems, including cable connections, in almost all areas of the plant apart from the diesel emergency power supply building.

- Absence of passive fire protection measures in the control rooms, including the associated cable distribution arrangements and electrical systems, from which all safety relevant functions proceed. The absence of emergency control stations increases the significance of this weak point to a considerable extent.

- The arrangement of safety-relevant systems (e.g. feedwater and emergency feedwater supply) in the turbine building in close proximity to significant fire hazards (lubricating oil). The turbine building for 8 units of the nuclear power station is built as one fire area, which is approximately 1000 m long and in which all turbine/generator sets are arranged in the longitudinal direction.

Separation, for fire protection purposes, of safety-related systems and of cable connections from one another and protection from effects due to fire resulting from significant fire hazards is missing. Moreover, there are deviations from the currently applicable standards of the Federal Republic of Germany (DIN, KTA 2101) and of the GDR (TGL). These deviations are demonstrated to be weak points in the following presentation of the fire protection measures implemented.

- Structural fire protection measures

There is no consistent separation of the power station units by fire walls, or of the individual buildings from one another, and no subdivision of the buildings into fire areas or fire-resistant areas. In particular, there is no subdivision of the turbine building into fire areas. Where massive concrete structures form the boundary of buildings or compartments, it is possible to admit internationally conventional fire resistances for the separation.

In the region of doors, cable and pipe penetrations as well as ventilation channels do not fulfil the fire resistance requirements for barriers of fire protection purposes in the Federal Republic of Germany and in the GDR as they are prescribed due to the current state of knowledge. Consequently, there are no correctly designed measures to prevent the propagation of fire from one compartment to another. This weakness, however, is in many cases of secondary importance, since even within the fire area failures are possible with a significant effect on redundancy or system operation. Thus a fire propagation to other compartments does not intensify the system-related consequences to a significant extent.

Structural fire protection measures which go beyond the measures planned in the design have been backfitted only sporadically, (e.g. laying of cables in sand beds).

- Technical fire protection measures

Large areas of the plant, with some exceptions (e.g. some compartments in the reactor building, frontage and turbine building), are monitored by fire detection systems with automatic fire alarms. In addition to these automatic fire detection systems, it is possible to raise the alarm manually and by telephone. Essential areas containing safety-related systems in which automatic fire detection systems are missing include, for example, emergency core cooling systems, 0.4 and 6 kV switching systems, the feedwater area and the service water system. The density of detectors in the monitored areas appears adequate.

Automatic fire-fighting systems are provided only in very few areas of the plant. These are the solder-releasing CO_2 installations in the diesel emergency power supply building and the fire-fighting systems of the high oil tanks in the turbine building and of the unit-connected transformers. On the other hand, fire-fighting systems with deluge sprinkler systems without automatic activation are used in cable systems and in specific transformers. These systems are manually activated by the fire brigade on site, since false alarms cannot be ruled out and secondary damage is possible. The manual activation of deluge sprinkler systems is also customary in the Federal Republic of Germany.

The water supply for fire-fighting purposes is provided by means of an annular conduit system, which at the same time serves as a drinking water, utility water and fire-fighting water network.

Water can be taken from this network via above-ground hydrants outside and in buildings via wet risers and wall hydrants.

The supply to the annular conduit is effected via three feed lines from the public water distribution, each of them equipped with a water tank, which is operated as a throughflow tank. Should an emergency occur in the public distribution, this stock of water will be sufficient for approximately one hour. The pressure in the annular conduit is maintained by six fire-fighting water pumps, which, in the event of emergency power being required in more than one unit, can also be switched to busbars supplied with power from the emergency supply. Their power requirement, however, has not been taken into account in the current capacity power level of the diesel emergency generator set. Furthermore, it remains to be checked whether as a result of fire an emergency power situation overlapping more than one unit can be initiated and whether a complete failure of the emergency power supply to the fire-fighting water pumps is possible in a unit-related situation involving the emergency power supply on account of the switching of this power supply to the internal supply busbars of the units.

At the present time, it is unclear whether the pressure of the fire-fighting water network is sufficient to supply the above mentioned areas of the buildings.

- Operational fire protection measures

The concept of fire-fighting in the installation is essentially based on the plant fire brigade and relies upon their quick response. The plant fire brigade of the nuclear power station is specially trained for fire-fighting within the installation. Their number and their equipment appear to be adequate. Fast response has been demonstrated in many reviews. Consideration should be given to the idea that in using the plant fire brigade the monitoring for radiation protection purposes within the active area takes place exclusively by the operating personnel and the plant fire brigade has not been issued with their own radiation measuring instruments and warning systems.

In order to fight incipient fires, the operating personnel have available an adequate number of hand extinguishers at appropriate places. The training of the operating personnel in dealing with hand extinguishers should be improved.

Periodic tests of the fire protection systems take place regularly, in some cases during the operation of the nuclear power plant and, where this is not possible, once annually as part of the major maintenance programme.

- Safety related assessment

The individual requirements applicable in accordance with KTA rules, especially with regard to the priority of structural measures and the separation of redundancies from safety devices, cannot be a decisive yardstick for assessing fire protection as applied to installations.

In this case, as in the case of older installations in the Federal Republic of Germany and in other Western countries, it is necessary to undertake an assessment based on protection objectives, since the structural conditions do not allow a consistent application of the KTA requirements, for example with regard to spatial separation. The absence of structural measures is, in the case of older installations in the Federal Republic of Germany and in other Western countries, compensated by

- surface monitoring of all relevant areas of the installation using automatic fire detection systems, which guarantee a fire alarm in the incipient phase of the fire,

- the intensified utilisation of fix fire-fighting systems to fight the fire at an early stage (water spray, CO_2, halon), where possible with semi-automatic or automatic activation,

- separate emergency systems which are isolated for fire protection purposes, including an emergency control room, which come into action in the event of redundancy overlapping failures of safety systems within the installation caused by fire.

Comparable measures in units 1 to 4 are provided only to a certain extent. Accordingly, it cannot be ruled out that in consequence of a fire accidents may be initiated in the sequence of which the removal of afterheat is jeopardized. The failsafe function of the scram system limits any threat to the scram function. A matter of importance concerns transients due to redundancy overlapping fires, with failure of the feedwater supply including emergency feedwater, the supply of service water and the intermediate coolant circuit. In the event of a fire in cable distribution systems of the power supply system or in the area of switching systems, it is furthermore possible that system overlapping failures occur and cause the entire failure of the control room or of partial functions of the control room (instrumentation, drive).

Where such failures lead to transients without primary-side leaks or without secondary-side pressure relief and the turbine building is not affected comprehensively, it is possible, by reason of the large feedwater contents of the steam generators, to control these transients by accident management measures, for example the support of the emergency feedwater supply via other units using existing pipe connections (time window up to 6 h).

In consequence of non-closure of safety valves after prior opening caused by the transient, primary-side leaks may occur. In this case, accident management measures are successful only where the emergency cooling systems are not substantially impaired (cf. accident 1975).

In the case of transients due to fire with secondary-side pressure relief (e.g. on account of failure of main steam or feedwater lines due to fire damage in the turbine building), there are only slight possibilities for accident management measures.

In general, in the event of a fire in the installation it is possible for sequences of events to occur in which the removal of afterheat is no longer ensured. Accordingly, it is necessary within the short term to implement appropriate improvement measures. Furthermore, a comprehensive backfitting of the fire protection systems is required.

8.1.1.2 Floods

Areas in which there is the possibility of flooding affecting safety include the building of the cooling water and service water pumps, the turbine building, the area of the emergency cooling pumps, the reactor building and the diesel emergency power supply building. Since the effects are not redundancy overlapping within the diesel emergency power supply building, flooding of the latter is not considered any further at this point.

- Building of the cooling water and service water pumps

In this building, 4 reactor coolant pumps and 5 service water pumps are arranged in one chamber for two reactor units.

The associated drives are situated on the next higher level of the building, which is however connected spatially to the lower region. Leaks due, for example, to corrosion and erosion damage, defects in maintenance and assembly work and the dropping of loads can be detected by means of various displays, such as pressure drop, reduction of throughput, penetration by water and sump pump activity, by the operating teams.

In the case of large leaks, it cannot be assumed that there will be timely isolation, so that flooding of the drives is to be expected. At the latest at flood levels of approximately 1.15 m below the water line of the intake channel, failure of the pump drives is to be expected. Accordingly, larger, not lockable leaks in the intake structure lead to failure of the operation of the service water system, which is important from the safety point of view, of the cooling water system and of the entire heat removal system.

- Turbine building

In this building there are at height level -2.1 m (zero height being equivalent to ground level) two pumps of the emergency feedwater system and 5 pumps of the main feedwater system set up alongside one another, in blocks in each instance, together with associated electrical supply systems. Flooding, which may lead to the failure of these pumps, is only possible if a main cooling water line fails and the associated pump is not shut down at the appropriate time.

Large leaks in a cooling water line, e.g. in consequence of internal events such as turbine or tank failures or on account of falling loads, are detectable by means of system failure displays in the turbine and condenser regions.

On shutdown of the affected cooling water pump within approximately 20 min, there should be no flooding of systems which are important with regard to safety.

Since feedwater pumps and emergency feedwater pumps are not spatially separated, a pipe failure in the feedwater system can lead to the failure of the emergency feedwater system. Impacting lines and jet forces, as well as the action of water and steam on electrical systems, may in this case lead to consequential failure.

- Area of the emergency cooling pumps

6 emergency cooling pumps also needed for boration, 3 spray water pumps and 2 spray water coolers are set up in this area. Leaks are detectable with reference to the switching frequency of the sump pump and by means of a filling level probe.

At a water level of 45 cm, flooding of the chamber leads to the loss of operation of the emergency boration pumps and at 57 cm to the failure of the spray water pumps.

Flooding is possible by:

- the non-lockable region of the output pipe of the service water system
- the emergency boration tank together with the pump suction lines
- the fuel tank filling line.

In the event of relatively large leaks, the failure of the service water output line which is constantly subjected to the system pressure leads to failure of the pumps as a consequence of flooding within approximately 10 min. The running down of the emergency boration tank (water content approximately 800 - 900 m^3) or a failure of the filling line of the fuel element storage pond (water content approximately 300 m^3) likewise lead to the operational failure of the pumps.

On account of the relatively small flood capacity of at most approximately 180 m^3 before failure and the small transport capacity of the sump pump, there is only a little time (approximately 10 min) for countermeasures where medium-sized and relatively large leaks are concerned.

- Reactor building

Cold water leaks in the chamber region below the steam generators, main coolant lines and pumps are collected and led away in a controlled manner. In the case of leaks which are below the exhaust pumping capacity of the drainage pumps, penetration by this water into the reactor well provided with overflow gutters and the inlet to the emergency boration tank can be prevented.

In the event of failure of cold flow lines, such as the fuel tank filling line, there may be flooding of the overflow gutters. For the reactor well, the result of this would be that in the case of leaks above the capacity of the well pump of 16 m^3/h external wetting of the hot reactor pressure vessel with cold water would take place. This results in an intensive cooling of the reactor pressure vessel wall and formation of steam with pressure build up in the pressure resistant compartment system.

Leaks in the adjoining valve compartment may lead to the flooding of valves which are important from the point of view of safety, but these leaks have no effect, since the drives are located outside the hazard area by the use of drive rods. With reference to the sump pump activity and the penetration of water within the compartment region, the control room team can detect the leaks and then initiate effective isolation measures.

In the event of water from operational systems penetrating within the region of the annular water tank and ventilation system cooler, a failure of the reactor well sump pump may also lead, in the event of failure to take countermeasures, to the described effects within the reactor well. The possibilities of formation of leaks in this region which may lead to wetting of the reactor pressure vessel and the consequential effects are to be subjected to a more detailed investigation.

- Safety-related assessment

In the case of larger leaks, especially in the intake structure, in the machine hall and in the emergency boration unit, the failure of safety systems cannot be ruled out.

Since even greater leaks can be assumed*, backfitting measures for the abovementioned areas are to be regarded as necessary as part of the reconstruction. The measures which are necessary in the short term will be discussed in section 8.3.

8.1.1.3 Other internal events

Other internal events - fragments due to turbine failures and shock waves due to bursting of tanks - were not taken into consideration in the design of the nuclear power station, since their probability of beginning is estimated as low. This assumption is to be checked in further investigations.

Likewise, no consideration is given in the design plan to accidents affecting a number of units. Their effects were investigated, and measures for reconstruction were inferred (reserve control room or fire protection measures within the turbine building).

8.1.2 External events

As regards the location of the Greifswald nuclear power station, the expert opinions required in the GDR for approval of the site are available (e.g. the hydrological, hydrogeological and seismic expert opinion). With re-

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* according to the German Risk Study for nuclear power stations phase B, the probability of a large cooling water leak is $p=5 \cdot 10^{-3}/a$ with the main proportion being accounted for by incorrect operation in the course of maintenance.

ference to the example of the intermediate store for spent fuel, the effects of external influences were investigated. The result of these investigations shows that the structures withstand the external influences specific to the location (with the exception of impact by aircraft). These investigations were carried out for

- the 10,000 year earthquake (MSK-5),
- the 10,000 year wind,
- shock waves possible at the site due to explosions in the area of the hydrogen store.

In the case of earthquake intensities up to MSK-5, there is no expectation of danger to the installation (even without special design).

Explosion shock waves derived from incident locations outside the installation can be ruled out, on account of the site conditions.

As far as these installation areas are concerned, the arrangement of the reactor buildings and of the turbine building prevents any danger due to external flooding. However, significant safety-related service water pumps within the pump building may be endangered by extreme levels of flooding. There are at present no findings available concerning the frequency of such water levels.

The lightning protection systems satisfy the statutory regulations of the GDR. Operational experience shows that there has to date been no lightning-induced failure of safety related systems. With more widespread utilisation of electronic system components in the future, it is necessary to improve lightning protection.

The probability of occurrence of an aircraft crash on an installation has been reduced by a prohibition on over-flying (2 km radius, 2 km altitude).

Further investigations concerning external effects are not considered necessary within this safety assessment, by reason of the conditions applicable to the site and probabilities of occurrence.

8.2 Fire protection measures required in the short term

In order to ensure fire alarm surface coverage, a consistent fire restrain system and rapid firefighting at all safety-related installations, the following additional fire protection measures are necessary:

8.2.1. Early fire detection

In order to construct a surface covering automatic fire detection system in all compartments containing fire loads, backfitting in the following areas of the installation is necessary:

- Electrical engineering operating compartments (E 103, E 105, E 107, E 108, E 113, E 114),
- Cable shafts and cabling lines
  (General internal main distribution bus, general internal DC main distribution bus, series of columns "B" at -3.60 m, B003, A008),
- Cabling lines within the turbine building,
- Oil supply areas of the turbo generator sets,
- Oil supply areas of the feedwater pumps.

In addition, manual initiation is to be provided for all areas.

The need for the use of hydrogen detectors in the turbine building and a pressure drop interlock in the hydrogen system is to be examined.

8.2.2 Passive fire protection

In order to prevent the propagation of fire within electrical operating compartments, the non-fire-resistant steel dampers currently fitted are to be replaced by fire dampers.

Fireproof bulkheads in cable shafts do not have the required fire resistance 90 and are therefore to be backfitted by means of appropriate measures.

In all other areas, the fireproof bulkheads are to be subjected to an assessment. Where the fire resistance 90 is not achieved, measures to increase the fire resistance are to be implemented. Particular attention is to be paid to the bulkheads beneath switching systems and in the cable gallery.

The cabling line along the series of columns "B" at level -3.6 m within the turbine building is to be protected against oil-induced fire and the action of steam in sections endangered by fire and steam (at least in the area of the feedwater platforms and the high-pressure heaters). The flanges of the oil pressure lines within the feedwater region are to be secured by caps, so that in the event of oil leaks in these regions the uncontrolled spraying of oil will be prevented.

In order to restrict the consequential effects of oil fires in the turbine building, facilities for drawing off smoke and heat and protective measures required in connection therewith are to be provided in the head truss region.

In order to reduce the probability of an oil fire in the turbine building, measures are to be implemented which prevent penetration by leaking oil from the turbine shaft bearings into isolations and into the cable channel below the turbine.

The protection of the safety-relevant installations (main steam and feedwater lines, emergency feedwater pumps) in the turbine building as well as the protection of the adjoining cable galleries in the frontage is to be further investigated with regard to the effects of a large-scale fire or an H_2 explosion in the generator. In this investigation, consideration is also to be given to consequential events such as the fall of head trussing material. Furthermore, the power and control cable trenches leading to the emergency feedwater pumps in the region of the machine hall are to be protected against effects of fire on the cabling line with a fire resistance of 30. Short circuits, which may themselves lead to the ignition of the cables, are to be prevented by a selective fuse system. With regard to such possibilities of ignition in other cabling lines of the installation, a check is furthermore to be made, within the terms of the system- related investigation, as to whether short circuit currents in power cables (especially in cables which lead through fire-resistant bulkheads) can be ruled out with a sufficient degree of certainty.

8.2.3 Fire-fighting

In order to achieve rapid and reliable fire-fighting in all compartments involving high fire hazards, the backfitting of deluge sprinkler systems in the following areas of the installation is necessary:

- Cable shafts and cabling lines (general internal DC main distribution bus, machine hall, E 102, E 104),

- Oil supply regions of the turbine,

- Oil supply regions of the feedwater pumps.

In this connection, it is to be ensured that no consequential failures can be caused to significant safety-related systems as a result of sprayed water and the accruing quantity of fire-fighting water.

A check is to be made as to the acceptance parameters (e.g. discharge heights) of the fire-fighting water network according to TGL 10685. If TGL 10685 is not complied with, backfitting measures are to be implemented.

The emergency power supply to the pumps of the common drinking water, utility water and fire-fighting water network is to be ensured. In the event that a fire-induced failure of the power supply to all pumps cannot be ruled out, a mobile supply facility for this water network is to be provided.

8.2.4 Special fire protection measures for the cable gallery (E 102)

The probability of a fire and its effects are to be reduced by protective measures against unauthorized access to the cable gallery and by administrative and organisational measures for mobile fire-fighting.

Over and above this, on account of the particular safety-related significance of the cable gallery, the installation of a fixed gas extinguisher is considered to be necessary.

8.2.5 Spatial separation of redundant safety-related systems

As a result of the system-related assessment, the improvement of the spatial separation of specified safety-related redundant systems and of their cable connections is required (see also chapter 8.4). By way of a short term measure, the separation for fire protection purposes of the protected DC supply is considered to be necessary.

The fire protection measures necessary for this purpose are to be based on a detailed assessment on site.

8.3 Flood protection measures required in the short term

In general, maintenance work which involves the opening of pipes with a potential for flooding is not to be carried out during power operation.

As regards the building of the cooling water and service water pumps, short-term measures to preserve the operation of the service water system are considered necessary. Transport processes above the components and pipes are permissible only where there is no danger to the operation of the service water system as a result of falling loads.

As regards the emergency boration tank, short-term precautionary measures are considered necessary for the effective isolation of leaks in the service water outlet line. Regarding the possible penetration by water into the reactor well with external wetting of the reactor pressure vessel, further investigations are to be carried out in the short term, and if appropriate countermeasures adopted.

8.4 Safety-promoting measures for backfitting

The scope and nature of further measures for protection against redundancy overlapping events are dependent upon the overall backfitting concept. Due to unalterable conditions regarding structure and systems technology, priority is to be given to a separate emergency system complying with the current safety standard in terms of fire protection technology (emergency feedwater supply, service water system, where appropriate significant safety-related emergency cooling functions, emergency control room, reactor protection with instrumentation and control systems, and power supply). This system is to be accommodated within its own building and to be incorporated into the installation in such a manner that the backfitting for fire protection purposes in the present building may be restricted to a few compartment regions.

9. OPERATING EXPERIENCE

9.1 Documentation and reporting of accidents within the Greifswald Nuclear Power Station

Since commissioning of the first nuclear power plant in the GDR, incidents and accidents have been reported in accordance with the plant monitoring specification which is currently applicable in the industrial sector (TGL 190-113).

The safety monitoring systems were built up following introduction of the Radiation Protection Ordinance of 26.11.1969 (Official Gazette Part 1 No. 99, page 627). The Guidelines on Abnormal Occurrences (AO) were amended on several occasions (Table 9-1). The modifications related to the classification and deadlines for the reporting of events led - in connection with SAAS Guideline 1/88 - to a broadly worded reporting specification in the sector of nuclear safety.

This guideline is based on the Ordinance concerning the assurance of atomic safety and radiation protection of 11 October 1984 (Official Gazette Part 1 No. 30, page 341), the Ordinance concerning the physical protection of nuclear material and nuclear installations of 7 April 1982 (Official Gazette Part 1 No. 21, page 410) and the Ordinance concerning the transport of radioactive substances of 12 April 1978 (Official Gazette special No. 953). The following considerations relate only to the field of nuclear safety.

Although the Guidelines concerning the reporting of abnormal occurrences were implemented into operating regulations, only since 1983 there has been systematic reporting of safety-related events to the SAAS. At the

Table 9-1:
Summary of reporting procedures for abnormal occurrences in the nuclear power station (nuclear safety)

Year/Reporting Category	SAAS 3/74	SAAS 1/83	SAAS 1/88
		- divided into abnormal occurrences in the nuclear power station - other abnormal occurrences	- divided into 4 areas - nuclear safety
AO-1	Radioactive substance released uncontrolled from control or radiation protection areas - not locally restricted - immediate reporting	- Nuclear and radiation incidents/accidents · Nuclear incidents · Nuclear incidents/accidents · Nuclear hazards - Immediate reporting	Occurrences which lead or may lead with high probability to damage of nuclear fuel or its safe containment - immediate reporting
AO-2	AO-1 - locally restricted - immediate reporting	- reductions of redundancy in safety systems - cases of activation of safety systems automatically or manually - reporting during working hours by a maximum of 4 weeks after onset of the occurrence	Occurrences in the case of which damage to nuclear fuel or its safe containment can be ruled out with high probability - reporting during working hours on the day of the occurrence
AO-3	- do not represent an occurrence of the same type as AO-1/2, but do affect the radiation exposure situation of locally areas - annual report	--------	Occurrence of general importance with regard to guaranteeing nuclear safety - Monthly collective reporting

Status: 2.5.1990

same time, there was a qualification of the power stations internal regulations for the purpose of ensuring nuclear safety with the implementations of the operating regulation "Limiting values and conditions of operation to maintain nuclear safety" (BdsB).

The internal regulation, drawn up by the operating utility, concerning the handling of incidents (KA 40-1059 of 1.11.1983) is, until today, based essentially on applicable regulations for availability-oriented plant supervision, and includes the requirements of the SAAS guideline 1/88 only in part. The reporting in accordance with SAAS guideline 1/88 is made separately and there is a special employee of the utility for this purpose.

All occurrences were assessed by the operating utility. Occurrences of categories AO 1 and AO 2 were analysed and assessed by investigating committees, in order to establish and to eliminate the causes of the occurrence.

9.2 Selection of events

In order to compile the present report, documents concerning operational transients and safety-related incidents within the period 1980-1990 were reviewed and a selection of these was evaluated. Restriction to this period of time appeared necessary on account of the great number of occurrences which have taken place since 1974. Therefore, certain significant occurrences have not been included in the evaluation (e.g. cable fire in unit 1 in 1975 and SG tube rupture in unit 1, 1978).

The selection was based on records of the plant operator (unit specific failure statistics for the years 1974-1985) as well as the occurrences recorded by the SAAS within the period 1986-1990. The selection was made with reference to summary information which was compiled by the operator. Therefore, it does not necessarily represent all safety-related problems. Within the time available, it was not possible to check whether the documents submitted were present in their entirety.

Table 9-2 gives an overview of the frequency of transients which occurred.

Table 9-2
Overview of the frequency of reactor scrams, initiations of the reactor protection system (RPS) and power reductions (PR) caused by incidents for each unit in the Greifswald Nuclear Power Station, units 1-4

Year/ transient	Unit 1, 2 1974-1978	Units 1-3 1978	Units 1-4 1979-1989
RPS 1/2	7.75	4.33	3.068
RPS 3/4	0.5	0.3	0.65
PR, incident-induced	4.125	2.66	1.15

For the evaluation of the selected transients and incidents, the operator provided detailed occurrence reports and investigation reports respectively.

The transients and incidents which occurred may be classified as follows (on average, two typical occurrences were selected and evaluated for each category):

- Failure of one or both turbo generators
- Failure of the feedwater supply
- Steam generators overfeed
- Pressurizer overfeeding
- Primary loop leakages
- Secondary loop leakages
- Failure of reactor coolant pumps
- Restriction of the shutdown reactivity
- Unlocking of the emergency protection system
- Failures in the emergency power supply
- Failures in the protected 380 V AC power supply
- Failures in the protected 220 V DC power supply.

The transients and incidents are at first briefly described and then assessed with respect to the following aspects:

- sequence in accordance with the design
- compliance of the conditions of safe operation
- adequate redundancy
- manual measures
- consequential failures which occurred
- effects on safety
- practical usability of the operating manuals

The causes of occurrences are presented, and the required backfitting measures are presented.

The backfitting measures are listed in summary for chapter 9.4, divided into measures which are to be implemented in the short term and the long term.

9.3 Selected occurrences

9.3.1 Failure of a turbo generator

Occurrences in this category involve transients with sequences not in accordance with the design and which lead to a rapid power reduction in the reactor core, with possible subcooling in the primary loop.

Causes of a turbine trip are the following:

- automatic initiating actions on reaching specified technical limit values

- manually initiated action in the event of exceeding limit values

- failure of the automatic protective systems, which may result in incidents and damage to equipment when operating the turbine and the power station.

The operating experience during the operation of the units shows that only 50% of cases of load shedding are controlled in accordance with the design.

From the spectrum of occurrences, the following representative example will be discussed.

- Turbine trip with pressure drop in the primary loop and cold injection in unit 3 on 13.12.89 (Occurrence No. 220/89)

During power operation there was a sudden power reduction on turbo generator set 6 (TS 6) from 210 to 155 MW, with subsequent initiation of turbine trip by the criterion "vacuum drop (0.28 KPa)". At the same time, two feed pumps were activated manually.

Sequence of the occurrence:

As a result of reduced main steam flow, there was a pressure increase in the main steam header of the steam generators 1-6. Pressure and temperature in the primary loop rose for a short period, whereupon the injection regulator on the pressurizer opened. The main steam pressure rose to 4.9 MPa, the automatic power regulator (APR) switched over from "Regime S" (power-limiting mode) to "Regime R" (main steam pressure regulation), and the scram control assembly K6 was lowered as a result of the signal "power reduction on two RCPs" due to the failure of turbine 6. After approximately 30 sec, the APR switched to "hot reserve", whereby in accordance with the design specification RPS 3 (reactor power reduction with normal speed of the control assemblies) was initiated and the assemblies K6 and K5 were further inserted over a period of 50 sec. In parallel, the power on TS 5 was reduced by the personnel, in order to reach the allowable power.

After 1 min, there was the initiation of RPS 2 (reactor power reduction with maximum speed of all scram control assemblies) as a result of a reduction of the pressure in the primary loop to $p < 11.3$ MPa; 1 min later, the emergency boration pumps were activated as a result of a drop in the level in the pressurizer $L < 2600$ mm. They were deactivated again manually after 1 min.

One minute later, turbine was tripped manually on TS 5, whereupon, in accordance with the design specification, RPS 1 (scram by simultaneous dropping of all 6 scram control assemblies) were activated by "failure of last turbo generator set".

Causes:

The cause of the occurrence was found to be the fracture of the stem of a control valve of TS 6. The consequence - failure to control the unit dynamics in the event of turbo generator failures -represents a recurrent failure, influenced by several factors. These include the following:

- switching of the automatic power regulator from "Regime R" to "hot stand by", additionally with a power reduction lasting for a period of 50 sec as a result of voltage fluctuations on the secured main bus of 0.4 kV (SMB) after load shedding of generator 5

- possible erroneous actions by the personnel, which cannot be precisely reproduced

- an excessively rapid power reduction by the automatic power regulator together with an excessively slow power reduction on the second turbo generator, so that manual control became necessary.

Similar transients, which led to subcooling of the primary coolant loop with additional initiation of the high-pressure injection system, occurred during of abnormal occurrences in the feedwater system with a mean frequency of 0.5 per year per unit.

Backfitting measures:

Transients of this type can be avoided if the main steam pressure is stabilized by the automatic power regulator following turbine trip in accordance with the design specification. Therefore the following measures are recommended:

- establishment of new settings of the automatic power regulator for power reduction in the primary coolant loop

- more rapid reduction of the power of the second turbo generator

- avoidance of unnecessary operating actions by consumer switchovers during the transients

- backfitting of the control concept, so that manual actions by the operating personnel do not become necessary during rapid power transients, for example, during a turbine trip.

9.3.2 Failure of the feedwater supply with injection of emergency feedwater

The safe operation made of the nuclear power plant is to a large extent dependent upon the reliability and availability of the feedwater system, especially for supplying the steam generators with coolant during power operation, abnormal occurrences and during the cooling or heating of the primary loop during start-up and shutdown.

Abnormal occurrences within the feedwater system, which, depending upon their cause, frequently lead to the cutting in of the emergency feedwater pumps (via L < - 140 mm - SG level interlock), can principally be attributed to the failure of the feedwater control valves and the failure of feedwater pumps.

The feedwater control valves (double cone regulators) often fail as a result of mechanical seizure, break of internals (stem, conical shaft) etc.

Accordingly, the operator introduced a revision programme involving annual exchange of regulators; in some cases, this involved considerable expenditure on repairs.

In order to reduce the mechanical loading of the feedwater control valves, the low-load control system was incorporated in the installation. In this connection, there is discussion of appropriate operating modes using 3 feedwater pumps as operating pumps (2 feedwater pumps preselected as back-up), whereby a reduction of the pressure difference by approximately 1 MPa and thus a certain relief of the regulators could be achieved.

The operator is carrying out vibration measurements at the feedwater control valves and further measures such as the rinsing of the regulator system before the feedwater control valves are taken into operation in order to avoid deposits, this being done in order to operate the feedwater control valves under optimal conditions.

In the course of the occurrences observed within the observation period, there were a total of 3 cases of the failure of feedwater pumps involving a fall in the level in the SG and cutting in of the emergency feedwater pumps. All 3 cases occurred as a result of the intentional isolation of a feedwater tank under conditions of power operation or voltage loss.

- Failure of all feedwater pumps on 28.11.1981 in unit 1 (Occurrence No. 285/81)

Sequence of the occurrence:

During the separation of the feedwater supply in order to isolate a feedwater tank on 28.11.1981 in unit 1, the feedwater pumps D and E were taken out of service and were no longer preselected as back-up. As a result of a cabling fault at the gate valves of the feedwater suction line, the feedwater pump C failed. At a unit power level of 300 MW, the two feedwater pumps A and B which were still in service reached their power limit and ceased to operate by action of the system protection interlock $p < 4.7$ MPa (in the pressure nozzle).

The generators were isolated from the grid, and the isolated operation of the two turbine sets was maintained using one emergency feedwater pump (the other emergency feedwater pump did not start, as a result of a power switch defect). Via the criterion $\Delta L < -500$ mm - interlock of the feedwater tank, the feedwater pump A which had meanwhile re-entered into service ceased to operate and after 20 min TS 1 tripped as a result of the vacuum interlock. TS 2 was shut down by the control room a short time later.

Causes:

The cause of the occurrence were cabling faults at the isolating valves as well as a defective power switch on an emergency feedwater pump. As regards the further sequence of events, it was of decisive importance that an automatic power reduction in the unit was not provided for in accordance with the design specification in the event of failure of feedwater pumps.

Backfitting measures:

The required automatic system is to be installed in conjunction with the LSG < -500 mm - interlock (item 7.6.1) recommended in chapter 7.

The failure of only one emergency feedwater pump, as occurred on 23.9.1985 in unit 2 as a result of wear and excessive mechanical overload, led to a reduction of the safety level, in spite of an immediate cross-connection of the emergency feedwater pressure line to the adjacent unit.

According to the operating specification, the emergency feedwater pumps are used not only for the emergency supply to the steam generators but also in start-up and shutdown processes in the unit as well as to cool down the primary coolant loop in the steam-water phase. Accordingly, it is necessary to evaluate, for the emergency feedwater system, appropriate solutions to relieve the emergency feedwater pumps. An interim solution would be to connect the emergency feedwater systems between the units. In the long term, it is desirable to achieve an injection system independent of the main feedwater system, with adequate reliability.

9.3.3 Steam generators overfeeding

For the horizontal steam generators, there is only little tolerance with regard to control of the steam generator level.

When taking units 3 and 4 into service, a steam generator level interlock in all units was introduced for the first time; this interlock operates on the feedwater control system and, when exceeding a permissible value, initiates turbine trip for both turbines. During the years of operation, several attempts were made to optimize the settings applicable to level interlock and feedwater control. However, as the described abnormal occurrences show, unnecessary unit failures during irregular operations in the feedwater control could not be prevented effectively.

- Unit failure on 11.11.1985 due to overfeeding of steam generator 1 (Occurrence No. 312/85)

Sequence of the occurrence:

Due to repair work in the feedwater system, the power was reduced to 35%. The feedwater system was operated using only one feedwater pump. As a result of oscillatory power fluctuations by the turbo generators, the second feedwater pump was cut in by the interlock "p < 5.4 MPa". The feedwater regulator 49 M should have been initiated by the level interlock "L > 75 mm". However, the control valve stalled at 90% aperture. At the time of the fault, actuation of the control valve from the control room was not possible either.

At "ΔL > 100 mm" the pilot valve was automatically shut. Nevertheless, within one minute the limiting value "ΔL > 200 mm" was reached and, in accordance with the design specification, the trip of both turbo generator sets as well as RPS 1 were initiated. As the pilot valve closed, it can be ruled out that the level "ΔL > 200 mm" had actually been reached. The probable reason for the excitation of the trip is the arrangement of the level measurement system. In the course of the backfitting of the interlock "ΔL > 200 mm", which is designed as a "2 out of 3 circuit", additional volume control surge tanks were placed at existing connections. These are fitted at various positions of the steam generator and at different heights. The adjustment was done under nominal load. Tests under partial load conditions have revealed deviations of up to 150 mm between the (measuring) devices, so that higher levels are feigned.

Causes:

The cause of the failures of feedwater control systems proved to be foreign bodies in the feedwater system. In addition, problems arise from the imprecise measurement of the level of the steam generators, and these problems can be ascribed on the one hand to inadequate measurement techniques and on the other hand to the unfavourable arrangement of the measurement sensors.

Backfitting measures:

The feedwater control valves are to be optimised with regard to their construction and to loading by the constant excitation of the regulator (caused by the small control deviation). In addition, a level measurement system suitable for the horizontal steam generators is to be provided.

- Failure of unit 4 on 18.1.1985 (Occurrence No. 21/85)

Sequence of the occurrence:

The initiating event was the failure of the high pressure pre-heater (HPH) No. 3 of turbo generator 8 and non-opening of the bypass valve for HPH 3. This led to a drop in the level in all steam generators, so that the feedwater control valves automatically opened fully. The power of the unit was reduced.

In the course of the incident, the turbine operator opened the bypass valve from the local control console. This resulted in a large rise in the level in the steam generators. As a result of two further failures (mechanical failure of a feedwater control valve and a failure in the measuring instrument for the interlock "$\Delta L > 100$ mm"), the steam generator 5 was overfed. The overfeeding was terminated by the interlock "ΔL > 200 mm". The result was turbine trip and shutdown of the reactor by RPS 1.

Causes:

The occurrence can be attributed to the absence of a parallel key at the coupling between motor and gear. In addition, the display of the measuring instrument M 260 (interlock "ΔL > 100 mm") jammed in the lower end position. As a result of the single-channel construction of the interlock, it was possible for the failure of a measuring instrument to have a direct effect.

Backfitting measures:

Such failures can be avoided by adequate quality control during maintenance (test following repairs). In the secondary loop as well, multi-channel interlocks are to be provided, especially for "steam generator level high or low".

9.3.4 Pressurizer overfeeding

- Formation of a "hard loop" in unit 1 on 23.9.1986 (Occurrence No. 236/86)

Sequence of the occurrence:

During the start-up, complete filling of the pressurizer with water led to the formation of a "hard loop". In the course of the present occurrence, the control function of the pressurizer was neutralized by complete filling with water.

During the start, the concentration of boric acid in the primary loop was reduced by injection of clean condensate in order to correct the setting of control assembly K 6. This feed operation was performed using the feed pumps.

At the same time, the power levels of turbo generators TS 1 and TS 2 were increased to 65 MW and 175 MW respectively. At this point in time, the display "approximate level in the pressurizer" was not available from the instruments to monitor the pressurizer due to repair. The recorder for the fine level was not operational and the level interlock for the feed pump (initiation pressurizer level > 5500 mm) was ineffective. In the course of the start-up, the levels in the pressurizer were recorded several times. In spite of a level of 6300 mm, which is already outside of operating specification, the feeding was continued by the operators. As a result of failing the coarse measurement system, the levels could only be monitored via the fine measurement system. However, on account of its connection into the pressurizer, the latter system only has a reliable measurement range of up to 6200 mm. As a result of these defects, feeding continued without a further increase in level having been detected by the measuring system. Further difficulty was also created by the failure of the indication "pressurizer level > 6000 mm". After 25 min the pressure in the primary coolant loop rose and consequently, water was injected from the cold leg by opening the pressurizer spray system. Since at this time the steam filling no longer existed and the injection nozzles were already under water, it was not possible to compensate the pressure rise. At a pressure of 13.2 MPa, RPS 4 was initiated. As a result of a further pressure rise, at 13.7 MPa RPS 3 was initiated, which correctly after 20 sec transferred to RPS 2. Accordingly, the unbalanced heat balance led to an intense cooling of the primary coolant loop and, consequentely, to the cutting in of the heating elements of the pressurizer and to the closure of the pressurizer spray valve. Nevertheless, it was not possible to stop the pressure drop, so that at

10.3 MPa the emergency boration pumps automatically came into service. Tripping of both turbo generator sets was "hammered" (by hand) and in doing so RPS 1 was initiated.

In the following period, the operating pressure in the primary loop was again restored using the emergency boration pumps, feed pumps and pressurizer heating.

Causes:

During the start-up, important safety values were missing. Moreover, the measuring instruments were not provided with markings which clearly indicated the reliable measurement range. There is no emergency protection interlock for the criterion "pressurizer level high".

Backfitting measures:

In the event of failure or repair of a level measurement for the pressurizer, injection into the primary coolant loop must be interrupted, in order to prevent the formation of a "hard loop" with the possible consequence of overloading the reactor pressure vessel. By means of a safety-related planning of revision, it is to be ensured (and this should also be specified in the conditions for safe nuclear operation), that all measuring devices and interlocks required for start-up purposes are effective.

Moreover, an appropriate emergency protection interlock by the criterion "pressurizer water level high" is to be installed.

9.3.5 Leaks in the primary loop

The WWER-440/W-230 plant is designed to cope with a leak in the primary loop of nominal diameter 32. This corresponds to the break of the injection line leading into the pressurizer with a nominal diameter 100 and a flow limiter of nominal diameter 32 in the non-lockable part of the primary loop.

Reactor scram in the event of leaks in the primary coolant loop takes place by the interlock p (primary coolant loop) < 11.8 MPa and L (pressurizer) < 2400 mm.

According to the utility's operating specifications the personnel has to carry out manual measures to locate and to isolate the leak, e.g. by alternate isolation of 3 primary circulation loops at a time, and to determine the leak rate by reference to the injected quantity and to the pressurizer level. Until now, smaller leaks have occurred at the tubes (capillaries) of the steam generators, and these were coped with by the personnel in accordance with the operating specifications.

Two occurrences involving leak in the primary coolant loop are described in the following:

- Leak of the regenerative heat exchanger in unit 2 on 7.3.1981 (Occurrence No. 40/81)

Sequence of the occurrence and causes:

The largest abnormal leakage which occurred in the primary loop and which has been reported during the period under review is the (overload) rupture of the regenerative heat exchanger of the water processing system 1 during the cooling phase of the primary loop for the revision of unit 2 on 7.3.1981.

In this case, inadequate procedure for cooling the primary loop led to the hermetic isolation of the pipe compartment in the regenerative heat exchanger and, as a result of the removal of the cooling water via the purification line, to a pressure rise in the regenerative heat exchanger up to 28.5 MPa (break limit). The leak was controlled using 4 emergency boration pumps. Under power operation, such a leak due to the same cause is not possible. The cooling technology of the primary loop was changed, and the design of the regenerative heat exchanger was modified.

- Leak in the pressurizer injection nozzle in unit 2 on 1.10.1984 (occurrence no. 264/84)

Sequence of the occurrence and causes:

This abnormal leakage occurred at the pressurizer injection nozzle when performing the 4.0 MPa tightness test during the start-up of unit 2 after a brief outage period.

Inspection revealed a circumferential crack having a length of 40 mm at the injection nozzle (nominal diameter 90), which was caused by low cyclic fatigue and promoted by remaining of a rotational edge at the internal surface. The crack configuration tended towards the "leak with break" type of crack and could have led to an abnormal occurrence exceeding the design specification. The defective component was replaced. The injection nozzles of units 1, 3 and 4 were examined. In doing so, similar findings were made in unit 3 as well.

Backfitting measures:

In order to prevent the recurrence of such an event a loading analysis will be made, measures to reduce the thermal cyclic loadings will be taken and a consistent examination of material will be performed. New components with improved design have been used. Nevertheless, the occurrence of small leaks must be expected. Since the location of small leaks requires costly and time-consuming manual measures, the installation of a leak detection system is necessary for a more rapid and safe detection and isolation of leaks.

9.3.6 Leaks in the secondary loop

A design problem in all units is the inlet of heating condensate into the feedwater tank. The condensate originates from the high pressure heaters (HPH) and the moisture separator reheater. The critical point is the junction to the lines from the moisture separator reheater into the heating condensate line.

At this location and at a segment pipe bend, considerable erosion corrosion was detected. Moreover, pipe oscillations having an amplitude of up to 10 cm were observed. This led to damage to the fixed bearings and sliding bearings of the pipe and to the connection of the line into the feedwater tank. In the four units, damage was found at almost all cantilever beams of fixed bearings.

- Leak in the heating condensate line to the feedwater tank 2 on 4.2.1984 in unit 1 (Occurrence No. 19/84)

Sequence of the occurrence:

The level control of high pressure heaters 1 and 2 was defective. Accordingly, all three HPHs should have been taken out of service to permit repair work. Thus, the condensate from the second stage of the moisture separator reheater was passed into the feedwater tank 2 instead of into the HPH 3. Consequently, the steam-water mixture passed via a leak in the feedwater tank 2/feedwater tank 1 region into the machine hall. The leak could not be precisely located immediately. The turbo generator 2 was shut down and the feedwater tanks were isolated. Subsequently, both turbo generators were taken out of operation by manual tripping, since as a result of water in-rush in the electrical distribution boards the situation there was also ambiguous. The reactor was shut down by RPS 1.

The leak was situated in the last segment of the bend (nominal bore 400), immediately upstream of the feedwater tank. There was an opening of approximately 200 mm x 300 mm. The cause is perceived to be erosion due to two-phase flow (steam-water mixture). The estimated maximum pressure in the heating condensate line was 0.8 MPa (design specification: 0.57 MPa).

Causes:

The main condensate line was not designed for loadings by two-phase flow systems. Erosion corrosion with subsequent leaks occurred repeatedly, in some cases even in less then one year of operation. Some design improvements have been made to the line. Furthermore, it has been proposed that the pipes should be constructed from austenitic steels. The monitoring of wall thickness, which was carried out on the heating condensate line, showed considerable defects both in methodology and also in performance. Since only a few positions were inspected on the pipe (not precisely the same measurement points), it was not possible to make precise statements as to the weakening of these positions since the last check, nor was it possible to determine the position of minimum wall thickness.

The discharge of steam into the machine hall caused two serious problems. On the one hand, certain valves which were required to isolate the feedwater tank could be lever-operated only from the machine hall. These levers were situated in the area affected by the steam. On the other hand, the steam atmosphere in the machine hall could impair the function of the electrical instruments installed therein.

Backfitting measures:

The valves which are required for the isolation of the feedwater tank should be capable of being operated from the control room. Furthermore, the valves of the emergency feedwater cross-connection to the adjacent units should be remotely controllable.

The electrical systems of the emergency feedwater system should be designed to withstand the loadings which may arise as a result of the escaping steam. In this connection, attention should principally be given to the operating capability of drive systems and motors under conditions of high atmospheric humidity (accident-resistant drives).

The need for this is supported by an occurrence in unit 4 on 18.5.1988 (Occurrence No. 87/88), in which the failure of both emergency feedwater pumps took place as a result of a steam leak in the preheat line at the feedwater pump.

The pipes in the main steam system, especially the heat condensate line, should be checked regularly by means of non-destructive material testing, for early detection of erosion, corrosion and damage due to water hammer. The suspensions of the pipes in the systems should be backfitted to withstand possible water hammer (fixed point forces to be increased).

- Rupture of a 4.7 MPa main steam heating line due to crane travel with a low-pressure turbine casing in unit 2 on 1.8.1980 (Occurrence No. 212/80).

Sequence of the event:

Unit 1 was out of service for major maintenance, and unit 2 was in normal operation. The turbo generator set 1 was uncovered for inspection. The housing of a low pressure part of the turbine was carried by the crane in the hall to a setdown position between turbo generator 4 (unit 2) and turbo generator 5 (unit 3) on the ± 0 m level in the machine hall. At the bottle-neck of the transport corridor, the 4.7 MPa main steam cross-connection from unit 2 to unit 3 is situated. While lowering the low-pressure casing, there was a collision with the heating line for this cross-connection, which consequently broke. The discharging steam turned the low-pressure casing against a crossover line of turbo generator 4. There two bellow joints were destroyed and the line and its supporting structure were bent.

In the operating room, the abnormal occurrence was detected only by the noise generated; deviations of parameters were not observed. For the turbo generator 4, the control operator initiated locally the turbine trip by hand; the automatic power regulator reduced the reactor power in accordance with the design specification. In the further course of the abnormal occurrence, the main steam was controlled manually. Due to the required manual measures the pressure dropped in the primary loop below 11.3 MPa and RPS 2 was initiated. In parallel the main steam pressure fell below 4 MPa. Consistently therewith, a turbine trip for the turbo generator 3 occurred, and the reactor was shut down by RPS 1. The leak was isolated by separation of the main steam header.

Causes:

The transport of bulky parts within the machine hall over operating components in other units of the installation represents a source of danger. The transport of the low-pressure casing was secured only by the personnel. The travel of the crane with its load swept over approximately one half of the machine hall.

The location of the leak took place after shutdown of the reactor only. In the unit control room there were no indications available suggesting that the line had broken.

Backfitting measures:

Transport operations using the hall crane over sections of the other units in operation should be avoided.

Independently, it should be checked whether a separation of the machine halls per unit can be implemented, since hazards do not arise only from operations involving transport of heavy loads but also, due to other large leaks, consequential damage is to be expected as a result of discharging steam and therefore all units may be endangered.

9.3.7 Failure of reactor coolant pumps

In the event of failing individual reactor coolant pumps, the reactor power regulator reduces the power by the insertion of control assemblies (when failing one MCP, to approximately 65% of nominal power, or, in the case of failing two MCPs, to approximately 35% of nominal power). If the operation of the regulator is abnormal, RPS 3 is initiated and the control assemblies are inserted into the core.

If a failure occurs of more than two MCPs during power operation, the reactor is shut down by RPS 1. The signal for this purpose is generated in the so-called automatic MCP system, which supervises a diversity of criteria for the MCP (power monitoring relay, pressure difference across the MCP, monitoring of the number of turbines in operation).

In the period under consideration, no failures of MCPs were found in the evaluated documents: an indication of their high degree of reliability. The tripping of one turbine results, in accordance with the design specification, in the rundown of the two MCPs driven by the internal power supply generator of the turbine within 3 min.

In unit 2 on 7.9.1987, there was an incorrect shutdown of an MCP. The cause and the possible consequences will now be set forth.

- Incorrect response by the interlock for the protection against break of a steam generator main steam line and failure of the MCP 3 in unit 2 on 7.9.1987 (Incident N° 221/87 and 222/87).

Sequence of the event:

The interlock 6.4.19 (protection of the reactor pressure vessel in case of break of a steam generator main steam line) should respond if the pressure in the SG falls below a value of 3.5 MPa and the differential pressure between the main steam header and the SG exceeds a value of 0.5 MPa. The measures resulting from this are the following: shutdown of the MCP associated with the SG and isolation of the SG on the feedwater side and on the main steam side.

During the period between 31.8 and 7.9.1987, unit 2 was operated at a main steam pressure of 4.0 MPa. On 4.9.1987 the signal was pending at one channel of the interlock for SG 4 (the interlock consists of three channels). The switching point for Δp was newly adjusted. On 7.9.1987 the interlock shut down the MCP 3 because two channels of the interlock had incorrectly triggered. At this time unit 2 was in cooling mode. The main steam pressure was 3.0 MPa.

The investigations performed showed that Zener diodes had been connected with incorrect polarity in the electronic system of these interlocks. This led to a situation in which, as the main steam pressure declined, the switching point for the pressure difference between the header and the steam generator fell to 0 MPa and the interlock became effective with only <u>one</u> criterion present.

Causes:

As a result of the incorrect function of the interlock, there was a probability that at a main steam pressure of < 3.5 MPa all MCPs would be shut down before having finished their rundown and that this would lead to film boiling in the core, to a damage of fuel elements and to an excess pressure within the primary loop. The diodes had initially been connected with correct polarity in all units, but then during the major maintenance in 1987 had been inserted with incorrect polarity in units 2 and 4. This was done by the maintenance personnel and was also amended in the schematic wiring diagrams, without any investigation having been made as to the effect of this change on the circuit. It was not possible to detect the error in the course of the operational test carried out subsequently by the power station personnel.

The error is to be ascribed to errors in management. Of particular importance is an error in relation to a test concept in accordance with the requirements which is able to detect faults in the interlock.

Backfitting measures:

Prior to the implementation of modifications at safety-related systems, a thorough evaluation and inspection is necessary. The areas of competence and responsibility must be clearly regulated. Any broadening of the scope of working orders without proper authorization must be prevented. An incorrect shutdown of all MCPs in the power operation mode could lead to an abnormal occurrence outside the design specification. The logic of the interlock and drive of the MCPs should be examined, this way ensuring that not more than 1 MCP can be shut down.

9.3.8 Reduction of the shutdown reactivity on outage of the reactor due to injection of chemicals and demineralized water

When evaluating the operational experience, several cases could be detected in which the boron concentration was reduced by injection of chemicals. Moreover, in one case, in order to start up the reactor, demineralized water was injected into the primary loop when the shutdown assemblies had been inserted. In doing so, the shutdown reactivity was no longer assured. In this connection, it is also necessary to take into account the deboration of the primary coolant loop. This occurred by erroneously exchanging the pipes of the intermediate coolant circuit of the control and protection system (ICC-CPS) with the air venting line of the reactor as consequence of a maintenance fault on 24.5.78. However, this occurrence took place before the selected period of 1980-1990 and is therefore not subjected to further analysis here.

- Inadequately borated addition of chemicals during outage of the reactor in unit 3 on 11.6.1984 (incident No. 156/84).

Sequence of the event:

At the time of the inadequately borated addition of chemicals, the installation was in revision phase; loop 2 was in natural circulation, and the decay pool was connected to the reactor pressure vessel via the refuelling canal. The boron concentration in the primary loop was 12.38 g/kg. On 11.6.84, 1,500 l of ammonia (NH_3) were to be added into the primary loop.

The shift personnel pumped 1.5% NH_3 from the chemical plant into the primary loop, using the make-up pump. In the course of this procedure, they failed to determine the initial boron concentration by means of sampling. As far as the primary loop was concerned, this resulted in an unknown and inadequate boration level which was not permissible within the terms of the "Conditions for the safe operation of nuclear safety systems during reactor outage". Under these conditions, it is generally established that the boron concentration of any feed must be higher than that present in the primary coolant loop.

During the subsequent investigation, it turned out that, depending upon the particular staff working the shift, different procedures were used with regard to the feed of chemicals.

On the same day, another shift carried out addition of chemicals by means of the seal water system, using the associated seal water pumps. In the past, the seal water system served principally for the internal tightness of the closed main isolating valves. Prior to the injection of 50 l of 21 % NH_3, three samples were correctly taken from the seal water tank, in order to determine the boron content.

Which of the two procedures was followed depends upon whether the shift concerned had or had not been previously employed in units 1 and 2. In units 1 and 2, feed of chemicals was possible only via the seal water tank, due to backfitting of the chemical plant.

In the further course of the investigation, previous and subsequent to this occurrence infringements of the general operating specification with regard to the feed of chemicals were discovered. In all cases, no determination was made of the boron content of the chemicals to be injected.

Unit 3:

- 14.6.84
  Addition of 150 l of 1.5% NH_3 via suction-side make-up pumps

- 19.6.84
  Addition of 900 l of 0.65% hydracine via suction-side make-up pump

Unit 4:

- 12.9.83
  Addition of 20 l of NH_3 into the decay-pool and into the reactor well using buckets!

- 1.10.83
  Addition of 150 l of 1.9% N_2H_4 via suction-side make-up pumps

- 26.10.83
  Addition of 120 l of N_2H_4 via suction-side make-up pumps

Causes:

The frequency of such incidents shows that the participating shifts were clearly unaware of the safety-related significance of correct addition and the maintenance of subcriticality in the primary coolant loop during outage of the installation.

The feeding of non-analysed chemicals by means of buckets into the cooling pond is proof of this.

In the course of the investigations of the nuclear power station, it emerged that the special operating instructions for the feeding of chemicals were totally inadequate. There is no special specification concerning such feeding. Only in the instructions regarding feeding by means of the seal water system is there some indication that samples must be taken.

In contrast to this, there are no instructions in the operating manuals "chemical plant" and "feed system". The general references in the operating manuals concerning the primary loop are of little help to the shift in performing a feed operation under the correct conditions.

Backfitting measures:

It is necessary to revise the operating manuals, particularly since the installations must be run virtually by manual means. Feeding without boration is a threat to safety, since there is no continuous measurement of the boron concentration in the primary coolant loop. Moreover, it appears that there is a need for increased training for the shift personnel. The fact that shifts previously employed in units 1 and 2 perform different operations in comparison to those shifts which were employed from the outset in units 3 and 4 poses a safety risk, since the units differ with regard to certain items of technical equipment and procedures.

- Unborated feed during reactor outage in unit 3 on 16.4.89
  (Incident No. 62/89)

Sequence of the event:

When the event took place, the installation was in the following condition: 120°C, 20 bar, MCPs 4 and 6 in operation.

Prior to commencement of the start-up, the "control and protection system (CPS) test part II" test procedure was carried out. In the final step of this test procedure, the six groups of the emergency, control and compensating assemblies are successively withdrawn and reinserted. After a test period of 1 h 50 min, deboration of the primary loop was commenced at the same time with cutting-in the make-up pump P9A, and the primary loop at this time had a boron content of 10.5 g/l. A few minutes later, the CPS test was successfully completed and the scram control assemblies including the last group 6 were incorrectly run into the lower end position.

Under this condition, the deboration was continued for a further 2 h using 2 operational reactor coolant pumps, until the minimum concentration of 8.0 g/kg was reached. The minimum concentration is applicable to start-up when the control assemblies are withdrawn. Since, however, the control assemblies were in the lower end position, the required shutdown reactivity was no longer available. This represents an infringement of the conditions for safe operation and the operating manual "reactor". The supply of reactivity by injection of demineralized water without adequate shutdown reactivity is classified as a particularly serious case.

Causes :

The incorrect action by the operator was made possible because there are no supervisory systems and interlocks which prevent deboration when the scram control assemblies are fully inserted. It further emerges from the investigation report relating to Occurrence No. 62/89 by the VE Kombinat Kernkraftwerke that at the date of the occurrence the applicable specifications to "make the reactor critical" were inadequate.

Backfitting measures:

An interlock should be provided which prevents a further deboration, after the integral control rod position has fallen below a load-dependent insertion limit value and, by way of a countermeasure, initiates an increase in the boration level of the reactor. By this interlock, a deboration should only be possible if the shutdown rods are above the insertion limit. By this means, deboration during start-up with the rods inserted is impossible. In addition, it would also be useful to provide a so called "10 t limit" as is customary in the FRG pressurized water reactors, to cover this case. This limiting system stops the feeding after the injection of 10 t of demineralized water. The operator must then perform a conscious switching operation to achieve any further injection of demineralized water - opening of the injection slide valve and starting up the pumps.

In addition, the "conditions for safe nuclear operation" should clearly state that the test of the control and protection system must in no circumstances be carried out simultaneously with deboration.

9.3.9 Unauthorized unlocking of initiating criteria in the reactor protection system and unlocking/ switch-over during maintenance

In this power station, interlocks are understood to be initiation criteria for the emergency protection system. "Unlocking" signifies that protective actions by the protection system are prevented. This applies both to scram and to protective actions of the other protection systems such as, for example, the emergency boration system and the emergency feedwater system.

In some modes of operation, the operating manual and the "conditions for safe nuclear operation" permit the unlocking of initiating criteria. Apart from the reactor period and the "power reserve" (thermal overpower protection), all initiating criteria may be unlocked in the control room using key switches. In evaluating operational experience, the unlocking of initiating criteria was observed in some cases. It is difficult to estimate to what extent the nature and, in particular, the frequency of such cases is representative. Particular attention must be paid to the unlocking of initiating criteria, above all because only the unlocking of the first criterion is signalled as a general notification on the control room panel; all further unlocking actions then remain undetected on account of the general notification already in existence. The unlocking switches arranged behind the control room panel and therefore not within the field of view of the operating personnel are a considerable safety risk.

The intended protection of these unlocking switches is made ineffective in that the keys stay on the available key switches even during operation. This was also observed during an on-site inspection. The interlocks are tested during operation and, for this purpose, are in some cases unlocked. Responsibility for the test is incumbent upon the "Operational Measurement, Control and Regulating Technology" department (BMSR). As a result of communication faults between the control room personnel and the BMSR, there were cases of unintentional unlocking, even over relatively long periods of time. The possibility of unlocking criteria applicable to emergency protection without restriction from the control room is a threat to safety. The two examples given below demonstrate the many possibilities of unlocking and switchover of criteria.

- Failure of unit transformer 4, with subsequent unit failure in unit 2 on 14.10.1980 (Incident No. 264/80)

Sequence of the event:

On commencement of the event, unit 2 was operating at 440 MW under full load. After a Buchholz protection system warning on the transformer BT4, the transformer protection system was triggered and opened the power switches. The generator 4 passed from the grid and the unit busbars 1/2 and 3 were switched over to the standby supply of the automatic standby actuation system. As a result of the loss of generator 4, the main steam pressure rose. The automatic power regulator reacted to this by inserting the shutdown control assembly 6 to reach the lower electrical end position. At the same time, the steam bypass dump system opened for a short period.

As a result of the grid disturbance of generator 4, the speed of rotation of the internal supply generators increased to such an extent that a voltage of 468 V was present for a short period on the subordinate 380 V busbars.

This increased voltage led to partial failures of the low-frequency converters which are required for the power supply to the drives and retaining magnets of the control assemblies. As a result of this, the scram control assembly 5 fell into the lower intermediate position. As a result of further faults in the drive and automatic switchover system of the low-frequency converters, the scram control assembly 4 also received a command to move downwards. When the scram control assembly 5 reached the lower mechanical end position, RPS 4 responded.

The interlock "RPS 3 on failure of the automatic power regulator" was unlocked by the operator. Following this, the automatic power regulator was taken out of service, in order to move the control assemblies upwards again manually.

Meanwhile, the reactor became subcritical and the pressure on the primary side fell to 11.3 MPa. In order not to acquire an RPS 2 command as a result of the criterion "pressure < min" the operator also unlocked this criterion.

The generator 3 automatically dropped to 170 MW. At this time, the main steam pressure was 4.2 MPa. A drop in the level in the pressurizer to below 2560 mm led to a situation in which four emergency boration pumps commenced operation via interlocks. The power of generator 3 was reduced to 50 MW.

The main steam pressure fell further to 4.05 MPa. Since at main steam pressure < 4 MPa turbine trip would have been initiated for both turbines, the operator also unlocked this initiating criterion, in order to prevent trip taking place. As a result of this, he also indirectly blocked RPS 1, since this RPS 1 criterion is established from the trip of the last turbine operating.

As a consequence of further voltage drops due to switch-overs to the standby distribution board, the neutron flux instrumentation, inter alia, also failed; this finally led to RPS 1.

The emergency boration pumps were deactivated by the operator after 10 min.

Causes:

In addition to the multiple failures in the voltage and internal power supplies, safety is affected in particular by the three unlockings of initiating criteria.

Although it is forbidden to deactivate "interlocks which serve to protect the installation", the following interlocks were disarmed:

- Primary loop pressure < 11.3 MPa
- Failure of automatic power regulators
- Trip of the last turbine operating

As a result of this, the staggered emergency protection sequence was interrupted for this transient development.

The unlocking of three important shutdown criteria within only two minutes during a transient with simultaneous failure of internal power supply busbars and incorrect cutting-in of the reserve supply system represents the most serious of the cases observed by us.

Such an infringement of the operating specifications is only possible because the majority of the unlocking switches are situated on the logging desk and the remainder on the reverse side in the room next to the control room.

Backfitting measures:

Since, according to the "conditions for safe nuclear operation" and the operating manuals, unlocking is permissible in specified cases and is also necessary for the operation of the installations, it is not possible at the present time to remove all switches from the control room.

A check should be made as to whether within the given 220 V/380 V control technology as an intermediate solution, unlocking of criteria could be sufficiently reliably effected only as a function of specified process parameters. In the long term, it would be necessary to construct a logic system with an entirely new instrumentation and control and protection concept.

- Impermissible switching condition in the neutron flux measurement system during reactor outage in unit 4 on 25.2.1988 (Occurrence No. 37/88)

Sequence of the occurrence:

At the beginning of the occurrence, the plant was in the condition "reactor assembly after charging with fuel elements", natural circulation using circulation loop 2, 30°C, level in the reactor 3.50 m.

In order to monitor the reactor period, the source range channels 15 and 23 of the neutron flux measurement system were in service, using the acoustic signal generators. The preamplifiers of the ionisation chambers of the "intermediate range" had been removed for examination. Therefore no acknowledgable emergency protection signals were available for the "intermediate range", the channels of which being switched to "monitoring" (test position).

On 28.2.88 a test was to be carried out on the automatic MCP system, i.e. the initiation of RPS 1 in the event of failing more than two MCPs. For this purpose, it was necessary to close the emergency protection initiation loops. The BMSR shift controller responsible for emergency protection accordingly switched over the measurement range switching device of the neutron flux measurement system from "source range" to "power range" on 25.2.88 at 18.30 hours in the control room after having informed the unit manager. This rendered the displays and the limiting value indication of the "source range" in the control room ineffective. The recorder for the reactor period in the "energy range" switch setting obtains its signals from the ionisation chambers of the "intermediate range", the preamplifiers of which were under inspection. Accordingly, the period recorder simply showed "∞" at all times. As a result of this, the monitoring of the reactor in the unit control room was greatly restricted. Only the acoustic signal generators were still in operation. No special measures were introduced, nor was the switching condition passed over to the following shift. Only during the early shift on the following day was the incorrect switch setting detected and rectified at 07.45 hours. Thus, the reactivity of the reactor had not been monitored for a total of 13 h 15 min. This represents an infringement of the "Conditions applicable to the safe operation of nuclear safety systems during reactor outage" and is a safety risk in as much as any reactivity increase could not be detected.

Causes:

The incorrect action underlying this occurrence was made possible by organisational and technical defects. On the one hand, the responsibilities of the technical support and the production department are overlapping each other. Evidently there is no distinct isolation concept, nor is there a work instruction system which guarantees the safety requirements of a nuclear power station to be respected. Thus, the shift supervisor of the BMSR technical support department can undertake work on the emergency protection system without the unit manager having previously issued a corresponding work instruction. Lacking here is an unrestricted direct responsibility on the part of the unit manager. The defects of the isolation concept became evident, inter alia, in that the emergency protection test concerning the "automatic MCP system" was to take place in parallel with the inspection of the preamplifiers for the intermediate range detectors.

In addition to this, there are no interlocks nor automatic control systems which prevent the unauthorised switchover of the neutron flux measurement ranges. Even during outage of the installation, it must remain impossible to bypass the monitoring of reactivity. Some of the personnel lack the safety awareness to appreciate the need for constant monitoring of the nuclear parameters of the reactor core during outage.

Backfitting measures:

The responsibilities for work on the installation must be clear. The final responsibility for performing any measure must rest with the unit manager. Without his consent, no work should be undertaken. To this end, it is necessary to formulate an appropriate work authorisation and isolation regime, to be related to a concept for the chronological sequence of inspection and maintenance work as well as functional tests, especially during inspection.

If at the beginning of the inspection a safety-oriented concept for the individual work operations had existed in this case, the BMSR shift supervisor would have had no reason to switch over the measurement range for the neutron flux measurement.

Within the control room or in the "control and protection system room" there must be no switches able to switch over the neutron flux measurement ranges. As is customary with the more recent pressurised water reactors, the switchover must take place automatically as the power of the installation is increased. In order to prevent an insensitive measurement range from being selected, there must be an interlock which initiates a reactor scram on falling below the measurement range.

9.3.10 Occurrences in the auxiliary power supply, including the emergency diesel generator set (6 kV)

The structure of the power supply of the Greifswald nuclear power station is described in Chapter 7.3.

In the period under review, no occurrence was found in which both emergency busbars were without voltage and thus all unit-associated diesel emergency power supply systems were called into service (total loss of power).

The events which occurred concerned the failure of the supply to an auxiliary power supply transformer, with subsequent automatic switchover to the standby grid supply system or - in the event of failing the cut-in to the standby grid - the activation of the associated diesel emergency power supply system for the auxiliary supply busbar affected.

- Failure of the operational feed, failure of the switchover to the standby distribution board (6 kV) and failure to start the diesel generator permanently associated with the unit emergency power supply distribution board in unit 1 on 29.4.1985 (Incident No. 104/85)

Sequence of the occurrence:

Triggering of the switches took place as a result of a double down short in the relay control system for the supply to the internal unit transformer feeding the unit distribution busses 1A and 2A. The automatic switchover of the emergency supply distribution boards 1A and 2A to the standby distribution board failed to operate. The subsequent examination of the relays and switches which accomplish this switchover revealed no indications of defects. The permanently associated diesel generator DG 1A did not switch to its emergency supply busbar BV 2A. The causes were subsequently found to be a fault in the oil pump control system for the diesel (the motor of the oil pump was protected at a rating of 15 A instead of 50 A) and a contaminated contact in the electrical path for the start control relay of the diesel. The reserve diesel DG 2 started correctly and supplied the emergency power supply busbar. As a result of further relay faults, the emergency lighting for the operation 1 and the switchover of the 0.4 kV busbar (illumination main distribution bus) to the standby supply failed.

Causes:

The relay contact failures were attributed to inadequate relay maintenance. The incorrect activation of the relay control system for the supply to the unit distribution busses occurred as a result of a double earth leak in the 220 V DC distribution system.

Backfitting measures:

The use of non-earthed power supply systems for control or protection systems means that earth leaks must be quickly detected and eliminated immediately. If an earth leak is not detected (as in the present case) and not eliminated, the possible occurrence of a second earth leak may result in incorrect operation of the control or protection system as a result of voltage doubling at the twin-fed consumers. This particularly applies to circumstances in which the DC grid is not constructed throughout with selective facilities, but where interconnections between the individual segments exist, for example, via diode-decoupled supply systems.

It is necessary to ensure that an earth leak is detected and immediately eliminated by means of appropriate monitoring arrangements.

The relays used in the control and protection system should be of a type protected against contamination (sealed relays).

Only three diesels are available for the twin 6 kV emergency power supply. The standby diesel switches either to the preselected busbar or, with a time delay, to the dead busbar, if one of the diesel fails to start. Since the emergency power consumers are arranged asymmetrically on the two busbars, a check should be made as to whether, in the event of failure of one diesel under emergency conditions, the capacity of the remaining diesels is adequate in all possible switching conditions.

9.3.11 Failures in the protected AC supply (0.4 kV)

The protected main distribution bus is available in twinned configuration for each unit. In the normal switching condition, the protected main distribution bus is supplied from the normal main supply via a thyristor power switch. In the event of failing the auxiliary power supply, the protected main distribution bus is supplied by batteries. The connection between the DC and AC power supplies is formed by reversible motor generators. In the normal switching condition, these motor generators supply the DC busbars in rectifier mode. If the AC supply fails, these generators supply the protected main distribution bus in inverter mode. Five reversible motor generators are provided for each double unit; one of these is a reserve set, which can be switched to each one of the four protected main distribution bus of the double units.

A strictly twinned configuration is not provided in the protected main distribution busses, since the consumers can be switched to both protected main distribution board busbars via the subdistribution busses. In the event of voltage failure on one busbar, this permits the consumers to be switched over automatically to the other busbar.

There is a further interconnection via the charging motor generator, one such motor generator being provided for each double unit. Feedback effects may also arise as a result of switching operations via the DC busbar (general internal DC main distribution bus) which is common to the units. Several failures of one protected main distribution busbar and one failure of both protected main distribution busbars were observed. The reversible motor generators proved to be areas of particular weakness. Accordingly, the switchover of the reversible motor generators from rectifier to inverter mode is particularly significant.

- Failure of both protected main distribution busses of unit 3 with uncontrolled shutdown of the reactor on 12.11.1988 (Incident No. 205/88).

Sequence of the occurrence:

In the course of a routine check of the reversible motor generator 1C in inverter mode, large voltage fluctuations on the protected main distribution bus 1C were detected in the unit control room. Following this finding, the reversible motor generator 1C was cut out and the protected main distribution board 1C busbar became dead, since for the purposes of this check the voltage supply from the normal main supply and thus also from the diesel emergency generators had been isolated. In accordance with the design specification, all consumers on the protected main distribution 1C busbar were switched over to the parallel protected main distribution 2C busbar. To replace the defective reversible motor generator 1C, the standby charging motor generator C should be taken into operation. However, the power switch to the protected main distribution 1C busbar had not yet been closed. At this time, a short circuit occurred in the protected subdistribution bus 31C at the protected main distribution 2C supply point, as a result of a loose terminal on the supply magnetic switch. The loose terminal was the result of an assembly error. As a result of the short-circuit, the corresponding power switch effected selective triggering for the normal main supply. In accordance with correct procedure, the power switch of the subdistribution bus was also opened via the short-circuit protection system. In this operation condition, the reversible motor generator 2C should switch over from rectifier to inverter mode. As a result of erroneous settings at the control of the motor generator, the protection system of the reversible motor generator 2C responded and this led to blocking of the thyristor 2C. The protected main distribution 2 C busbar became dead.

As a result of these events, both AC busbars were dead and the reactor was shut down via RPS 1 in accordance with the design specifications.

As a result of the voltage failure on the protected main distribution busses, all measuring instruments in the control room remained static, showing their instantaneous value. The interlocks derived from the measuring instruments were ineffective. The display of the position of the shutdown control assemblies failed. The personnel performed certain switching operations, which could be controlled only in situ. Four minutes after onset of the incident, the consumers connected to protected main distribution bus 1C were cut in again manually, and thus the normal switching condition was restored on the busbar.

As a result of the incident, the atmospheric steam dump and the condenser steam dump were actuated in consequence of the high main steam pressure.

Causes:

As a result of the twinned structure of the protected main distribution bus, the supply to a large proportion of the measurement systems and interlocks is assured only by one busbar in the course of the shutdown of one train of the protected main distribution bus which is permitted for a period of up to 2 h (conditions for safe nuclear operation 4.6). If a fault occurs on the remaining busbar during this period, and this fault leads to failure of the busbar, the result is an uncontrolled shutdown of the reactor. If both protected main distribution busses fail, the following systems become inoperative:

- the entire metrological system, and thus the entire technological emergency protection system (except for the automatic MCP system). This means that various interlocks which initiate RPS 1 are not operative

- displays of the position of shut down/regulation control rods and quick-shutoff valves in the main steam system

- as a result of all protected subdistribution busses becoming dead, all drive systems supplied from those busses can not be actuated

- the interlocks for emergency boration pumps and emergency feedwater pumps and also for the spray water pumps are not operative.

If this fault takes place in conjunction with a leak in the primary or secondary loop, it would no longer be possible to monitor the reactor, since none of the display instruments would operate. Since, in the same way, the neutron flux display in the energy and intermediate ranges is situated outside the display range and all ionisation chambers are no longer capable of operation, such faults cannot be detected by reference to reactivity changes either.

The only safety systems which are still operative are the mechanically acting pressurizer and steam generator safety valves and the hydraulic oil-operated protection systems for the turbines.

The defective control loop setting of the reversible motor generator following a short circuit in a subdistribution bus led to failure of the protected main supply. Selectivity problems also occurred in the course of other events.

To prevent the shutdown of the reversible motor generators in the event of short circuits, whether close to or distant from the generator, on consumers connected to the protected main distribution bus, tests were carried out for the purpose of optimizing the control parameters in the event of voltage drop and improving the stability of the reversible motor generators.

Backfitting measures:

In spite of the changed set parameters, failure of the protected main distribution board can still not be ruled out. The weak points are:

- double function of the reversible motor generators (these act as rectifiers under normal conditions of operation, and as inverters in the case of an incident)

- a low degree of redundancy of the system

- components of low reliability.

On account of the low degree of redundancy, no protected main distribution bus must be isolated in the course of power operation of the reactor.

It is necessary to separate the functions of inverter and rectifier, and to employ different components which are reliable for these purposes. This would involve a modification of the structure of the protected power supply: e.g. supply to the DC main distribution bus via rectifiers directly from the emergency supply main distribution bus, bypassing the protected main distribution bus, and supply to the protected main distribution bus from the DC main distribution bus via inverters.

- Multiple failures of a reversible motor generator, involving shutdown of unit 2 between 5.10 and 26.10.1980, and implementation of special switching arrangements to maintain power operation (Incident No. 258/80 and No. 270/80)

Sequence of the occurrence:

In the course of the events mentioned above, the reversible motor generator 2B in unit 2 failed on a total of 5 occasions. The cause of this was not discovered, but it was assumed that the fault lay in the excitation control system, since the abnormal occurrences indicated the appearance of overvoltage conditions. In the course of these events, the protected main distribution bus 2B failed on four occasions and at the same time the general internal DC main distribution bus A failed twice. Further effects of the overvoltage conditions were a total of four failures (triggering of fuses) in the supply units of the neutron flux measurement system (NFMS) (1 x both units, 1 x unit 1 and 1 x unit 2). On 5.10, the switches of the reversible motor generator 2B and charging motor generator A on the AC side and the DC side tripped simultaneously. As a result of this, all feed units of the NFMS unit 2 failed, and this resulted in an RPS 1 shutdown of the reactor. The standby charging motor generator A was used for the operational converting reversible motor generator 2A (reversible motor generator 2A had been taken over to unit 4). In addition, the power switch of the battery AEB A tripped and as a result of this the general internal DC main distribution A busbar became dead. Since the battery AEB A is associated with both units, four channels of the neutron flux measurement system also tripped in unit 1. Since this case did not involve 2-out-of-3 initiation, unit 1 continued to be the only unit operating in the power mode (units 3 and 4 were out of service). According to the currently applicable "conditions for safe nuclear operation", (item 4.8), both units of a double unit must be run down to minimum controllable power at maximum possible travel rate following failure of the general internal DC main distribution board. The general internal DC main distribution A busbar was dead for 12 min.

Following the 4th and 5th failure of the reversible motor generator 2B, an instruction to continue power operation of unit 2 was issued by the plant management. According to the conditions for safe nuclear operation, item 4.5, power operation with only one unit is permitted only for a period of less than 2h. A "special switching status" was created in the 6 kV internal supply system, and the unit distribution bus 2 and 6 were isolated from unit distribution bus 1 and 5 respectively and fed through the standby distribution and the associated diesel emergency generators. The diesel emergency generators were in operation for approximately 8 h in the 4th case and approximately 9 h in the 5th case. Following the 5th occurrence, the special switching condition was not cancelled until 4 days had elapsed, due to the likelihood of no lower power requirements from the grid prior to this.

Causes:

In three cases, power operation had priority over the maintenance of nuclear safety; on two occasions, "special switching status" were created in order to be able to continue power operation. In one case, unit 1 ought to have been shut down.

In view of the large number of failures of a reversible motor generator, the electronic system for initiation control in the automatic reversible motor generator panel should have been changed as a preventive measure, even without precise knowledge of the cause of the fault.

Where defects are present in one unit, the double unit system DC power supply, which is not separated into trains, may have effects on the other unit. The uncoupling of the electrical systems of the units is required as a matter of urgency.

Backfitting measures:

The nuclear safety of the installations has unconditional priority over power operation. Accordingly, it is necessary to observe the conditions for safe operation on a consistent basis.

The reversible motor generators and their control system have frequently defects. Accordingly, a comprehensive review of the protected power supply should be undertaken (refer to the occurrence on 12.11.1988: failure of both protected main distribution boards).

9.3.12 Failures in the continuous power supply (220 V DC supply)

The structure of the uninterrupted DC distribution bus is described in chapter 7.3.Within the period under review, on several occasions unreliable rotary converters gave rise to defects and failures as a consequence of the structure of the DC supply (one distribution bus being permanently associated with the unit, and the second being used by both units of the double unit) and its supply.

Two occurrences are described below.

- Failure of the DC main distribution board D in unit 4 on 5.1.1980 (Incident No. 7/80)

Sequence of the event:

The general internal main distribution bus C, which is associated with the two units 3 and 4, is supplied by two rotary converting reversible motor generators 2C (unit 3) and 1D (unit 4). The converting charging motor generator C is used as standby. Before the incident, the converting reversible motor generator 1D was taken out of service in order to be inspected and the standby converting charging motor generator C was taken into service. In the course of synchronising the charging motor generator C, excitation of the charging motor generator C was lost, and this led to the protective shutdown of the converter by the overcurrent protection system. At the same time, the power switch of the unit battery BD tripped non-selectively. Since the converting reversible motor generator 1D was out of service, the DC main distribution D busbar (unit 4) was dead. After 3 min, the DC main distribution D busbar was again supplied with voltage by reclosing of the power switch by hand. During the time interval when the busbar was dead, reactor scram was initiated by the emergency protection system RPS 1 in accordance with the design specification (criterion: voltage failure on 2 out of 3 sets to generate the emergency signal). During this period of 3 min, significant protection and safety systems were partially or entirely ineffective:

- impossible to actuate the 6 kV power switches and the 0.4 kV supply switch for all 0.4 kV main distribution busses

- failure of most displays of the settings of pumps, valves and power switches

- no initiation of trip for turbines 7 and 8 by the unit control room

- no possibility of shutdown of the main feedwater pumps

- no possibility of automatically starting the following safety systems when required: emergency boration pumps, emergency feedwater pumps and spray pumps

- failure of the electrical drive of the pressurizer safety valves (mechanical opening possible).

However, the entire monitoring system was available during the occurrence (supplied from the 380 V system).

When, after 3 min, the voltage was again available, trip of the turbines and grid isolation were initiated manually. The emergency boration pumps started automatically as a result of the low pressurizer level. According to the falling main steam pressure, the turbine actuating valves had been closed by the main steam minimum pressure controller.

Causes:

The failure of the excitation for the reserve converter was due to unreliable contact on the part of one relay (open construction, contamination).

The non-selective tripping of the power switch of the battery feed system was attributable to inadequate setting of the excess current trip device.

During the failure of the DC busbar, the emergency feedwater and spray water pumps but not the emergency boration pumps could have been cut in manually from the control room. An automatic feeding was not, in general, possible.

Backfitting measures:

The effects of the failure of a DC power supply permanently associated with the unit make it necessary to reduce the probability of such an event occurring and to minimise its possible consequences. In this connection, the following are contributory factors:

- consistent multi-train configuration, i.e. at least actual duplication of the DC power supply for each unit (separation of the general internal DC main distribution bus and installation of two batteries for each unit).

- power supply to each unit battery by means of two redundant rectifiers. This would overcome a disadvantage of the present circuit (4 reversible motor generators and a standby reversible motor generator which can be selectively cut in).

- safety-related systems must continue to be available, even in the event of failure of a DC main distribution bus; there is thus a requirement for diode-decoupled dual supply systems for consumers which have previously been supplied only from one DC main distribution bus.

- Failure of the turbine protection system for turbo generator 8 in unit 4 on 8.7.1986 (Incident No. 169/86)

Sequence of the occurrence:

Turbo generator set 8 was operated with serious steam leaks, with the result that susceptible operating, measurement and control systems (BMSR) had to be protected by covers and metal sheets. Unit 3 had been shut down for major maintenance. The unit-associated DC distribution bus C (unit 3) and D (unit 4) as well as the common DC main distribution bus, were in the normal switching conditions, i.e. there was a conductive connection between all three main distribution busses via the diode decoupling systems.

In the control room, a ground short of DC main distribution bus D, was indicated and the signal representing voltage loss in the field P 9. The fuse for the feed to the turbine protection system of TS 8 had tripped. Since repeated exchange of the fuse was not successful, an attempt was begun to locate the fault. The measurements revealed that an alternating voltage of 170 V was superposed on the supply from the DC main distribution bus D.

After this, in order to attempt to locate the ground short, the DC grids between the units were successively isolated and the ground shorts located. A ground short (minus to earth) was discovered at a relay of the control circuit of a pump in the intake structure; the second ground short (plus to earth), which led to the short circuit, occurred at a connecting cable for the overspeed protection system of turbine 8. This was attributable to the steam leaks at the turbine. After approximately 8 h, the search for the ground shorts had successfully been completed. The cause of the superposition of the alternating voltage was not discovered until the next day. It was caused by a cabling fault due to a design drawing error, as a result of which an alternating voltage (control voltage) of 220 V was superposed upon the DC supply.

Causes:

Until the isolation of the DC grids and reconnection of the turbine protection system one hour after onset of the event, the 3.5 MPa interlock of the primary main steam semi-unit (closure of the main steam quick-shutdown valves in the event of break of a main steam header line) was out of service: an infringement of the "conditions applicable to the safe operation of nuclear installations".

The search for a ground short must be commenced immediately when the ground short alarm has been given (in the present case, this took place after a time delay).

In the case of ground short alarms are given and fuses trip, it is necessary to undertake the immediate unit separation of the DC power supply (this also took place after a time delay).

Backfitting measures:

It is necessary to provide independent DC grids for each unit by using additional unit batteries. This means that, in place of the general internal DC main distribution bus, two batteries permanently associated with a particular unit are to be provided and the cross- connection between the units should be eliminated.

The disadvantages associated with grids operated without ground connections would, however, continue to exist. This event and others which have taken place underline the need for a low-tension instrumentation and control system utilising a grounding arrangement.

9.4 Recommended backfitting measures

Although some of the events which have been reviewed are also known from other plants, their multiple occurrences in the installations having been investigated does, however, necessitate technical and administrative measures as a matter of urgency. In order to achieve a lasting improvement, the following measures in the areas of plant management, process technology, electrical engineering, instrumentation and control and civil engineering are required on the basis of an evaluation of operational transients and safety-related abnormal incidents.

9.4.1 Immediate measures required

Process technology

- Improvement of the control system design in order to cope with rapid power transients

- The automatic power reduction of the reactor in the event of failure of individual main feedwater pumps should be implemented

- Automatic reactor scram in the event of total failure of the main feedwater supply

- Optimization of the feedwater control valves and the feedwater control system

- Cross-connection of the emergency feedwater systems of all four units, using remote-control and protected valves

- Refitting of locally actuated valves to enable remote control from the control room, to improve the isolation of leaks in the secondary loop (the scope and design still remain to be investigated).

Instrumentation and control

- More reliable measurement of levels in the steam generators, with the shutdown criterion: SG level < min

- The existing signals for controlling the emergency feedwater pumps are to be designed as multi-channel system

- Backfitting of a brittle break interlock for the reactor pressure vessel

- A check as to whether initiation of a reactor scram is possible when the pressurizer level "high", and possible backfitting

- A check of the safety-related sensors in the primary and secondary loops to establish that their design is accident resistant

- The logic system of the interlock and drive of the MCP should be examined, in order to ensure that no more than 1 MCP can be shut down

- Installation of a continuously operating boric acid measuring instrument

- Backfitting of automatic measures to maintain the shutdown reactivity, e.g. quantity limits for the feed of demineralised water into the primary loop

- Installation of a leak detection system

- Should with the present technology an automation for the unlocking switches - such as start-up unlocking systems - not be realizable, it would also be feasible to make use of key switches which are locked under normal operating conditions. In addition, for each unlocking switch there should be a clear display on the operating room panel, so that the operating personnel knows at any time which switches have been unlocked. This solution would be meaningful only if the operator teams can obtain the keys only when required through the safety engineer on duty. With that a situation would be prevented in which the operating team inhibits automatic protective actions by unlocking initiating criteria during a transient phase

- Automatic switchover of the measurement range of the neutron flux measurement system on starting up the reactor. By way of an interim solution to the problem, it would be feasible to undertake backfitting for an RPS 1 trigger at a "lower" limiting value for the neutron flux (this value has to be determined).

With regard to the scope and complexity of the recommended backfitting measures in the area of instrumentation and control, a check should be made as to whether this procedure can be implemented with the instrumentation and control system existing in the Greifswald nuclear power station (220 V DC for the control system using relay technology, and 380 V AC for the measurement system) with an adequate degree of reliability.

If the existing technology is retained for a transitional period, an independent accident instrumentation system will be necessary.

Electric system

- Examination for an adequate degree of protection for electrical installations (motors, subdistribution boards, sensors) in the machine hall

- Specification of realistic power balances for the emergency power supply systems and, where necessary, implementation of required measures (e.g. 4th diesel, and an increase in the battery capacity)

- Strict train separation in the area of the protected 220 V DC distribution bus by using two batteries and busbars permanently associated with a particular unit (cross-connections to the adjacent unit only in an emergency case)

- Replacement of the unreliable reversible motor generators by separate inverters and rectifiers

- Reduction of the interconnection of the emergency power supply distribution bus on the 380 V voltage level, and, as far as possible, fixed allocation of the consumers to the main distribution bus

- An adequate standby of inverters and rectifiers to allow the isolation of one set for repair or inspection.

- Improvement in the detection and localisation of ground shorts

Plant management, administrative measures

- Regulation of transport operations using the hall crane over operative parts of the installation

- Revision of the operating manuals to provide unambiguous rules applicable to all operational actions

- Provision in the operating manuals of a clear outline of the switching condition and of the process variables applicable to outage or to the operation of installed systems

- Improvement of the systematic evaluation of operating experience

- Improvement of the internal organisation of operations in order to optimise the feedback of experience

- No exchange of shifts between the individual units without thorough training being given in advance

- Introduction of an appropriate work instruction and isolation system in order to improve the coordination of operations between a shift and the radiation protection, fire protection, maintenance and technical departments

- Immediate shutdown of the unit in the event of failing one of the two protected main distribution busses

- Revision of the operating instructions with regard to

  - improvement of the examination procedures after maintenance
  - optimization of the maintenance system and of the planning of revisions
  - implementation of measures for modifications at safety-related systems, to ensure that these modi-

fications will be thoroughly prepared, approved, and tested after having been performed.

- the prevention of the unlocking of protective criteria in the emergency protection system during power operation with the purpose of avoiding protective actions
- the operational status of the safety-related instrumentation required to monitor any condition of the plant (even during inspection and start-up) has constantly to be maintained.

9.4.2 Long term measures required

Process technology, civil engineering

- Installation of an independent emergency feedwater supply
- Prevention of erosion corrosion in the secondary loop, by various means including optimized pipe installation
- A check as to whether separation of the machine halls is possible

Instrumentation and control

- Replacement of the existing instrumentation and control system by a low-voltage instrumentation and control system including grounding
- Unlocking switches should be removed from the control room. Any required unlocking functions should, where possible, be automated (start-up interlock)

Plant management, administrative measures

- Revision of all operating manuals

10. SUMMARY

10.1 Assessment of the results

In assessing the safety-related design of the system and the protection of the installation against redundancy overlapping events, the first interim report which was published in February of this year contained no more than initial estimates. In that publication, safety-related deficiencies were found in differing degrees in almost all areas, in comparison with current safety requirements in the Federal Republic.

This second interim report contains the findings of the continued safety investigations. In the course of these investigations, more detailed tests were carried out on the safety deficiencies which had previously been found, concentrating on the safety-related design of the system and the evaluation of operational experience. The investigations covered the question of to what extent the existing deficiencies can be compensated by safety- related improvements, improved organisational/administrative regulations and the safety-related advantages of the installation.

It is revealed that the evaluation of operational experience is of prime importance. The frequency of immediately obligatory reporting safety-related incidents has actually diminished during the last five years, but it has also been discovered that operating specifications do not comply with international requirements in the differentiated description of specific actions. In addition to this, serious infringements of the operating specifications over a period of 10 years up to very recent times have been established. Relevant examples are the following:

- inadequate monitoring of components when maintenance has been completed

- incomplete testing of alterations made to important safety-related systems

- bypassing of technological criteria (excitation conditions) applicable to the reactor protection system during power operation, for the purpose of avoiding protective actions

- incomplete operational capability of the safety-related instrumentation required to monitor the condition of the installation during inspection and start-up

- continued operation of the installation at full power with a leak in the primary loop purification system outside the pressure resistant compartment system.

In addition, defects have been found in the systematic processing of operational experience and considerable deficiencies in the utilisation of acquired knowledge in the practical operation of the installation.

Taking an overall view, it is revealed by the evaluations that there are substantial defects in the safety philosophy of the Greifswald nuclear power plant. Accordingly, it is necessary for all persons involved to implement measures, without delay, which will prevent the repetition of such infringements of correct proceeding. Furthermore, the operational structure concerned with production, maintenance, monitoring and technology should be reviewed. In order to achieve an improvement, it is necessary to assess the experience and practices derived from nuclear power stations in the Federal Republic of Germany. Participation in WANO (World Association of Nuclear Operators) should also be fully utilised in this connection.

10.2 Recommended backfitting measures

The deficiencies revealed in the course of the investigations concerned with safety-related design matters indicate the need for backfitting measures in almost every area. This is applicable both to technical alteration and reconstruction of systems and installations and also to operational organisation and plant management.

Among the backfitting measures derived as a whole from the investigations, a distinction is drawn between three categories:

Category I: Measures which are immediately necessary for achieving an adequate level of safety for continued operation in the near future (approximately 2 years)

These measures are derived from operational experience and from investigation of technical systems, where initiating events are not controlled with an adequate degree of reliability by means of the existing system technology and the intended mode of operation, having regard to their probability of incident. It is necessary to implement these measures immediately.

Category II: Measures which are required for achieving an adequate level of safety for continued operation over a fixed period

These measures should be implemented as quickly as possible - at the latest within two years after resumption of operation - in order to enable continued operation of the installation over a fixed period.

Category III: Measures which are required for achieving an adequate level of safety for long-term continued operation

These measures are considered necessary to bring the safety status of the installations up to the international level. They must be derived from comprehensive safety investigations, including the specification and analysis of the design basis accidents to be adopted as a starting point.

Within each category, the measures have been compiled under the following headings:

- Process engineering
- Instrumentation and control
- Electrical engineering
- Plant management, administrative measures
- Buildings
- Pressurized components
- Fire protection
- Flood protection.

When undertaking the detailed planning and preparatory work for the individual measures, it will be necessary to examine carefully the question of whether the safety of the installation will in any way be impaired by the introduction of the measures.

The results and conclusions of the investigations were discussed with Soviet experts from the Ministry for the Atomic Energy Industry, the Kurchatov Institute, the designer and the manufacturer of the installation on 22 and 23 May 1990 in Moscow. On that occasion, the Soviet side presented a comment which to a large extent agrees with the assessments made in the report. However, where certain individual measures are concerned, differing views continue to exist concerning allocation to particular categories. This comment is attached as an annex to the second interim report.

The next part of the present document contains a detailed compilation of all technical measures which have been derived from the investigations to date and their allocation to categories I, II and III. In this compilation, the measures have been identified by stating the chapter in which they were listed.

- Category I

  - Process engineering

1. Replacement of the EP 50 pumps by ZN 65 pumps in dependence upon the analyses concerned with the control of the leak of nominal diameter 32 with one pump and the leak of nominal diameter 200 with four pumps, possibly necessary only in the relatively long term (chapter 5.6; 7.6.2)

2. Emergency power supply and use of the reactor filling pump P 36 and the fuel cooling installation pump P 17 for low-pressure long-term injection. To this end, it is necessary to determine the delivery of the pumps by experiment (chapter 7.6.2)

3. Emergency boration tank heating independent of the emergency cooling system (chapter 7.6.2)

4. Assurance of the environmental conditions in accordance with the design specification for the emergency cooling and sprinkler-pumps (isolation

of pipes using heated emergency cooling water) (chapter 7.6.2)

5. Establishment of a cross-connection with shutoff device, on the pump pressure side, to the emergency cooling system of the adjacent unit (within one double unit) and of a spatially separated emergency cooling water return pump (chapter 7.6.2)

6. Prevention of the blockage of the return line from the sump by improving the sump cover (chapter 7.6.2)

7. A cross-connection with shutoff device between the emergency feed pressure lines of units 1 to 4, possibly with remotely controlled and protected valve drives with additional hose connections (chapter 5.6; 7.6.3; 9.3)

8. Provision of an injection facility to feed the steam generators, e.g. a facility for injection into the steam generator blowoff lines via a pipe spatially separated from the 14.7 m platform, likewise with additional hose connections (chapter 5.6; 7.6.3)

9. Demineralised water pump with self-sufficient drive for the emergency water supply to the steam generators from the demineralised water tanks (chapter 5.6; 7.6.3)

10. Use of the facility for injection from the feedwater tank by pressure gradient (chapter 5.6; 7.6.3)

11. A cross-connection with shutoff-device, on the pump pressure side, with a connection facility for mobile pumps of the service water systems of both double units (chapter 7.6.4; 8.3)

12. Structure drainage pump for the intake structure, with automatic activation in the event of a water incursion (chapter 7.6.4)

13. Improvement of the diesel cooling by provision of an independent cooling system (chapter 7.6.7)

14. Conversion of the cooling of at least two spare transformers to air cooling (chapter 7.6.7)

15. In order to prevent the direct release of primary coolant into the reactor hall (bypass of the pressure resistant compartment system), it is necessary to provide a design solution for the separation of the seal of AWT 1 and the pressure resistant compartment system in the area of the filling connections of the ion exchangers (chapter 7.3.6)

16. The number of isolating valves (air exhaust valves) of the ventilation plant W 2 which are open during operation should be reduced to the minimum number required to maintain the reduced pressure in the compartments of the pressure resistant compartment system. The other dampers should be closed and secured before the installation is put into service; the tightness thereof should be checked. The closing of the valves should take place in such a manner that the ventilation of the compartment system is possible after beginning of an accident. The open

isolating valves should be protected against penetration by foreign bodies, which may obstruct their closure (chapter 6.5.1; 7.6.5; Soviet Union)

17. Transmitter rooms in which the accident instrumentation has been installed should be protected from the action of steam by additional quick-closing valves (chapter 7.6.5)

18. The actual rate of leakage of the pressure resistant compartment system at start-up of the units should be drastically reduced by intensifying the maintenance work or by additional seals. The value of approximately 300 m^3/h was achieved already at a test pressure of 1.25 bar in unit 1 (chapter 6.5.1)

19. Performance of analyses concerned with the formation of cold water strands, based on the study carried out in Finland on the LOVIISA installation, principally concerned with asymmetric subcooling transients (chapter 5.6)

- Instrumentation and control

1. Automatic operating of the spare pump and of the motor-driven valve in the third sprinkler line in the event of failure of a pump or failure of a valve to open

2. Implementation of measures to improve the independence and monitoring of the power circuits to operate safety systems, to a large extent based

on the single failure criterion, especially with regard to operate the emergency feed pumps (chapter 7.6.6; 9.3.3; Soviet Union)

3. Establishment of an accident sequence recording system

4. Provision of a self-sufficient accident instrumentation system in the transmitter rooms spatially separated from the control room, which records at least primary pressure, primary temperatures, pressurizer level and steam generatorlevel, as well as selected radiological values (chapter 7.6.6; 9.3.9.)

5. Testing of the safety-related tansmitter in the primary and secondary loops for fault-resistant design (chapter 9.3.6)

6. An interlock to ensure protection against brittle breach at temperatures below the brittle fracture transition temperature during start-up and rundown procedures and tightness tests (chapter 4.3; 7.6.6; 9.3.4)

7. The automatic reduction of power of the reactor in the event of failure of individual main feedwater pumps should be implemented (chapter 9.3.2)

8. Automatic reactor scram in the event of total loss of the main feedwater supply (chapter 7.6.1; 9.3.2)

10. More reliable measurement of levels in the steam generators, using the shutdown criterion: SG

level in at least one steam generator < min (chapter 5.6; 7.6.1; 9.3.3)

11. A check as to whether initiation of the emergency protection system of the reactor is possible when the pressurizer level is "high", and eventual backfitting (chapter 9.3.4)

12. Optimization of the limiting values relating to the criterion "pressurizer level low" for shutdown of the reactor and for the activation of the emergency cooling pumps (chapter 5.6; 7.6.6.)

13. Optimization of the limiting values relating to the criterion "pressure in primary loop high" (reduction of the pick-up value for RPS 3 and introduction of an additional RPS 2 criterion) (chapter 5.6)

14. Improvement of the power regulation system to run down the installation in the event of failure of one or both turbo generator sets (grid failure) to the appropriate power level, where possible without response by safety systems (reactor shutdown, emergency cooling) (chapter 7.6.6; 9.3.1)

15. Installation of further water indicators in the emergency cooling pump room and in the intake structure of the service water system (chapter 7.6.6)

16. Impermissible switching conditions of the MCPs with respect to the permissible reactor power are to be signalled (chapter 7.6.6; 9.3.7)

17. Backfitting of automatic measures to guarantee the shutdown reactivity (chapter 9.3.8)

18. If it should prove impossible to provide an automated unlocking system such as, for example, start-up unlocking using current technology for the unlocking switches, it would be feasible to use key switches which are locked under normal conditions of operation. In addition, a clear display should be provided on the control room panel for each unlocking switch, so that the operating personnel know at any time which switches have been unlocked. This solution would be only advisable if the teams of operators can obtain the keys only through the safety engineer on duty when required. This would prevent a situation in which the operating team inhibits automatic protective actions by unlocking initiating criteria during a transient sequence (chapter 7.6.1; 9.3.9)

19. Automatic changeover of the measurement ranges of the neutron flux measurement of the reactor. By way of an interim solution, it would be feasible to implement backfitting for an RPS 1 initiation at a lower limiting value for the neutron flux which has not yet been specified (chapter 9.3.9)

20. Improvement of the detection and location of leaks for all areas not subject to large leaks or breaches (chapter 4.3; 7.6.6; 9.3.5)

- Electrical engineering

1. Elimination of the couplings between the unit

distribution boards 3 and 4, as well as between unit distribution boards 7 and 8, and permanent allocation of the MCP 3 to distribution board 3 and MCP 2 to distribution board 7 (chapter 7.6.7)

2. Formulation of realistic power balances for the emergency power supply systems and, where appropriate, derivation of measures required (e.g. a fourth diesel set, and an increase in the battery capacity)(chapter 7.6.7; 9.3.10)

3. Strict train separation in the area of the 220 V protected DC distribution board by using two unit-associated batteries and busbars, cross-connections to the adjacent unit only in an emergency (chapter 7.6.7; 9.3.10; 9.3.12)

4. Replacement of components which are susceptible to trouble or which require costly maintenance, such as motor generators operating in reversible mode (use of static inverters and charging rectifiers) (chapter 7.6.7; 9.3.11)

5. Extensive uncoupling of interconnections of the emergency power supply distribution boards at the 380 V voltage level (chapter 7.6.7; 9.3.11)

6. An adequate reserve of inverters and rectifiers in order to permit the isolation of one set for repair or inspection (chapter 9.3.11; 9.3.12)

7. Improvement of the spatial separation of important safety-related redundant cables, or equivalent measures (chapter 7.6.7; 8.2.5)

8. Installation of a separate, additional cable route for power supply to the emergency cooling pumps and emergency feed pumps (Soviet Union)

9. Examination and possible improvement of the switch-on programme of the emergency power supply consumers with regard to the priority of the manual commands, the blocking of automatic measures concerned with safety-significant consumers during diesel operation, and the conditions for the termination of diesel operation (chapter 7.6.7)

- Plant management, administrative measures

1. Introduction of a fifth shift, exclusively for qualification and training measures (chapter 7.6.8)

2. The permanent presence of an experienced, highly qualified safety engineer on the site to assist the shift personnel in situations beyond the design plan (chapter 7.6.8)

3. Establishment of an emergency call standby system for specialist engineers and provision of appropriate alarm systems (chapter 7.6.8)

4. Revision of the accident operating manual regard to ergonomic design principles, with the objective of quickly providing clear practical aids to assist the control room personnel, and the inclusion or clarification of instructions regarding specified incidents, e.g. detection of and behaviour in the event of the incident of leaks or impending grid breakdowns (chapter 7.6.8)

5. Formulation of instructions concerning the handling of accidents which do not proceed in accordance with the design specification, such as a steam generator leak with failure of the primary-side isolation of the defective steam generator; failure of emergency cooling systems in leak-induced accidents or their auxiliary systems, such as the service water system and intermediate coolant loops; leaks in connecting lines of the primary loop, which are situated outside the pressure resistant compartment system; failure of the secondary-side heat removal system (chapter 7.6.8)

6. Regulation of transport operations above parts in service of the installation, using the hall cranes in the reactor hall, the machine hall and above the intake structure (chapter 7.6.8; 8.3; 9.3.6).

7. More frequent inspection (several times per shift) to the service water system, especially after maintenance (chapter 8.3)

8. In general, maintenance work which involves opening pipes involving a flood hazard should not be carried out during power operation (chapter 8.3)

9. Provision of a clear display of the switching status and the process variables relating to standstill or the operation of systems in the operating manuals (chapter 9.3.8)

10. No exchange of shifts between the individual units without thorough training being given in advance (chapter 9.3.8)

11. Improvement of the concept for work order and enabling of equipment in order to achieve improved coordination of operations between a shift and the radiation protection, fire protection, maintenance and technical service, especially with regard to safety-related components and equipment (chapter 9.3.9)

12. Immediate rundown of the unit in the event of failure of one of the two protected main distribution boards (amendment to the conditions applicable to the safe operation of nuclear installations) (chapter 9.3.11)

13. The labelling of wall and console panels in the control room should be carried out on a standardized basis. To achieve this, a standardized plant identification code should be used. Labels should be readily legible, and clearly correlated with the component symbols (chapter 7.6.8)

14. Comparison of the requirements regarding in-service inspections (material and functional tests) between the test programs of the USSR, GDR and FRG (SU)

- Pressurized components

1. Before being taken into operation again, the reactor pressure vessels of units 2 and 3 should be subjected to a heat treatment to achieve substantial restoration of their ductility properties (chapter 4.1.2)

2. In the course of switching operations for the purpose of locating a leak in the primary loop, it may happen that the emergency cooling system is supplied only in three adjacent loops. Additional efforts should be made to calculate the load on the reactor pressure vessel in such a case, and the calculation should be presented for assessment (chapter 4.2)

3. The criterion applicable to lock damaged steam generator tubes should be reformulated; not only the depth of the defect but also the length of the defect should be taken into account. More comprehensive determinations of the rate of leakage are required (chapter 4.3)

4. An analysis should be carried out regarding the possibility and consequences of leakage phenomena, including their consequences for the main steam and feedwater lines as well as for the vessels situated within these systems (chapter 4.3)

- Fire protection

• Early fire detection

1. For the purposes of the early detection of fire hazards, the following areas of the installation should be inspected frequently (at least twice per shift); compartments containing electrical engineering equipment (E 103, E 105, E 107, E 108, E 113, E 114); cable chutes and cable routes, machine hall, the "B" line of columns on level -3.60 m, B 003, A 008, cable routes in the machine hall (chapter 8.2)

2. Automatic fire alarm systems should be installed in the oil supply areas of the turbo generator sets and the feedwater pumps, as well as in the compartments occupied by the electrical main distribution boards including the main service station DC distribution board (chapter 8.2)

3. The pressure drop interlock in the hydrogen system of the generator should be checked and, if required, backfitted (chapter 8.2)

- Passive fire protection

4. Fire walls in cable chutes do not have the required fire resistance 90, and appropriate backfitting measures should therefore be implemented (chapter 8.2)

5. In all other areas the fire walls should be assessed immediately, and where the fire resistance 90 level is not reached measures should be taken to increase the fire resistance (within category II). Particular attention should be paid to the walls below switching systems and within the cable floor (chapter 8.2)

6. The flanges of the oil pressure lines in the feedwater zone should be protected by caps, so that if oil leaks occur in these zones any uncontrolled spraying of oil is prevented (chapter 8.2)

7. In order to reduce the probability of an oil fire occurring in the turbine zone, measures should be taken to prevent penetration by leaking oil from the turbine shaft bearings into isolations and into the cable duct situated beneath the turbine (chapter 8.2)

8. The power and control cable tracks leading to the emergency feedwater pumps in the area of the machine hall should be protected against the effects of fire on the cable route by having a fire resistance of at least 30. Short circuits, which may lead to the ignition of the cables themselves, should be prevented by selective fusing (chapter 8.2)

9. A fundamental requirement following the assessment of the technical aspects of the system is the improvement of the spatial separation of specified safety-related redundant systems and their connection by cables. An immediate measure should be the fireproof separation of the protected DC supply. The fire protection measures which are required in this connection should be decided upon following a detailed on-site inspection (chapter 8.2)

- Fire-Fighting

10. The acceptance parameters (e.g. discharge heights) of the fire-fighting water system should be checked for accordance with TGL 10685 and if deficient, backfitting measures should be implemented (chapter 8.2)

11. It should be ensured that a fire-induced failure of the power supply to all pumps of the common drinking water, utility water and fire-fighting water system cannot occur; where appropriate, provision should be made for a commensurate and efficicent supply facility for this water system (chapter 8.2)

12. The probability of incident of a fire and the effects of a fire in the cable floor (E 102) should be reduced by protection against unauthorised access to the cable floor and by administrative and operational measures concerned with mobile fire-fighting, e.g. provision of special mobile extinguishers on the spot (chapter 8.2)

13. Water spray systems should be installed in the oil supply areas of the turbo generator sets and the feedwater pumps (chapter 8.2)

14. In the case of all compartments in which water is provided as an extinguishing medium (fixed systems as well as mobile firefighting), it should be ensured that no consequential failures of important safety-related systems in adjacent areas can occur as a result of water spray and the quantity of water employed for firefighting purposes.

- Category II

  - Process engineering

1. Installation of a fast-acting boron poisoning system (dependent upon ATWS analyses) (chapter 7.6.1)

2. Cessation of the use of sealed motor-driven valves which must open when required. In this regard, it must be ensured that protection against leaks from the primary loop is not impaired (chapter 7.6.2)

3. Provision of a building drainage pump for the emergency cooling pump room with location of leak position by determination of the origin of the leaking water (service water or emergency cooling water) (chapter 7.6.2)

4. Conversion of the cooling of at least two spare transformers to air cooling (chapter 7.6.7)

5. Incorporation of other water reserves, e.g. for remote heating, drinking water, utility water and fire-fighting water system, in the feedwater supply system in an emergency (chapter 7.6.3)

6. Improvement of the ventilation of the battery compartment in order to reduce the temperature of the compartment (chapter 7.6.7)

7. Optimisation of the feedwater control valves and of the feedwater control system (chapter 9.3.3)

8. Backfitting of locally actuated valves to enable remote control from the control room to obtain better isolation of leaks in the secondary loop (the scope and implementation of such refitting still remain to be examined) (chapter 9.3.6)

9. Measures to shorten the closing times of the ventilation valves (especially of the air exhaust flap valve of the installation W-4, nominal diameter 1,000) should be adopted (chapter 6.5.1)

10. To date, the tightness tests on the pressure resistant compartment system have been carried out only at a test pressure of 1.25 bar. In order to clarify the pressure dependence of the leak rate, tightness tests at the maximum permissible test pressure should be carried out at least on one unit (chapter 6.5.1; SU)

11. It is necessary to demonstrate the adequate reliability in operation of the dampers of the pressure resistant compartment system (chapter 6.5.1)

12. The tightness and loading capacity of the penetrations through the walls of the pressure resistant compartment system (e.g of the cable alignment sections) should be tested at design pressure and design temperature (chapter 6.5.1)

13. Evidence should be furnished, showing that the long-term decay heat removal from the pressure resistant compartment system is assured by one sprinkler cooler. Should this not be the case, the redundancy of the coolers should be increased (chapter 6.5.1)

14. Extension of the design basis loss of coolant accident to cover the breach of the largest connecting line, with due consideration of total loss of power and single failures; in order to continue operation for a short period, it is necessary to show that under "best estimate" conditions the second design limit value according to Principles of Nuclear Safety 1982 (OPB-82) will not be exceeded as a result of leaks extending as far as breach of the largest connecting line (chapter 5.6)

- Instrumentation and control

1. Installation of a power density distribution monitoring system including an in-core instrumentation system (DNB signalling) and the repaired fuel assembly outlet temperature measurement system (chapter 7.6.6; SU)

2. Installation of continuously operating instruments for boric acid measurement (chapter 9.3.8)

3. A check should be made as to whether the administrative measures for ensuring that there are a sufficient number of rundown-protected main reactor coolant pumps ("table of permissible reactor power levels") should be replaced by electrical interlocks (chapter 7.6.6; 9.3.7)

4. The section between the first and second points of isolation of connecting lines leading to the primary coolant loop which are equipped with motor-driven valves should be monitored for leaks at the first point of isolation (chapter 7.6.6)

5. Reduction of the set protection criteria applicable to the diesel emergency power generation sets which under emergency conditions initiate shutdown of the set (chapter 7.6.7)

6. Installation of the RPS 3 criterion "activity in the main steam line high" (chapter 5.6)

- Electrical engineering

1. Improvement of the detection and localisation of ground faults in the DC grid (chapter 7.6.7; 9.3.10)

2. Testing for an adequate degree of protection of electrical systems (motors, subdistribution boards, transmitters) in the machine hall (chapter 8.2.2; 9.3.6)

3. Provision of a monitoring system for charging batteries (chapter 7.6.7)

4. Backfitting of synchronising systems for the diesel generator switches (chapter 7.6.7)

- Plant management, administrative measures

1. Testing of the knowledge of the control room personnel in the presence of the supervisory authority, with participation, if considered appropriate, of independent experts (chapter 7.6.8)

2. The system for monitoring the settings of the manually operated lock valves in the connections of low-pressure systems leading to the primary

loop should be designed for reliability (e.g. by using a key system) (chapter 7.6.8)

3. Review of the operating manuals with regard to unambiguous rules for all operating actions (chapter 9.3.8)

4. Improvement of the systematic evaluation of operating experience, with feedback of the experience gained (chapter 9.3.8)

- Pressurized components

1. In principle, the use of shielding assemblies is recommended for the operation of the reactor pressure vessels of all units (chapter 4.3)

2. The gain in safety level achieved by enclosing the emergency boration feed system in the hot leg should be analysed with respect to the operation of the reactor pressure vessels of all units (chapter 4.1.3; 7.6.2)

3. The loadings applicable to the reactor pressure vessel and other components on which the safety analyses were based should be supplemented on the basis of specified requirements (chapter 4.1; 4.2)

4. Appropriate test programs should be implemented in order to specify the nonsteady cyclic temperature load

   - at the spray line of the pressurizer injection unit and
   - at the connections of nominal diameter 200 leading from the pressurizer to the primary

coolant loop (media pushing and, where appropriate, temperature stratification) (chapter 4.2)

5. In order to supplement the calculations relevant to the design specification, displacement measurements should be made

   - at the primary loop
   - at the main steam line and
   - at the feedwater line

   (chapter 4.2)

6. The conditions applicable to the performance of tightness tests at a high pressure level and at temperatures exceeding 100°C should be reviewed (chapter 4.3)

7. Analyses should be presented regarding effects of accidents during a pressure test, including any possible consequential damage to adjacent installations (chapter 4.3)

8. The surface crack examinations on the internal surfaces should be carried out, in order to obtain further information regarding the nominal diameters 100 and 200 and the results of ultrasonic tests on the primary coolant loop (chapter 4.3)

9. Appropriate loads should be specified for the leak-before-break analyses, e.g. including the closure of the main isolating slide valves etc. (chapter 4.3)

10. A check should be made as to whether the corrosion problems on pressurizer tubes, pipes and

vessels can be solved by sealed condensers and adjusted water chemistry (chapter 4.3)

- Fire protection

• Early fire detection

1. In order to provide a covering automatic fire alarm system in all compartments involving a fire hazard, backfitting is required in the following areas of the installation: compartments containing electrical equipment (E 103, E 105, E 107, E 108, E 113, E 114); cable chutes and cable routes in the compartments, or areas of compartments, the "B" line of columns in the machine hall on the -3.60 m level, B00 3, A00 8; cable routes in the machine hall (chapter 8.2)

2. Manual initiation should also be provided for all areas (chapter 8.2)

3. The possibility of using hydrogen detectors in the machine hall should be examined (chapter 8.2)

• Passive fire protection

4. In order to prevent the propagation of a fire in compartments with electrical equipment, the steel doors in existence in such compartments which are not fire resistant should be replaced by appropriate fire doors (chapter 8.2)

5. Measures to increase fire resistance should be adopted in all areas in which the fire walls do not achieve the fire resistance level 90 according to the assessment which has been made. Particular attention should be paid to the

separations beneath switching systems and in the cable floor (chapter 8.2)

6. The cable route along the "B" line of columns on level -3.6 m in the machine hall should be protected against oil fires and the action of steam in sections subject to a fire and steam hazard (at least in the area of the feedwater platforms and the high-pressure preheaters) (chapter 8.2)

7. In order to restrict the propagation of heat in the event of oil fires in the machine hall to one unit, it is necessary to provide smoke exhaust and heat removal facilities and appropriate protection measures in the roof truss area (chapter 8.2)

8. The protection of the safety-related systems (main steam and feedwater lines, emergency feedwater pumps) in the machine hall, as well as the protection of the adjoining cable floors in the frontage area should be further investigated with regard to the effects of a large-scale fire in the machine hall or of an H_2 explosion at the generator, taking into account consequential events, such as the collapse of the roof truss (chapter 8.2)

9. Having regard to the possibility of ignition of cables, a thorough check should also be made, as part of the investigation of the technical aspects of the system, as to whether short circuit currents in power cables (those which penetrate fire walls) can be prevented with a sufficient degree of reliability (chapter 8.2)

- Fire-fighting

10. In order to achieve rapid and reliable fire-fighting in all compartments involving extensive fire hazards, the backfitting of water spray installations is necessary in the following areas of the installation: cable chutes and cable routes (service center DC main distribution board, machine hall, E 102, E 104) (chapter 8.2)

11. The emergency power supply to the pumps of the common drinking water, utility water and fire-fighting water network must be assured in accordance with the rating (chapter 8.2)

12. As a result of the safety-related significance of the cable floor, the installation of a fixed gas extinguishing system is considered necessary (chapter 8.2)

- Flood protection

1. Further investigations should be carried out regarding the possible penetration by water into the reactor well with external wetting of the reactor pressure vessel, and, if appropriate, effective countermeasures must be formulated (chapter 8.3)

- Category III

  - Process engineering

1. Improvement of the capacity, of the degree of redundancy and of the spatial separation of the

emergency cooling and sprinkler systems (hydraulic accumulator, low-pressure and high-pressure systems, second sump drain) (chapter 6.5.2; 7.6.2)

2. Improvement of the capacity, of the degree of redundancy and of the spatial separation of the emergency supply and main steam dump systems, and of the required auxiliary systems (chapter 7.6.3; 9.3.2; 9.3.6)

3. Improvement of the degree of redundancy and the spatial separation of the service water system and ICC-NPS (chapter 7.6.4)

4. Matching of the auxiliary systems of the diesel sets to the redundancy and circuitry of the main sets (chapter 7.6.7)

5. Reduction of erosion-corrosion in the secondary loop, inter alia by optimised pipe-tracking; prevention of pipe failure by optimised periodic testing (chapter 9.3.6)

6. The lock valves for pipes which are connected to the primary coolant loop and penetrate the pressure resistant compartment system should be designed with redundancy (chapter 6.5.2)

7. Following accidents involving loss of coolant, the formation of combustible gas mixtures (hydrogen) in the pressure resistant compartment system must be prevented. Investigations should be carried out into the limitation of local hydrogen concentrations, and any required backfitting measures should be implemented (chapter 6.5.2)

8. Within the terms of a comprehensive backfitting, it will be necessary to demonstrate adequate emergency cooling (no exceeding of critical values for evacuation) to cover the case of the breach, at both ends, of a main coolant line (chapter 5.6)

9. Twin locking devices for all ventilation lines using quick-closing flap valves (chapter 7.6.5)

- Instrumentation and control

1. Replacement of the existing instrumentation and control system with due regard to the applicable rules and guidelines (chapter 7.6.6; 9.3.9; 9.3.12)

2. Unlocking switches should be removed from the control room. Where possible, any required unlocking mechanisms should be automated (start-up interlock) (chapter 9.3.9)

- Electrical engineering

1. Examination of possibilities for improvement to the grid supply with regard to functional and spatial independence (connection to the power grid network, main and reserve grid connections, independent grid power supply where an emergency power supply is required, grid coupling between the 380 kV and the 220 kV grids) (chapter 7.6.7)

2. Improvement of the capacity, degree of redundancy and spatial separation of the emergency power supply system (chapter 7.6.7)

- Plant management, administrative measures

1. Revision of all operating manuals (chapter 9.3.8)

2. Consistent application of ergonomic principles within the terms of the backfitting of the instrumentation and control system (chapter 7.6.8)

3. Matching of the existing training simulator to the specific conditions applicable to the WWER-440/W-230 reactor type, especially with regard to accident training (chapter 7.6.8)

- Buildings

1. Investigation of whether it is possible to subdivide the machine hall (chapter 9.3.6)

2. Analysis of the dynamic stresses of the pressure resistant compartment system and safety systems installed therein (shock waves, jet forces, flying fragments etc.) as well as of possible consequential damage upon breach of connecting lines up to nominal diameter 200 (2F); implementation of any required backfitting measures (chapter 6.5.2)

3. For an expanded accident spectrum going as far as breach of a line of nominal diameter 200 (2F), it is necessary to demonstrate the required tightness of the pressure resistant compartment

system. Any measures required to increase the effectiveness of the tightness should be implemented (chapter 6.5.2)

4. In order to control an expanded accident spectrum (leak > nominal diameter 32), it is necessary to guarantee the operation of the pressure resistant compartment system : rapid pressure relief in conjunction with reliable, early closure of relief valves. Any required measures in the area of the dampers should be prepared and implemented (chapter 6.5.2)

- Pressure boundary

Measures under category III have not yet been analysed.

- Fire and flood protection

1. As a result of unalterable constructional and technical features, as far as comprehensive effects are concerned, priority should be given to the construction of a separate emergency system complying with the current safety standard with regard to fire protection (emergency feedwater supply, service water system, where appropriate, significant safety-related emergency cooling functions, emergency control room, reactor protection with an instrumentation and control system, power supply) in its own building. This system should be integrated into the existing installation at an appropriate point, so that the fire protection backfitting in the existing building may be restricted to a few compartments (chapter 8.4)

10.3 Continuation of the investigations

At the current stage of the work, it is not possible to make a conclusive assessment of all safety questions relevant to the Greifswald nuclear power plant, units 1-4. In order to answer a number of individual questions, it will be necessary to examine further documents. Because of the limited time available, it has not yet been possible to address a number of problems and topics. This applies, for example, to investigations concerning the strength of safety-related building structures. In addition, it will be necessary to carry out supplementary investigations concerned with behaviour of materials, dynamic analyses concerned with the effectiveness of safety systems and more detailed evaluations of existing operational experience.

It is intended that investigations should be carried out in a joint GRS-SAAS program of work, with the participation of Soviet and French experts.

The objective of the continuing investigations is to monitor the realisation of the measures classified in categories I and II, and/or to provide a more detailed specification of these measures. In addition to this, the intention is to supplement and to complete the measures set out in category III by means of these investigations and to assess the level of safety which can be achieved if those measures are implemented.

ANNEX 1

Comments by the Soviet experts on the Second Interim Report on the Safety Assessment of the Greifswald Nuclear Power Plant Units 1-4 (WWER-440/W-230)

Comments by the Soviet experts on the "Second Interim Report on the Safety Assessment of the Greifswald Nuclear Power Station, Units 1-4 (WWER-440/W-230)"

The investigations which were carried out for the purposes of the safety assessment of the Greifswald nuclear power plant, units 1-4, were discussed at various working sessions with the participation of Soviet and French experts. Detailed discussions took place, in particular, on 3 and 4 May 1990 in Berlin and on 22 and 23 May 1990 in Moscow, involving Soviet experts, representatives of the Ministry for the Atomic Energy Industry, the Kurchatov Institute, the designer and the manufacturer of the installation. In the course of these discussions, the Soviet delegation formulated comments on the investigations and their results, and these comments were discussed in detail at the joint sessions.

The text which follows sets out the comments by the Soviet experts on the second interim report, comments on the technical chapters, chapters 4-9, as well as on the summary of the report, chapter 10.

- Comments on chapter 4
  Assessment of the pressurised components of the primary and secondary loops

It is clear from the available material that a great deal of effort went into analysing the condition of the equipment and the materials in the primary and secondary loops. The estimate is objective and gives detailed consideration to the relevant questions.
The remarks made in the report and the problems which still remain to be investigated bear witness to the desire on the part of the experts in the FRG and GDR to improve

the operating parameters of the equipment for the purpose of achieving a further increase in the level of safety.

The Soviet experts approve all measures mentioned in this connection, and are prepared to participate in the relevant programmes and investigations.

There are no fundamental objections to this material.

A certain number of remarks, detailed comments and explanatory observations should however be made.

1. Page ... the following sentence should be inserted: This information was obtained from the results of the investigation of the surveillance specimens from the reactors in the Loviisa nuclear power station, in unit 2 of the Armenian nuclear power station, in units 3 and 4 of the Kola nuclear power station and in units 1-2 of the Rovensk nuclear power station.

2. Page ... item 4.1.2

As regards the currently available data concerning the effect of annealing on the restoration of the properties, reference is made also to the results of the investigations of the specimens taken from units 1 and 2 of the Nord nuclear power plant (Greifswald).

3. Page ... item 4.3. should be supplemented as follows: With regard to the reactor pressure vessels of units 1-4, the safe operation of the reactor pressure vessel after implementation of the measures, including the annealing of the vessels in accordance with the recommended procedures, will also be assured for a period extending beyond the planned service life.

In the view of the Soviet experts, there is no need for the annealing of the third unit before the end of the 1990 cycle.

4. Fig. 4-7 should be replaced by Figure 1 from the "proposals and supplements" relating to the "First interim report on the safety assessment of the Greifswald nuclear power station" units 1-4 (WWER-440,W-230), Cologne, 15.2.90".

5. Page ... item 4.1.2. The Soviet experts regard the sampling of metal from the weld of the pressure vessel of unit 3 of the Nord nuclear power plant, with damage to the plating and subsequent repair, as being inadvisable, since the analyses which have to date been carried out provide adequately precise confirmation of the calculated values of the chemical composition of the specimens from unplated pressure vessels.

6. Page ... item 4.1.5. The operating regime differing from the standard was caused by substantial incursion of sea water into the technological condenser in conjunction with oxygen. The elimination of this principal defect together with other established defects resulted in a reduction in corrosion.

- Comments on chapter 5
  Accident analysis

The Soviet experts recognise the large volume of work which was undertaken by the "accident analysis" working group in order to systematise the calculations, carried out at present in the USSR and the GDR, appertaining to various design basis accidents and beyond design basis accidents. Similarly, a welcome is to be given to the measures which are proposed for 1990 and subsequent years with regard to the performance of more accurately specified computations using the program ATHLET, as well as to the corresponding work for the verification of the computer programs for the WWER. The soviet experts regret that they were unable to participate in all discussions which took place in the working group; such participation could have served to provide a more detailed understanding of the positions adopted by the various delegations.

As far as the materials presented are concerned, a number of remarks, proposals and explanatory comments should be made.

1. Page ... "steam leak of an equivalent diameter 90 mm". In our view, the operation of two pumps would be sufficient to cool the core in this case. Accordingly, the computations should be continued and the temperature conditions of the fuel elements should be determined.

2. Page .... item 5.1.3. First paragraph.
Explanations are required as to how a response by the safety valves of the pressurizer is possible in the case of accidents involving loss of coolant from the secondary loop after the cooling and activation of the emergency cooling pumps.

3. Page ... "Break of the main steam line in the SG box".
Upon response of the protective logic 6.4.19, only the MCP of the circulation loop affected is shut down, and not all MCPs as is stated in the text. It is not possible to understand the sequence of events which leads to the opening of the safety valve of the pressurizer. According to our understanding of the situation, in the course of this process the fast-acting atmospheric exhaust systems initially begin to open, and, in the event of their failure, the safety valve of the SG begins to open.

4. Page ... "Break of the feedwater header or failure of all feedwater pumps"
This accident is not taken into consideration in the design. The process which leads to damage to the fuel elements has not been described. It is possible that this accident will be controlled when a protective logic is available. Irrespective of this, the Soviet experts support the introduction of a protection signal for the response of the emergency protection system at "SG level low".

5. Page ... "Break of the feedwater line between SG and non-return valve"
It is possible to achieve an increase in the safety of the units in the case of such an accident either by means of an additional emergency feedwater system or by isolating the feedwater tank by an interlock mechanism. In order to avoid an erroneous response by the emergency protection system, the excitation criterion "SG level low" in more than one steam generator is considered to be necessary for reactor scram.
Having identical initiation criteria for the spray system and reactor scram in the event of an increase in the pressure in the boxes is also considered by us to be advisable, although perhaps this should happen at a lower value.

6. General remark concerning section 5.1
The successful course of an accident is characterised by the words overcome, compensated, controlled etc. In our view, it should be specifically stated that the relationship to the degree of damage to the fuel elements is intended by such expressions.

The following criteria are possible:

- Non-attainment of film boiling in the hottest fuel element with a probability of 95%
- Assurance of the tightness of the fuel elements, i.e. fuel cladding temperature ≦ 600°
- Prevention of such damage to the core which can hamper its unloading (observance of the second design limit value within the meaning of the Principles of Nuclear Safety 82).

7. Pages ... supplements to the 35-point programme.
The discussion concerning measures for backfitting and reconstruction. In the nuclear power plant plan involving W-230, the incident of burnout in the core at the surface of the hottest fuel element was not permitted in design basis accidents. In the current standards (Principles of Nuclear Safety 82) applicable in the USSR, the second design limit value (1200°C at the fuel cladding etc.) is permitted in the case of the worst design basis accident.

Having regard to this criterion, the breach of a pipe of nominal bore 100 could be assumed to be the worst design basis accident for W-230.

In the USSR, investigations have been made concerning the probability of a spontaneous breach of pipes of large diameter (nominal bore 200, nominal bore 500), which are made of austenitic steel, with regard to the W-230 units of the nuclear power plant in Armenia and the Kola nuclear power plant.

Assuming the scheduled performance of the water pressure tests, correct inspections according to the plan and not exceeding of the prescribed cycles for normal rundown and reactor scram, this probability is not greater than 10^{-6}/reactor year.

Proceeding on the basis of the international recommendations (see Safety Series 75-INSAG-3, IAEA, 1988, page 9), the frequency of core meltdown should be below 10^{-4}/reactor year for operating nuclear power plants. In this context, the fraction of breaches of the pipes of nominal bore 200 and nominal bore 500 is negligible for the abovementioned value in the case of W-230 reactors. On these assumptions, we take the view that only breachs ≦ nominal bore 200 can be assumed to represent the worst design basis accident after backfitting. In order to obtain a final determination of the worst design basis accident, it is necessary to consult experts on materials science.

A reduction in the power of the units or outage of one unit with the objective of using its systems for the adjacent unit is regarded by us as inadvisable. The WWER-440 reactors operate with very high stability and reliability at nominal power. It is inevitable that operating the system at 50% power will give rise to a large number of failures and to a reduction in the reliability of the installation. As far as safety is concerned, the use of systems forming part of other units is also possible without outage of the adjacent units. The question how to increase the reliability of the emergency cooling by 1992 must, in our view, be solved within the frame of the 35-point programme of the SAAS.

If necessary, it will be possible for the EP 50 pumps to be replaced by ZP 65. The investigation of the possibility of using the existing low-pressure pumps for the emergency cooling of the core is also supported.

- Comments on chapter 6
  The pressure resistant compartment system as confinement

It is evident from the material presented that a great deal of work was done by the working group concerned with assessing the correctness of the assumed concept of the design of a localisation system, concerning the analysis of the operation of the systems and equipment under the conditions of design basis accidents and a number of accidents exceeding the design specification.

The measures and investigations put forward by the group will contribute to a further increase in the operational safety of the units.

The Soviet experts support the following as priority measures:
An increase in the reliability of the isolation valves and the quick-closing valves and measures to enhance the tightness of the compartments.

As regards the preparatory work ahead of backfitting, we consider the expansion of the accident spectrum to include a leak of nominal diameter 200 with bilateral outflow (2 F) to be extremely beneficial. This will permit the derivation of demands on the emergency cooling system and on the spray system.

In addition, the following proposals and remarks should be made with regard to the material submitted:

1. In order to realise the operations under item 6.5.1c), e), it is necessary to perform in conjunction with the manufacturers a detailed examination of the test regime with regard to the operating capability of theequipment and the condition of the penetrations at a pressure

exceeding 1.25 bar, and to ensure the controlled raising of the parameters. The testing of the condition of the penetrations through the pressure resistant compartment system cannot take place in the units which are in service at design pressure and design temperature. As a rule, such investigations are carried out on test benches.

The Soviet experts are prepared to participate in the planned tests on the equipment of the pressure resistant compartment system, in the complex computational analysis, including the radiological calculations appertaining to loss of coolant accidents, and in the discussion of measures to enhance the level of safety of the units on the basis of the results of these calculations.

- Comments on chapter 7
  Systems technology

The Soviet experts acknowledge that the accident analyses undertaken by the "systems technology" working group was based on a wide range of initiating events, including beyond design basis accidents.

The expansion of the considered range of initiating events which has been undertaken is considered to be expedient.

There are no fundamental objections to the report of the systems technology working group.

A few remarks relating to the accident analysis are contained in the comments on chapter 5.

In the course of the safety assessment of the units of the Greifswald nuclear power station, a comparison is made with requirements as to the international level of safety. In order to satisfy the requirements of this level of safety, it is in our view expedient to consider for units

1-4 those measures intended to enhance the level of safety which have already been implemented in units WWER-440/ W-213.

In general terms, the safety-enhancing measures recommended by the working group are considered to be acceptable.

The text which follows contains a few comments on particular measures:

concerning section 7.6.1.:
The need for backfitting a fast boron poisoning system has to be demonstrated on the basis of analyses concerning reliability and consequences of failure of the reactor scram system.

concerning section 7.6.2.:
According to the available results of the accident analyses, it is possible to classify the replacement of the EP 50 emergency cooling pump by ZN 65 pumps in category II. Before installing a connection between the emergency cooling systems of a double unit, it is necessary to assess the specific technical design. The possibility of the classification of this measure in a different category should exist, depending upon the production deadlines.

concerning section 7.6.3.:
In order to assess the emergency feed to the steam generators via the drain lines, it is necessary to examine the specific technical designs. Measures concerning the use of further supplies of water for the emergency feed to the steam generators should not be classified in category I.

concerning section 7.6.5.:
The closure of the valves must not lead to the infringe-

ment of design specifications. Detailed examination of this matter is required.

concerning section 7.6.6.:
The signalling of DNB is, in our view, unnecessary.

The technical design of the brittle fracture protection interlock must be investigated in detail.

The consolidation of the set points for the initiation of reactor scram and the activation of the emergency cooling pumps or spray water pumps is possible on the basis of the criteria "pressuriser level low" as well as "pressure in the confinement high"

In the short term, it is possible to monitor the observance of the table of permissible reactor power levels as a function of the electrical switching condition of the reactor coolant pumps by more stringent administrative measures, e.g. by a safety engineer.

The requirement for early leak detection and leak location is emphatically supported.

concerning section 7.6.7.:
A large proportion of the requirements in the area of electrical engineering still remain to be discussed in detail by experts in the field.

concerning section 7.6.8.:
The revision of the operating manuals with regard to accident management can take place after completion of the 1990 general overhaul, considering at the same time the results of the accident analyses which have been undertaken.

- Comments on chapter 8
  Redundancy onverlapping events

The report dealt with the questions of internal and external events on the Greifswald nuclear power station. Particular attention was given to the effects of fire on the safety systems of the installation and to fire protection measures. This involved an estimate of the current plan. A number of defects relevant to fire protection were found; these were due to the fact that the nuclear power station was designed and built to the standards of the 60s.

The major weaknesses in the fire protection system were convincingly presented by the authors of the report:

- Lack of separation of the cables of the safety system and of other systems;
- Absence of required fire proof separations at many places (including absence of fireproof walls which subdivide the turbine building into fireproof sections based on units);
- Absence of fireproof cables;
- Inadequate fire resistance of the cable ducts;
- Alarm and fire fighting systems are designed for excessively large areas;
- Not all required positions are equipped with automatic fire fighting systems;
- There is no subdivision of the water supply system into individual drinking, utility and fire fighting water networks;
- Absence of information concerning the adequate supply of fire fighting water.

On the basis of the analysis carried out concerning the actual threat of fire to the installation, measures were recommended which are directed to the reduction of the probability of the incident of fires and to the enhancement of the effectiveness of the fire protection arrangements. These measures, which in practical terms should be implemented as soon as possible, include the following: backfitting of a series of compartments with fire alarm systems and of the turbine building with hydrogen sensors; equipping of the fire sections with fireproof doors; an increase in the limiting value of the fire resistance of the cable ducts to 90 minutes; protection of the oil lines against oil spraying out through a jacket; provision of a system for removing dust from the machine hall; fire resistant cables for the safety systems to guarantee a limiting value of their fire resistance of up to 30 minutes; estimation of the dynamics of the development of a fire in the turbine building; reduction of the probability of the incident of fire in the cable wells and cable galleries; subdivision of the safety systems on a compartment basis.

The measures which have been enunciated are certainly beneficial and are sufficient for a brief period of time. They comply with the current Soviet practice of guaranteeing the fire protection of operating nuclear power plants.

The implementation of the proposals of the working group will permit a substantial reduction in the risk of occurring a fire in the Greifswald nuclear power station.

- Comments on chapter 9
  Operational experience

Because of the limited time available to examine the draft report, it was not possible to carry out a detailed investigation of the correctness of the analysis given in the report, of cases which occurred during operation of the Greifswald nuclear power station, or to examine the validity of the recommendations for perfecting the nuclear power plant systems.

The analysis involved a description of 19 specific incidents, 10 of which were connected with errors by the personnel, including 4 involving errors by the technical maintenance staff. Moreover, according to the estimation of Atomenergoprojekt, 5 cases (No. 236/86, No. 40/81, No. 205/88, No. 7/80 and No. 169/86) were likewise associated with errors by the personnel. Cases No. 205/88, No. 258/80 (270/88), No. 7/80 and No. 169/86 require detailed examination in conjunction with the development engineers concerned with the equipment and the corresponding experts within the design organisation.

With due regard for the large number of errors by the personnel, many of the proposed measures are directed to the prevention of such errors.

A preliminary estimate by Atomenergoprojekt concerning the proposed technical measures is given in the attached table.

The administrative measures are considered to be entirely acceptable, with the exception of the third and of the final measure.

In order to guarantee the safe transport of loads, it will be necessary to formulate a set of administrative supervisory measures which will make it impossible for hazardous events to take place.

The final measure should be supplemented by the requirement to specify precise monitoring actions which must be implemented by taking out of service a part of the system

Table

Measures

to prevent the causes of failures and incidents which have taken place

No. Item	Proposed measures	Preliminary assessment
1	2	3
1	Improvement of the control concept to ensure that fast transients can be controlled	The operational experience gained from Soviet nuclear power stations using W-230 systems does not substantiate this as a problem of special interest. A detailed investigation of the problem is required.
2	Automatic reduction of the reactor power on shutdown of one or more feedwater pumps as well as reactor scram on loss of feedwater	In principle acceptable. Details of the specific technical solution must be worked out.
3	Optimisation of the feedwater control valve and of the feedwater regulator	Agreed; an improvement is required.
4	Improvement of the SG level measurement system	Desirable. Specific proposals are required.
5	Connection of the emergency feedwater systems of all 4 units	Acceptable, but only to guarantee the removal of heat from the reactor installation on failure of its own emergency feed pumps
6	Provision of a system for leak detection	Required

7	Conversion of the valves which are locally controlled to remote control from the control room in order to improve the isolation of leaks in the secondary loop	This proposal has various deficiences, but is acceptable.
8	Multichannel design of the safety control systems. This must include the safety-related systems of the primary, secondary and tertiary loops	This will be possible in the course of a comprehensive backfitting.
9	Backfitting of the interlock against embrittlement of the reactor pressure vessel	The text of the report does not indicate the substance of the backfitting.
10	Initiation of reactor scram on "pressurizer level high"	Acceptable. A check of feasibility without reconstruction of the pressurizer is required. Experience gained in the USSR has not revealed any similar cases.
11	Accident-proof design of safety-related sensors in the primary and secondary loops	Not clear what abnormal incidents are involved.
12	Alteration of the logic system for the interlock 6.4.19 (protection of the reactor pressure vessel on break of the steam line from the SG) to exclude an incorrect shutdown of all MCPs	This should be treated separately.
13	Automatic restriction of the quantity of demineralised water to be fed into the primary loop	This is restricted by the limited power of the feed pumps - 6.3 t/h.
14	Provision of an instrument for continuously measuring the boron concentration	Required.

15	Backfitting of an interlock to guarantee the shutdown reactivity on outage of the reactor	Further consideration is required.
16	Removal of the unlocking switches from the control room while maintaining the current technology. Any unlocking systems required should be automated	Necessary. Systems to protect against unauthorised access can be used. The question of automation should also receive consideration.
17	Automatic switchover of the measurement ranges for the neutron flux during start-up of the reactor	Currently, a modernised system with a broad range is available.
18	An increase in the level of safety of the electrical systems (motors, distribution boards, sensors) in the machine hall	A detailed examination of this matter is required.
19	Testing of the emergency power supply system for adequate power on shutdown of one of three diesel sets	Adequate.
20	Strict train separation of the protected 220 V DC distribution bus using 2 batteries and busbars permanently associated with each unit. Prohibition of cross connections to the adjacent unit	What is desirable is an installation with an additional battery and subdivision of the busbars. The prohibition of cross-connections demands consideration within the terms of the entire problem.
21	Replacement of the unreliable reversible motor generators by separate inverters and rectifiers	Replacement is possible. Backfitting using static inverters.
22	Extensive uncoupling of interconnections of the emergency power supply distribution bus on the 0.4 kV level by fixed allocation of the consumers to the main distribution boards	Possible.

23	Adequate redundancy of inverters and rectifiers to guarantee repair and inspection	The redundancy can be increased.
24	Provision of an independent emergency feed-water supply system	Possible.
25	Reduction of erosion and corrosion processes by using austenitic steel and optimisation of the pipes of the secondary loop	This requires detailed examination while considering periodic monitoring and the current positions.
26	Examination of a possible subdivision of the machine hall	The suitability of this is doubtful, with the exception of the basement area.
27	Replacement of the existing instrumentation and control system by a low-voltage instrumentation and control system using an grounded design	This matter requires examination in conjunction with the problem of the exchange of the entire low-voltage instrumentation and control system (0.4 kV).

for periodic tests, where the period of the test does not correspond to the period of the planned outage of the unit.

- Comments on chapter 10
  Summary

In principle, the experts are in agreement with the proposed measures designed to enhance the level of safety of the units of this nuclear power plant. The following remarks and proposals are made with regard to particular measures:

CATEGORY I

- Process engineering

5. The expediency of the technical design should be investigated by experts.

13. The proposal should be examined in detail.

16. The atmospheric conditions prescribed in the project should be observed.

19. The implementation of this measure is not a condition for resumption of operation of the units. The measure is proposed for category II.

- Instrumentation and control system

1.,2.,6.,14.,18. The proposals should be examined in detail.

7. The implementation of this measure should be transferred to category II, since

- the measures required under items 8 and 9 are sufficient to guarantee the feedwater quantity in the steam generators, by way of short-term measures;

- reactor scram due to the excitation condition "SG level low" takes place by way of a 2 out of 6 selection circuit;

- the horizontal steam generators have large reserves of feedwater;

- a number of short-term measures are proposed for steam generator feed from additional sources, including emergency measures.

8.,9. The introduction of these measures is advisable, but not absolutely necessary.

3.,16.,17.,19. These measures should be classified in category II. In order to guarantee the shutdown reactivity (item 17), additional organisational measures should be provided under category I.

5. The accident conditions adopted as a basis for the requirements should be more precisely specified.

- <u>Electrical engineering</u>

1.,5.,9. The proposals should be examined in detail.

3. Separation of the protected distribution boards is present on the 220 V DC level. The provision of additional batteries can be classified in category II or III.

4.,6. The measures should be classified in category II.

- <u>Plant management, administrative measures</u>

4. The measures should be classified in category II.

12. The sequence of actions on shutdown should be specified.

- <u>Fire protection</u>

8. It is recommended that the fire resistance should be increased.

9. An additional examination of the local conditions is required.

CATEGORY II

- <u>Process engineering</u>

10. The proposal should be examined in detail.

12. The test should not be carried out at design temperature.

14. Evidence should be provided regarding the breach of connecting lines of nominal diameter 200.

• <u>Instrumentation and control system</u>

1. The following supplement should be added: backfitting and modernisation of the fuel assembly outlet temperature measurement system.

• <u>Pressure boundary</u>

9. The requirements regarding representative loadings should be more precisely specified.

The Soviet experts regard the measures of category I as having priority; however, an immediate shutdown of the units is not necessary.

ANNEX 2

Brief description of the analytical tools employed

1. Brief description of the program BRACO-1
2. Brief description of the program COFLOW
3. Brief description of the program RALOC

Brief description of the analytical tools employed

1. Brief description of the program BRACO-1

BRACO-1 is a program to compute atmospheric parameters in containments of nuclear power stations following an accident involving loss of coolant.

The physical basis for the program is formed by the energy, mass and volume balance, as well as the equation of state for ideal gases.

Thermodynamic equilibrium is assumed for the two-phase two-component mixture to be computed; it is thus possible for water and vapour to be present in the saturated or superheated state.

Nodal discretisation takes place by subdivision of the technological installation into model compartments which participate in the further analyses on a point by point basis (lumped parameter model).

Mass flow between the individual compartments is computed using the diaphragm equation for a quasi stationary state, taking into account the carry-over of water.

The heat flow at walls and internals of the pressure resistant compartment system is taken into consideration. This heat flow and the masses and energies introduced by the spray system into the pressure resistant compartment system appear as terms in the energy balance.

2. Brief description of the program COFLOW

COFLOW is a program to compute the locally different pressure and temperature progressions over time in a closed structure (e.g. the containment of a nuclear power plant) during an accident involving loss of coolant. The program is a multinode "lumped parameter" model offering the possibility of superheated an saturated zone conditions. Compartmental systems of real structures are discretised by subdivision into model nodes. Thermodynamic equilibrium is assumed for the homogeneous two-phase two-component mixture in a node.

The mass flows between the zones are taken into consideration

a) On a quasi stationary basis in accordance with the diaphragm equation with loss-dependent contraction coefficients (flow compressible, homogeneous, 2-phase with 2 components),

b) on a non-steady basis in accordance with the momentum equation, taking into account the frictional loss (incompressible, homogeneous, 2-phase with 2 components).

The water carry-over is also taken into consideration.

A one-dimensional heat transfer model for cylindrical and slab geometry is available for the differential consideration of heat supply/removal to/from structures. To describe the heat transfer by condensation and convection, it is possible to select various correlations, which are different for each structure.

In order to verify the program, the series C and D of the RS-50 program "pressure distribution in the containment" with reference to the Battelle model containment, the RS 50 CASP-2 problem, the German standard problem No. 6 (ISP 16) and various HDR blowdown experiments were calculated in advance and posteriori.

3. Brief description of the program RALOC

RALOC was developed to describe long-term hydrogen distribution processes due to pressure compensation and convection flows in the multiply subdivided containment of a nuclear power plant. Within the bounds of the expanded range of application, inter alia in risk studies, also loss of coolant accidents and other transient release processes are simulated together with a series of active systems. Time-dependent and node-dependent pressures, temperatures and gas distributions are calculated in this way.

Compartmental systems of real structures are discretised by subdivision into model nodes on the basis of the lumped parameter principle.

The zones contain homogeneous two-phase multicomponent mixtures (besides steam, up to 3 different gases) in thermodynamic equilibrium, in a saturated or superheated state. Thermodynamic non-equilibrium between vapour/gas and water can be taken into consideration by simulating sump zones. The mass exchange between the zones is calculated using the momentum equation for a system which is non-steady and incompressible, frictional losses being taken into account.

A one-dimensional heat transfer model for cylindrical and slab geometry is available for the differential consideration of the heat supply/removal to/from structures. The heat exchange by convection and radiation is handled separately from heat and material exchange by condensation.

In order to verify the program, a multiplicity of experiments were calculated beforehand and/or afterwards on the basis of various experimental programs. These procedures include calculations using the FIPLOC code, which belongs to the RALOC group of codes: it has the same thermodynamics part and calculates aerosol distributions.

With regard to the matter of pressure build-up in containments, mention should especially be made of the RALOC calculations in advance concerning the standard problem No. 6 (superheated steam reactor test V 44, pressure build-up after blowdown) and concerning ISP 23, superheated steam reactor test T 31.5, likewise involving pressure build-up after blowdown with a different break location.

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