

Gesellschaft für Anlagenund Reaktorsicherheit (GRS) mbH

Reactor Safety in Eastern Europe



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Preface

One of the most important tasks the East-European countries have to cope with while establishing their new administrative and economic structures is to improve the safety of their nuclear power plants. Because of the international significance of reactor safety, the solution of this task is also of vital interest to Western industrial states. Before the reactor accident at Chernobyl very little was known about the design and operation of WWER and RBMK reactor types in the West. Only after the political change has a closer expert co-operation become possible.

To improve the safety of nuclear power plants in Eastern Europe, national and international programmes have been initiated by Western industrial states. First successes have been achieved. Urgently needed measures to increase safety have been taken in several plants.

But it has also shown that the upgrading measures necessary proceed considerably more slowly and that the problems are more diverse than originally assumed. Many tasks can only be solved by long-term co-operation.

It is an important objective of this co-operation that both sides gain a joint understanding of the basic questions of nuclear safety. Safety precautions have to be taken already at the preliminary stage of a possible endangerment. To ensure and further develop safety the Eastern experts are to be integrated into the international projects as partners.

Within the framework of projects run by the Federal Government, GRS started in the second half of the 80s to establish a co-operation with experts of the former Soviet Union in the areas of scientific and technical co-operation as well as expert support for the safety authorities. Since the beginning of the 90s, the number of joint projects has increased considerably, especially by programmes of the European Union.

To facilitate day-to-day work and as a sign of lasting co-operation, GRS and its French partner IPSN, with the support of the German and the French governments, opened an office in Moscow in 1993 and in Kiev in 1994. These offices are operated by the joint subsidiary RIS-KAUDIT. They represent information, contact and co-ordination bureaus between RIS-KAUDIT, GRS, IPSN and the Western, Eastern and international organisations participating in the projects. The following lectures were presented on the occasion of the annual GRS Colloquium on November 24, 1994 in Munich. They provide an overview over the work progress achieved in the co-operation relating to reactor safety in Eastern Europe. The strong participation of the Eastern experts shows that considerable progress has been achieved on the way to co-operation as partners.

Cologne, January 1995

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Co-operation with Eastern Europe

F.W. Heuser, M. Banaschik, I. Sütterlin, H. Teske

1 Introduction

One of the most important tasks the East-European countries have to cope with while establishing their new administrative and economic structures is to improve the safety of their nuclear power plants. Because of the international significance of reactor safety the solution of this task is also of vital interest to Western industrial states. For this reason the co-operation with East-European states in the field of reactor safety is extremely important.

To improve the safety of nuclear power plants in Eastern Europe, national and international programmes have been initiated by Western industrial states. First successes have been achieved here. Improvements increasing safety have been made in several plants, as, for example, in the Kozloduy Nuclear Power Plant in Bulgaria. At the same time the safety authorities have been strengthened and the infrastructure has been improved.

On the other hand it has turned out that the development is much slower and that the problems are more diverse than originally assumed. There still exist considerable safety deficiencies and insufficient safety equipment of the plants, inadequate management and problems connected with the procurement of spare parts. In some countries, e.g. in the Ukraine, the situation has even been aggravated by the bad overall economic situation.

Today it is clear that many problems can only be solved by long-term co-operation with the East-European countries. The co-operation with Russia is particularly important here. The experience and the competence of Russian scientists and engineers is imperative for the improvement of reactor safety in Eastern Europe. Safety can only be improved in co-operation with Russia, its scientific and technical institutions and its nuclear industry. This co-operation is also important for nuclear safety in the other East-European countries.



Fig. 1 Kozloduy Nuclear Power Plant (Photo: IPSN-DEMAIL)

2 Objectives

The problems to be solved are diverse. Measures for improving the safety of the plants but also for strengthening infrastructure are still necessary. Now as before the support of the safety authorities and expert co-operation with the scientific and technical institutions have a high priority. The following objectives are pursued in detail:

Improvement of the Infrastructure

The improvement of communication, the establishment of a communication network between authorities, scientific and technical institutions and power plants, but also to foreign countries are extremely important for co-operation with the Eastern partners to shorten the previous time-consuming information paths;

Furtherance and Support of Safety Authorities

The expert competence of the safety authorities and their technical organisations must be furthered and supported so that they can accomplish their responsibilities towards industry as equal partners;

Scientific and Technical Investigations

Joint safety investigations and research and development studies are to be carried out with East-European scientific and technical institutions;

Safety Assessment of Nuclear Power Plants

Safety assessments are to be carried out in co-operation with the safety authorities and industry to develop backfitting measures for upgrading the plants;

Technical Investment and Equipment

Where necessary, Western technology and equipment are to be provided to achieve the urgently required improvements quickly;

- Training

Training and qualification of the staff working for the authorities and nuclear power plants represent an essential part of the co-operation.

Germany takes a major share in the co-operation with the East-European countries. The Federal Minister for the Environment, Nature Conservation and Nuclear Safety (BMU) and the Federal Minister for Research and Technology (BMFT) thus provide considerable financial means for these purposes. The scientific and technical co-operation is strongly promoted by the BMFT. The BMU repeatedly took initiatives which led to comprehensive bilateral as well as international support programmes for East-European countries.

In the co-operation with East-European countries, GRS assumes a leading role in national as well as in international support projects. There is a close co-operation with our French partner, the Institut de Protection et de Sûreté Nucléaire in Paris here. This, in particular, applies to various joint activities within the framework of the support programmes PHARE and TACIS of the European Union.

In the other lectures of this meeting different assignments which are being worked on together with the East-European partners will be reported in more detail:

- Dr. Asmolov of the Kurchatov Institute, Moscow is going to give an account of the co-operation from the Russian perspective;
- the two following lectures are going to deal with safety improvements for WWER plants, improvements for the older W-230 reactors and for W-213 and WWER-1000 plants, taking the Rovno Nuclear Power Plant at Rovno, Ukraine as an example;





- the joint studies on accident analysis and on the adaptation of GRS computation codes to Soviet reactors are then going to be discussed;
- the two final lectures are going to deal with safety aspects of RBMK reactors and the state of the investigations on the encasement of Unit 4 at Chernobyl.

In this contribution the general areas of the co-operation shall be dealt with first, i.e. in the field of infrastructure with the establishment of a data processing communication network and with the training level of seminars, workshops and visits which are carried out in co-operation with the Eastern partners.

In the second part of the lecture, technical measures for ensuring and improving safety within the plants are going to be discussed. Safety precautions, i.e. preventive safety measures which can prevent damage right from the beginning, are the primary concern here.



Fig. 3 Kozloduy Nuclear Power Plant: cooling pond (Photo: IPSN-DEMAIL)

3 Communication Technology

The first contacts with Eastern authorities and institutions already showed that the use of efficient data processing equipment and the improvement of the communication structure were essential requirements for co-operation. The connections within Russia itself and especially to foreign Western countries were particularly bad. Here the first investments had to be made. GRS, commissioned by the BMU and the BMFT, has achieved quite a few results in Russia and the Ukraine in the meantime.

The most important results are summarised here:

- The voice and fax connections of the authorities and institutions in Moscow to nuclear power plants and regional centres of the authorities, as well as those from Moscow to foreign Western countries, were improved.
- data processing equipment (hard- and software) for accident analyses was provided and advanced computation programmes, like ATHLET, RALOC and DRASYS were transferred to scientific and technical institutions in Russia and the Ukraine.
- Local-area networks (LANs) with PC workstations, output equipment and an eMail system for the authorities and their scientific and technical centres were established and connected to wide area networks (WAN).

Fig. 4 illustrates the current state of the present communication network for voice, fax, data and eMail communication with its participants.

- In addition to the normal telephone network the GRS/IPSN offices in Moscow and Kiev, the local authorities and their scientific and technical centres as well as a series of nuclear power plants and research institutes are tied into the special telephone network ISKRA.
- In addition, the Moscow and Kiev authorities as well as the Rovno and Balakovo Nuclear Power Plants can be reached via satellite.
- Finally, a commercial satellite network will also be used. The network can be used to carry
 out an efficient exchange of data between the Russian and German partners. Nuclear
 power plants can also be integrated into the satellite connection via the Russian operator
 concern Rosenergoatom.

The Kurchatov Institute and the GRS/IPSN office in Moscow were chosen as the hub for the communication network and the data processing support as the technical know-how and the infrastructure here represented the best preconditions.

Communication and data processing equipment have been improved considerably. Apart from the improved voice communication, an efficient communication structure for the scientific and technical transfer and processing of data were created.



Fig. 4 Telecommunication network: voice, fax, data, eMail

4 Seminars and Workshops

A further emphasis of the projects deals with training. Commissioned by the BMU, GRS together with the Eastern partners runs seminars, workshops and audits for training employees of the authorities, scientific and technical institutions and operators.

The topics of these training events are:

- basic safety requirements,
 - · the safety concept for nuclear power stations,
- bases of atomic law
 - legal bases, regulations,
 - · safety guidelines,
 - · nuclear regulations,
- nuclear licensing and supervisory procedures,
 - responsibilities and functions of partners, authorities, experts, manufacturers and operators taking part in the licensing procedure and supervision,

furthermore, operational, organisational and technical issues, for example regarding

- quality assurance during operation,
 - analysis of operating experience,
 - · acquisition and reporting of abnormal incidents,
 - improvement of plant documentation,
- technical instruction and qualification of the staff working in the plants.

Since 1992 about 40 seminars, workshops and audits have been carried out in Russia, the Ukraine and in Germany. About 1000 experts participated in these events. In addition thereto, more than 200 reactor operators of WWER-440 plants have so far been trained at the simulator in Greifswald.

The good response of the participants demonstrates the usefulness and success of these events. Seminars and workshops represent a good opportunity for meeting and informing

each other, and for exchanging ideas. They offer a unique opportunity for discussing and conveying fundamental safety concepts.

The licensing and regulatory authorities in Russia and the Ukraine also asked for the setting up of training courses for young engineers and for assistance with in-service training of teachers at their training centres. Corresponding courses and lectures are currently being prepared.

5 Measures to Increase Safety

Technical measures to increase safety, taking the example of WWER pressurised water reactors, are illustrated below:

Compared to Western pressurised water reactors there are deficiencies in the safety design of all three types of WWER reactors.

	WWER-440/W-230	WWER-440/W-213	WWER-1000
emergency cooling NWA (primary side)	insufficient no independent NWA system	3 x 100 % no CCS sump blockage	3 x 100 % no CCS sump blockage
confinement/ containment	insufficient depressurisation system	depressurisation via pool-type pressure suppression	safety containment
emergency injecti- on (secondary side)	insufficient redundancy and physical separation	3 x 100 % insufficient physical separation	3 x 100 % insufficient physical separation
Protection against spreading impacts	deficiencies - fire - retroactive effects, events, turbine hall	deficiencies - fire - retroactive effects, events, turbine hall	deficiencies of design

Table 5-1 Safety design of WWER reactors

The most severe deficiencies exist in the old WWER-440/W-230 plants. The plants do not have a sufficient emergency core cooling system and no pressure-tight safety containment.

As a safety enclosure the plants only have a confinement system which is designed against a pressure of only 0.1 MPa. These plants cannot be practically backfitted to a level which corresponds to international safety requirements.

Plants of the second and third generation, i.e. the later WWER-440/W-213 and WWER-1000 reactors are assessed more favourably. Although plants of these types also exhibit deficiencies in their safety design, these deficiencies can, apart from a few exceptions, largely be removed by backfitting measures.

W-213 plants and WWER-1000 plants have a 3 X 100 % redundant emergency and residual heat removal system which can also control large leakages in the reactor cooling system. The safety enclosure is also guaranteed; WWER-440/W-213 have a pool-type pressure suppression system, WWER-1000 have a pressure-tight safety containment.

Important deficiencies in W-213 and WWER-1000 plants which can, in principle, however, be backfitted, concern:

- the insufficient physical separation of main steam, feedwater and emergency feedwater lines in the transition area from the reactor building to the turbine hall,
- meshing in the injection lines of the emergency feedwater system and
- the insufficient protection against spreading impacts, especially the protection against fire and internal flooding.

It would be unrealistic to say that the old W-230 plants should be shut down immediately and the newer WWER-440/W-213 and WWER-1000 should be backfitted to international safety standards. The experiences at Kozloduy (Fig. 5) have shown that this is not possible.

To maintain power supply in Bulgaria it was necessary to backfit even the oldest units 1 and 2, two W-230 reactors, at Kozloduy to such an extent that these could continue their operation for a few years. The Western side, industry and independent expert organisations gave considerable support to the regulatory authority in its efforts to substantially increase the safety of the units. Measures which could increase operational safety, control more frequent operational transients and prevent beyond-design-basis accidents were mainly implemented. In some cases, necessary plant improvement measures could not be taken immediately. In these cases temporary conditions, e.g. special inspections or supervisory measures had to be established until the improvements in the plants had been realised. In the newer plants too, backfitting measures cannot be realised at once, especially when these are major constructional changes or improvements of the system for which there are no financial funds. But here too, like for the old plants, selectively targeted improvements can be accomplished with relatively restricted efforts, thus achieving a high safety benefit. These are, above all, preventive safety measures detecting incidents early and preventing accidents.



Fig. 5 To ensure the power supply in Bulgaria, even the oldest units 1 and 2 (two W-230 reactors) of the Kozloduy Nuclear Power Plant were backfitted with Western help, to such an extent that they can be operated for a few more years (Photo: IPSN-DEMAIL).

The main points are

Measures for Plant Diagnosis and Monitoring

Recurrent inspections monitoring the integrity of the reactor cooling system, its components and the pipes are particularly important here;

- Improvements of the Electrotechnical Facilities for Reactor Protection and Accident Instrumentation

he analyses of operating experience in WWER-1000 plants exhibit a high proportion of incidents which are due to deficiencies in the reactor protection, due to restricted reliability of the control, limitation and protection devices;

 Preventive Measures against Spreading ImpactsHere, improved fire protective measures, measures against internal flooding, e.g. upon leakage in a residual heat exchanger, and against consequential damage from pipe ruptures in the turbine hall are necessary for W-213 plants, but also for WWER-1000 plants.

But not only the technical measures are concerned here. Further investigations and studies on various issues are required to elaborate the bases and prerequisites for specifying, classifying and assessing the technical improvements necessary.

Material Inspections

Now as before, material inspections are necessary, e.g. regarding radiation embrittlement in the weld seams of the 440 reactor pressure vessels. In the long run, there is also a problem for the pressure vessels of some WWER-1000 plants where the material of the weld seam close to the core exhibits a high nickel content;

Accident Analysis

(Design basis accidents, beyond-design-basis accidents) The accident analyses for WWER reactors must be further completed. Considerable developments have already been made in the application and verification of advanced computation codes for WWER plants, e.g. in adapting the thermohydraulic programme ATHLET. The analyses contained in the technical project documentation have to be checked and supplemented with advanced computation codes, especially for the plants of the WWER-1000 type. Plant-specific data for the reactor core, the reactor protection system and the process control systems are to be used for the computations here. Respective accident analyses are required to more exactly examine and determine the requirements to be met by reactor protection and the safety systems;

Improvement in Plant and Operational Documentation

(Operation manuals, accident procedures, AM measures)Within the plants themselves accompanying measures are also necessary. An important point refers to the plant and operational documentation. Improvements requiring comparatively little effort which support operating personnel in their functions are possible here.

Within the framework of the studies financed by the BMU, a pilot project is currently being carried out together with the operator for WWER-1000 plants, using Unit 3 of the Rovno Nuclear Power Plant in the Ukraine as an example. The co-operation is extremely successful. Thus, records of components and descriptions of the system were prepared. At present, the revision of the operation manual and of selected accident procedures is being started.

Other Ukrainian and Russian nuclear power plants are very interested in the work performed. Thus, in July 1994 the Balakovo Nuclear Power Plant in Russia and the Rovno Nuclear Power Plant agreed to co-operate in the field of plant and operational documentation. Actual work at Balakovo, with the participation of GRS, will start at the beginning of 1995. The example shows how a pilot project can provide the impetus for corresponding projects in other plants.

6 Technical Equipment

In the safety inspections of WWER reactors, which have so far been performed, different areas have been determined in which safety technical improvements are urgently required. In a special investment programme for Russia and the Ukraine the BMU also provides financial funds for technical equipment.

The investments are intended for plants of the WWER-1000 type. For this reason the Balakovo Nuclear Power Plant in Russia and unit 3 of the Rovno Nuclear Power Plant in the Ukraine were selected as reference plants. To a limited extent funds are also employed for the establishment of an operational telemonitoring system in the Ukrainian nuclear power plant at Zaporozhe. A total of 21 million DM is provided for each of the two countries. The investment measures were co-ordinated in close co-operation with the authorities, organisations and the nuclear power plants involved. Special attention was paid to ensuring that the intended investments fit smoothly into the backfitting programmes of the two countries.

Equipment for

- diagnosis and monitoring procedures,
- in-service inspections
- fire protection measures and
- telecommunication
- represent the main areas of investment.

Fig. 6 outlines the scope of the technical equipment for in-service inspections at the Balakovo Nuclear Power Plant. These essentially are:

- a central pole manipulator for internal inspections of the reactor pressure vessel (Fig. 7),
- a manipulator for material testing of the steam generator collectors,
- equipment for ultrasonic inspections of weld seams in pipes,
- as well as manual ultrasonic test units and x-ray equipment with assessories.



Fig. 6 Balakovo Nuclear Power Plant: scope of the technical equipment for in-service testing

Of course, this equipment provided by the German side signifies technical progress. The following two remarks must still be made:

 the investments represent one element of the co-operation. Their selection was closely connected with the present safety investigations and their results; of course, the German side cannot provide technical aid and equipment for all plants. This
would exceed by far the purpose and the possibilities of technical support. On the contrary,
the technical equipment intended for the plants at Balakovo and Rovno will demonstrate in
an exemplary way that safety improvements can be achieved with a justifiable effort. Like
the joint projects on plant and operational documentation, this project is also a pilot project.
This pilot project will achieve a broad effect, providing an impetus so that the countries
themselves also take the corresponding measures in other plants.



Fig. 7 Central pole manipulator for the reactor pressure vessel

7 Summary

This lecture was not intended to give an account of all projects in co-operation with East-European countries. However, some important aspects and emphases of the co-operation should be highlighted:

- An efficient communication network and good data processing equipment are an imperative precondition for the scientific and technical co-operation;
- Training events are an important element of the co-operation. They offer a unique opportunity to impart a fundamental understanding of safety issues.
- Backfitting of the plants takes longer than originally assumed. To improve the safety of the
 plants, preventive safety measures are primarily required in the short term, i.e. measures
 which detect incidents early and which prevent accidents. For this purpose technical
 equipment is also supplied by the German side on the basis of the results of the safety
 investigations.

In general it is very important that the current projects create the bases for a long-term cooperation. The East-European partner states, with the experience and the competence of their scientists and engineers, have to be completely integrated into the international projects on the further development of reactor safety.

Russian-German Co-operation in the Field of Safety Research

V. Asmolov*

On the basis of the agreement on the scientific and technical co-operation in the field of peaceful utilisation of nuclear energy between the Soviet Ministry for Atomic Energy and the Federal German Ministry for Research and Technology, the Russian scientific centre "Kurchatov Institute" has co-operated closely with German organisations in the field of reactor safety research since 1988.

Technical Progress Consequence Changes in the Nature and Structure of technology-related Dangers

- yesterday high frequency low consequence accidents
- today large consequence accidents

Fig. 1 Effects of technological progress on the nature and the structure of the resulting dangers

At the beginning of this co-operation there were projects on the transfer of German methods of analysis for accident analyses and probabilistic safety investigations for nuclear power plants as well as the preparation of joint experimental research.

Research Areas

- Analysis of accident processes that precede the core disruption
- Studies of physical and chemical processes of fission product release
- Studies of the main core damage processes
- Studies of the molten core-reactor vessel interaction
- Studies of the molten core-concrete interaction
- Studies of the hydrogen behaviour
- Studies of the containment filtration system

Fig. 2 Research fields

* Editorial revision by GRS

After an intensive familiarisation of the Kurchatov Institute employees with the accident analysis system code ATHLET of the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH in Germany this programme was transferred to the Kurchatov Institute.



Fig. 3 Accident codes of the different countries

Since 1989 joint projects for adapting and verifying this code for Russian reactors of the WWER and RBMK type have been carried out. Within the framework of these projects, verification computations have been carried out with the help of experimental results from Russian and international experimental plants and new computation models considering the characteristics of Russian reactors have been developed together.



Fig. 4 Co-operation partners for the adaptation and verification of computer codes



An example of this is the creation of a coupled code complex with the German thermohydraulics code ATHLET and the Russian 3D neutron kinetics code BIPR-8 for analysing accidents in WWER reactors. On the basis of this joint work and the verification matrix developed for verifying the ATHLET code for the peculiarities of WWER reactors, the Kurchatov Institute is going to prepare a final report for the Russian licensing and supervisory authority Gosatomnadzor. On the basis of this final report, the application of this programme can be permitted in the licensing and supervisory procedure for Russian nuclear power plants.

	Severe Accident Conditions
Question	What is the possibility for the description of the core processes?
Answer	You should use some of the codes:
	ATHLET-CD, SCDAP/RELAP5, ICARE2.
Question	What code is most realistic?
Answer	?
Conclusion	A careful comparison analysis of the results is needed to be carried out.
Decision	International verification procedure performance (ISP).

Fig. 6 Prediction of reactor core behaviour under severe accident conditions

From the beginning of the co-operation it was intended to make the results of the joint studies and the methods of analysis available to other Russian organisations which work in the field of reactor safety. The THERMOCODE User's Club of the Kurchatov Institute, which is responsible for the transfer of computation programmes to other Russian organisations and which is to organise the exchange of experiences among the code users in Russia, has an important function here.

GRS Codes	Russian User Organizations		
ATHLET 1.1 c. A/1110	RRC KI, RDIPE, SC NSA, Gidropress		
ATHLET-CD / 4110	RRC KI, NSI AS		
DRASYS MOD 3.3	RRC KI, RDIPE		
RALOC MOD 2.2	RRC KI		
WECHSL	RRC KI		
QUABOX/CUBBOX-HYCA / 2065	RRC KI		
RALLY	RRC KI		

Fig. 7 THERMOCODE User's Club

In the meantime, the number of work fields as well as the number of computation codes transferred and the number of organisations participating directly in the scientific and technical co-operation have increased substantially, so that almost all important areas of reactor safety research are covered and there are direct contacts to all important organisations in this field in Russia.

The initial phase of the co-operation had shown that the effectiveness of the work carried out in the Kurchatov-Institute was seriously impaired by the lack of an appropriate computer basis. The BMFT, being the commissioner, and GRS found an opportunity for providing an "Amdahl"-computer and a workstation to the Kurchatov Institute and they supported the Russian colleagues during the planning and establishment stage. In addition to the joint work within the scientific and technical co-operation programme, the BMFT provided one-off financial funds for supporting scientists of the Kurchatov Institute and of the Institute for Reactor Safety of the Academy of Sciences in 1993 and 1994. This aid permitted additional work of mutual interest to be carried out at the institutes mentioned, accident analyses and the development of new methods for the safety assessment of WWER and RBMK reactors representing the main emphases.

ATHLET Code for VVER, RBMK Reactors

1. Verifikation

- Development of VVER verification matrices within multilateral working group by GRS
- Verification matrices for RBMK are still under development
- Assessement:
 - on seperate effect tests
 - * GE-13, -15 blow down
 - * KC-VVER partially uncovered core
 - * VERTICAL CANON depressurization
 - * KCB-RBMK stop flow rate
 - · on integral tests
 - * SPE-1, -2, -3, -4 SBLOCA
 - * ISB-VVER SBLOCA
 - * KS-AST NC
 - * PACTEL NC
 - on NPP experiments
 - * Sosnovy-Bor NPP pump coast down
 - * Smolensk NPP power scram
 - * Sosnovy-Bor NPP power scram

2. Development of New Models

- Implementation of 3-D VVER kinetics module and development of ATHLET-BIPR-8 package
- Development of the horizontal steam-generator model and it's validation on EDO "Gidropress" test data
- Development of WAST water-steam properties package
- Development and implementation of radiation heat transfer module for RBMK
- Development of the core melt and reactor vessel interaction module for further implementation into ATHLET-CD code

3. Application

- Accident analysis for current NPP and new generation design

Final report on verification of ATHLET code for VVER reactors will be prepared for Nuclear Safety Center (GAN).

Fig. 8 ATHLET code for the safety assessment of WWER and RBMK reactors

In parallel with the transfer of ATHLET, the co-operation regarding the performance of experimental research work for Russian reactors between the Kernforschungszentrum Karlsruhe, GRS and the Kurchatov-Institute at the BETA and CORA experimental plants began. The experiments and their assessment have been completed. At present, an international standard problem for code verification is being prepared on the basis of the CORA/WWER experiments.

1989	Preparation of technical requirements	
1990	Designing of WWER/Cora bundles	Scoping test performance in IGR
1991	and CORA facility modified elements.	reactor.
	Manufacture of elements for WWER bundles and CORA facility.	Experimental determination of original
1992		Russian material properties.
1993	Performance of WWER/CORA-W1,	
_	W2 tests.	
1994	Post-test examinations of W1, W2 bundles for ISP-36 tasks.	
	Preparation of the joint report with expe	rimental results.

Fig. 9 Time schedule of WWER/CORA programme

Content of Experimental Database on WWER-CORA-W2 Test

- Bundle elements temperatures
- H₂ generation
- Time of fuel rod rupture
- Time and elevation of first molten material relocation
- Zr oxidized parameters
- B₄C dissolution parameters
- SS melting characteristics
- UO2 dissolution and relocation characteristics

Fig. 10 Objectives of the WWER/CORA-W2 tests



Fig. 11 CORA test facility















Fig. 14 CORA-ISP 36: comparison of the fuel temperature (K) between calculated and experimental data (experiment completed after 4300 seconds)



experimental data (status after 4700 seconds)

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Fig. 16 CORA-ISP 36: comparison of the fuel temperature (K) between calculated and experimental data (experiment completed after 4700 seconds)





ω



	List of original properties
1	Zr-1% Nb - steam interactions
2	Zr-1% Nb-SS (X18H10T) interactions
3	Zr-1% Nb-UO ₂ interactions
4	SS-B ₄ C interactions
5	Zr-1% Nb melting point





Fig. 20 Hydrogen research programme

The international co-operation in the field of experimental research is continuing within the framework of the "RASPLAV" project financed by the OECD. The experimental programme to be carried out by the Kurchatov Institute concerns the examination of phenomena occurring in the in-vessel phase of a core melting accident. The main emphasis of the further joint studies on the development of methods of analysis is the development of models and the verification of computation codes for the analysis of core melting accidents (e.g. ATHLET-CD) and the adaptation of the existing codes for accident analyses to recent Russian reactor projects.

Project Participant	Country
JAERI	Japan
KAERI	Korea
ERF	Netherlands
CSN	Spain
Statens Karnkraftinspection	Sweden
Paul Scherrer Institute	Switzerland
AEA	United Kingdom
NRC	USA
RRC "Kurchatov Insitute"	Russia
AIB-Vincotte Nuclear	Belgium
AE	Canada
IVO, VTT and Sateilyturvakeskus	Finland
CEA, IPSN	France
GRS	Germany
ANPA, ENEL and ENT	Italy

Fig. 21 RASPLAV project participants



Possible Scope for Future Joint Activities Experimental "Quench" facility RASPLAV Steam explosion Hydrogen research - large scale experiments with steam atmosphere (RVT-facility) Theoretical Implementation advanced model to the existing codes (core degradation, thermohydraulic, cavity and containment behaviour) Development of the RASPLAV-V Code Turbulent deflagration 3D model development Advanced integral code's system development

Fig. 23 Future emphasis of bilateral co-operation

Safety Improvements for WWER-440/W-230 Nuclear Power Plants

W. Kelm, A.M. Kiritschenko, W. Wenk

1 Introduction

GRS has been involved in the safety assessments of nuclear power plants with WWER reactors for several years. These studies were commissioned by

- the Federal Minister for the Environment, Nature Conservation and Reactor Safety,
- the Council of the European Communities in consortia to support the Bulgarian and the Ukrainian supervisory authorities,
- The European Bank of Reconstruction and Development in a consortium to support the Slovenian supervisory authority and
- the International Atomic Energy Agency within the framework of consultants' meetings and local missions.

The results of these studies on the W-230 type and the present safety status of the plants concerned in Russia (two units each in Kola and Novovoronnezh), Slovakia (two units at Bohunice) and in Bulgaria (four units at Kozloduy) are described below (Table 1-1).

The Armenian nuclear power plant Medzamor is also listed in the Table. In spring 1993 the Armenian government decided to start-up the second unit of the power plant again. Intensive work is being performed here at present.

Site/Number of Units	Start of Operation	Operational Period Projected
Bohunice 1	1978	2005
Bohunice 2	1980	2006
Kola 1	1973	2003
Kola 2	1974	2004
Kozloduy 1	1974	2004
Kozloduy 2	1975	2005
Kozloduy 3	1980	2010
Kozloduy 4	1982	2012
Novovoronezh 1	1971	2001
Novovoronezh 2	1972	2002
Armenien	1976	
Armenien	1979	

Table 1-1 WWER-440/W-230 units (operating in 1994)

2 Status of the Safety Upgradings

The GRS safety assessment of the Greifswald Nuclear Power Plant, Units 1 - 4, of 1990 represents one of the first comprehensive descriptions of the safety status of this type. The deficiencies described there and the classification into measures of different priority still apply.

GRS representatives in 1990/91 also played a prominent part in the IAEA investigations on W-230 as well as in the relevant safety review missions up to the elaboration of the final report TECDOC 640.

A further step was the leading participation of GRS in a consortium of Western expert organisations supporting the Bulgarian supervisory authority BNSA and the safety assessment of units 1 and 2 of the Kozloduy Nuclear Power Plant. These activities have already been reported earlier. The upgrading measures have not been finalised yet. The authority and the consortium do, however, agree that the work of the consortium at Kozloduy should be continued, giving priority to units 3 and 4. In the meantime, the most severe deficiencies are being remedied in all plants, for further short-term operation, and part of this work has already been completed. Table 2-1 provides a general idea of the status of this work at the individual locations.

Independent of the location, the measures which have been taken so far concentrate on the following areas:

- re-establishment of the project state (housekeeping),
- improvement of operational management,
- improvement of the technical safety of the plants.

Table 2-1 Steps for upgrading safety

Country	Period	Measures				
Slovakia	1989/1990	Safety inspections				
NPP Bohunice 1/2	1991	Upgrading programme for short-term operation until 1995				
	1991/1992	Implementation of upgrading programme				
	1993	Provisional safety report				
	1994	Planned measures for operation until 2000				
Bulgaria	1991	Shutdown of units after IAEA-Safety Review Mission				
NPP Kozloduy 1/2 NPP Kozloduy 3/4	1991/1992	Three-step upgrading programme for the operation of units 1/2 until 1997 at the latest				
	1992/1995	Upgrading, operational licence restricted to one year each				
	1993	EBRD-support for units 3/4 (24 million ECU)				
	end of 1994	Operator's concept for further operation				
Russia	from 1990 on	"Special operational regime"				
NPP Novovoronesh 3/4 NPP Kola 1/2	1992	Two-step upgrading programme for further operation of these plants until EOL; time schedule or completion endangered by financial problems				

Table 2-2 shows examples of how some significant deficiencies in the individual plants have been remedied or compensated. Bohunice thus represents the best upgraded power plant with a W-230 reactor apart from units 3/4 of the Kozloduy NPP, the original design of which is better.
 Table 2-2
 Remedy or compensation of individual deficiencies in the different nuclear power plants

Measure:	Sepa (mal	aration of the emergency core cooling system from the volume control keup) system					
NPP Bohunice	1/2	not separated, manual valves replaced by motorised valves					
NPP Kozloduy	1/2	no separation, analysis for controlling this accident presented, extended ISI, supplementation of accident manual					
NPP Kozloduy	3/4	irrelevant, improved design					
NPP Kola 1/2		systems separated (hot leg injection), installation of motorised valves in the make-up system including the respective interlocks is planned					
NPP NW 3/4		systems separated (hot leg injection)					
Measure:	Insta	llation of quick-acting steam isolating valves in the main steam system					
NPP Bohunice	1/2	Installation in the collector and in the main steam lines					
NPP Kozloduy	1/2	Installation planned					
NPP Kozloduy	3/4	Installation planned, equipment ordered					
NPP Kola 1/2	-	Installed					
NPP NW 3/4		Installation scheduled for the 2nd stage					
Measure:	Eme	rgency Injection of steam generators					
NPP Bohunice	1/2	Independent additional system with pumps and containers outside the turbine hall					
NPP Kozloduy	1/2	Mobile pumps for HP and LP injection, in LP area also use of other operational systems					
NPP Kozloduy	3/4	Like unit 1/2, independent system planned, equipment via EBRD					
NPP Kola 1/2		Mobile pumps, independent system planned in 2nd stage					
NPP NW 3/4		Mobile pumps, independent system planned in 2nd stage					
Measure:	Sepa	ration and redundancy separation of electrical supply (6kV; 0.4 kV)					
NPP Bohunice	1/2	Installation of two independent 6-kV systems					
NPP Kozloduy	1/2	Newly installed 6-kV lines from the DG to the distributions					
NPP Kozloduy	3/4	3 independent trains in the original design					
NPP Kola 1/2		Redundancy separation planned, being prepared					
NPP NW 3/4		Mobile DG has been ordered					

Measure: LBE	for pressuriser surge line
NPP Bohunice 1/2	3 independent leakage monitoring systems, additional shock absorbers, one (upper) surge line (SL) cut
NPP Kozloduy 1/2	1 leakage monitoring system installed, LBB study for unit 2, adaptation to unit 1 planned. Additional fixings at unit 1 and 2 installed, however pipes not fastened
NPP Kozloduy 3/4	like unit 1
NPP Kola 1/2	1 leakage monitoring system installed, generic LBB studies for nominal diameter 58-200 exist, not verified
NPP NW 3/4	1 leakage monitoring system installed, generic LBB studies for nominal diameter 58-200 exist, not verified

Those measures which were carried out in the individual plants to ensure the integrity of the reactor pressure vessel and the main pipes, as well as to prevent large leakages, are described in more detail below.

3 Integrity Assurance of Components and Main Pipes

The system upgradings performed can only improve the safety status of the plants if the integrity of the pressurised components can be ensured and in particular if large leakages can be excluded. The measures for the major components and pipes concentrate on damage prevention (Fig. 1). This is to ensure the integrity of the reactor pressure vessel, the main pipes and essential screwed connections as well as to monitor and detect potential damaging mechanisms.



Fig. 1 Integrity assurance of the pressurised component

The integrity of the primary system and of the safety-relevant parts of the secondary system are frequently reduced to the neutron embrittlement of the reactor pressure vessel. Fig. 2 below shows how the problem is inadmissibly simplified by this approach. The integrity of the reactor pressure vessel is also very strongly determined by the integrity of the two circulation systems. A post-qualification of the main pipes to exclude breaks can improve the safety status of these plants substantially.



Fig. 2 Integrity assurance (schematically) KIC > KI

3.1 Integrity Assurance of the Reactor Pressure Vessel

The reactor pressure vessel of the type W-230 was built in accordance with the regulations valid in the former Soviet Union in the sixties.

As internationally common at this time, the phosphorus and copper contents of the weld seams were not sufficiently restricted.

The gap between the outer edge of the core and the RPV wall in WWER types is much narrower than in comparable Western vessels, as rail transport of the RPVs was intended.

The high neutron flux density in the reactor wall resulting therefrom, together with the increased sensitivity of the weld seams to neutron irradiation, led to the fact that the brittle fracture safety of the RPV according to the manufacturing regulations, in principle, could no longer be demonstrated after one third of the planned service life.

The integrity of the RPVs of the type WWER-440/W-230 in the medium and long term can only be ensured (Fig. 3), if

- the load-bearing capacity, i.e. the strength of the weld seam close to the core, is regenerated by annealing and is subsequently kept high by lowering the neutron flux density and if
- the local loads in the RPV wall upon cooldown transients are kept low by warming the emergency cooling water, injection into the hot legs, installation of quick-closing valves in the main steam lines or
- if cold injections can even be avoided by requalifying the main lines to exclude breaks.

The totality of these measures has so far not been realised in any of the plants mentioned (Table 3-1).



Fig. 3 Measures to ensure the integrity of the RPV

Table 3-1 WWER-440/W-230 RPV

Site / Unit	BOL	EOL	EC	cs	MSI	Cladd	Annealed	Fluxred	Тко	TKEOL	тка
			IP	Tı₩/°C		ed			[°C]	[°C]	[°C]
Bohunice 1	1978		cold leg	55	1992	Y	1993	LL 1983 D 1992	0 (MC)	198	152
Bohunice 2	1980		cold leg	55	1993	Y	1993	LL 1984 D 1984	49 (MC)	200	152
Kola 1	1973		hot leg	55	1993	N	1989	D 1985	30 (C)	148	177
Kola 2	1974		hot leg	55	1993	N	1989	D 1985	60 (C)	186	177
Kozloduy 1	1974		cold leg	55	N	N	1989	D 1987	52 (MC)	> 200	179 LDH
Kozloduy 2	1975		cold leg	55	N	N	1992	D 1988	50 (MT)	175	163 LDH
Kozloduy 3	1980		cold leg	55	N	Y	1989	D 1987	50 (MC)	202	> 179 LDH
Kozloduy 4	1982		cold leg	55	N	Y	N	N/LL 1986	5 (C)	108	> 179 LDH
Novovoronezh 3	1971		hot leg	55	N	N	1987 1991	N/LLAA	55 (MT)	194	175
Novovoronezh 4	1972		hot leg	55	N	N	1991	N/LLAA	15 (MT)	158	175
Legend: BOL Begin of EOL End of L LDH Low Dec MC Calculat RPV Reactor TrKA Maximum Transitio	Life ife ay Heat on chen Pressure n allowa	n. al. e Vesse ible	91	C C ECCS E LL L MT F TKO I	Calculat Emerge Low Lea Measure nitial Cr of Brittle	ion Yerma ncy Core kage Core d Value, itical Tem Fracture	akow From. Cooling Syst re Tamplets perature	D IP LLAA MSI TKEOL	Dummie: Injection Low Lea Main Ste Critical T Fracture	s Point kage after / am Isolatio emperature at End of L	Annealing n of Brittle ife

3.1.1 Neutron Embrittlement of the Weld Seam Close to the Core

So far the RPVs in all plants concerned have been annealed in the weld seam area close to the core. Before and after the annealing samples were taken from the wall of some un-

cladded RPVs. Testing of the small samples taken confirmed the suitability of the annealing technology applied which had been developed and tested several times in the Soviet Union. In addition to the Soviet side, an EdF/Siemens-KWU-consortium commissioned by WANO, as well as Czech and Finnish institutions, also participated in these investigations.

At present, the re-embrittlement after annealing is additionally being tested in an international test programme.

After annealing the neutron flux density in the RPV wall is kept low by the use of shielding elements on the edge positions of the core or by low leakage core management so that the embrittlement of the material of the weld seam close to the core proceeds much slower.

3.1.2 Security against Fracture of the Reactor Pressure Vessel

There is security against fracture of the RPVs as long as the actual value of the brittle fracture transition temperature for the weld seam close to the core is lower than the maximum allowable value. At present this condition is fulfilled by all RPVs mentioned apart from Unit 1 of the Kozloduy Nuclear Power Plant in Bulgaria. But the remaining service life until the limit of the individual units is reached differs considerably. For the Kozloduy 1 RPV only a restricted proof of security against fracture is possible. Based on the results of RPV in-service inspections the operation of Kozloduy 1 was restricted to the current fuel cycle.

The maximum allowable value of the brittle fracture transition temperature for the weld seam close to the core depends on the plant-specific circumstances and is principally not transferable (Table 3-2).

The decay heat of the core also exerts a strong influence. With lower decay heat the maximum permissible values of the brittle fracture transition temperature are considerably lower.

Analytical as well as numerical procedures are used to calculate the maximum permissible brittle fracture transition temperature. To qualify these computation procedures, GRS together with the IVO and the Kurchatov Institute carried out model computations for cladded and uncladded RPV in WWER plants. This comparison showed that the "temporary methodology" applied to verifications in the former Soviet Union is conservative in comparison to the more sophisticated 3-D-FEM computations. It is still to be examined whether this statement always holds true.

To increase the safety level of this type, consideration was given among other things to upgrading the emergency core cooling systems for primary leakages equivalent to D_{NOM} 200, which corresponds to the rupture of a volume control line. The influence of this leakage accident on the RPV integrity was investigated by the EdF/Siemens consortium commissioned by the WANO. The results show that the weld seam above the core is also to be integrated into the brittle fracture analysis of the RPV. This example shows that the upgrading measures in different areas, which are seemingly independent of each other, must be co-ordinated.

Table 3-2	Maximum allowable transition temperatures in emergency situations with regard
to RPV-2 (c	cold leg injection); (Gidropress 230-P-045, 1992)

Kozloduy RPV-2	P [%]	Cu [%]	NI [%]	Annealed	Тко cal.	Тко meas.
A BARRANS STORAGE	0,0375	0,18	0,20	1992	10 °C	50 °C
				TK,92 AA = 70 °C		
				Т к,95 = 140 °С]	
Cooling Tra	nsient	T _k A I	n °C Weld	No. 0.1.4	T _K A in M	°C Base etal
	Decay Heat	Smooth Wall	G	rinded Wall	Smooth Wall	Grinded Wall
Primary Break	Yes	237		214		209
DN 32	No	233		179		176
Primary Break	Yes	192	145		191	141
DN 20,2 ECCP	No	127		102		97
Primary Break	Yes	230	208		225	202
DN 20,1 ECCP	No	224	1	193		193
Steam Line Rup-	Yes	199	183		193	173
ture without FAV	No	179	1.1-	163		155
with FAV	No	194		178		170
BRU-A Failure		277	1012 12 0	204	221	199

FAV Fast Acting Valve in Main Steam Line

3.2 Exclusion of Fracture in the Main Pipes

The main pipes of WWER-440 reactors consist of austenitic steel. Leak and break assumptions for the design of the pipe system and for the design of the supports as pipe whip restraints are not sufficiently known. At the time of design in the sixties the main pipes were not qualified to exclude breaks.

Different institutions, like the VNIIAES in Moscow, the Nuclear Research Centre at Rez and an EdF/SIEMENS/Energoprojekt-consortium examined whether a leak-before-break behaviour can be assumed for the main pipes and the volume control lines. In this case large leaks could be excluded from the design basis. The investigations which were partially carried out independently show similar results (Table 3-3). During normal operation a leak-beforebreak behaviour is to be expected for both pipe systems. In case of additional loads from the design basis earthquake a leak-before-break can only be demonstrated for the main coolant lines. The proof cannot be provided for the volume control lines as the critical defect size is smaller than the defect size which can be detected by the leakage control systems installed. Changes at supports and damping devices of the pipes concerned become necessary here. Main steam and feedwater lines have so far been included into these investigations to a limited extent only.

 Table 3-3
 Critical and detectable crack sizes of the main coolant and volume control lines in

 WWER-440/W-230 plants - examples (according to EdF/Siemens-KWU/ENERGOPROJEKT

 S-514/92)

Pipe	Crack	Wall	Wall-Penetrating Crack Length 2c [mm]				
	length	thick- ness [mm]	detectable	2c _{krit} during normal operation	2c _{krit} during normal operation + design basis earthquake		
Main Coolant	Line						
hot leg	scope	32	90	536	417		
cold leg	scope	32	90	566	435		
hot leg	axial	48	140	176	166		
cold leg	axial	46	140	162	159		
cold leg	axial	70	200	322	317		

Pipe	Crack	Wall	Wall-Penetrating Crack Length 2c [mm]					
Atary !	length	thick- ness [mm]	detectable	2c _{krit} during normal operation	2c _{krit} during normal operation + design basis earthquake			
Volume Contr	ol Lines	_						
leg A	scope	18	50	170	6			
leg B	scope	18	50	248	36			
leg C	scope	18	50	240	57			
leg A/B/C	axial	25	70	106	63			

3.3 In-Service Testing

Previous operating experience has shown that it is possible to control the design deficiencies of the pressurised components by purposefully adapted in-service inspections. Here the existing test restrictions, in particular, are to be analysed and assessed critically.

As the following example shows there are, however, also considerations of exceeding the limits of non-destructive test procedures:

According to the design regulations, brittle fracture safety of the RPVs for assumed cracks at the inner surface is to be demonstrated up to a depth of one quarter of the wall thickness.

For advanced embrittlement of the weld seam close to the core this proof can only be provided to a limited extent, i.e. for limited crack configurations.

The dimensions of cracks close to the surface which are already inadmissible thus, for example, reach the range of reference errors of the test procedures applied. In our opinion this approach exceeds the limits of permissible integrity verifications and can only be accepted in exceptional cases and for a limited period of time.

4 Conclusion

There have been several independent investigations relating to the safety status of nuclear power plants with WWER-440/W-230 reactors in the past few years. The safety deficiencies of the design are known and they have been classified according to their significance for safety.

During the last 10 years supplementary investigations to improve the safety status and to assess the integrity of components and pipes have been carried out. The results of these investigations have so far only in exceptional cases been considered for the preparation of updated plant-specific safety reports for this type, so that uncertainties in supervision and licensing have occurred again and again.

Apart from the measures for re-establishing the project status, upgrading measures for ensuring the integrity of the components and the pipes are to be mentioned in the first place.

Ensuring the integrity of the components is not only a problem of RPV embrittlement but, as shown, also a problem of damage prevention. This has partially not always been considered.

The main emphases of further projects are:

- to verify RPV brittle fracture assessment,
- to upgrade the exclusion of fracture of main pipes and areas with screwed connections,
- to upgrade the in-service testing programme.

Upgrading measures are to be checked with respect to their balance. For certain plants RPV integrity can become the decisive weakness for upgradings.

Concepts for long-term upgrading can thus differ from site to site and also from unit to unit. Costly systems engineering and structural changes are generally to be assessed with respect to the safety increase for the overall plant. The means could perhaps better be used for damage prevention, for example, for upgrading the main pipes for break exclusion. The example of a break of the volume control line with a subsequent RPV failure at the weld seam above the core clearly demonstrates this context.

Situation and Measures to Improve Safety at the Rovno Nuclear Power Plant, Rovno/Ukraine

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1 Situation and Problems

The Rovno Nuclear Power Plant is one of five nuclear power plant sites in the Ukraine.

On four sites, 12 nuclear power plants with Russian pressurised water reactors of the newer types WWER-1000/W-320 and WWER-440/W213 are operating. The power installed is 10880 MW. On the fifth site, the Chernobyl Nuclear Power Plant, two RBMK-1000 type reactors with 1000 MW each are in operation.

Nuclear Power Plant	Туре	No. of Units	Planned Life-time	Status
Rovno	WWER-440/W-213	2	2010, 2011	in operation
	WWER-1000	1	2016	in operation
	WWER-1000	1	all Million	80 % completed
Chemelnitzky	WWER-1000	1	2017	in operation
	WWER-1000	3		80 %, 5 %, 2 % completed
Zaporozhe	WWER-1000	5	2014 - 2019	in operation
	WWER-1000	1		95 % completed
Southern Ukraine	WWER-1000/W-302, 338, 320	3	2012 - 2019	in operation
Unidino	WWER-1000	1		50 % completed
Chernobyl	RBMK	2		in operation

Table I-I Nuclear power plants in the Okrain	Table 1-1	Nuclear	power	plants i	n the	Ukrain
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Owing to the decreased energy consumption as a consequence of the economic depression and the simultaneous reduction of oil and gas imports from Russia, a considerable part of the newer thermal power plants are no longer operated. The decreased energy demand is therefore predominantly met by nuclear power plants and coal-fired power plants which has led to a relative increase of the share of coal and nuclear energy in the electrical energy production.

	1990		1991		1992		1993	
	GWh	%	GWh	%	GWh	%	GWh	%
Thermal power station	211.6	70.9	191.6	68.8	170.8	67.6	142.5	62.2
Hydro-electric power plant	10.7	3.6	11.9	4.3	8.1	3.2	11.2	4.9
Nuclear Power Plant	76.2	25.5	75.1	26.9	73.7	29.2	75.2	32.9
Total	298.5	-	278.6		252.6		228.9	

Table 1-2 Share of the sources of energy in the electrical energy production of the Ukraine

The energy saving potential, currently hardly developed, must increasingly be used for decreasing environmental pollution in the future. A large proportion of the coal-fired power plants is worn out and inefficient; 20 % are older than 30 years. Flue gas dust collectors are not installed in any thermal power station. New thermal or hydro-electric power plants to replace the old ones are not being constructed.

The Ukraine does not want to and cannot do without nuclear energy. For this reason the Ukrainian parliament renounced the two resolutions to close down the Chernobyl Nuclear Power Plant in 1993 and the five-year moratorium on the further construction of new nuclear power plants on October 21, 1993.

In the current transitional period there are political, economic and social conditions which impair the power plant operation and which delay the realisation of upgrading measures. These are in particular:

Inflation and economic regression

During the last four years the Coupon has experienced a substantial loss against the Rouble and the Western currencies. (In 1991 1 US\$ was about 20 Coupons, nowadays the relation is 1 US\$ = approx. 70,000 Coupons).

Fluctuation of staff

The staff move to Russian nuclear power plants, or elsewhere, because the salaries are higher there.

- Production costs covered to a limited extent only
 Only up to about 50 % of the total production costs are covered by the selling price for electrical energy.
- Bad payment behaviour of energy consumers.
- Access of Ukrainian power plant operators to information from industry, science and research in the Russian part of the former Soviet Union has become more complicated because of different currencies and changed responsibilities.
- Difficulties in the procurement of fresh fuel and in the recycling of spent fuel and radioactive wasteThe situation has improved after the American-Russian-Ukrainian agreements on arms plutonium, but the Ukraine nevertheless plans the establishment of its own fuel cycle.
- Shortage of spare parts because of financial and procurement difficulties.
- Slow change and adaptation of social structures and customs to new challenges.
- Power plant management is weighed down by every-day problems;
- Establishment phase of state supervision, licensing and monitoring.

Under the present conditions it is impossible for the nuclear power plant directors to build up an efficient operational organisation following the Western pattern. The director of the Rovno Nuclear Power Plant thus does not only have to care for energy production, but also for the construction of the new unit 4, for services for the power plant and for the residential town including the supply of a total of 9,000 employees.

In his business organisational decisions he has to consider that the town of Kusnetzovsk located close to the nuclear power plant with its 40,000 inhabitants depends on the nuclear power plant.



Fig. 1 Rovno Nuclear Power Plant: reactor cavity of Unit 3 (Photo: H.-J. Burkhard)

2 Concept for Increasing the Safety of the Rovno Nuclear Power Plant

On the Rovno Nuclear Power Plant site there are four pressurised water reactors of the newer types WWER-1000/W-320 and WWER-440/W213. The first two units of the WWER-440/W-213 (like units 5-8 at Greifswald) have been in operation since 1980 and 1981 respectively, the third unit (like units 1, 2 at Stendal) since 1986. The fourth unit will probably start its operation in 1997.

2.1 Licensing Situation

Until the Atomic Energy Act comes into force the dealings with radioactive substances and the use of nuclear energy are regulated by statutory rule. At present the Atomic Act is being discussed in parliament.

The atomic licences legally required are granted by the Ukrainian supervisory authority established in 1991, the Ukrainian state committee for Nuclear Safety and Radiation Protection (UkrSCNRS). The UkrSCNRS state committee has been incorporated into the new ministry for environmental protection and reactor safety since the beginning of 1995.

The operation of the Ukrainian nuclear power plants is still based on the commissioning permits of the former Soviet atomic energy ministry. Since 1994 the annual start-up after the revisions and the individual initial start-up phases have required the permission of the Ukrainian authority.

The former UkrSCNRS intended to introduce the atomic licensing procedure by the end of 1995 and to grant operating licences on the basis of the Ukrainian Atomic Energy Act. As a preparation, the UkrSCNRS asked the nuclear power plants to establish safety status reports.

To speed up the process, the UkrSCNRS and its Scientific-Technical Centre themselves want to participate in the preparation of the first safety status report. Since 1992 the work has been performed within the framework of the national project "Rovno" under the management of the Rovno Nuclear Power Plant and with the participation of the Scientific-Technical Centre of the UkrSCNRS as well as further Ukrainian and Russian organisations.

With its participation, the authority, being aware of the different Western practice, wants to train its own employees in an independent safety assessment role as well as in the specification of licensing guidelines and regulations.

The State Committee for Atomic Energy (Derschkomatom) acts on behalf of industry. It represents the umbrella organisation for Ukrainian nuclear power plants, nuclear industry and the project planning institutes in Kiev and Charkov. In spring 1995 Derschkomatom will be incorporated into the energy ministry.

The restructuring by the government is also to achieve a better separation of the responsibilities between operators and manufacturers on the one hand and supervision and licensing on the other hand.

2.2 Safety Status

Both types of power plants (WWER-440/W-213 and WWER-1000/W-320) were designed at the end of the sixties and at the beginning of the seventies, respectively. They were designed for controlling a range of design basis accidents up to the rupture of the main coolant line under consideration of a single fault in a safety train.

The design was based on the basic safety regulations OPB-73 and PBJa-74, valid at the beginning of the seventies, and their subordinate rules.

During the design phase the WWER-1000/W-320 was adapted to the requirements of the revised OPB-82, valid from 1982 onwards.

The results of the safety assessments for the Greifswald, Stendal and Rovno Nuclear Power Plants confirm that the Rovno units meet the basic safety requirements and that the existing deficiencies can largely be removed or compensated by backfitting measures, which, among other things, is also due to the relatively good-natured plant behaviour, which is a consequence of greater water volumes, smaller power densities in the core and the possibility of isolating the steam generators on the primary side.

Quality deficits have arisen during the realisation of the design in the real plant and they do exist in a number of components.

The further adaptation of the units in operation to the current basic safety regulations OPB-88 and PBJa-89 enforced in 1988/89 has been started with initial measures. The rapid continuation of necessary safety improvements, among other things, has been delayed because of the structural conditions and economic constraints. To ensure conservation of the design basis state of the plants and components requires everyday efforts.

To evaluate the safety status, there are now more and more methods and programmes for safety assessment. The Ukraine receives German, French and American support in the performance of safety assessments (transfer of codes, computation technology and training of applicants), especially within the framework of bilateral agreements.

The TACIS programme of the European Union (EU) for the "Technical Assistance to the Republics of the Former Soviet Union (TACIS)" is going to provide for the transfer of further codes for accident analysis. The effectiveness of the individual upgrading measures can be checked with the Western codes and the respective Russian computation programmes. The operators' umbrella organisation Derschkomatom worked out upgrading programmes together with the nuclear power plant. Unit-specific upgrading programmes already do exist for units 1 and 2 of the Rovno Nuclear Power Plant.

2.2.1 WWER-440/W-213

For the period between 1994 and 1998, 16 measures increasing the safety of the urgent categories 3 and 4 according to the IAEA classification and approx. 50 operational upgradings were determined for each of the two units.

- The preparation of the upgrading programme was based on the following documents:
- GRS Safety Assessment Greifswald, Unit 5 ("green book"),
- Russian upgrading concept for WWER-440/W-213 reactors,
- TACIS-"Safety Assessment Rovno",
- operating experience.

Essential upgradings concern:

- brittle fracture protection of the RPV,
- additional emergency injection the steam generator (bypassing the turbine hall),
- improvement of tightness and examination of the confinement system (confinement with pool-type pressure suppression system),
- upon actuation of emergency cooling, it shall only be switched over to emergency power upon failure of the station service power supply,
- common cause failure prevention, e.g.:
 - obstruction of the emergency cooling sump return line with insulation material,
 - · same fault pattern of measuring transducers for the actuation of reactor scram and
 - fire protection,

procedures for design basis accidents and beyond-design-basis event sequences. (The
accident "opening of the collector lid" in unit 1 in 1982, apart from corrosion problems at
collector stud bolts and blind threaded holes, also drew attention to necessary improvements in procedures).

A number of improvements which were recommended in the Greifswald report have already been implemented:

- Modification of the nuclear fuel core loading plan and the heating of emergency cooling tanks and accumulators to decrease brittle fracture,
- Exchange of the neutron flux measurement systems,
- Automated reactor protection inspection,
- Prevention of start-up air failure of the diesel generators,
- New filling level control of steam generators,



Fig. 2 Rovno Nuclear Power Plant: reactor hall with 2 WWER-440 units (Photo: H.-J. Burkhard)

- Gradual exchange of:
 - · Measurement transducers for reactor scram measurements,
 - · Safety automats for power supply,
 - · Cable bushings through the confinement.

2.2.2 WWER-1000/W-320

There is also an upgrading programme (1994 - 1998) for this type. It has not yet been determined in detail for the individual nuclear power plant units. Similar to the upgrading concept of the Russian operator concern Rosenergoatom, the upgrading programme is based on the following documents:

- Summary of the deviations of the existing design from the regulations applying today, as well as the safety assessment of the existing deviations,
- the recommendations SM-90-WWER of the former Soviet Atomic Energy Ministry to increase the reliability and safety of nuclear power plants in operation,
- Upgrading suggestions of the design institute Teploenergoprojekt,
- Safety assessment of the Temelin Nuclear Power Plant,
- PSA-level 1 for the Balakovo-4 Nuclear Power Plant.

The most urgent measures (category 3 according to IAEA classification) are to be realised gradually within the next four years. The new components and systems generally are to meet the current regulations. However, for economic reasons, equipment which does not meet entirely the current requirements of the regulations will also be used.

The safety gain achieved and the contribution of the individual upgrading measures are to be evaluated by a PSA study. The realisation of the upgrading programme is, however, not dependent on the existence of the PSA results.

First agreements on the establishment of a PSA have been made. The results are, however, not expected within the next four years. Within the framework of the TACIS programme it is planned to perform a PSA for Rovno, Unit 3.

To prepare a PSA for a WWER-1000/W-320, GRS together with Russian and Ukrainian organisations carry out methodical examinations of different initiating events and the event sequences resulting therefrom, e.g. loss of off-site power, main steam lines fracture and steam generator heater tube fracture, taking the Zaporozhe Nuclear Power Plant, Unit 5 as an example.

The four-year programme for upgrading the WWER-1000/W-320 comprises about 50 technical and 30 organisational upgrading measures.

- Essential measures, among other things, concern:
- Optimisation and supervision of the core,
- design basis insertion of all shutdown rods,
- subcriticality control and the prevention of coolant dilution,
- the prevention of common mode failure, e.g.:
 - blockage of all emergency core cooling and residual heat removal system with insulation material; joint impulse lines for several reactor scram criteria,
 - · joint ventilation for main and standby control room and
 - fire protection,
- gradual exchange of steam generators (crack formation in the collectors),
- procedures relating to beyond-design basis sequences of events, among other things for larger steam generator leakages,
- renewal of the neutron flux measurement system and the station service power supply,
- automated material inspection facilities,
- investigations of the application of the "leak-before-break" criterion for primary system and secondary system pipes and investigations of the brittle fracture safety of the reactor pressure vessel (RPV),
- diagnostics for monitoring leaks, component oscillations and valve positions.

The realisation of the upgrading programme and also the preventive maintenance to a decisive extent depend on whether the equipment can be financed and procured. 70 percent of the equipment components are produced in Russia, but the power plant only possesses limited amounts of Roubles and Western currencies for buying this equipment.

The nuclear power plant also tries to improve plant safety, parallel to the investments in machinery, process technology and instrumentation and control, by plant organisational measures with comparatively low expense, e.g.:

- plant and operational documentation,
- operational and accident training and further reinforcement of the safety culture,



- quality assurance, work organisation, house-keeping measures.

Fig. 3 Rovno Nuclear Power Plant: control room of Unit 3 (WWER-1000); (Photo: H.-J. Burkhard)

3 International Co-operation

In their efforts to increase the safety of nuclear power plants the Ukraine is supported by biand multilateral programmes. These essentially are programmes of the U.S.A., Germany, France, the EU, the G7 and the IAEA. On behalf of industry, there are also a number of activities and participation like the twinning programme of WANO between Western and Ukrainian nuclear power plants.

3.1 Bilateral Support

The American support concentrates among other things on fire protection, the improvement of operational documentation, especially as regards symptom-oriented accident procedures in the Rovno Nuclear Plant, units 1 and 2, the training of staff by supplying power plant simulators and training schemes for Ukrainian experts in the United States.

France performs joint safety investigations with the Scientific-Technical Centre of the UkrSCNRS, supplies computation codes and workstations and offers visits to French nuclear facilities. In addition, EdF contributes additional funds within the framework of the TACIS Rovno project.

For better co-ordination of their activities and for direct support, IPSN and GRS established a joint office in Kiev in 1993. The German side makes considerable contributions to the improvement of safety. The German programme comprises a total of approx. \$35 million, the supply of equipment representing about 70 % of the costs.

The main emphases of the German support are the joint safety assessments with the supervisory and licensing authority as well as with the nuclear power plants. The findings of the GRS safety studies on Greifswald and Stendal are used here.

The UkrSCNRS suggested subjects for the BMU co-operation programme and proposed the Rovno Nuclear Power Plant as the reference plant. Commissioned by the BMU, GRS performs the work jointly with the Ukrainian partners or co-ordinates the work of other German partners. The support programme is intended to indirectly strengthen the expert competence of the authority and its Scientific-Technical Centre or to directly contribute to the improvement of the technical safety of the plant. The entire programme is designed in such a way that the individual measures supplement each other and fit into the Ukrainian programme. The objectives of the BMU co-operation programme are to create examples, which can be continued independently in the Ukraine, and the initiation of important safety-related functions, like

- the consistent analysis and application of operating experience,
- realistic accident investigations as a basis for the establishment of well-secured emergency measures.



Fig. 4 Rovno Nuclear Power Plant: view from Kuznetzovsk, the residential town for the power plant staff (Photo: H.-J. Burkhard)

In the German programme the safety investigations are followed by an investment programme. The measures required have been initiated so that the realisation can take place in 1995.

The following points of the BMU programme refer directly to the Rovno Nuclear Power Plant:

 joint safety assessments of the accident behaviour of the reactor plant and the safety confinement; of reactor physical core computation; of accident instrumentation; of the reliability of electrical power supply; of material inspection and the applicability of the break exclusion concept and of the assessment of operating experience,

- the improvement of the plant and operational documentation and communication,
- the training of operating personnel at the Greifswald simulator,
- seminars and visits to German nuclear power plants and other facilities,
- joint safety investigations by German firms with the nuclear power plant and the authority commissioned by the BMU/GRS, to prepare technical and organisational support measures for increasing plant safety in the fields of
 - · fire protection,
 - · plant diagnostics,
 - operational organisation,

the German investment programme including supplies of technical equipment until the end of 1995 for

- material inspections,
- · plant diagnostics,
- · fire protection,
- internal communication.

3.2 Support Programme of the European Union

Two forms of co-operation concerning the Rovno Nuclear Power Plant have been initiated within the framework of the TACIS programme:

- Support of the UkrSCNRS with the projects
 - Safety assessment of the Rovno Nuclear Power Plant, units 1, 2 and 3 by a consortium of six Western expert organisations under the direction of IPSN and GRS,
 - Transfer of West-European practice and methods of supervisory safety assurance;
- Local support of the Rovno Nuclear Power Plant by a Western operator, the EdF (similar support also exists for the other Ukrainian nuclear power plants).

The TACIS-project "Rovno Safety Assessment" in the first one-year phase pursued the following main objectives:

- to impart the experience of Western methods and requirements of the deterministic and probabilistic safety assessment to employees of the authority,
- to identify the safety technical state of the three plants, detecting missing verifications and technical and organisational safety deficiencies, and to recommend upgrading measures.

On the basis of these findings and its own knowledge the authority requires the nuclear power plant to upgrade the plant and to submit the missing verifications.

The final report (English and Russian edition) states the basis for and the performance of the assessment, the upgrading recommendations and the verifications to ensure that the results of the safety assessment can be reconstructed. The results confirm or complement the Ukrainian and Russian upgrading suggestions on individual points. Appendix A2 comprises a selection of the upgrading suggestions.

It is the object of the TACIS-Project "Local Support of the Rovno Nuclear Power Plant" by the EdF to check the upgrading programmes together with the operator on the basis of the safety assessments performed and to suggest measures for Western investment. The financial budget of this Rovno project is 11 MEcu from TACIS-92, -93 and 94. 40 % of these means are intended for the supply of equipment.

On the basis of the project currently financed via TACIS-92 (4 MEcu), three supply contracts with a total budget of approx. 1.6 MEcu for units 1 and 2 have been concluded recently:

- Improvement of fire protection by exchange of floor coverings (Geholit),
- Oscillation monitoring of MCPs (Brühl & Kehr),
- Steam generator-manipulator (Intercontrol, Siemens).

Furthermore, TACIS-93 provides for the information system MADAM-S for units 1 and 2 and TACIS-94 for the exchange of all pressuriser safety valves of units 1, 2 and 3.


Fig. 5 German visit at the Rovno Nuclear Power Plant to initiate the investment programme. Model of the Rovno site: the WWER-440 double unit and the two WWER-1000 units can be seen from left to right. Unit 3 is in operation and unit 4 is being constructed.

3.3 G7 Initiative

During the preparation of possible support for the Ukraine by the seven biggest industrial states (G7), the financial need for the closedown of the Chernobyl Nuclear Power Plant and the completion of the WWER-1000 units at Zaporozhe-6, Rovno-4 and Chmelnitzky-2 was estimated to be about \$1.5 billion. About \$300 to \$400 million were indicated for Rovno-4. These estimates also included safety-related upgrading measures going beyond the original design. The upgrading measures supported by the West should in any case be based on the results of the Western bi- and multilateral WWER safety assessments. The initiation of actual support measures still requires further clarifications between the two parties.



Fig. 6 The Zaporozhe Nuclear Power Plant - the biggest nuclear power plant in the world (6 units of the WWER-1000 type) (Photo: H.-J. Burkhard)

4 Examples of Safety Improvements

4.1 The WWER-1000 Core

The operational and core monitoring experiences require:

- to increase the effectiveness (by about 15-20 %) and to ensure rapid insertion (4s) of the shutdown rods,
- to ensure negative reactivity feedback during increasing moderator temperature and density (zero power with absorber rods being withdrawn at the beginning of the loading),
- to lower recriticality temperature 100°C,
- to lower the neutron flux density on the RPV wall by approx. 25 to 30 %,

- to improve core monitoring and to limit the power density distribution,
- to monitor subcriticality reliably.

The following improvement measures are suggested:

- change the fuel element spacer grids and guide tube material from steel to zirconium,
- enlargement of the diameters for the shutdown rod guide tubes (further measures for ensuring insertion are listed in Appendix 1),
- the application of burnable uranium-gadolinium absorbers,
- low-leakage fuel element loading scheme,
- change the in-core measurement system,
- improved instrumentation for monitoring subcriticality.

The completion of the design and the gradual introduction will begin in 1995. Some new fuel elements are already being used for trials.

With German support, the UkrSCNRS is creating the preconditions for performing independent reactor physics core computations. The German programmes developed and tested for WWER reactors by K.A.B. Berlin and the Rossendorf Research Centre, together with Ukrainian physicists, are being adopted. Presumably from 1995 on the authority will check the core with these programmes prior to the annual start-up of the plants after the core reload. The nuclear power plants determine the loading scheme for the core with Russian reactor physics programmes.

As the familiarisation of the Ukrainian physicists with the German reactor physics programmes has not been completed yet, GRS and K.A.B. Berlin, subcontracted by GRS, carried out investigations on the shutdown safety, for the deviations which occur during the insertion of the shutdown rods.

The effects of the shutdown rods getting stuck in the lower third of the core or of their delayed insertion (8-10s) on the safe shutdown of the reactor were examined for four different accidents with a reduction of the coolant flow rate and fast reactivity injection. The results were compared with Gidropress results.

4.2 Plant and Operational Documentation

In 1992 GRS and K.A.B. Berlin, subcontracted by GRS, started their work on the pilot project "Improvement of the plant and operational documentation for the Rovno Nuclear Power Plant, Unit 3" together with the Rovno Nuclear Power Plant and the UkrSCNRS. (Appendix A3).

Taking into account the ideas of the Ukrainian partners and GRS experience with documentation concepts for German nuclear power plants, documents meeting Western standards will be developed after a joint programme has been established.

Exemplary sample documents have been produced for central components of the plant and operational documentation:

- Inventory lists for the emergency cooling and core flooding system including statements on design, operation, functional testing and maintenance,
- improved system diagrams by use of the local CAD technology (Fig. 7) and of the CAD-diagrams produced by K.A.B. for the Stendal Nuclear Power Plant. In addition to the high quality of the drawings, with a uniform plant characterisation and symbol selection, the information content is also greater, among other things because boundary limits, local and remote displays, basic and standby settings are indicated.
- Descriptions for the emergency cooling and the core flooding system, subdivided according to the structure and operating mode of the system, including a description of all essential aspects, locks, controls, monitoring, etc. The descriptions are also used as training documents,
- Accident decision tree and accident guiding scheme,
- Procedures (detailed process descriptions and instructions for operation) for normal operation, disturbances and accidents; a sample procedure is being prepared for the small leak; the outline version has already been completed.

Referring to the documents relating to the accident decision tree, the accident guiding scheme and the accident procedures, the operator decided to follow the example of the Konvoi plants. During the preparation of the system diagrams differences between the plant and the drawing could be detected and corrected. The operator is convinced of the quality and the benefit of the present results and continues the system documentation independently.



Fig. 7 Local area network in Rovno

If the work is continued without delay, the documentation will fulfil essential preconditions for the safety status report required by the authorities.

It was decided to immediately produce all drawings for unit 4, which is currently being constructed, following the high quality of the sample documents. Other Ukrainian and Russian nuclear power plants have already interested themselves in Rovno. Together with the experts from the Rovno Nuclear Power Plant they took over diagrams or adapted these diagrams for their own plants.

From the beginning of 1995 this project will be continued trilaterally with the Russian Balakovo Nuclear Power Plant, which is the same type of plant and the reference plant for the German co-operation with Russia. This division of labour and the mutual exchange of experience will speed up the preparation and harmonisation of the documents. For this purpose, GRS is going to provide data processing equipment to the Balakovo Nuclear Power Plant which corresponds to the equipment of the Rovno Nuclear Power Plant. In an additional project supported by the BMU, the Ukrainian authority is supported by GRS in the preparation of drafts for national guidelines for plant and operational documentation (secondary, safety and licensing documentation). The authority here uses its experience from its participation in the Rovno Nuclear Power Plant project to produce further practice-oriented drafts of standards for the kinds of documents involved.

4.3 Continuation of the Support within the Framework of theTACIS-Programme

At present the proposals for the second phase for improving safety in the Rovno Nuclear Power Plant are being examined by the European Union. The proposals have been prepared by the four participating sides, the Rovno Nuclear Power Plant and its partner, EdF, as well as the authority and its partner, the Western consortium led by IPSN and GRS.

The proposals are based on the results of the first phases of the two TACIS projects related to Rovno.

he UkrSCNRS will examine the safety assessment completed during the first TACIS phase and will request the verifications and analyses still missing from the operator. The Rovno Nuclear Power Plant was already requested to present an upgrading programme and an actualisation of the TOB (report to substantiate technical safety) within the framework of the safety status report to be prepared.

The UkrSCNRS will base its licence to be granted for the Rovno plant on these documents.

The consortium will support the authority during the examination of the upgrading programme, the verifications, the actualised TOB and the implementation of the upgrading measures. Verifying computations will be performed for selected accidents. The PSA to be presented by the nuclear power plant will be checked together with the authority and possibly further developed methodically.

Taking the verifications as an example, Western practice and methods concerning safety examination, the content of the safety report and the licensing procedure are made accessible to the Ukrainian authority. In addition, the verification of the upgrading programme by the consortium as an independent expert organisation is a precondition for the Western financial support during the realisation of individual measures.

EdF actively supports the Rovno Nuclear Power Plant in the fulfilment of the requirements of the authorities and provides local consultations in safety-related operational issues. In addi-

tion, EdF prepares the invitations to tender for the supply of equipment and the incorporation of the measures in the plant (see Section 3.2).

5 Summary

Never before has a nuclear power plant of the former Soviet Union opened its gates as widely for Western experts as the Rovno Nuclear Power Plant during the various bi- and multilateral co-operations in the last two years. It was, however, not always easy to convince the nuclear power plant officials that Western investments are normally supported by joint safety investigations.

Nevertheless the German side in its programme tried to strike a balance between safety investigations and investments.

Nuclear power plant management knows about the need to upgrade its facilities. The realisation of the first measures and the continued identification of further project solutions have already been started. The implementation of the improvements is, however, impaired by the difficulties in procuring quality equipment.

The supply of equipment promised within the framework of the German investment programme for unit 3 and the TACIS project "Local Examination of the Rovno Nuclear Power Plant" for units 1 and 2 show the operator and the authorities that the safety examinations are followed by investments. In addition, nuclear power plant management underlines the benefits of a series of other support measures, e.g.:

- Simulator training of the 440-MW unit shift staff at Greifswald. Apart from the regular training at the Greifswald simulator, a large number of employees get the opportunity to familiarise themselves with Western practice, for example in the Brokdorf Nuclear Power Plant or by travelling abroad at all. According to the director, this final point is a contribution to the comparatively low fluctuation of staff in the Rovno Nuclear Power Plant, which must not be underestimated.
- Documentation project with CAD technology (see Section 4.2).
- Free provision of spare parts from the Greifswald Nuclear Power Plant, among other things of a 220-MW generator.

The Western partner states put great emphasis on the involvement of the authority in the safety improvements and on the separation of the responsibilities between the authority and the operator. As a support measure, the authority is familiarised with the methods of Western safety examinations. In addition, the authority receives computation programmes and equipment for independent safety examinations.

The work on the TACIS project Rovno, for example, shows that operator, technical designer and project engineer also pay much attention to the qualification and expert commitment of the employees working for the authority.

According to all experts, future safety assessments, to a greater extent than in the past, should be based on the findings of prior WWER investigations and operating experience.

Although the Western support shows deficits in the concentration of activities and in co-ordination, it has nevertheless after two years begun to show visible and recognised contributions to ensuring and improving nuclear power plant safety at the Rovno Nuclear Power Plant.

Basic improvements do, however, require the continuation of the Ukrainian upgrading programmes, the exchange of outdated components and the creation of an effective operational organisation without delay, as well as a consequent attention to operating experience and alterations in the operational organisation which promote safety-awareness.

If, in addition, the Western support is continued successfully, the Rovno Nuclear Power Plant will be able to achieve an exemplary safety status in the Ukraine.

For this purpose, the nuclear power plant requires good basic conditions with a stable economy.

Appendix A1

Compilation of Possible Causes and Suggestions for Solutions to the Problem of Delayed Insertion of WWER-1000 Shutdown Rods or these Rods Being Stuck

Problems with the shutdown rods have occurred since 1992 after the introduction of the 3year fuel cycle. On the occasion of the assessment of the Russian WWER-1000 upgrading programme Gidropress provided information about:

Possible Causes

Increased friction between shutdown rod and guide tube due to bowing of the fuel element guide tubes with an increased service life. This also led to the distortion of the entire fuel element. Reasons for this are:

- Increased axial load on the fuel element by 18 guide tubes from the protecting tube unit after closure of the reactor lid,
- additional temperature gradients in the boundary fuel elements due to increased power imbalances within and outside the fuel elements,
- the free spring movement as the fuel element head approaches zero,
- the axial load absorbability and the geometric stability of the fuel elements decrease with an increasing service life, as the 15 steel spacer grids and the Zr-jackets of the fuel elements expand differently and the spacer grid cells lose their elasticity.

Gidropress calculated that reactor shutdown can be ensured for all accidents even with insertion times of 8-10s (the design basis insertion time being 4s).

Suggestions for Ensuring the Functional Insertion of the Shutdown Rods

- Shortening of protecting tube unit and thus reduction of the pressure against the fuel elements,
- monitoring of the spring movement at the fuel element heads,
- elimination of those fuel elements which fail a guide tube bore-gauge test,
- extension of the spring movement of new fuel elements,
- increasing the weight of the shutdown mechanisms,
- no fuel elements with an increased burn-up at the shutdown rod positions.

Appendix A2

Selected Results of the TACIS Safety Assessment of the Rovno Nuclear Power Plant (Consortium directed by IPSN and GRS)

Table A2-1				
Core	WWER-440	WWER-1000		
Core power distributions	Improvement of the adjustment between calculated results and measured data from in-core and ex-core detectors for automatic control.			
	The operational use of the control elements h optimised to avoid the initiation of xenon instat			
Reactivity Control		Blockage of control rods events.		
	Automatic insertion limitation.			
Criticality during refuelling	The possibility to install additional subcriticality monitoring should be considered during the refuelling.			
Stability		Two reactor trips should be generated:		
		- on low DNBR		
- on high W/cm				

Table A2-2				
Plant Operation	WWER-440	WWER-1000		
Management, organisation and administration	 Improvement of the safety culture safety aspects by personnel. Implementation of a new organisat evaluation group and a site safety Installation of a project manager controlling and co-ordinating th modifications. Increase of personnel motivation process which will be to the benefit Improvement of the documents, include the experience feedbac preservation of the acquired know- Review of the punishment syste distribution of responsibilities and the safety of the safe	by realising a better understanding of the tional structure with an independent safety independent surveillance committee. ment organisation with the aim of better ne acitivities related to the technical n and indroduction of an improvement t of reliability and safety of the plant. especially operational procedures (also sk based on own experienc and the how of the personnel in the plant). em and review of the organisation, the he corresponding procedures or orders.		
Quality assurance	Installation of an own QA-program, taking into account personnel practices, working conditions and qualifications, used technology, encountered problems and any other boundary conditions.			
Maintenance	 Improvement of the preventive maintenance program. Personnel training before they carry out maintenance or repair on equipment in the controlled area. 			
Technical specification for operation	Creation of a specific document that would clearly define the operational limits and limiting conditions for operation for each unit and that would be used as an operational document in the control room.			
Incident reporting systems	The punishment system should be reviewed, otherwise no efficient improvement of the incident/defect reporting system can be expected. A trend analysis system cannot be stared up without this improvement.			
Emergency planning	 Installation of a post accidental sampling system. Review the duties of the shift supervisor regarding the calculation of the impact on the environment of an accidental release, so that he can use most of his time for helping to fight the accident and mitigate its consequences. 			
Training	 The small simulator for WWER-4 centre, should be used more intens The acess of full scope simulators is supported. 	440/213 that is available at the training sively and be improved where necessary. for WWER-440 and WWER-1000 reactors		

Table A2-3				
Pressurized Components	WWER-440	WWER-1000		
Reactor coolant system and connected systems under pressure	 Prevention of low temperature Leakage detection system and 	re pressurization. nd early detection of cracks.		
Reactor pressure vessel	 Reduce the EOL fluence in order to reduce the material embrittlement. Long-term preservation of the present safety reserves (low leakage loading, shielding elements). 			
		High nickel content in the weld.		
Pumps	 Regular inspection of examinations. Locations to be inspected s stress analysis results. 	pump flywheel by ultrasonic hould be selected on the basis of		
Steam generators	Regular inspection of V-tube by eddy current technique.			
Primary side safety valves	Qualification testing for discharging	ng of water or 2-phase mixture.		
SG valves				
BRU-A				

Table A2-4				
Operating Experience Feedback	WWER-440	WWER-1000		
Events related to the perfor- mance or design of systems	The leak detection system should be improved. It should be checked whether specific criteria exist for reporting of a RCS (Reactor Cooling System) leakage and RCS pipe cracks (as used in western reporting criteria).			
Main steam and feedwater system	 The operating procedures should be revised regarding isolation of a stuck-open BRU-A. After reconstruction of main steam system (one BRU-A in each steam line), an isolation valve should be provided upstream of each BRU-A. 			
Emergency power supply	The consequences of long term rator) at partial load during an a	operation of DG (Diesel Gene- ccident should be studied.		

		Table A2-5
Electrical Supply	WWER-440	WWER-1000
Station/Grid electrical connections	Determination of optimum conditions, to maintain a stable grid system.	
Diesel generators		 Reduction of the excessive cable lengths (high risk of damage by external causes and common cause failures auch as bus bar failures in the main building 6 kV bus bar chamber/OCB panel). Protection of the cable from all postulated accident scenarios including seismic disturbances and external event failures in the air tank rooms. Improvement of the lubrication system of the engine during the start-up phase.
Electrical power require- ments	The minimum of electrical power requirements of identified safety equipment needed for each initiating fault within the design basis should be established and these results should be presented in a safety schedule, from which the generator size, the diesel engine size and the loading schedule will be identified.	
DC installation	 Isolation of battery terminals. Installation of hydrogen detection and monitoring, for providing an automatic control of the ventilation equipment. 	
The cable installation	Upgrading of th	e cable installation (fire protection, segregation).
		Replacement of the containment wall penetration (moisture ingression).
6 kV switchbar	Replacement of circuit breakers,	
0.4 kV switchbar		
Equipment qualification	A QA program should be instituted in line with the operating experi-	
Electrical protection	Improvement of the existing electrical installation reliability.	

Table A2-6				
Instrumentation and Control	WWER-440 WWER-1000			
Neutron flux instrumen- tation	Istallation of modern, rate limited, variable trips on the source and operating range echelons.			
Safety protection functions	 Labeling or otherwise distinction of their sensors and cables). Improvement of the security of System), key control etc Improvement of the segregation be Removal of unnecessary equip vincinity of the SPS and its associated 	of these components (including the SPS (Safety Protection tween control and protection. ment, packing etc. from the ted equipment.		
	 Provision of fire protection and such isolation as it is possible for the components of the SPS. Provision of protection of the open terminals in the CR equipment room. 			
Control functions	Clear identification of the compone distinction of these components. Provision of fire protection and such is	nts and labeling or otherwise olation as it is possible for them.		
	Provision of fire protection and pro- tection of the open terminals in the CR equipment room.			
Power limiter and prelimi- nary protection system		Provision of a MCR (Main Control Room) display which indicates the allowed and actual power.		
Main and emergency con- trol rooms	Setting-up of up an accident decision room, separate from the RCR (Reserve Control Room).			
System cables, containment penetrations and equipment segregation	Improvement of the fire protection, suppression and fire door discipline and the cable overload protection.			
Testing and calibration	Setting-up of a program of regular EMC (Electromagnetic Compability) testing.			

Table A2-7			
Containment	WWER-440	WWER-1000	
Description of the de- sign,structural ana- lysis and status		Evaluation of the influence of prestress load variations on the tendons once the instruments are installed	
Penetration including design, execution, be- haviour		Study of the possibility of the loss of containment isolation, because the ECCS suction pipelines have only a single isolation valve.	
Leaktightness test	 Continuation of the liner weld repair program. Improvement of the seal of the personnel air locks to the confinement and the adjacent chambers by removing elements which could affect leak-tightness. 		
Operational experience		 Continuation of the process for re-instrumenting the pre- stressed tendons. 	
		 Development of a reliable and precise mathematical model to verify the effect of any variation on the con- dition of the tendons. 	
		 Development of a protec- tion against tendon corro- sion. 	

Table A2-8				
Internal Events	WWER-440	WWER-1000		
Main sources of combustible materials in safety areas	 The "plastikat" floor lining should be removed from all areas of the plant and replaced by non-combustible material. Turbine generator building roofing material should be removed and replaced by non-combustible material. The planned replacement of the main reactor coolant pumps with pumps using an integral lubrication system should be carried out. Cables with fire retardant insulation should be used where the store t			
	 existing cables need replace In order to reduce fire ha should be provided (e.g. dra leaks and spillages of oil from Hydrogen detection systems systems should be fitted in t 	ment or new cables are added. zard in the turbine hall, facilities ained trays or enclosures) to retain m the turbine and control systems. and temperature sensing he safety battery rooms.		
Passive fire protection	 Repair of the fire doors in compartments and fitting of a linstallation of remote indica left open. All fire doors between safet uprated to the 1.5 hr resistant 	the emergency diesel generator self-closing mechanisms. ators signalling that fire doors are by and non-safety areas should be nce.		
Fire suppression system includ- ing qualification and testing	 Improvement of the fire protection of turbine/generator because there is a significant safety hazard due to the presence of four turbine generators in one building using large quantities of oil and hydrogen. Installation of fixed suppression systems in the building of diesel generators. Installation of an environmentally acceptable fire suppression systems to protect electrical and electronic equipment taking into account characteristics and hazards in specific plant areas 			
Turbine hall	Removal of the emergency feed due to seismic, flooding hazard	dwater pumps from the turbine hall, and fire considerations.		

Table A2-9				
Site Conditions and External Events	WWER-440	WWER-1000		
Hydrogeology	 Improvement of the coordination between different organizations involved in the groundwater and soil monitoring and karstic problems. Continuation of the groundwater monitoring and control of the evolutions of this parameter. 			
Ground and utilization	Continuation of the monitoring of building settlements with a high precision leveller and theodolite for measurements of heights and tilts.			
Reference of the radiological situation	The two water wells should be included in the control program of the Radiological Monitoring Laboratory. The quality of the water for human consumption should be verified.			
Earthquakes	 It is recommended that the plant defines the earthquake level at the site S1 and S2 by the following parameters: Horizontal and vertical ground acceleration (zero period). Ground response spectra. Acceleration time-histories. 			

Table A2-10				
Accident Analysis	WWER-440	WWER-1000		
Applied computer codes	 Improvement of the documentation with graphinodalization for each analysis, information about developemnt and code verification in each case. The analysis should be performed until reaching a an alternative there should be an argumentati further sequence (including operator actions) of the calculation until reaching the safe shutdown conditional transible values of dose rates recieved including the delimitation of the considered excludible given. 	cal representation of the t the major steps of code safe shutdown state or as on describing clearly the e accident from the end of tions. d by external irradiation, ded zone boundary, should		
List of events, classification and acceptance criteria	 Definintion of the classes of events (operating caraccount in the safety analysis. A list of beyond design basis accidents must be or probabilistic approach taking into account the relia Study of the initiating events which can happe (primary breaks during shutdown, loss of residumnass inventory in the primary circuit, dilution by primary breaks during shutdown, loss of residumnass inventory in the primary circuit, dilution by primary breaks during shutdown. 	drawn up on the basis of a bility of the components. In during shutdown states ual heat removal with low ure water slug,).		
Decrease in reactor coolant inventory	 Study of the long term phase until reaching the final safe shut d conditions (including necessary operator action). It should be demonstrated that the LBLOCA with the double-ended guillo break with full flow area on both sides is really the worst case with regar flow stagnation in the core. 			
	The LBLOCA in the hot leg should be considered with regard to the load on the bubble condensor suppression facility.			
Leaks from primary coolant system to the secondary system	A complete analysis should be performed for the accident "Leakage via the steam generator collector cover", taking into account conservative assumptions with respect to radioactive release and operator actions according to emergency procedures. Subsequently an analysis of the radiological consequences should be performed.			

Table A2-11				
System Analysis	WWER-440	WWER-1000		
Scram signals	Installation of a low primary pressure signal.			
Emergency core cooling system	 Monitoring of the internal leakting the boron concentration. Avoidance of large release outs the ECCS when it is used to coordinate the section of the se	ghtness of the heat exchangers and also side the containment in case of rupture of I down the primary circuit.		
Make-up system	Automatic back-up by diesel generat	tor.		
		 Development of the boron concentration measurement device. Connection of the suction of the pumps to the boron tank. 		
Allowance for a primary break and the primary pump seals	Review of the hermetic compart- ment isolation logic to allow for primary pump packing integrity problems.			
Isolation of the containment	 The non-isolable sections of the containment sump suction lines including the nozzles have to be equipped systematically with a double envelope. The monitoring of the leaktightness of the main pipes has to be foreseen. The risk of flooding of the three separated rooms housing the safeguard systems via the drain connection. Indication of the means available for localising and isolating a leak in the facilities outside the containment. 	The improvement (double envelope on the containment sump suction lines) implemented on the 440 MWe plants has to be realized on the 1000 MWe plant. Moreover, means to control the tightness of this part of the circuit have to be installed.		
	Periodical control of all containment	isolation equipment.		
Emergency feedwater	Installation outside of the turbine hal			
Primary transmitter lines	Specific procedures have to be elat the passive means used to isolate a	borated to perform periodical testings on rupture of a measurement channel.		

Appendix A3

Plant and Operational Documentation (Long Version by G. Farber)

Plant safety cannot be increased by investments in machinery, processes and instrumentation and control alone. With comparatively little effort, improvements which support the operating personnel in plant operation as well as in the various technical and organisational measures are also possible. In this connection, two projects are to be mentioned in the first place which were carried out successfully with the financial support of the BMU and in cooperation with the German authorities:

- The simulator training for the responsible shift personnel of 440 MWe units at the nuclear power plant simulator at Greifswald. As the Rovno Nuclear Power Plant does not have its own plant- or type-specific simulator, the opportunity to practice strategies and measures for accident control was offered here. To design the three-week stays at the simulator effectively, plant-related training material for the performance of exercises as well as for theoretical training were developed. Western safety philosophy and safety standards can also be taught with these documents in a practice-oriented way.
- The pilot project for plant and operational documentation using the 1000 MWe plant Rovno, Unit 3 as an example. As the importance of this project extends beyond this plant (and possibly also beyond the country) and because of its proven success, it will be dealt with in more detail.

Project Targets

The overriding objective is to assist the plant operator as well as the supervisory and control authority in the preparation, completion and verification of the plant and operational documentation for a reference plant. Sample documents for central components of the plant and operational documentation are prepared in an exemplary way, on the basis of concepts mutually agreed upon by the operator, the authority and GRS. These functions require:

- the preparation of inventory records,
- the improvement of system diagrams,

- the creation of system specifications,
- the elaboration of an accident decision tree and an accident guiding scheme
- the development of procedures (detailed process descriptions and operational instructions) for the operational modes of normal and anomalous operation, as well as for accidents.

Taking the ideas of the Ukrainian partners and the GRS findings on documentation concepts for German nuclear power plants into account, documents are being or have been developed, respectively, which largely meet Western standards with regard to content, scope and depth. Because of the pilot nature of the project, the authority was incorporated into all work from the initial concept. With the help of the sample documents, the nuclear power plant staff can improve and complete the plant and operational documentation independently.

Project Implementation

GRS presented the project targets and the proposals for co-operation to the Ukrainian authorities, the Scientific-Technical Centre, all Ukrainian nuclear power plant operators and their umbrella organisations and discussed these objectives with them. The Ukrainian authorities acknowledged that the support could contribute to a substantial improvement of the operation.

Following a detailed project planning phase and intensive concept co-ordination with the authority in Kiev and at the Rovno Nuclear Power Plant, a computer-aided documentation and communication technology was provided to and installed at the Rovno Nuclear Power Plant as well as at the Scientific-Technical Centre. The documentation technology for the Rovno Nuclear Power Plant, which has meanwhile been extended by one CAD workstation, can be seen in Fig. 7.

Kraftwerks- und Anlagenbau (K.A.B.) AG, commissioned by GRS, carried out several basic and advanced training courses in CAD technology. Staff from other nuclear power plants (of the Ukraine and Russia) have participated in these training measures in the meantime.

The existing plant and operational documentation was examined, drafts of sample documents for inventory records, system diagrams and system specifications were presented, discussed and, if necessary, modified and finally adopted in regular one-week meetings between experts of the Rovno Nuclear Power Plant, the Scientific-Technical Centre, GRS and K.A.B.

Visits to German nuclear power plants and participation of the operating personnel of the Rovno Nuclear Power Plant in a simulator course for the shift team of a Konvoi plant were arranged, especially to teach the practical use of the operation manual and the emergency manual as well as the event- and protection-oriented courses of procedures.

Work groups have been and will be established to perform complex tasks (accident decision tree and guiding scheme, operation manual procedures) in which all participants work together at the respective most expedient place for up to four weeks.

Parallel to the preparation of the documentation, the drafts of national guidelines for plant and operational documentation are being prepared by the project participants.

Results and Benefits

For all forms of documentation dealt with useful results have been achieved in the first two years of the planned overall project time of three years.

Inventory Records

These contain the components existing in the systems of a nuclear power plant including the features, data, etc. characteristic of them. With this information it can be seen at any time which components are present in a system, by which operational parameters they are affected, which basic settings they must have, which equipment belongs to them and which design requirements, examination, maintenance and exchange intervals have been established for them. The information, stored in a data processing system, therefore supports fault diagnosis, process planning, safety isolation, the planning of in-service testing and the statistical analysis of operating experience. In accordance with the final determination of the form and content of the inventory records for the Rovno Nuclear Power Plant, the documents were elaborated using the example of the emergency boron system TQ and the core flooding system YT, following the German pattern implemented by GRS. From this basis, the operating personnel continues the work independently.

System Diagrams

The acquisition and improvement/actualization of the old drawings/system diagrams, like the efficient preparation of those diagrams still missing, are meanwhile being processed with the CAD technology installed locally. The C.A.D. diagrams prepared by the K.A.B. for the Stendal plant were provided, recommendations relating to the design of diagrams were given and sample diagrams were drawn for the TQ and YT systems. A substantially improved standard of drawings was achieved because of a far-reaching agreement on all problem areas, like, for example, on formatting, system delimitation, unique plant characterisation, the use of graphical symbols, room numbers, small and boundary limits, the indication of design limits, the identification of local displays, remote displays and basic valve settings or standby settings. Convinced of the higher quality level, the operator therefore decided to prepare all drawings of unit 4, which is in the final erection stage, to this improved quality immediately. The reactor circuit and all safety systems have already been documented accordingly. Within the framework of the independent preparation of system diagrams for unit 3, local examinations by inspecting the plant have already been carried out. Various faults were detected which could be removed.

System Specifications

System specifications do not only support orderly system and plant operation. They further represent important training documents for plant and operating personnel. At the beginning of the project they only existed in an abbreviated and incomplete form. For this reason concepts and organisational drafts were presented, the system specifications of the nuclear power plants in the old and new states of the Federal Republic of Germany serving as the base material. Together with representatives of the Ukrainian authorities and the operator these proposals were discussed and confirmed for further processing. In principle, a subdivision according to the subordinate aspects "systems structure" and "operational modes" is intended. All essential aspects, like design criteria, technological systems structure, systems arrangement, main components, operational monitoring facilities, operational transients, signal formation, controls, locks, controllers and operational parameters, for example, are being dealt with. A specification for the systems TQ and YT of unit 3 was completed as a sample document. The operator staff is going to continue the system documentation according to this sample.

Operational Manual/Procedures

Repeated, detailed descriptions and explanations of the concept and the structure of the operation manual using examples of German plants were necessary to explain the structures and procedures of the operation manual. The main emphases of the discussion here were the accident decision tree, the accident guiding scheme and the accident procedures. After a difficult reorientation process, the operator decided upon procedures following those of Konvoi plants. These are to be realised as soon as possible for 1000 MWe-units. In a work group consisting of experts from the Rovno Nuclear Power Plant, the Scientific-Technical Centre, GRS and K.A.B., the accident decision tree for unit 3 was completed in October 1994 after several weeks of intensive co-operation and work on the accident guiding scheme and the sample procedure for the small leak in the primary system has started. These work items are not only very important because of the new safety strategy for the reference plant, but also because of the problem recognition relating to design requirements and accident control dealt with by the expert team.

Requirements of the Authorities to Be Met by Documentation

Although the requirements to be met by plant and operational documentation are also dealt with in a different project, supported by the BMU in a co-operation between GANU/Scientific-Technical Centre and GRS (secondary documentation, safety documentation, licensing documentation), contributions to this also necessarily resulted from the preparation of sample documents during the performance of the project introduced here. The operator thus completed standard drafts for the symbols and abbreviations to be used. In the Scientific-Technical Centre the experience from the participation in the project is used to examine the operator's proposals and to prepare further practice-oriented standard drafts for the kinds of documents dealt with, with the objective of their recognition as national guidelines.

Outlook

The successful co-operation in the above fields is to be continued in 1995. The test manual or the test specifications are integrated into the work programme as a new kind of documentation. After a letter of intent referring to a "Co-operation between GRS, the Rovno Nuclear Power Plant (Ukraine) and the Balakovo Nuclear Power Plant (Russia) in the field of improving operational documentation" was signed, followed by bilateral and trilateral negotiations on the specification of the content, the co-operation can start from the beginning of 1995. It promotes the mutual exchange of experience, the harmonisation of documents and a faster establishment of documents by an exchange of data and a division of labour which is particularly relevant for plants of the same type. As a precondition, the Balakovo Nuclear Power Plant shall be provided with data processing technology which corresponds to the one at the Rovno Nuclear Power Plant.



Adaptation of GRS-Computation Codes to Soviet Reactors

S. Langenbuch, A. Petry, J. Steinborn, I. Stenbock, A. Suslov

1 Introduction

The studies and results presented have been performed within the framework of Scientific-Technical Centre projects of the BMFT. These are the two projects:

- RS941

Development and verification of computation codes for the analysis of accidents in WWER reactors;

- RS942

The further development of methods for safety-oriented analyses of RBMK reactors.

It is not the objective of this lecture to report on the individual points of the projects, but to provide a general account of the model adaptations performed within GRS computation codes. The main emphasis here is on the reactor physics models, as there are further descriptions on the use of ATHLET for transient analyses.

It is the object of the two projects of the Scientific-Technical Centre to make a contribution to the improvement of the safety of WWER and RBMK reactors by first providing suitable computation methods for accident analysis.

This is achieved by examining the applicability of existing GRS computation methods, like the thermohydraulic fluid dynamic system code ATHLET and the 3D-core model QUABOX/CUB-BOX-HYCA and their further development by plant-specific computation models.

A further important objective of the project of the Scientific Technical Centre is to promote cooperation between German and Russian institutions. For this reason, the studies are carried out in close co-operation with the Kurchatov Institute and OKB Gidropress for WWER and with NIKIET and the Kurchatov Institute for RBMK. For this purpose Russian employees were sent to Berlin and Garching for working stays lasting several weeks. The division of work between the Russian side and GRS is about 2 : 1.

2 Adaptation of ATHLET to WWER

Because of its modular structure the ATHLET system code can represent different reactor concepts. The computation model is used for pressurised water reactors and boiling water reactor plants and is, ins principle, also suitable for the representation of WWER plants.

However, the use for accident computations requires the examination of applicability with a reactor-specific verification matrix. For this purpose verifications of tests in experimental plants are carried out, like for

- PACTEL (OECD/CSNI standard problem ISP-33),
- the Russian integral test standard ISB,
- the Hungarian PMK-2 plant.

In addition, transients of WWER reactor plants are verified.

2.1 Supplementation of WWER-Specific Component Models

One of the peculiarities of WWER plants is the construction of the steam generators with heater tubes in a horizontal position between the headers on the hot and cold side. In the meantime a multi-channel description with the existing ATHLET components has been developed which permits a description of the circulation flow on the secondary side. Detailed modelling is required to simultaneously describe the states in the steam generator for pressure, water level and heat transfer in a sufficiently exact way. In addition, it is planned to supplement the correlations for vertical and horizontal flow in tube bundles and drift-flux-correlations for vertical two-phase flow in tube bundles.

In ATHLET, the models for the regulators in the WWER-440 were represented with GCSM. The setting of the parameters was verified by examining several operational transients.

2.2 Coupling of the 3D Core Model BIPR-8

The computation code BIPR-8 was developed at the Kurchatov Institute for WWER reactor cores with hexagonal fuel elements. To couple a 3D-neutron kinetics programme, one general interface was added to ATHLET. This interface can be used for different 3D models, e.g. for BIPR-8 as in this project. Corresponding calculations are, however, also being performed for the DYN3D computation code of the FZR Rossendorf, which also describes hexagonal fuel elements for WWER plants. An adaptation for coupling with ATHLET is also being developed for QUABOX/CUBBOX, the GRS 3D neutron kinetics programme for rectangular fuel elements in pressurised water reactors and boiling water reactors.

For transients with a close connection between neutron kinetics in the core and fluid thermodynamics in the cooling system there is a need for a coupled computation. This, for example, occurs for ATWS transients, demineralised water accidents or a break of a main steam line. A further advantage of such computation models also is that the direct 3D core computation is easier than the derivation of consistent data for point kinetics or 1D-kinetics. The coupled computation with the 3D core model requires a longer computation time but significantly less staff, with simultaneous elimination of modelling uncertainties, which again increases the reliability of the result.



Fig. 1 Rovno Nuclear Power Plant: instrumentation and control room of Unit 3 (Photo: H.-J. Burkhard)

The essential coupling properties are characterised in the following way:

The fluid thermodynamics in the core are described by ATHLET components. This permits uniform time integration methods in the core area and in the cooling system. The time integration of the neutron diffusion equation takes places in the 3D core model BIPR-8. The programming of the coupling makes possible a flexible division between fuel elements and cooling channels with fuel rods. There is a connection to GCSM to describe control rod movements. Complete restart capability can be used to carry out the computations. Power and pressure variation during an ATWS-loss of off-site power event illustrate this type of computation.

3 The Adaptation of ATHLET to RBMK

The ATHLET system code is also, in principle, suitable for the representation of RBMK plants. As already discussed in connection with the use for WWER plants, a plant-specific verification must, however, be performed. For this purpose the establishment of an RBMK-specific verification matrix is being worked on. So far it has only been possible to discuss parts so that an overall description is missing. As a first step towards verification, it is, however, possible to recalculate tests in RBMK-specific experimental plants, for example for the parallel channel test stand at Elektrogorsk or the integral test stands KSB and KS at the Kurchatov Institute. Available measurements of plant transients, like after a power increase or a pump leakage, can also be re-calculated. Accident analyses for accidents like:

- break of a group distribution header,
- break of a pressure collector,
- break in the main steam system or
- break of a pressure tube.

are the objective of the computations.

The modelling of the steam separator and of the control systems required special attention, but they could be performed with the existing ATHLET components. RBMK-specific models were added, like the model for radiation heat in the pressure tube after draining, for example. Here the radiation exchange between the central rod, fuel rods and wall of the pressure tube is described. In addition, a model for the heat transfer at the gas gap between pressure tube and graphite block was developed. The integration of a thermomechanical fuel rod model into ATHLET is also planned.

4 Adaptation of the 3D Core Model QUABOX/CUBBOX-HYCA to RBMK

The reactor physical behaviour of RBMK plants is of special importance for the assessment of the technical safety of these reactor plants. The essential backfitting measures for improving RBMK plants refer to the reactor core, like, for example, to backfitting a fast shutdown signal. The installation of additional absorbers or increase in the proportion of inserted control rods reduces the positive void reactivity effect. In the same way, increase in the initial enrichment from 2.0 % to 2.4 % makes a similar contribution. The effectiveness of such measures can only be examined by nuclear computations. For this reason the use of the 3D core model QUABOX/CUBBOX-HYCA developed at GRS was examined for the computation of the RBMK core arrangement. An additional code is necessary first to provide the activation cross section libraries for RBMK conditions. For this purpose, data were provided by the Kurchatov Institute. In addition, further new sets of data were produced by NIKIET. Both sets of data are based on computations with WIMS-D, the British code system which has been available in the Russian institutes for guite some time. Within the EU project on the safety assessment of RBMK plants further sets of data were computed with the newer version WIMS-E. These different activation cross section rates for RBMK are meanwhile available for QUABOX/CUB-BOX computations so that it is possible to compare the assessment of different sets of data.

The preparation of sets of input data for RBMK reactor cores with more than 1600 pressure pipes with fuel elements or absorber rods is costly. An essential simplification results from establishing a connection to the Russian database for operational measurements at special times with data on the core loading, the burn-up distribution, the control rod positions well as to the displays of core instrumentation. The models were further supplemented to compute the graphite temperature as well as to describe the complex control rod structure.

In the meantime, the following computations have been carried out to examine the 3D core model QUABOX/CUBBOX-HYCA for RBMK conditions:

- computation of the void reactivity effect for macrocell arrangement,

verification of void reactivity measurements in the cold state of Chernobyl-3,

- verification of void reactivity measurements in the operational state of Smolensk-3.

Parallel to the verifications, comparisons with the Russian computation models were carried out, like, for example, with the STEPAN model of the Kurchatov Institute which is available for RBMK or the SADCO model which is being developed by NIKIET.

The objective of this work is to compute typical accident sequences like the inadvertent movement of control rods, the failure of the cooling flow in several pressure tubes after break of a group distributor or a pressure decrease after the break of a pressure pipe. The sequence of these accidents is decisively influenced by the combined effect of neutron kinetic and fluid dynamic processes, with the current void reactivity coefficient. As the pressure characteristic and the coolant conditions are dependent on the entire system behaviour, coupled computations of the 3D core model with ATHLET will also be necessary.

5 Summary

The results which have been achieved so far can be summarised as follows:

For WWER the use of ATHLET for accident computations has been secured by numerous verifications. A further need for adaptation can, however, result from application computations, especially for WWER-1000 plants. The coupling of ATHLET with the 3D core model BIPR-8 available for WWER cores allows for clearly improved examinations of the influence of neutron kinetics.

For RBMK the present computations with ATHLET show that typical accident sequences in complex RBMK reactor plants can be calculated, but that the verification is to be completed further.

The results of the 3D core model QUABOX/CUBBOX-HYCA show a good match for verifying measurements. Essential influencing factors on the RBMK core behaviour could be examined with the computations which have been performed so far, in particular the dependence of the effective void reactivity on the number of control rods inserted.

The co-operation relevant to the adaptation of GRS computation codes has led to a clearly improved computation methodology for accident analyses.

It must also be emphasised that an intensive exchange of information on systems technology and on questions of model development has developed during the co-operation and that numerous employees of Russian institutes were introduced to the handling of GRS computation codes.



Accident Analyses of WWER Plants with ATHLET and DRASYS

R.L. Fuks, W. Richter, H. Wolff, O.M. Kowalewitsch

1 Introduction

Table 1-1 provides an overview of GRS programmes for the analysis of accidents and transients in reactor systems and the confinement/containment.

Programme	Field of Application	Reactor Type
ATHLET	thermohydraulics of the reactor cooling system for - transients - design basis accidents - beyond-design-basis accident without severe core damage	PWR, BWR, WWER, RBMK
ATHLET-CD	thermohydraulics, heat-up of the core and core destruction, fission product release, fission product transport for - beyond-design-basis accidents with severe core damage - core melting accidents	PWR, BWR, WWER, RBMK
QUABOX/ CUBBOX-HY CA	neutron kinetics and power distribution in the core for static nuclear design reactivity accidents transient accidents 	PWR, BWR, RBMK
DRASYS	thermohydraulics in the safety enclosure with pool-type pressure suppression - design basis accidents	BWR, [PWR], WWER (W- 2131), RBMK
RALOC	 thermohydraulics in the safety enclosure with hydrogen distribution, burn-up, recombination design basis accidents beyond-design-basis accidents with severe core damage (fire, chemical reactions) 	PWR, [BWR], WWER (W-230, W-320)

Table 1-1 GRS programmes for the reactor cooling system and the confinement/containment.

All programmes listed are, in one way or another, the subject of co-operation projects with East-European partners. These projects range from the bilateral scientific and technical co-operation for adapting and verifying the GRS programmes to Russian reactor types, to BMU support projects for the licensing and supervisory authorities of these countries which are being established, to the transfer of codes within the framework of multilateral EU projects.

First, selected results of applying the ATHLET and DRASYS codes to Russian pressurised water reactors of the WWER type are presented. Then, further results are presented which were achieved by experts of the Russian licensing and supervisory authority Gosatomnadzor during the application of ATHLET to unit 3 of the Balakovo Nuclear Power Plant, a WWER-1000/W-230 unit, within the framework of a bilateral support programme of the BMU in cooperation with GRS.

The objectives of the joint work on accident analysis, the state of the preparation of the ATH-LET set of input data for Blalakovo-3 and the work carried out for its verification, as well as the results of accident analyses for small leaks in the primary cycle, are reported in detail.

Finally, the activities on the use of DRASYS for computing parameters in the safety enclosure for WWER-reactors of the WWER-440/W-213 type with a depressurisation system are briefly discussed.

2 Objectives of the Joint Works on Accident Analysis

The start of direct co-operation with the scientific and technical centre of the Russian authorities in the use of ATHLET for WWER plants dates back to the year 1992. In this year two employees of the Scientific-Technical Centre of Gosatomnadzor of the Russian Federation (Scientific-Technical Centre GAN RF) during their stay in Germany were familiarised with the programme and its application for the first time. The programme itself was left to the authority for use.

Since 1993 this co-operation has been continued within the framework of the BMU support programme already mentioned on the basis of a three-year joint work programme for assessing technical safety problems for the WWER-1000 Balakovo-3 Nuclear Power Plant. The work on accident analysis with the GRS programmes ATHLET and RALOC here only represents one of the six main areas of co-operation. The main objective of the work on accident analysis is the expert support of the Russian authority in its efforts to create a capability for the **performance of accident analyses** in the licensing and supervisory procedure for Russian nuclear power plants which is **independent from industry**. Together with our Russian colleagues we foresee the following main directions for the use of ATHLET:

- expertise on accident analyses in the safety report,

- expertise, improvement and review of accident-operating instructions
- elaboration and proof of effectiveness for AM-procedures,
- support to the event sequence analyses during the performance of probabilistic safety analyses (PSA).

Within the framework of the familiarisation of the Russian experts with the use of the ATHLET code, an ATHLET set of input data is being prepared and verified for the Balakovo-3 Nuclear Power Plant and first accident analyses are being carried out in addition to practising the code handling. The results can be used for the realisation of safety improvement measures in the nuclear power plant.

For this reason we have chosen a form of co-operation which tries to consider all aspects mentioned. In each phase of the co-operation the realisation of certain enlargements of the set of data, the examination of the new set of data by verification of operational transients and the performance of accident computations have been and are being planned. In the second and third phase of the projects, a pilot analysis is carried out by experienced ATHLET users of GRS. Apart from the extension of ATHLET's area of application to new accident classes, the objective of this pilot analysis also is a detailed examination of the set of data worked out by the Russian colleagues.

3 Preparation of the Collection of Data for Balakovo-3 and the Preparation of an ATHLET Set of Input Data for WWER-1000/W-320

In the first phase of the co-operation, a collection of data was carried out on the basis of the project documents of the Balakovo-3 Nuclear Power Plant. In the next step, this collection of data was used for the derivation of a set of input data for the ATHLET programme. There was
a GRS set of data for the Stendal plant as an initial basis for this work. This two-loop set of data had been prepared for the analysis of a large break of the main coolant line, therefore the description of a large number of safety and operational systems and of the main control system of the power plant unit was missing.

The following new models were integrated into the ATHLET set of input data which had been prepared for Balakovo-3:

- Systems for influencing pressure in the primary system,
 - pressuriser safety valves,
 - · pressuriser injection,
 - pressuriser heater,
- systems for influencing pressure in the secondary system,
 - blow-off control valve (BRU-A),
 - steam bypass station (BRU-K),
 - steam generator safety valves,
- turbine control system,
- feedwater control system,
- quick-closing main steam isolating valve,
- reactor protection system,
 - reactor scram (HS 1),
 - fast power reduction,
- reactor power limiter (ROM),
- reactor power control system (ARM),
- emergency core cooling and residual heat removal system,
 - HP-emergency cooling pumps,
 - LP-emergency cooling pumps.

The ATHLET set of input data was designed as a two-loop nodalisation scheme (1+3).

4 Verification of the ATHLET Set of Data

After the completion of the preparation of the new set of data for the ATHLET code, a series of test and verification computations were carried out. The objective of these computations was to test the set of data in its respective stages of development and, in particular, to test the newly integrated models by re-calculating actual operational transients in WWER units. Starting out from these objectives, the following four transients were selected during the re-calculation of which practically all control systems newly integrated into the set of data could be tested:

- Examination of the self-regulation behaviour of the reactor plant upon insertion and withdrawal of the control rods,
- shutdown of one of four main coolant pumps in operation,
- closure of the quick-closing valves of the turbine,
- shutdown of one of two turbo-feedwater pumps.

In the case of a **shutdown of one of four main coolant pumps in operation** a power reduction to 63 % of the rated power is realised because of the interference of the reactor power limiter (ROM). After the completion of the power reduction the automatic reactor power control system (ARM) switches in, which stabilises the power at the new level (Fig. 1).

The turbine control system maintains the pressure in the main steam collector by decreasing the electrical power of the turbo-generator to 500 MW at the end of the power decrease of the reactor plant. During the transition process the feedwater control systems keep the levels in the steam generators within the permitted limits.

The computed parameter characteristics for the pressure and the level in the pressuriser and the temperatures in the hot and cold legs of the coolant loops (Fig. 2, 3 and 4) on the whole correspond satisfactorily to the experimental values. In the transient examined there is reverse flow of the coolant in the loop with the shutdown main coolant pump, which leads to a cooldown in the hot leg of the intact loops.

In general, the measured and computed data of all verifications corresponded satisfactorily. It has thus been demonstrated that it is possible to use this set of data for accident analyses for WWER-1000 plants.







Shutdown of one of four main coolant pumps: temperatures in the intact loops







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5 Results of ATHLET Accident Analyses for Small Leaks in the Primary System

In 1993 our Russian colleagues performed computations for some accidents which, being design basis accidents, are a part of the safety report and for which there are comparative results of Russian codes.

Fig. 5 shows selected results of the analysis of a **coolant leak DN 109** (approx. 93 cm2) in the cold leg of a coolant loop. The results of the ATHLET computation are compared with the results of the computations with the Russian programmes DINAMICA and UROWEN, which are contained in the safety report on unit 3 of the Balakovo Nuclear Power Plant.

The Russian computation in the safety report was carried out in two stages; until the loss of circulation with the DINAMICA code and then with the UROWEN code. The transition from the first to the second code in this approach is, of course, determined by the exactness of the modelling of the phase separation and the level movements in the DINAMICA code. The Russian colleagues here had grave doubts about the procedure described, as DYNAMICA only locally models the phase separation processes.

The use of the ATHLET code provided the opportunity of using a single model for the computation of the entire accident sequence and to carry out the analyses over a longer period without a number of simplifying assumptions. The comparison of the results shows that in both computations the safe cooling of the core is ensured and that a qualitative match between the results of the two analyses can be seen. But at the same time there are also some differences. The start-up of the LP-emergency cooling pumps in the ATHLET computation, for example, takes place much later than in the computations contained in the safety report.

The computation described represents an example of the application of independent expertise to a design computation in the safety report presented to the authority. In the present case the doubts in the computation presented could largely be removed.

This year the set of data prepared was used for a number of analyses of loss-of-coolant accidents with smaller leaks in the primary system. It was the objective of these projects to create knowledge as a basis for expertise on the existing accident operating instructions for leaks in the primary system of nuclear power plants, for further development and the derivation of new regulations.



6 The Influence of the Leak Size on the Accident Sequence

To determine the influence of the leak size on the character of the processes taking place, a loss of coolant out of the cold leg near the reactor inlet nozzle was examined under the assumption of additional failures assumed in the project:

- Opening of a leak with a simultaneous loss of off-site power,
- failure of one channel of the emergency core cooling system (accumulator, HP-pump, LP-pump).

Fig. 6 shows the pressure characteristics in the primary and secondary system for leak sizes having equivalent diameters of 80, 50 and 25 mm.

It is obvious that the pressure decrease is higher for greater leak cross-sections. While the pressure in the primary system drops until the accumulators respond and then until the LP-residual heat removal pumps start injection in the case of the leaks DN 80 and DN 50 (45 minutes after the start of the accident for DN 80, 108 minutes for DN 50), the injection of two HP-pumps is sufficient for DN 25 to compensate the losses from the leak, maintaining high pressure so that the accumulators are not actuated.

The loss of pressure of the DN 25 leak after 10,000 seconds is combined with the start of the largely emptied pressuriser being refilled and the associated condensation processes. After the pressuriser has been refilled, the initial pressure is recovered.

Between 6,200 and 7,500 seconds there is a similar, though less distinct phenomenon for the DN 50 leak.

In all three accidents there is a cooldown of the secondary system by the colder primary system (owing to the injection of cold water from the emergency core cooling and residual heat removal system) in the long-term phase. For the leak DN 80 the pressure in the primary system is much lower than in the secondary system, for the leak DN 25 it is the other way round, while both pressures are about the same for the leak DN 50. The relative values of the two pressures are determined by the relation between the throughput of the HP-emergency injection and the leak mass flow rate. The parameter characteristic for each actual leak thus depends on the number of HP-emergency cooling injection pumps in operation.

The different characters of the processes taking place for the leak sizes examined require a corresponding treatment in the accident operating instructions, which, at present, is not entirely the case.



and secondary system

Ξ

Influence of HP-Emergency Cooling Injection Pump and Emergency Feedwater Pump Failures

7

For each of the above mentioned leaks (DN 80, DN 50 and DN 25) variants with operation of three, two or one HP-emergency core cooling pumps and the failure of all three pumps were analysed. In addition, the processes with a failure of all emergency feedwater pumps were computed.

Furthermore, accident sequences with a complete failure of the emergency feedwater admission during the operation of the HP-emergency injection pumps were analysed for the leak sizes DN 80, DN 50, and DN 25. As the temperature of the secondary system is higher than in the primary system during the major part of the process time and the injection of emergency feedwater is only requested in the initial phase, the emergency feedwater failure here had no clearly detectable influence on the accident sequence. Table 7-1 shows the times of some characteristic events for the leak accidents examined. The analysis of these data shows that a heat-up of the fuel rod-cladding tubes to temperatures above the second project limit (1200°C) for the leak sizes analysed can only be expected in cases with a total failure of the HP-emergency cooling injection.

Leak Size	DN 80			DN 50			DN 25					
	3 HPIP	2 HPIP	1 HPIP	0 HPIP	3 HPIP	2 HPIP	1 HPIP	0 HPIP	3 HPIP	2 HPIP	1 HPIP	0 HPIP
Event		Time after start of accident, in seconds										
Period analysed	2000	3100	18000	5900	11000	14500	15000	5700	4500	31000	11500	9500
Start of Injection HP-emergency injection pumps	42	42	42		62	62	62	•	464	433	570	-
Start of Injection Ac- cumulator	790	1013	2250	975	1200	2625	13200	3400	•	5	ca. 1400	1
Accumulator empty	1693	3100	17500	5830	5100	•	-			-	•	-

 Table 7-1
 Times when characteristic events occur in the leak accidents analysed (with simultaneous loss of off-site power)

Leak Size	DN 80			DN 50			DN 25					
	3 HPIP	2 HPIP	1 HPIP	0 HPIP	3 HPIP	2 HPIP	1 HPIP	0 HPIP	3 HPIP	2 HPIP	1 HPIP	0 HPIP
Event		Time after start of accident, in se						t, in se	conds			
Start of Injection LP-residual heat re- moval pumps	1470	2747	17400	3000	4870	6500	•	•	•	•	•	
Pressuriser empty	95	91	85	80	600	546	524	150	900	966	830	816
Pressuriser refilled	•	•	•	•	5050	7500	•	•	•	14030	•	•
Start of fuel rod heat-up	•			900	•		•	2200	•	•	3	6600
2nd project limit (1200°C) is exceeded	•	•	•	?			•	4300	•	*	•	8800

8 Influence of the Leak Location

To assess the influence of the location of a leak on the sequence of an accident, the DN 50 leak was analysed under design conditions at the following leak positions:

- In the cold leg of the coolant system
 - · at the reactor inlet nozzle,
 - at the exit of the main coolant pump,
 - · at the lowest point of the loop (between main coolant pump and steam generator),
 - · at the exit of the steam generator,
- in the hot leg of the coolant system
 - · at the reactor outlet nozzle.

Table 8-1 shows the characteristic times for the cases analysed. As can be seen from the Table, these times only differ slightly for all cases examined. As expected, the pressure decrease of the leak at the reactor inlet nozzle is highest and lowest for the leak at the reactor outlet nozzle.

		Hot leg				
Leak position	13	10	4	1	1	
Event	Time after start of accident, se					
Start of injection HP-pumps	62	63	64	66	70	
Start of injection accumulator	2625	2366	2300	2430	2700	
Accumulator empty	-		6967	•	9600	
Start of injection LP-pumps	6500	6160	6700	5460	5370	
Pressuriser empty	546	589	617	595	824	
Pressuriser refilled	7500	6930	7070	6250	7490	

 Table 8-1
 Influence of the leak position on the accident sequence (leak DN 50 with simultaneous loss of off-site power, 2 HPIP)

9 Analysis of the Influence of Staff Actions on the Accident Sequence

To investigate and illustrate the potential of the staff to influence the course of accidents with smaller leaks, a DN 25 leak, with failure of all three HP-emergency core cooling pumps, is examined. As one possible staff action for accident control, it is possible to change the blow-off valves (BRU-A) manually from the preselected operational mode "pressure stabilisation" to the operational mode "automatic cooldown of the primary system at 60K/h". Fig. 7 and 8 show the pressure and cladding tube temperature characteristics for the case analysed. Two variants were analysed:

- without interference of the operator,
- with a change-over of the BRU-A to the operational mode "cooldown of the primary system" 2000 s after the start of the event.

The analysis of the results illustrated shows that the accident without operator interference in variant 1 leads to core melting, whereas by manual change-over of the BRU-A in variant 2, a pressure decrease can be achieved without core damage until the LP-residual heat removal is available.





10 Analyses without the Assumption of Simultaneous Loss of Off-Site Power

During the performance of accident analyses to verify the safety of nuclear power plants, the simultaneous loss of off-site power is assumed in Russian plants for the analysis of design basis accidents with a loss of coolant from the primary system. All analyses presented so far have been carried out under this boundary condition. As this assumption exerts a great influence on the further course of the accident (immediate reactor scram, turbine shutdown, failure of many operational systems, like, for example, the main coolant pumps, BRU-K, turbo injection pumps) it is also necessary to analyse the leak accidents without assuming a loss of off-site power.

Loss-of-coolant accidents with leaks (equivalent diameter 80, 50 and 25 mm) at the reactor inlet with the failure of one channel of the emergency core cooling and residual heat removal system, but without failure of the off-site power supply were analysed.

In these cases the actuation of the reactor protection, the shutdown of the main coolant pumps and the turbine shutdown do not occur simultaneously with the initial event due to power failure, but at a later time, when the respective technological limits are reached. The turbo injection pumps remain in operation and keep the steam generator level in the permissible range with the help of feedwater control systems.

A comparison of the accident sequence with and without loss of off-site power for the leak DN 80, for example, shows the different time sequence of the two events. Thus, for the variant with a loss of off-site power, 19 or 46 minutes, respectively, pass until the injection of the accumulator or the LP-emergency cooling pumps, respectively, and for the variant without loss of off-site power 15 or 30 minutes pass, respectively. It can be concluded that the transition to stable residual heat removal from the core takes place much earlier in the variant without loss of off-site power.

In addition thereto, a pressure increase in the steam generators up to the opening setpoints of the BRU-A or of the steam generator-safety valves is avoided, owing to the later closure of the quick-closing turbine valves when the station service power supply is available. Therefore there are no losses of inventory from the secondary system during the course of the accident, in accordance with the design.

It can thus be said that the assumption of a simultaneous loss of off-site power for the cases considered is conservative. Both alternatives are, however, to be analysed and considered for the derivation of accident operating instructions.

11 The Application of DRASYS to WWER-440/W-213

Finally, some information is provided on the status of the application of the DRASYS code to WWER reactors. The main field of application of the DRASYS code is the computation of parameters in the safety enclosure of nuclear power plants during leakage accidents, with a detailed simulation of the physical processes in the depressurisation systems (like, for example, free-blowing, water throw-up, condensation oscillation, etc.). DRASYS, which was originally developed and validated for German boiling water reactors of the 1969 and 1972 types, suggests itself for the analysis of the newer WWER-440 reactors of the W-213 type. These power plant units have a confinement with a pool-type pressure suppression system which limits the pressure built-up in the safety enclosure by condensation of the evaporated coolant.

The suitability of the code in principle for WWER-440/W-213 was demonstrated in co-operation with expert colleagues of the Ukrainian and Russian licensing and supervisory authorities. Fig. 9 shows traces of pressure in different compartments of the confinement of the Rovno Nuclear Power Plant, Unit 2, as well as a schematic illustration of the set of data prepared for the computations for this power plant. Fig. 10 shows the dependency of the maximum pressure on the room temperature before the start of the accident, as a result of a parameter study. The comparison of the two results shows that the initial conditions of the Ukrainian analysis had not been selected conservatively.

One essential safety problem of W-213 reactors is the insufficient proof of the functional capability of the pool-type pressure suppression system which so far has only been examined in small-scale experimental plants. One test in such a small-scale experimental plant (KSNV -Zugres, Ukraine) was recalculated successfully with DRASYS. The lack of suitable experimental plants essentially limits the possibilities for verifying DRASYS.

This situation will change fundamentally only after the successful termination of the international efforts relating to a joint performance of experiments in newly constructed experimental plants.



Fig. 9 Example for the application of DRASYS to WWER-440/213 (Rovno Nuclear Power Plant, Unit 2)



Fig. 10 Influence of the initial temperature on the maximum pressure (DRASYS, Rovno-1/2 Nuclear Power Plant)

12 Conclusion

With the present results of the co-operation between GRS and the experts of Russian and Ukrainian authorities in the field of utilisation of the ATHLET and DRASYS codes, a test was derived, made of the possibilities of using the ATHLET code and the set of input data derived, for the analysis of accidents and transients in nuclear power plants with WWER-1000 reactors by East-European experts.

All the objectives developed at the beginning of my lecture have been demonstrated with the help of selected examples. The objective of the further co-operation with the Scientific-Technical Centre GAN RF in 1995 is to further improve the quality of the present set of data and to enlarge this set of data to a reference set of data, m which is available to all East-European ATHLET users for the performance of ATHLET analyses for WWER-1000. The projects on the analysis of small leaks without loss of off-site power are continuing and two further initiating events are being included into the scope of the analysis.

An important task for increasing the operational safety of WWER reactors is seen as the creation of the broadest possible basis for the performance of thermohydraulic accident analyses in the operator countries in co-ordination with the Russian and Ukrainian expert colleagues. The accident procedures for the design and beyond-design-basis range, which are currently being revised or newly developed, represent the main emphasis here.

RBMK Safety Issues

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1 Introduction

Nuclear power plants play an increasingly important role in the power supply of the countries of Central and Eastern Europe (Table 1-1). The governments of many of these states cannot and do not want to do without these plants in the short term. The RBMK share of the nuclear power production in Lithuania in 1993 was 100 %, in the Russian Federation 55 %, and in the Ukraine 16 %. The RBMKs are a Soviet type of graphite-moderated, light water cooled pressure tube reactors, which were developed largely independently of any international exchange of experience and for which there are no Western parallels.

Country	1991	1992	1993	rel. share of RBMK, 1993
Bulgaria	34	32,4	36,9	
Lithuania	60	80	87,2	100
Slovakian Republic	51,5	46	53,6	
Czech Republic	26,4	28	29,2	
Ukraine	27,1	29,1	32,9	16
Slovenia	6,3 ^{*)}	34,6	43,3	
Hungary	46,8	46,4	43,3	
Russia, i.e.	11,4	11,8	12,7	55
- Compound system Volga	13,6	17,9	16,4	
- Compound system Centre	21,3	22,7	23,9	
- Compound system North-West	47,4	43,9	47,8	

Table 1-1 Proportion of nuclear energy in the power generation of East-European states in %

*) former Yugoslavia

The exchange of information on the RBMK concept between East and West, even after 1986, started hesitantly. The information available in the West has only improved recently, after the most modern plants (Smolensk-3 and Ignalina-2) had been examined within the framework of bilateral projects, like the Barselina study conducted by Sweden as well as international projects of the European Union and the International Atomic Energy Agency (IAEA). But the state of knowledge of Western institutions is still not satisfactory, especially for the older RBMKs. It is to be borne in mind here that the RBMK reactors are particularly complex plants with respect to systems technology.

The results and assessments relating to safety issues of RBMK plants described were predominantly worked out within the framework of the "RBMK Safety Review Project" /1,2/, which for the largest part was financed by the TACIS programme of the European Union.

At first an account of the "RBMK Safety Review Project" including important overall results will be given below. After that a few specific problems of the plant design, like

- reactor shutdown,
- core cooling and
- confinement

will be examined more closely. Some recommendations for upgrading measures are given as examples. Owing to the restricted time available, aspects of reactor operation and management cannot be dealt with.

2 RBMK Safety Review Project

Organisations of five EU states (Great Britain, France, Germany, Italy and Spain), three further Western countries (Canada, Finland and Sweden) as well as Russia, Lithuania and the Ukraine participated in the first phase of the above mentioned safety assessment of RBMK reactors by an international consortium (Fig. 1 and 2). Phase 1 of the project was officially completed on March 31, 1994. On June 10 the results of the project were presented to the public by the Commission of the European Union in Brussels. It was the objective of the project to examine all essential aspects of the design and operation of RBMK reactors, to evaluate the effectiveness of all upgrading measures which had been installed so far, to identify safety deficits, as well as to derive recommendations for upgrading measures and improvements in operational management. This examination performed by the Western consortium was based on information and documents from the RBMK operator countries and also included results from bilateral programmes of co-operation. The units Smolensk-3 in Russia (start of commercial operation in 1990) and Ignalina-2 in Lithuania (1987) were the reference plants for the project.

Western Consortium

- Atomic Energy Authority (AEA) Great Britain
- Commissariat à L'Energie Atomique (CEA) France
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Germany
- Ente per le Nuove Tecnologie, l'Energia e l'Ambiente (ENEA) Italy
- Statens Kaernkraftinspektion (SKI), ES-Konsult (ES) Sweden
- Sateilyturvakeskus (STUK) Finland
- Atomic Energy of Kanada Ltd. (AECL) Canada
- Empresarios Agrupados (EA) Spain

Fig. 1 Western consortium

Eastern Partner Organisations

- Lithuania: Lithuanian Energy Institute
- Russia: Atomenergoprojekt, GAN, Kurchatov-Institut, Minatom, MOAEP, Rosenergoatom, RDIPE, Smolensk ETC, VNIIAES, VNIPIET
- Nuclear Power Plants with RBMK:
 - · ChAES (Chernobyl) Ukraine
 - IAES (Ignalina) Lithuania
 - · KAES (Kursk) Russia
 - · LAES (Leningrad) Russia
 - · SAES (Smolensk) Russia

Fig. 2 Eastern partner organisations

The examination took place in a total of nine work groups, each with four to eight experts on the Western as well as on the Eastern side (see Fig. 3), who covered all essential design and operation aspects. GRS was represented in five of these work groups. In these work groups, recommendations for measures improving safety were formulated on the basis of the deficits determined and priorities were set. This served as a basis for the derivation of general recommendations in the areas of management, systems engineering and further analyses.

Working Groups

- Systems Engineering and Accident Analysis
- Control and Protection Systems
- Reactor Core
- External Impacts
- Quality of Design and Components
- Operating Experience
- Human Factors
- Licensing and Supervision
- Probabilistic Safety Analysis

Fig. 3 Working groups



Fig. 4 Ignalina Nuclear Power Plant, Lithuania

3 General Results of the "RBMK Safety Review Project"

An assessment of the final reports of the individual work groups led to the following essential evaluations of the RBMK plant safety:

- The upgrading measures which have been carried out since 1986 mainly concentrate on the core design and the shutdown signal (Fig. 5). The design deficits which had decisively contributed to the Chernobyl catastrophe have thus partially or wholly been removed.
- The safety level of the first-generation RBMK plants represents a cause for particular concern. Further measures to improve safety are urgently required for these plants. In particular, the situation at the Chernobyl Nuclear Power Plant needs to be clarified immediately.
- Improvements within the management of the individual nuclear power plants as well as the entire nuclear industry of the states concerned represent a particularly efficient way of improving the safety level.
- On the basis of the deficits determined a large number (more than 300) recommendations for improving safety were formulated. Even in the most modern plant, Smolensk-3, the reference plant of this study, which has only been in operation since 1990, major safety deficits were detected.
- A steady implementation of the recommendations made by the consortium is considered necessary. The adverse economic situation in Russia and Lithuania does, however, endanger the implementation of the upgrading programmes in these countries. For this reason international financial support for these programmes is very important.
- Considerable differences between the individual generations of RBMK reactors as well as between plants belonging to one generation were determined. For this reason, plant-specific safety reviews will be necessary to correctly assess the respective safety level and the effectiveness of the backfitting measures.
- The basic conditions for the licensing authorities in Russia, Lithuania and the Ukraine also give causes for concern. Thus, apart from an atomic energy act and a licensing philosophy adapted to RBMK plants, the financial preconditions are also frequently missing. The introduction of licensing procedures intended by the authorities is seriously impaired by the lack of the financial and technical preconditions.



Fig. 5 Upgrading measures implemented in RBMK

4 Specific Results of the "RBMK Safety Review Project" Relating to the Design

A large proportion of the individual recommendations referring to the design of RBMK plants necessarily results from work group 1 "Systems Engineering and Accident Analyses".

On the basis of the documentation relating to the Smolensk-3 plant provided by the Moscow NIKIET Institute, this work group analysed in detail a total of 12 systems and about 50 accident sequences. In addition, independent computations were carried out for three accident sequences.

As a result of the examination it was determined that the RBMK concept, apart from its safety deficits, also has some advantages (Fig. 6).



Fig. 6 Positive properties of the RBMK-1000

The relatively low thermal load of the fuel elements during normal operation, for example, thus represents a valuable contribution to accident prevention. The procedure of on-load refuelling furthermore offers the possibility of replacing defective fuel elements within a short period. The comparatively large coolant inventory in the "primary system" as well as the high heat capacity of the graphite cause a considerable thermal inertia of the plant. Some technical particulars of the system offer starting points for developing plant-internal emergency protection measures. The potential for such measures is first to be explored by analyses.

The most important design deficits of RBMK plants are dealt with below (also see Table 4-1). It must be admitted here that on the Western side there is only a comprehensive knowledge of Smolensk-3, a plant of the third generation. Starting out from the deficiencies determined, a number of recommendations for improving systems engineering were formulated, which can contribute to accident prevention and diminishing possible radiological consequences of accidents. It is presumed here that upgrading recommendations are secured by extensive analyses prior to their implementation and that possible negative influences of other systems are to be excluded.

	RBMK-1000 1st Generation	RBMK-1000 2nd Generation	RBMK-1000 3rd Generation	RBMK-1500					
Reactor protection	(There practically is on operational system, insi individual components; lack of diverse para	ly information or ufficient separatio deficient local pr meters for initiat	n the 3rd generation on of redundancies, otection system ("da ing important protect): meshing with low reliability of angerous rods"), ctive actions					
Reactor scram	no diverse second shutdown system; high reactivity insertion (4-5 B) upon loss of water in rod cooling system								
	low subcriticality reserve								
Emergency cooling	ow capacity; partially no separate emergency core cooling system	partially in	3 x 50 % sufficient redundan	cy separation					
Residual heat removal	?	?	not designed as safety system	?					
Emergency feedwater	not designed as safety system	?	not designed as safety system	?					

Table 4-1 Assessment of the safety systems of RBMK plants

	RBMK-1000 1st Generation	RBMK-1000 2nd Generation	RBMK-1000 3rd Generation	RBMK-1500
Fuel elementre- actor core	larger s	afety reserves		core design not optimised (e.g. local effects)
Confinement	Primary system: 5000 c	0 pipes, 3000 flar ontraction; qualit	nged joints, 2000 va y assurance?	alves, graphite
	no containment; limited strength of the reactor building	partial pro depressurisa	essure confinemen ation system desigr nominal diameter	t system with ned for 2F-break 900
Protection against - internal impacts	insufficient fire protection	?	insufficient fire protection	partially insufficient
 external impacts 	?	?	eactor hall and crane are endangered by earthquakes	?

4.1 Shutdown Function

Shutdown Signal

After the Chernobyl accident the Western side does not consider the reactor scram system introduced in all RBMK to be a system independent of the original shutdown signal. Therefore an additional, diverse and independent shutdown signal should be developed for all RBMK and introduced in the plants. This additional system should be able to ensure short-term as well as long-term subcriticality and to control all cases with a partial or a complete failure of the existing shutdown system.

Local Control and Protection System

Smolensk-3 is equipped with a local automatic control and protection system which has nine radial zones (Fig. 7). Other RBMK are operated with a similar system which is composed of 7 or 12 zones. The disadvantage of the 9-zone system is that there are 40 so-called "potentially dangerous" control rods in a certain annular core area (Fig. 8). The power increase of adjacent fuel elements occurring with the inadvertent withdrawal of one of these rods is not detected by the local protection system. Under conservative boundary conditions this power in-



crease can be more than 100 %. Consequently, it is recommended to equip all RBMK with the more effective 12-zone system for local power control and limitation.







Shutdown Signals

To prevent the simultaneous failure of several pressure tubes and thus a failure of the reactor vessel, an additional automatic shutdown signal should be introduced which is to be derived from a reduction of the flow in several connecting lines assigned to one common group collector. This requires an increase in the reliability of the flow measurement at each pressure tube, combined with an automatic self-monitoring of the system.

Furthermore, another diverse shutdown signal should be introduced for the case of a pressure tube failure in the reactor vessel.

Control Rod Cooling System

An essential disadvantage of the cooling system for the control rods, the fission chambers and the reflector is the fact that this system is partially designed as one train. A complete loss of coolant in this system leads to a positive reactivity admission of 4 - 5 β_{eff} assuming an additional failure of the shutdown system. As some incidents with losses of coolant out of this system or gas admission into this system are known from operating experience, it is considered necessary to reduce this positive reactivity effect, for example by multiple trains or by using modified control rods. In addition, the recommended additional diverse shutdown system should be able keep the reactor in the cold and unpoisoned state, allowing for the potential positive reactivity contribution of the rod cooling system.



Fig. 9 Ignalina Nuclear Power Plant: main control room

4.2 Core Cooling

Emergency Feedwater Supply System

The emergency feedwater supply system at Smolensk-3, like in the other RBMK plants, is not classified as a safety system. A failure of the main feedwater supply does not directly lead to the actuation of a safety system. As the three trains of the emergency feedwater supply system are partially not separated spatially and as they are in the direct neighbourhood of the

components of the main feedwater system (Fig. 10), they are very sensitive to common cause failures. In addition, the connection of the emergency feedwater requires the operation of pumps connected in series for clean condensate. The design of this system thus does not meet the more recent Russian safety requirements. This is confirmed by the accident in unit 2 of the Chernobyl Nuclear Power Plant in 1991. Because of a fire a part of the turbine hall roof collapsed. The parts falling down caused a total failure of the feedwater supply and thus a serious impairment of the core cooling.

Consequently, it is recommended to install an additional, diverse emergency feedwater supply system which injects directly into the steam separators. The main components should be located outside the turbine hall. It is suggested to use high pressure pumps as well as the water reserves of the tanks for clean condensate. Such upgrading measures are presently being developed. In the short term it is urgently recommended to check the evidence for the actuation of the emergency core cooling system upon failure of the feedwater supply, as well as the possibilities of a better physical or constructional separation of the existing feedwater systems.



Fig. 10 Ignalina, feedwater pumps (7 pumps, 5 always in operation, no physical separation)

Emergency Core Cooling System

The design of the emergency core cooling system of the Smolensk-3 plant in general is assessed to be good as it has sufficient redundancies and an appropriate physical separation of the individual trains (Fig. 11). The subsystem for ensuring short-term core cooling consists of two trains with core flooding tanks and the main feedwater system. For long-term core cooling there is a second subsystem which distinguishes between half of the core affected by a leakage accident and the one not affected by this accident. There thus is an emergency cooling capacity of 3 x 50 % to control the maximum design basis accident, i.e. a 2F-break of the largest main coolant line with a nominal diameter of 900.



Fig. 11 Smolensk-3 Nuclear Power Plant: emergency core cooling system

The existing accident analyses cover the entire range of possible break positions in the "primary system" as well as in the main steam and feedwater system. It must be mentioned here that the Russian analyses are generally carried out on the basis of conservative initial and boundary conditions. Apart from a few exceptions appropriate measures for accident control have also been taken by the protection system. Nevertheless, there are design weaknesses in some areas of the emergency core cooling system as well as a backlog demand for accident analyses. This applies to the failure of the first signal for actuation of the emergency core cooling system for common cause failure, for example. Accordingly, it is recommended to repeat the accident analyses under the precondition that the first actuation signal fails. It should be demonstrated in particular that the safety criteria are fulfilled under this boundary condition for all relevant break positions in the downcomers as well as the feedwater and main steam lines.

Only one signal, the pressure increase in the confinement, is used for the actuation of emergency cooling in case of breaks of pipes or collectors within the containment including the maximum design basis accident. Especially for the control of large loss-of-coolant accidents where the tolerable time interval between break and injection of the emergency core cooling system is particularly small, it must be required that a diverse technique for measuring the same parameter is employed or that a further parameter is used additionally for activating emergency cooling, like, for example, the rate of the pressure decrease in the circulation system.

It was further found out that the physical separation of the emergency cooling trains of the Smolensk-3 plant is inconsistent (see Table 4-1). 3 pumps, which are intended for long-term cooling of the undamaged half of the core during a leakage accident, are located in one room and are connected with the clean condensate tanks via a common collector on the suction side. It therefore is to be ensured that no initiating event to be assumed can impair the proper service condition of more than one of these pumps.

Residual Heat Removal System

The residual heat removal system of RBMK plants does not fulfil the single failure criterion, in particular because of the lack of double isolation valves, missing redundancies for some components, as well as a deficient physical separation of redundant equipment.

In accordance with the generally acknowledged standards it must, however, be ensured that a reactor can be transferred to a cold, subcritical state using the safety systems only. Accordingly, the residual heat removal system has to be modified in such a way that its proper service condition is not impaired by single faults or the failure of auxiliary systems. Further analyses are required.

Service Water System

Although the Russian RBMK-specialists classified the service water system as a safety system, important deficiencies were identified. Thus, in the Smolensk-3 plant there is still an injection to operational systems which are not required for accident control and not designed according to the respective safety standards even under accident conditions. As the design
concept of the service water system does not provide for automatic isolation of these operational systems, a loss of integrity in these systems under accident conditions can impair the safety-relevant cooling function of the service water system substantially. In addition, different legs of certain safety systems, like the sprinkler system of the accident localisation system, are connected to one common distributor of the service water system. The single failure criterion is thus not met. For this reason, it was recommended to upgrade the service water system with respect to redundancies and the separation of redundancies, to increase the availability of this system.

4.3 Confinement

Starting out from the Russian analyses on the behaviour of the accident localisation system, which represents a partial confinement with a pool-type pressure suppression system (see Fig. 12), it can be said that there is sufficient difference between the maximum calculated pressure in the confinement and its design pressure in the case of the maximum design accident (2F-break nominal diameter 900). In comparison, the margin between the calculated maximum pressure and the design pressure in the hermetic rooms positioned below the reactor vessel is smaller for the break of a group collector.

One important weakness of the partial confinement is that it is not designed for the retention of radioactive releases in case of breaks of main steam lines, of the pipes between the pressure pipes and the steam separator drums, as well as the upper sections of the downcomer and the feedwater lines. The defense-in-depth concept thus was not consistently applied to these compartments, nor to the reactor hall (also see Table 4-1). For this reason, it is recommended to pipe the coolant released into these compartments during pipe breaks into the pool-type pressure suppressions system or alternatively via a filter system.



Fig. 12 Cross-section of an RBMK-1500 (Ignalina Nuclear Power Plant)

5 Summary

Although the work expenditure of the "RBMK Safety Review Project" conducted by the international consortium was 60 man years, not all safety aspects of RBMK reactors could be analysed sufficiently. Large parts of the study had to concentrate on the newest RBMK plants Smolensk-3 and Ignalina-2, in particular. The examination identified a large number of weaknesses which were expressed in about 300 individual recommendations (Fig. 13). Many of these weaknesses had also been detected by the developers and operators: 15 % of the recommendations have already been implemented by relevant improvement measures. A further 69 % of the measures recommended have already or are being integrated into action plans for improving safety. 16 % of the recommendations were new for the Eastern partners. Only 5 % of the recommendations made did not receive the approval of the RBMK developers.

The overall importance of the project is also to be seen in the fact that this project represents the first comprehensive examination of RBMK reactors performed jointly by Eastern and Western partners. Although there was no conclusive statement on whether these plants are acceptable or not, the technical basis for decisions of the Russian authority in the licensing and supervisory process could be clearly strengthened. The fact that the members of the individual work groups from East and West have developed a mutual understanding and respect for the skills of their counterparts within the course of the project should not be underestimated. While the Western experts reached a better understanding of the RBMK design concept, the specialists from Russia, the Ukraine and Lithuania gained a deep insight into Western safety philosophy. Among other things this led to the fact that, apart from a few exceptions, an agreement on all deficits determined and individual recommendations could be achieved. This also applies to the basic recommendations in the areas of management, systems engineering and continuing analyses to the implementation of which a high priority was assigned.

Phase 1 of the "RBMK Safety Review Project" thus represents a sound basis for the continuation of the co-operation in the area of RBMK safety. Apart from the open questions remaining from phase 1, the RBMK of the first and second generation will above all be dealt with in the second phase of this project. The experience which has been gathered so far is also extremely valuable for the establishment and assessment of the safety report for the Ignalina plant according to the Western pattern. This work is currently being carried out within the framework of the Nuclear Safety Account administered by the European Bank for Reconstruction and Development (EBRD).



Fig. 13 Results of the "RBMK Safety Review Project"

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Radiological Situation and State of the Sarcophagus at Chernobyl

G. Pretzsch, A.A. Borovoy

1 Introduction

At present GRS, together with the respective institutions in Kiev, is conducting examinations for further protective measures in unit 4 of the Chernobyl Nuclear Power Plant within the framework of the BMU-sponsored support programme for the establishment of independent licensing and supervisory authorities and their scientific and technical organisations in the CIS countries.

These investigations started out from an assessment of the present structural and radiological situation, required for the planned long-term safe enclosure (Sarcophagus 2).

A part of the new findings made during the determination of the current status is reported below.



Fig. 1 Chernobyl Nuclear Power Plant: the Sarcophagus of Unit 4 during the construction phase



Fig. 2 Unit 4 of the Chernobyl Nuclear Power Plant after the construction of the Sarcophagus

2 Structural State of the Sarcophagus

The Sarcophagus was erected, partially without detailed planning, as an external cover around the damaged unit 4 of the Chernobyl Nuclear Power Plant within a few months, using parts of the destroyed reactor building and the turbine hall. A comprehensive characterisation of the safety of the site and the expected life time of the individual components as well as the complete structure are not available yet. Essential components as well as recognised weaknesses of the structural design will be dealt with briefly.

The construction of the Sarcophagus (see Fig. 2 and 3) can be outlined as follows:

A monolithic wall with a thickness of 2.3 m was erected in the turbine hall between units 3 and 4 up to a height of +19 m. In the de-aerator aisle the erection of monolithic partition walls of reinforced concrete with a thickness of 1 m followed. In the reactor building, a partition wall up to the height of +12 m was executed by grouting of the transport corridor with concrete. At other points the existing walls and partition walls were used after appropriate repair of openings and gaps, etc.



Fig. 3 Construction of the Sarcophagus (section along axis 47)

The northern stepped cascade protection wall is built of concrete. It also contains a large proportion of the radioactively contaminated debris thrown out of the interior of the reactor building during the explosion. From the outside a wall with buttresses (height 50 m) was erected in front of the largely preserved western wall.

The monolithic wall which had remained on the western side, the newly erected cascade wall on the northern side, the two remaining air shafts of reinforced concrete on the eastern side and the newly erected foundations on the debris of the de-aerator aisle on the southern side are used as the bearing foundation for the most important supporting structures.

The cover consists of steel girders which were inserted alongside the central reactor building, on which steel tubes with a diameter of 1220 mm and a length of 34.5 m rest. Above the tubes there is a roof construction of shaped roofing plates. A new cover was erected above the destroyed part of the turbine hall.

During the erection of these constructions, because of the existing radiation exposure, many installation processes could only be carried out by remote-control. The steel girders, mentioned above, which were used in the central area of the reactor building, could only be laid on, so that there are no bracings. Potential horizontal loads can only be absorbed via frictional forces. In the meantime a scientific research institute for structural design in Kiev has been commissioned by the Ukrainian authorities for a prognosis on the stability of the Sarcophagus.

Challenges to the stability of the Sarcophagus can be derived from the partially contradictory information on the constructional state. These are essentially connected with the following causes:

- Deficiencies during the installation (arrangement of essential supporting beams),
- Corrosion processes caused by humidity within the Sarcophagus,
- External impacts caused by extreme atmospheric influences (storms, snow loads and earthquakes),
- Intrusion of rain water (joints and gaps in the surface of the Sarcophagus).

In the last few years, technical measures to diminish these dangers have been carried out by the Ukrainian side. These measures include the installation of a groundwater holding system and a spray system for binding dust, as well as work to seal the openings in the outer skin of the Sarcophagus.

3 Aspects of Radiological Protection

3.1 Radiological Situation in the Surroundings of the Object "Shelter"

The radiological situation in the surrounding of the object "shelter" (Sarcophagus and destroyed unit 4) was determined with a variety of measurements. Fig. 4 shows a cartogram of the local dose rate at a height of 1 m above ground.

The measurement grid 8 m x 8 m in the southern zone behind the turbine hall of unit 4 is tightest. These values of about 0.2 mSv/h to 0.6 mSv/h remain about the same up to a distance of 100 m south of the turbine hall (position of the spur tracks for the working railway) and they only decrease at larger distances. South of the tracks the dose rate falls to about 0.2

to 0.3 mSv/h. In the west the measurement field extends from the western wall of the reactor building to about 120 m and north of the cascade wall of the reactor building up to about 30 m. The dose rates are here from 0.1 to 0.3 mSv/h. The measuring points in the north and west are scattered and much lower in number than in the south. The gamma dose rate is predominantly determined by long-lived nuclides like Cs 137.

In the entire area examined, there are several large areas up to 5 m x 10 m with strongly increased local dose rate of up to 50 mSv/h. At these points there are either retention tanks with highly radioactive fluids or stores of radioactive substances which were established immediately after the accident to contain highly contaminated structural components and fuelbearing materials thrown out of the reactor building.

The area around the reactor building and the turbine hall is very uneven. Because of earth banks or ditches, etc. there can be differences in height of up to several metres. So far the measured values have not been assigned to one definite uniform reference height.



Fig. 4 Cartogram of the gamma dose rate in mSv/h 1 m above ground in the surroundings of the Sarcophagus in September 1991

3.2 Type and Amount of the Radioactive Material inside the Object "Shelter"

Until the explosion of the reactor there were 1,659 fuel elements with a total of 190 tons of uranium in the active zone. About 3.8 % of this was released to the environment during the explosion, so that there are still about 183 tons of uranium in the plant. This predominantly fluid uranium flowed through openings to the compartments situated below the reactor cavity and solidified there to a fuel-containing mass, after having mixed with other substances. Under the influence of radiation, heat and humidity this mass gradually changes from a glass-like to a porous state. In the fuel-containing mass with an average density of 3.7 g/cm³ there is up to about 18 % by mass of UO₂ in the form of fine particles . Further components are, for example, Al₂O₃, Fe₂O₃ and Si₂O₃. The temperatures at the surfaces of the fuel-containing mass have generally reached room temperature.

The spatial distribution of the solidified fuel is basically known. Thus, there are 100 to 130 tons at the lower altitudes of 0 to 9 m. A large part, especially of the lower compartments was partially or completely filled with concrete after the accident so that an exact determination of the uranium mass is practically impossible. The remaining part of about 50 to 80 tons is assumed to be located in the destroyed reactor hall and in the northern cascade wall.

In the lower compartments of the object "shelter" there are about 1000 m³ of water (extinguishing water, rain water from outside). The fuel-containing mass is, however, in the compartments at higher elevations, so that it normally cannot get into contact with the water. On the basis of numerous examinations, the possibility of recriticality was assessed to be very unlikely.

The type and amount of the radioactive dust in the object "shelter" is particularly interesting, as it can leave the Sarcophagus in the form of airborne aerosols and can thus contribute to human radiation exposure. Under normal conditions the release of radioactive aerosols is low due to natural air circulation. The airborne radioactive release from the object "shelter", according to Ukrainian statements, of 1.1×10^{10} Bq/a (plutonium fraction 0.6 %) is clearly below the permitted values for a an RBMK-1000 unit in operation.

However, during accidents due to parts falling down, the dust can be whirled up and larger amounts of aerosols can be released into the environment.

The total mass of releasable, radioactive dust above the destroyed reactor cavity and the directly adjacent compartments was estimated to be about 1 ton with a specific activity of 4.3×10^9 Bq/g. The particle size distribution and the activity concentration of individual nuclides in the radioactive dust are known from analyses. Sr 90/Y 90 with about 47 % and Cs 137 with about 30 % of the overall activity represent the main fractions. The amount of unbound dust remains relatively constant from a temporal point of view, although the dust is bound periodically by spraying with a latex solution, via a spraying system installed in the object "shelter". Spraying does, however, not cover the whole area. In addition, the formation of new dust by the disintegration of the radioactive fuel-containing mass (radiation, effects of humidity and temperature variations) oppose the dust binding process.

3.3 Radiological Situation Inside the Object "Shelter"

In most compartments which had been accessible manually or using robots, the radiological situation is well known. It strongly depends on the degree of contamination with burnt fuel, radioactive dust, the degree of destruction and filling with concrete, sand, etc. after the accident and therefore differs to a large extent. According to the average gamma dose rate the compartments were subdivided into 5 classes which are illustrated in Table 3-1.

 Table 3-1
 Subdivision of the Compartments Inside the Object "Shelter" according to the

 Local Dose Rate LDR [mSv/h]

Class	LDR [mSv/h]	Percentage	Radiation Caused By
1	<1	15	radioactive dust
2	1 10	20	like 1 + radioactive water
3	10 100	40	like 2 + UO ₂ -containing concrete
4	100 300	10	like 3 + UO ₂ -containing mass + direct radiation from neighbouring compartments
5	; 300	15	like 4

Compared with the compartments of the 4th reactor unit, the contamination in the compartments of the deaerator floor is relatively low as most of the walls there had remained intact and thus hardly any radioactive material had penetrated. The situation is different for the turbine hall, the roof of which was destroyed. During the fire-extinguishing process, radioactive dust, fragments of the active zone and radioactive water had penetrated via the openings which had formed. The average dose rate in the turbine hall is 10 ... 20 mSv/h.

The dose rate on the roof above the destroyed reactor reaches values of up to 390 mSv/h.

Because of the previous groundwater analyses, the danger of an intrusion of radioactivity from the object "shelter" into the groundwater is assessed to be low.

3.4 Present State of the Object "Shelter" and Its Monitoring

The object "shelter", i.e. the damaged unit 4 surrounded by the Sarcophagus, is situated right next to unit 3 of the Chernobyl Nuclear Power Plant, still in operation. But only slight vibrations caused by the operation of unit 3 can be felt in the object "shelter", which do not exert an influence on its stability.

At present, monitoring systems for thermal control, for radiological protection control and for controlling neutron flux are used in the object "shelter". An extension of radiation and dose control is planned. As well as the stationary monitoring systems, controls are performed with the help of mobile systems and by taking samples.

It can generally be estimated that the state of the object "shelter" can be considered stable for a limited period. According to Ukrainian experts of GANU, planned upgrading measures are directed at pumping the radioactive water out of the inside, sealing further leaks in the roof and stabilising the supports of the two main roof beams on the ruins of the western wall.

4 Perspective of the Object "Shelter"

In 1992 the Ukraine organised an international contest on how to transfer unit 4 into an ecologically secure state. Within the framework of this contest walling-in with a second Sarcophagus was favoured.

Based on the proposals of the contest, a concept for a new encasement was elaborated. This concept contains the following points:

- stabilisation of the present Sarcophagus,
- design of a new protective structure (Sarcophagus 2),
- design and concept of a deposit for radioactive wastes close to the surface,
- clearing and conditioning of the waste from the present Sarcophagus and subsequent storage within the new Sarcophagus 2.

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In 1993 the European Union put out a feasibility study for tender referring to upgrading the existing Sarcophagus and building a new Sarcophagus. As the result of the invitation to tender, an international consortium directed by the French concern Bernard was granted the order. The feasibility study is to be presented in 1995 and to contain statements on the most favourable design of the Sarcophagus 2 as well as on the associated costs. The Ukrainian side already expects the costs to lie within a range of \$200 to \$400 million.

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