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WWER and RBMK reactors

Extrabudgetary Programme on WWER and RBMK Safety

In 1991, the Agency started an Extrabudgetary Programme to assist countries in Eastern Europe and the former Soviet Union in evaluating the safety of WWER and RBMK reactors. International consensus has been established on the major safety issues for these reactors, and their safety significance. The programme was completed in 1998, and a final report was published by the IAEA. An International Conference on the Strengthening of Nuclear Safety in Eastern Europe was held 14-18 June 1999 in Vienna, Austria.

Papers presented: Bulgaria - Czech Republic - Hungary - Lithuania - Russia

For further information please contact NS Coordination

Resources

WWER RBMK Database

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RANET

IAEA Safety Standards
IAEA Security Guidelines
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IAEA-EBP-WWER-15

FINAL REPORT OF THE PROGRAMME ON THE SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS

A PUBLICATION OF THE
EXTRABUDGETARY PROGRAMME ON THE
SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS

February 1999

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FOREWORD

This report presents an overview of the activities of the Extrabudgetary Programme (EBP) on the Safety of WWER and RBMK Nuclear Power Plants during the period 1990-1998. The Programme background, its objectives and its scope are described in Sections 1 and 2.

The report focuses on the wide scope of the activities aimed at identifying safety deficiencies, ranking their safety importance on the results of the safety missions, review of safety improvement programmes and on areas where future work is necessary. The information in the report reflects, to a large extent, the situation as it stood when the individual International Atomic Energy Agency (IAEA) tasks actually took place. Since then, however, work has been continuing in the framework of national, bilateral and international programmes. In order to bridge this gap, information recently brought to the attention of the IAEA has been included in the report.

This report deals with IAEA related activities. Therefore, references to other national, bilateral and multilateral programmes to improve WWER and RBMK safety are made only in specific cases where the results of such programmes have been used in the framework of the IAEA activities.

In this context, it should be mentioned that an effective mechanism of co-ordination between the IAEA and other programmes helped to minimize any undue duplication of work.

Individual sections describing results related to WWER and RBMK power plants are structured in a format which reflects the EBP tasks.

The report discusses selected safety issues and safety review results as they apply to each reactor type; generic issues which are common to more than one reactor type are addressed separately. This report addresses only those issues which had been selected for review by nuclear safety experts and which were subsequently discussed in dedicated IAEA technical meetings. Indeed, a primary target of the Programme was to facilitate the exchange of technical information among specialists, to consolidate the results obtained and to provide guidance on the work remaining to be done at the installations concerned. These results are discussed in Sections 3 to 8.

Section 9 provides an overview of the work which remains to be done and for which relevant IAEA assistance is planned.

Section 10 presents Conclusions to this report.

Finally, it should be noted that the results, recommendations and conclusions resulting from the IAEA's Programme are intended only to assist national decision makers who have the sole responsibilities for the regulation and safe operation of their nuclear power plants. The results, recommendations and conclusions do not replace a comprehensive safety assessment which needs to be performed in the framework of the national licensing process.

EDITORIAL NOTE

CONTENTS

SUMMARY

1. INTRODUCTION

In response to requests from Member States operating Soviet designed WWER model 230 NPPs for assistance through the IAEA's nuclear safety services, the IAEA launched a major international programme in 1990 to evaluate this first generation of Soviet designed reactors and provide safety assistance to plant operators and regulators. The Programme was undertaken as a complement to existing national, bilateral and international activities and was extended in 1992 to other types of WWERs and to RBMKs. The Programme concluded successfully in 1998. It was financed primarily by voluntary contributions from a number of IAEA Member States as an Extrabudgetary Programme (EBP), with some activities being funded through the IAEA's Regular Budget or by national and regional Technical Co-operation projects. Steering Committees provided independent advice on the conduct of the EBP.

The specific objectives of the Programme were as follows:

1. to identify safety shortcomings in design and operation of WWER and RBMK NPPs, either as deviations from IAEA standards and international practices or from operating experience, as safety issues and to evaluate the safety significance of these deficiencies with respect to their impact on the defence in depth of the plants;

2. to establish international consensus on priorities for safety improvements;

3. to provide assistance in the review of the completeness and adequacy of safety improvement programmes with respect to IAEA recommendations;

4. to undertake specific studies of unresolved topical safety issues.

Programme implementation was based on the IAEA practices and its well developed infrastructure for providing nuclear safety assistance. It also took full advantage of the results of safety evaluations completed and under way in the framework of many national, bilateral and international programmes.

The Programme included: identification and ranking of generic safety issues; review of the design concept, safety and seismic reviews; reviews of safety improvement/modernization programmes; studies of generic safety issues; preparation of guidance documents for safety analysis; code validation efforts; selected probabilistic reviews; review of operational practices, training workshops; establishment and maintenance of a database; and assistance to G-24 Nuclear Safety Assistance Co-ordination (NUSAC).

The greater part of the EBP centred on the development of the Safety Issue Books for the different types of WWER and RBMK NPPs. Safety Issue Books are lists of safety issues generic to all units of a plant type, ranking of their associated safety significance and the corresponding recommendations for safety improvements. The Safety Issue Books covered the WWER-440/230, WWER-440/213, WWER-1000/320, 'small series' WWER-1000 and the various generations of RBMK NPPs. The main sources for these Safety Issue Books were the Safety Review Missions carried out under the Programme and the evaluation of the results of other missions performed by the IAEA safety services. The complex process of producing the Safety Issue Books ultimately ensured the consensus of all donor and recipient Member States involved in the EBP.

2. RESULTS CONCERNING WWER-440/230 NPPs

Based on the results of a review of the design concept and safety review missions to Bohunice NPP Units 1-2, Kozloduy NPP Units 1-4, Novovoronezh NPP Units 3-4 and Kola NPP Units 1-2, some 100 issues of safety concern were identified and analysed by international experts. In 1992, consensus was reached on their safety significance and on the priority safety improvements necessary.

Since 1992, the IAEA has carried out periodic visits to these NPPs and to Medzamor NPP Unit 2 to provide independent advice on the safety improvement programmes and related actions. The safety status of individual plants as shown by these visits is reviewed against the generic safety issues.

In parallel to these activities, the EBP focused on selected issues of high safety significance, particularly the integrity of the primary circuit and confinement.

Primary circuit integrity

IAEA assistance addressed a wide range of technical issues related to material properties, pressurized thermal shock (PTS) analysis, leak before break (LBB) concept application and in-service inspection qualification.

Activities to address the issues are under way or completed at all plants. The results of the round-robin exercise on WWER-440/230 reactor pressure vessel (RPV) weld metal irradiation, embrittlement, annealing and re-embrittlement and the benchmark exercise on PTS analysis co-ordinated by the IAEA will serve as a tool for judging the reliability of results of plant assessments and to promote the use of state-of-the-art procedures.

The application of the LBB concept to large diameter primary piping of WWER-440/230 plants is a practical approach to restore some features of the original safety concept from the current point of view on maintaining primary circuit integrity. It was initiated as early as 1988 for Bohunice NPP Units 1-2. At all plants concerned, activities to address the issue are under way or have been completed. The application of the LBB concept, including all load conditions, has been completed at the Bohunice, Kozloduy, Novovoronezh and Kola NPPs, as recently reported to the IAEA. The in-service inspection (ISI) approaches have been reviewed and modified as necessary. At all of these units, acoustic emission leak detection systems have been installed.

Considering the multidisciplinary nature of the application of the LBB concept, a thorough review at individual plants would strengthen the confidence that the safety objectives are achieved. The IAEA has already provided peer review services to Bohunice NPP Units 1-2 and is prepared to assist other NPPs, if so requested. It has also developed guidelines on the application of the LBB concept.

Confinement

Concerning the confinement leaktightness improvements, the results achieved at Bohunice have demonstrated that significant improvements are possible in this area.

Concerning confinement improvement in the context of major upgradings, programmes exist for each plant; however, they all include new technological systems or equipment for which sufficient testing or experience is not available yet. The cost of such improvements and the need for additional testing leave some doubts as to the possibility of a timely implementation.

The major upgrading of the WWER-440/230 reactors with respect to extending the original design basis is still an ongoing process at all plants. This work needs to be completed with high priority. A proper balance of the necessary improvements needs to be achieved considering the upgradings necessary to cope with the new bounding LOCA DBA, as well as the design parameters of existing equipment and structures.

3. RESULTS CONCERNING WWER-440/213 NPPs

The identification and ranking of safety issues for these NPPs was completed in 1995. No category IV (highest safety significance) issues were identified, which confirms that WWER-440/213 safety had been improved in all areas of major concern, as compared with the earlier WWER-440/230 design.

The IAEA also reviewed the safety improvement measures planned or implemented at Dukovany NPP Units 1-4 and Paks NPP Units 1-4.

In response to EBP studies which showed a lack of demonstrated safety margins in the strength behaviour of the bubbler condenser structure, some countries undertook further studies on the issue and implemented structural modifications. An experimental thermohydraulic qualification test programme of the bubbler condenser was planned in the OECD/NEA group. A separate PHARE/TACIS project funded by the EC is under way to verify strength characteristics under accident loads using Paks NPP as the reference plant.

4. RESULTS CONCERNING THE WWER-1000 NPPs

The WWER-1000 NPPs were divided into two types of model 320 NPPs and the ‘small series’ NPPs. The identification and ranking of safety concerns for WWER 1000/320 NPPs was completed in 1995 and for ‘small series’ NPPs in 1998. As for the WWER 440/213 NPPs, no category IV issues were found. However, the operational experience revealed problems related to design solutions, quality of manufacturing and reliability of components.

The safety modernization programmes reviewed by the IAEA were in general found to be well developed and structured with respect to the design issues. Their implementation will make a major contribution to plant safety. However, the degree of detail of the individual measures addressed in the programmes and same descriptions need to be further elaborated. Considerable effort would still be needed to bring planned programmes to successful implementation.

In the past few years, events of delayed control rod insertion or even rod sticking have been observed at some WWER-1000 NPPs and PWRs worldwide. While measures have been implemented to ensure that safety design limits for control rod drop time are fulfilled the appearance of water gaps between fuel assemblies raises safety concerns which need to be addressed. Fuel safety criteria and methods currently applied to existing and new fuel designs under advanced performances are being reviewed for Western PWRs under a task force in the CSNI WG2. This topic as related to WWER fuel is included in the framework of the IAEA's Nuclear Safety Programme for the years 1999-2000. Progress has been made in understanding the root cause of the fuel assembly bowing, which led to the problem.

EBP activities related to WWER-1000/320 NPPs also addressed the material behaviour of the reactor pressure vessel, the related surveillance programmes and integrity assessment, and the problem of cracks in steam generator collector plates. Problems which surfaced with the steam generators while the plants were operating pointed up the critical importance of the timely implementation of preventive and corrective measures. These include: inspection, water chemistry, monitoring and control, leak detection, repair and replacement and the establishment of approaches to deal with primary to secondary leaks. In addition, the need to address “anticipated transients without scram” at all plants was demonstrated in an EBP review of this topic.

For the small series WWER-1000 units, the lack of physical separation and functional isolation of the most important components of the ECCS and its support systems was identified as the major concern. It may result in loss of control, i.e. loss of main safety functions in case of common cause failures. Compensatory measures have been taken to eliminate common cause failures: upgrading of the system (including upgrade of the support system); implementation of piping inspection; development, approval and implementation of the event based management procedures; staff training.

5. RESULTS CONCERNING RBMKs

Over the past decade, a considerable amount of work has been carried out by designers, operators and regulators to improve the safety of RBMK reactors and to eliminate the causes that led to the Chernobyl accident. As a result, major design and operational modifications have been implemented. However, safety concerns remain, particularly regarding the units of the first generation.

The consolidation of safety issues for the second and third generation RBMK NPPs was completed by the IAEA in 1995. The Smolensk NPP Unit 3 and Ignalina NPP have been used as reference plants. In 1997 similar work was completed for the first generation RBMKs using Leningrad NPP Unit 2 as the reference plant. Much use was made of the results of the other international activities, particularly those sponsored by the European Commission for RBMK NPPs.

As demonstrated during the Chernobyl NPP Unit 4 accident in 1986, the RBMK design had some serious deficiencies related to its reactivity control and shutdown system. Some of the more significant concerns were: the positive scram effect of the control rods when they were moved down from the topmost position, a large positive void coefficient and slow insertion of negative reactivity. Consensus was reached that the understanding of the causes of the Chernobyl accident is now sufficient to ensure the adequacy of measures taken to eliminate its reoccurrence.

Since the accident, a number of modifications have been made to all RBMKs to improve the ability to control the power and rapidly shut down the reactor. Additional changes and refinements are still being considered, the most significant being an additional shutdown system. There is now general agreement that a fully independent additional shutdown system is required for RBMK plants of all generations. The requirements to be fulfilled by the new system should be those of the IAEA safety standards.

Recently cracks were found in large 300 mm diameter austenitic piping in some RBMK units.

It was concluded that intergranular stress corrosion cracking of austenitic stainless steel piping is a generic safety issue for reactors operating with BWR type water chemistry. This type of degradation appears to be already well under control for vessel type BWRs, while it is a new safety issue for RBMK plants, which needs attention. The IAEA is reviewing, together with RBMK operators and regulators, the results of a workshop held in June 1998 to define further assistance.

Events leading to multiple pressure tube rupture have received considerable attention in the safety evaluation of RBMKs since they might develop into severe accidents. Studies by Russian specialists indicate that the probability of such events is very low. The partial break of a group distribution header was identified as a potential candidate based on modern Western computer codes RELAP5, ATHLET and CATHARE calculations. These analyses have shown that for a certain break size, flow stagnation accompanied by flow fluctuations might lead to excessive heat-up of the pressure tubes. Studies by Russian specialists show that a break of such size is larger that the “critical break”, and therefore will immediately grow further.

Out of the series of tests performed in Japan and made available to the IAEA, a group of international experts selected a test matrix addressing the phenomena of interest. Based on this material, in 1995 the IAEA launched an international code validation programme aimed at validating thermal-hydraulic computer codes for selected phenomena relevant to RBMK safety analysis.

The code validation exercise contributed to an improved understanding of the potential causes, nature and behaviour of flow oscillations in channel type reactors. Flow oscillations accompanying flow stagnation, as observed in previous analyses of the partial break of a GDH, might be real. Consequently, improved channel dry-out behaviour and heat transfer should be given credit, if adequately simulated by the computer codes in use.

6. RESULTS CONCERNING GENERIC TOPICS

The EBP has addressed several topics of general interest.

A review of WWER PSAs showed differences in results for similar NPPs, probably due to different assumptions, methods and scopes, which must be explained to allow for an adequate use of the insights shown by these PSAs. The ultimate objective should be to establish a validated “living” PSA which is used as a management tool in plant operation and decisions on plant modifications.

IAEA assistance on the development of PSAs and their peer review will continue.

Reviews and on-site studies of the seismic hazard at sites and seismic qualification of some plants have been conducted in the EBP, and can be continued if requested in future IAEA projects.

The requests for up-to-date guidance on some topics led to the preparation of several guideline reports on short notice. These reports may form the basis of future safety standard series documents.

Of supplementary benefit was the regular collecting of technical information into a database, which has become extensive and will continue to be maintained for the benefit of all countries involved.

All operating countries were encouraged to reconsider their safety analysis reports. Guidelines for In-depth Safety Assessment (ISA) of WWER and RBMK have been issued by Gosatomnadzor, Russia in September 1997. Actual work to prepare ISAs is under way in the framework of international projects as was done in 1996 for the Ignalina NPP in the context of the EBRD's Nuclear Safety Account grant agreement. It is of utmost importance that this work be continued, completed and ultimately peer reviewed for all NPPs.

Other countries operating WWER and RBMK NPPs have issued new requirements for safety analysis reports. The accident analysis part in the safety analysis report also needs to be upgraded commensurately with international practice.

It is expected that in the future the IAEA will be called on to play a major role in providing guidance on upgrading safety analysis reports for WWERs.

Other generic safety issues identified in the activities of the EBP which need to be addressed in the future include: accidents during low power and shutdown conditions, anticipated transients without scram, primary to secondary leaks, components qualification and fire protection.

IAEA's recent OSART and ASSET missions reported improvements in the area of operational safety. However, there is still room for improvement. The most important area where further work needs to be done are related to general management and safety culture, QA and document management, training and the development of emergency operating procedures. IAEA assistance will continue through its operational safety services. Focus is on promotion and review of self-assessment.

7. CONCLUSIONS

There is general agreement among countries providing and receiving assistance that the specific objectives of the EBP have been fully met. The main achievement of the Programme was the preparation the Safety Issue Books, which provide a clear picture of the design and operational safety issues and their safety importance for the plants under consideration. International consensus was reached on the safety issues related to WWER and RBMKs and on the priority measures required. As a result, Programme findings and related publications are being widely used as a technical basis for the development of safety upgradings for these plants, reviews by national regulatory authorities, and the establishment of safety priorities in national, bilateral and other international programmes.

The Programme also enabled a regular exchange of information among all countries involved and in-depth technical dialogue between Western and Eastern experts.

The organization of the EBP allowed great flexibility and the capacity to respond quickly to new requests and needs which emerged during the implementation of the Programme. In this context, the role of the Programme's Advisory Group and Steering Committees was instrumental to advise on priority actions and programme changes.

Despite the improvements in safety already achieved, much remains to be done at individual NPPs, particularly at the WWER and RBMK plants of the first generation. Safety improvement work for these plants is essential if they are not decommissioned in the near future.

Of utmost importance is to ensure that the operating organizations draw up a safety case for each NPP based on a plant specific safety analysis and that it is reviewed and approved by the national regulatory authorities. This will allow an assessment of the overall safety impact of plant modifications. This process - which has been initiated, but not finalized in several plants - needs to be completed as a matter of urgency.

Bilateral and multilateral assistance programmes will remain important to complement national efforts.

Upon completion of the EBP, the IAEA will continue to provide nuclear safety assistance to its Member States in the framework of both its regular Nuclear Safety Programme and its Technical Co-operation projects. A specific project on WWER and RBMK safety has already been included in the IAEA Nuclear Safety Programme for 1999-2000. Three ongoing regional Technical Co-operation projects are also being extended until the year 2000. An important element of this assistance is to strengthen the national regulatory authorities in the countries operating these NPPs on the basis of IAEA recommendations and good international regulatory practices.

An International Conference is being organized by the IAEA in co-operation with the EC and NEA in June 1999 to review the results of national, bilateral and other international programmes to enhance WWER and RBMK safety. Countries operating these NPPs are expected to provide a comprehensive overview of the actual achievements of all efforts to improve safety, and to identify those areas which require further work. The conference results should indicate future international assistance needed.

Safety upgrading projects have shown that substantial improvements in the safety of WWER and RBMK NPPs are feasible.

However, according to the Convention on Nuclear Safety, each Contracting Party "shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible. The timing of the shutdown may take into account the whole energy context and possible alternatives, as well as the social, environmental and economic impact".

1. INTRODUCTION

1.1. BACKGROUND

1.1.1. Early safety concerns

Nuclear power plants (NPPs) built to earlier standards, including those of the WWER-440/230 and RBMK design originating in the former Soviet Union (FSU), were designed in the 1960s and early 1970s. At that time, comprehensive international safety standards such as Nuclear Safety Standards of the IAEA (NUSS) were not available, and although Western safety practices similar to those currently applied were under development, the exchange of information between experts from the Organization for Economic Co-operation and Development (OECD) countries and countries of Eastern Europe and of the FSU was very limited. The objective of the reactors designed in the FSU was to produce electric power with high availability. Safety was oriented towards preventive features, while mitigation features were given lesser consideration. This design philosophy resulted, on the one hand, in inherent safety features for WWER-440/230 reactors such as low power density, large coolant inventories, resistance of the primary circuit material against cracking and a low impact of failures of individual equipment on the whole plant. On the other hand, it resulted in significant deviations from current safety standards, such as the insufficient capability of the emergency core cooling system (ECCS), the lack of a containment building, and insufficient separation of process systems from safety systems. Other design deficiencies, due to insufficient consideration of relevant hazards, included insufficient protection against internal and external hazards, the potential for common cause failures, and vulnerability of RPV to radiation embrittlement.

Starting in the seventies, safety upgrading tasks were initiated by the Russian design organizations. In July 1989, the regulators of the WWER-440/230 operating countries jointly drew up a position paper with 16 requirements to ensure a safety level which they considered acceptable during the remaining design lifetimes of the units and a requirement for a special operating regime until backfitting was completed. However, the paper recognized that the current accepted level of safety could not be fully reached.

The development of safety upgrading programmes was significantly accelerated after the political changes in the late 1980s, owing to strengthened international co-operation.

1.1.2. Genesis of the EBP

In response to requests from Member States operating Soviet-designed WWER model 230 NPPs for assistance through the IAEA’s nuclear safety services, a major international undertaking was launched by the IAEA in 1990 to evaluate this first generation of Soviet designed reactors and provide safety assistance to plant operators and regulators. This was intended as a complement to existing national, bilateral and international activities. The Programme was extended in 1992 to other types of WWERs and to RBMKs and concluded in 1998. It was financed by voluntary contributions from IAEA Member States as an Extrabudgetary Programme.

An Advisory Group (AG) composed of 42 participants from 19 Member States and international organizations met in September 1990 to consider requests and review existing information and activities. It agreed to the establishment of the Programme as an EBP because of the urgent need to set it in motion. A Programme was agreed to, which involved a compilation of design information, fact finding missions to each site, and studies on generic safety concerns, followed by recommendations for enhancing plant operational safety. Use was to be made of existing IAEA information and resources, particularly those of the Assessment of Safety Significant Events Teams (ASSET), Operational Safety Review Teams (OSART) and other safety services (e.g., seismic safety).

The AG also decided that a Steering Committee (SC) should be formed with representatives of the three countries operating model 230 WWERs, Member States providing technical and financial support, and a Secretariat from the IAEA. Representatives from international organizations were invited to attend the meetings of the Committee. Further Advisory Group Meetings (AGMs) were held in 1992, 1994, 1995, 1996 and 1998 for Member States to monitor progress and recommend future activities.

One of the key objectives of this Programme was to identify and clarify the outstanding safety concerns and to determine where additional assistance was needed. In this respect, the IAEA played a fundamental role in helping to identify the specific areas of assistance needed. In addition, the IAEA also served as a catalyst in facilitating information exchange between the various donor countries on the specific kinds of assistance each country was providing and where it was applied.

In this context, it is important to recognize that the IAEA was able to serve as a unique venue for the exchange of information required to achieve a broad international consensus, within a relatively short time frame, on technical aspects related to nuclear safety. This was an essential prerequisite for further actions, both within the framework of the EBP and of bilateral or multilateral assistance activities.

This EBP was unprecedented in that it enabled the regular exchange of information among all countries involved and an in-depth technical dialogue on the safety of nuclear power plants between Western and Eastern experts.

The results achieved by the Programme by 1992 and the requests from WWER and RBMK operating countries led the 1992 AGM to agree an extension of the EBP to review similarly the other types of WWERs and the RBMKs operating or under construction. The existing SC became responsible for all WWER activities and a second SC was formed for RBMK activities.

The financial and in-kind contributions to the EBP allowed it to complete its main tasks by 1996, at which time the decision was made for an orderly end of the EBP by 1998, with the understanding that outstanding activities would be either financed by the IAEA Technical Co-operation (TC) or incorporated in the 1999-2000 Nuclear Safety Programme under the IAEA's regular budget. In line with the recommendation of the AG, the IAEA General Conference in September 1996 approved the Programme and Budget of the IAEA for 1997-1998, which, under subprogramme H.1 (Nuclear Power Plant Safety Assessment), contains Project H.1.04 (WWER and RBMK Plant Safety), which is financed through regular budget, TC and extrabudgetary funds.

1.2. PROGRAMME OBJECTIVES

On the basis of the original requests from countries operating WWER plants and the recommendations of the 1990 AGM, the overall objective was defined as assistance to countries operating WWER model 230 NPPs in performing comprehensive safety reviews on these plants aimed at identifying design and operational safety weaknesses. The results of the review would be used in the technical basis for the national decisions to be taken to improve safety. Certain safety related subjects such as regulatory control, radiation protection, waste management and nuclear research were not explicitly included in the review.

The specific objectives of the Programme were as follows:

1. to identify safety shortcomings in the design and operation of WWER model 230 NPPs, either as deviations from current standards and international practices or from operating experience, as safety issues and to evaluate the safety significance with respect to their impact on defence in depth;

2. to establish international consensus on priorities for safety improvements;

3. to provide assistance in the review of the completeness and adequacy of safety improvement programmes with respect to IAEA recommendations;

4. to undertake specific studies of unresolved topical safety issues.

Information to meet the Programme objectives was compiled from activities of the IAEA, Member States and other international organizations, particularly the European Commission (EC), World Association of Nuclear Operators (WANO) and the OECD/NEA.

The same objectives applied to the later reviews of the other types of WWER and RBMK NPPs. Soon after the Programme began, co-ordination was established between the activities of the EBP and those of other international assistance programmes. The EBP thus evolved into a forum to establish international consensus on the technical basis for upgrading the safety of WWER and RBMK NPPs, to gauge the progress of the upgrading programmes and to discuss topical safety subjects. Of supplementary benefit to the Programme was the collection of technical information into a database for EBP activities, which grew to be extensive and was later used to support the work of the G-24 NUSAC Secretariat.

For the duration of the Programme, the results, recommendations and conclusions were intended only to assist national decision makers who had sole responsibility for the regulation and safe operation of their WWER and RBMK NPPs. Therefore, they did not replace the comprehensive safety assessment required in the framework of the national licensing process. Discussions at the AGMs and the Steering Committee Meetings (SCMs) helped the IAEA to ensure that the Programme remained focused on the main safety concerns of these NPPs.

1.3. INTERNATIONAL PROGRAMMES CONCURRENT WITH THE EBP

The Chernobyl accident led to a great deal of concern of Western countries over the safety of RBMKs and other Soviet designed NPPs. Following the discussions at the economic summits of the G-7 countries in 1991 and 1992, several international assistance activities were initiated. Considering their number, it was essential to establish co-ordination to ensure that available resources were invested effectively and that duplication of efforts was kept to a minimum. Co-ordination with EBP activities was achieved from the early stages by including in the SCs the representatives of the EC, OECD/NEA, G-24 NUSAC and WANO.

The EBP was conducted while other multilateral nuclear safety efforts were being deployed by the EC, OECD/NEA, G-24 NUSAC, European Bank for Reconstruction and Development (EBRD) and WANO in the countries involved. These activities, very briefly summarized below, were taken into account in setting up the work programmes of the EBP.

1.3.1. European Commission

The European Union (EU) has undertaken a great number of activities in the nuclear sector since 1990. Specific programmes were created with considerable budgetary appropriations. The EC was entrusted with the implementation of those programmes. The EC has since then established a certain number of instruments to promote nuclear safety and nuclear security in the Central and Eastern European Countries and in the New Independent States. Most active of those instruments are in the Poland and Hungary Assistance for Reconstruction of Economy (PHARE) and Technical Assistance to the Community of Independent States (TACIS) programmes, under which ECU 150 million and ECU 573 million have been committed, respectively, since 1990. The PHARE and TACIS programmes include design safety and operational assistance measures, support to regulatory regimes, radioactive waste and fuel cycle activities, on-site and off-site emergency preparedness assistance, safeguards and safety related research project.

The activities of the in the nuclear safety field have been detailed in a communication presented in March 1998 by the Commission to the European Parliament and to the Council.

1.3.2. The OECD Nuclear Energy Agency

The NEA of the OECD is carrying out a programme of co-operation with the Central and Eastern European Countries (CEEC) and the New Independent States (NIS) of the FSU under the auspices of the OECD Centre for Co-operation with Economies in Transition. The co-operation is in the field of nuclear safety and regulation, and the three main areas covered here are: enhancement and support of nuclear safety research capabilities, provision of nuclear safety information and support to regulatory authorities. The aim of the programme is to assist the CEEC/NIS in the planning, development and execution of safety research programmes with a view to building up capabilities in safety technology and safety analysis, in particular with respect to WWER reactors.

Three working groups, the so-called Support Groups, consisting of representatives from NEA Member Countries and from the CEEC/NIS, have been organized to evaluate technical proposals, to share experience accumulated in OECD Member Countries, to arrange for training of CEEC/NIS researchers in laboratories of NEA Member Countries and to assist in identifying sources of funding.

Moreover, CEEC and NIS experts are invited regularly to attend meetings of the Committee on the Safety of Nuclear Installations (CSNI), the Committee on Nuclear Regulatory Activities (CNRA), NEA workshops and specialist meetings and to participate in International Standard Problem exercises.

The Committee on Nuclear Regulatory Activities has established a link with the Association of Regulatory Bodies of Countries Operating WWER Reactors. Such a link is considered necessary to strengthen the authority and stature of CEEC/NIS regulatory bodies.

Some of the accomplishment to date are summarized below:

1. A group of specialists met several times to discuss in detail the ability of the bubbler condenser to perform its intended safety function following an accident. The group concluded that large scale confirmatory research is necessary, and they identified a number of unresolved technical issues that need to be addressed in a research programme.

2. A group of specialists is developing a matrix of experimental data and relevant phenomena which can be used to validate computer codes used in accident analysis for WWER NPPs. A similar matrix was developed in the past for LWR of Western design and is now widely used in the OECD countries

3. A group of senior experts from Russia and the OECD countries met several times to identify the research needs of Russian designed reactors. The group produced a report which lists the needs and the priorities, and formulates a number of recommendations . Some of these recommendations are now being implemented.

1.3.3. G-24 Nuclear Safety Assistance Co-ordination

The G-24 NUSAC mechanism is an intergovernmental arrangement of the world's most industrialized countries (the Group of 24, G-24), CEEC, and the NIS of the FSU. It aims to co-ordinate domestic and international programmes for improving the safety of Soviet designed nuclear power reactors being operated in the CEEC and the NIS. NUSAC seeks to enhance the effectiveness of bilateral and multilateral assistance and co-operation programmes. An important aspect of NUSAC is the better integration of international activities into the domestic nuclear safety upgrade programmes funded from the national budgets of the countries operating these reactors.

G-24 nuclear safety assistance commenced with an appeal by the G-7 London Economic Summit in July 1991 to the international community to develop an effective means of co-ordinating its response to the problems of nuclear safety of the countries of Central and Eastern Europe and to the then existing Soviet Union.

Short term technical improvements were intended primarily for the first generation of plants of WWER and RBMK design and to enhance their levels of safety as soon as practicable. Grant financing was not intended for improvements enabling long term operation, that is, until the end of design life.

Having evolved from the G-24 Working Group, the more structured mechanism established in 1992 and known as G-24 NUSAC was adapted continuously to the changing needs of its participants. The initial situation of pure technical assistance is evolving to an environment of co-operation and partnership where participants from G-24, countries of Eastern Europe and of the FSU seek to resolve jointly the persisting nuclear safety problems in this part of the world.

The IAEA acts as a technical advisor to the NUSAC Group and provides technical information on relevant issues, based on the results of its programme.

1.3.4. The European Bank for Reconstruction and Development/Nuclear Safety Account

At their Munich Summit (6-8 July 1992), the G-7 heads of state and government offered the countries of Central and Eastern Europe and the FSU a multilateral programme of action to improve safety in their nuclear power plants.

As part of the programme, the G-7 also advocated the setting up of a new mechanism complementary to the local efforts of the countries operating these reactors and to traditional bilateral assistance programmes. This resulted in the establishment of the Nuclear Safety Account (NSA) at the Bank. The NSA became effective in April 1993 for a period of three years. The term of the NSA was extended in April 1996 for a period of another three years. The activities financed through the NSA are carried out under the supervision of the Assembly of Contributors. The Bank as Fund Administrator provides technical, project monitoring, financial, legal and administrative services.

The purpose of the NSA is to finance, through grants, the implementation of projects designed to address immediate operational safety and short term technical safety improvement measures (safety equipment and associated engineering) for nuclear reactors having the highest level of risk pending their early closure. The findings of the IAEA have largely been used by the Bank for the identification of these measures.

On the basis of the mandates of the Assembly of NSA Contributors, the Bank negotiated with the recipient countries Agreements which include conditionality clauses to ensure successful implementation of the short term safety upgrades and the preparation of decisions on the early closure of reactors of older designs the safety of which can be significantly enhanced.

Since the inception of the Nuclear Safety Account in March 1993, the Bank has signed Agreements as NSA Administrator with the Governments of Bulgaria, Lithuania, Russia and the Ukraine, the relevant countries operating these reactors and the Russian Nuclear Safety Authority (Gosatomnadzor).

The Agreements include commitments for a total amount of ECU 261.5 million for the financing of six short term safety upgrading projects for seven RBMK and eight WWER-440/230 reactor units, the pre-decommissioning facilities for the Chernobyl NPP, the in-depth safety assessment project of Ignalina NPP Units 1-2 and a project to support RF Gosatomnadzor in the licensing review of NSA-financed safety upgrades.

The Agreements also call on the concerned countries to implement their commitments to: establish sound and transparent regulatory regimes for designated reactor units on the basis of plant specific in-depth safety assessments; execute least-cost planning activities and reforms in the power subsectors; and to the early shutdown of less safe reactors.

The implementation of a number of important provisions of the NSA Agreements is supported by the bilateral co-operation agencies of the NSA Contributors. These projects include: the in-depth safety assessment of selected reactor units in Russia; the preparation of the Russian North-west Region least-cost power sector investment study; and the licensing review process of the Ignalina NPP Unit 1. The projects are well co-ordinated with the NSA, there is an exchange of information on the progress and expediting measures are jointly carried out.

The Bank has indeed promoted a large number of projects in the countries of operations. All in all, the Bank to date has committed ECU 2.1 billion out of a total commitment of ECU 10.9 billion on projects in the energy area, positioning energy at the second rank of all sectors, after financial institutions. Not less than half of the ECU 2.1 billion in commitments relates to the four NSA countries of operations while the Bank operates in 25 countries of Eastern Europe. This calculation does not yet include the Khmelnitsky NPP Unit 2 and Rovno NPP Unit 4 (K2R4) project which is at present under due examination.

1.3.5. The World Association of Nuclear Operators

The World Association of Nuclear Operators (WANO), founded in 1989, unites all nuclear electricity operators in the world. It facilitates the exchange of operating experience so that its members can work together to achieve the highest possible standards of safety and reliability in operating their nuclear power plants.

Communications and the WANO website are essential services in general support of WANO and its members. Through WANO all nuclear power plant operators can communicate and exchange information with one another within a culture of unique co-operation and openness.

Membership of WANO is through its four Regional Centres - Atlanta, Moscow, Paris and Tokyo - and is determined by geographical location or reactor type. WANO’s programmes currently include

- The Operating Experience Programme
- Professional and Technical Development Programme
- Technical Support and Exchange Programme
- The Peer Review Programme.

WANO also recognizes the particular needs of many of its members in the Former Soviet Union and Eastern Europe and supports these organizations in achieving upgrades to enhance the safety of their plants and to share information on their projects and experience.

Review of an IAEA report on requirements to upgrading WWER 440/230 units led to the EU funding of a ‘Special Project’ to implement the recommendations at Kozloduy. A contract was placed with WANO-PC for a programme of work covering outage assistance, twinning and housekeeping, and multinational WANO teams were drawn together to implement the changes. Subsequently a contract for the Project Management Unit was awarded initially to Nuclear Electric of UK and later to DTN of Spain. A contract for on-site assistance was awarded to Electricité de France (EdF).

Operators of Soviet designed reactors in the FSU and Eastern Europe formed a Users Group with the aim of working together in areas of common interest relating to operational safety. The establishment of the Group was supported by WANO.

Early in 1994 the Users Group produced specific reports [1] identifying potential safety and operational modifications or backfits for the four basic Soviet reactor designs (WWER-1000, WWER-440/230, WWER-440/213, RBMK). The Group defined in some detail a priority listing of improvements for consideration by western funding agencies. This led to some western funding for specific projects.

In 1993 the EU launched an On-Site Assistance Programme under its general TACIS Programme with the aim of upgrading the safety of the Soviet designed reactors operated by member utilities of the Moscow Centre. Initially, this was at eight sites in the FSU (six in Russia and two in Ukraine). In 1994 two sites were added (one each in Ukraine and Kazakhstan). In 1997 Armenia (Medzamor) was added to the programme. WANO established a Co-ordinating Committee in response to an EU request for it to perform certain co-ordinating and advisory functions based on a recognition that both the EU utilities providing the assistance and the Eastern utilities receiving it are members of WANO of the Paris and Moscow Centres, respectively. The Co-ordinating Committee has functioned back to back with the committee originally formed to oversee safety matters at Kozloduy which has been extended to include all WWER 230s. Similar 'family groups' have been set up for the WWER-1000s and RBMKs under the sponsorship of WANO and with TACIS funding.

1.3.6. Bilateral programmes

A number of bilateral programmes were established between countries operating WWER and RBMK NPPs and OECD countries. The activities carried out in the framework of some of these programme were periodically reported to the IAEA during the meetings of the Steering Committees. This exchange of information was fundamental to ensure co-ordination and avoid undue duplication.

Most of the information obtained derived from bilateral programmes involving the following countries:

Germany

German bilateral co-operation is based on agreements between the Federal Government, i.e., the Ministry for the Environment (BMU) and the Research Ministry (BMBF), and Russia, Ukraine and other Eastern European countries.

Research co-operation with Russian institutions was begun in 1987 and has been extended to institutions in other Eastern European countries as well. BMBF funded co-operation projects focused on WWER reactors, mainly concentrating on:

- component integrity;
- code development for accident analyses (reactor physics, thermal hydraulics);
- severe accident analyses, man-machine communication and application of modern methods of diagnosis and risk and reliability;
- application of probabilistic safety analyses.

With concern to RBMK plants, codes for accident analyses (reactor physics, thermal hydraulics) have been developed further and verified.

The objectives of the BMU programme, which forms the major part of German assistance to enhance nuclear safety in Eastern Europe, largely correspond to the first three points of the action programme decided at the G7 summit in Munich in 1992.

Co-operation with the authorities focuses on:

- organizational and administrative assistance on site;

- improvement of working conditions and the technical infrastructure;

- seminars and training measures, transfer of methods used in Western safety analyses, provision of computer codes, and joint scientific and technical analyses of safety issues;

- support provided to the authorities on technical aspects of licensing and supervision;

- assistance in the preparation of safety-related codes and guides.

The technical work performed jointly with scientific and technical institutions as well as with the industry comprises of:

- safety analyses and recommendations for safety improvements
- comments on backfitting programmes
- analyses and proposals for the improvement of operational management
- technical assistance to the Balakovo (Russia) and Rovno (Ukraine) NPPs
- transfer of know-how concerning in-service inspections.

Japan

The Japanese bilateral collaboration focused on the safety practices and researches closely related to NPP operation which Japan would be able to share with the countries of Eastern Europe and the FSU. The bilateral programme consisted of both human resource development and equipment grants sponsored by two government ministries, the Ministry of International Trade and Industry (MITI) and the Science and Technology Agency (STA). MITI collaboration concentrates on the activities directly related to NPP operation, while the STA collaboration covers safety research and radioactive waste management.

The MITI thus set up a programme to invite to Japan 1,000 specialists Eastern Europe, FSU and China during the 10 year period since 1992 to share the safety practice and experience at NPP sites and manufacturers' laboratories. As of March 1998 a total of 577 specialists had been invited to various courses for general administrators, managers and supervisors, maintenance personnel, seismic designers and regulators. Another MITI collaboration is the donation of a high performance full-scope plant simulator for WWER-1000. With effective co-operation between Japan and Russia, the simulator development was successfully completed and installed at the Novovoronezh Training Centre in July 1996.

In parallel with the MITI activities, the STA also organized Nuclear Safety Seminars to invite to the Japan Atomic Energy Research Institute (JAERI) more than 1,000 specialists and experts from Eastern Europe, FSU and the Asian countries since 1992. The seminars covered topics on nuclear safety, radiation waste and spent fuel management and safeguards. For further assistance the STA also organized symposia, a workshop and an expert mission to enhance better understanding of the status and safety issues of NPPs being operated in Eastern Europe and FSU.

USA

The United States actively participates in international efforts to improve the safety of Soviet-designed NPPs. These activities are focused in three principal areas: near-term risk reduction, operational safety enhancements, and regulatory assistance. The US Nuclear Regulatory Commission (USNRC) is the principal U.S. Government implementing agency for providing regulatory assistance. The US Department of Energy (USDOE) is the principal implementing agency for near-term risk reduction and operational safety efforts. All activities are conducted in accordance with the guidance and policies of the US Department of State and the US Agency for International Development. Arrangements covering regulatory assistance activities have been established between the USNRC and the regulatory authorities of Armenia, Bulgaria, Czech Republic, Hungary, Kazakhstan, Lithuania, Russia, Slovakia and Ukraine. NRC’s goal is to assist these organizations in developing into strong, independent regulatory authorities. These activities are conducted using USNRC personnel and expertise, supplemented by US national laboratories, US commercial organizations and by personnel and expertise from the regulatory authorities in the host countries, as needed. Major technology areas where USNRC has provided regulatory assistance include: development of a legal basis for nuclear regulation, development of a licensing process, development of inspection procedures, development of an analytical support capability, establishment of regulatory training programmes and development of fire protection regulations and inspection techniques. The USDOE has established partnerships with eight "host" countries - Armenia, Bulgaria, Czech Republic, Hungary, Lithuania, Russia, Slovakia and Ukraine - to improve the physical conditions of plants, train operators and establish modern safety technologies and methods. The USDOE effort is managed by the Office of Nuclear Energy, Science and Technology. Pacific Northwest National Laboratory provides technical and management support. Additional support is also provided by other US national laboratories, US commercial organizations, and the NPPs and scientific and engineering organizations in the host countries, as needed. Major technology areas where the USDOE has provided assistance include: management and operational safety improvements (operational safety), training and maintenance, engineering and technology upgrades, plant safety evaluations, fuel cycle safety, and legislative and regulatory support.

Sweden

The Swedish bilateral co-operation in nuclear safety in the Baltic region is co-ordinated by the Swedish International Project (SIP) on Nuclear Safety. The project has been set up especially for this purpose by the Swedish Nuclear Power Inspectorate (SKI), considering the clear differences in the work approach and the financial base for the co-operation programme abroad, as compared with the regulatory work of SKI in Sweden.

From its inception six years ago, this programme has been focused on Lithuania and its two RBMK-1500 units at the Ignalina NPP. Co-operation with the Lithuanian authorities includes assistance to the Ministry for the development of the Nuclear Energy Law and to VATESI in developing their regulatory base, control methods and staff competence. Industry and NPP co-operation includes the following areas: material inspection, QA development, management development, the PSA Barselina project, emergency preparedness, fire hazards analysis, radioactive waste handling and the establishment of an information centre. Technical-hardware projects include upgrading of fire protection, physical protection, communication and non-destructive testing (NDT) equipment.

Some two years ago, co-operation involving also the US DOE and UK (AEAT) was started with Russia on a safety analysis project on Leningrad NPP Unit 2. Work has started to extend this collaboration into an in-depth safety analysis for Unit 2. Co-operation with Leningrad NPP has also recently begun in the area of in-service inspection. A co-operation programme with the Kola NPP in Russia is under development.

2. PROGRAMME IMPLEMENTATION

2.1. PROGRAMME DEVELOPMENT AND SCOPE

The years spanning the EBP can be subdivided into at least two distinct phases. The first commenced in 1990 with Member States operating WWER-440/230 NPPs requesting the IAEA for assistance and donor countries from the West agreeing to support such an assistance programme. The AG, which met in September 1990, established the technical scope of a work programme and agreed on a programme which included the following elements:

- review of the design concept to obtain an overview of the safety aspects of WWER-440/230 NPPs;
- safety review missions (SRMs) by teams of international experts to the individual NPP sites to identify plant specific deficiencies in design and the conduct of operations, maximum use being made of IAEA experience in the provision of safety services, especially OSART and ASSET missions, and
- studies on aspects of generic safety concern.

This first phase, which was limited to WWER-440/230 NPPs, was completed in 1992 [2]. It identified design and operational safety issues, evaluated their safety significance and assigned categories to them according to their potential to degrade plants' defence in depth. This evaluation helped the countries operating these reactors to establish and implement national and plant specific programmes to improve the safety of their plants.

Early in 1992, the IAEA started the second phase of the EBP for WWER-440/230 NPPs. The major objective of this second phase was to assist Member States to evaluate modifications (both with respect to hardware and to the conduct of operations) in order to verify that proposed modifications responded to the concerns identified and recommendations made during the first phase, to review matters of generic safety concern and to establish consensus on actions required.

In December 1992, an AGM was convened by the IAEA in Vienna to review the progress of IAEA safety programmes for countries of Eastern Europe and the FSU and to advise on future activities and co-ordination. In general it was agreed that the EBP had been very useful and that the activities planned responded to the assistance requested. In response to requests from Bulgaria, the Czech Republic and the Ukraine, the AG recommended that the EBP be extended to the other types of plants of Soviet design, namely the WWER-440/213, WWER-1000 and, at the request of Russia, the RBMK NPPs.

The Steering Committee originally established to advise the IAEA Programme related to the safety of WWER-440/230 NPPs was expanded to deal with WWER-440/213 and WWER-1000 NPPs. A second Steering Committee was established to deal with the RBMK plants.

In December 1994, an AGM reviewed the EBP activities from 1992 to 1994 and the AG reiterated that all results, recommendations and conclusions resulting from the EBP were only intended to assist national decision makers with sole responsibility for the regulation and safe operation of their NPPs. There was general agreement that (i) the results obtained in the framework of the EBP provided important insights for safety decisions to be taken by the national authorities, (ii) the EBP achievements had been numerous and proved to be a cost effective means of international assistance, and (iii) the EBP provided a forum for an effective and open exchange of international experience and could respond in a fast and effective way to requests made by countries operating WWER and RBMK NPPs.

The AG reviewed the scope and objectives of the EBP and agreed that further activities should focus on the preparation of guidelines, peer reviews, assistance for evaluation of plant specific safety improvements/modernizations (upon request) and organization of topical meetings to consolidate the state of knowledge and obtain international consensus on actions taken and required to resolve generic safety issues. It was reiterated that beyond the EBP activities a comprehensive safety assessment was required under the national licensing process. Assistance in the review of safety improvements of first and second generation RBMKs was also considered a matter of high priority.

The AG recommended that the detailed work schedule for 1995 be agreed by the Steering Committees considering the priorities of the assistance requested, the funding available and pledged, and national, bilateral or multilateral programmes.

The EBP activities during the year 1995 were reviewed at an AGM in December 1995. The AG noted with satisfaction that the IAEA had taken steps to incorporate part of the assistance to improve WWER and RBMK safety into its Regular Budget and TC, and urged the IAEA to continue this process. This was also supported by countries providing and receiving assistance.

At their last meeting in December 1996, the AG agreed that the EBP should be phased out and brought to a successful completion by the end of 1998. They recommended that during the period 1997-1998, the extrabudgetary part of the IAEA activities concentrate on topical meetings addressing the generic safety issues presented and on completion of the list of safety issues and ranking for the ‘small series’ WWER-1000s as well as RBMKs of the first and second generations. They also recommended that special consideration be given by the IAEA to strike a balance in the remaining activities of the EBP on the safety of RBMK and WWER reactors, ensuring that highest priority be given to the successful completion or effective transfer into the IAEA’s regular programme or other international programmes of the work remaining to be done on these reactors of older design.

They noted that further work was needed to promote safety culture, to exchange experience on implementation and regulatory assessment of safety modifications, to finalize guidelines and maintain the database on the plant specific status of safety improvements and to assist the G-24 NUSAC. In this context, they also noted that the amount of work to be carried out would depend on the availability of extrabudgetary funds and human resources.

It was understood that the work remaining to be completed beyond 1998 in the framework of IAEA, national, bilateral or other international programmes in co-ordination with G-24 NUSAC, would be transferred in an orderly fashion.

2.2. ADVISORY GROUP AND STEERING COMMITTEES

The EBP considered the results of other relevant national, bilateral and multilateral activities. It was therefore essential to have an advisory and co-ordination mechanism to ensure that the scope and the objectives of the programme were consistent with the expectations of the donor and recipient Member States.

The scope and results of the programme were discussed at five AGMs. The final meeting of the AG is scheduled for the second half of 1998, when the EBP will be phased out and brought to a successful completion.

Two Steering Committees were established, one for WWER and the other for RBMK NPPs. The members of the SCs were nominated by the Member States participating in the EBP (both recipients and donors). Considering the numerous assistance efforts under way as a result of the discussions at the economic summits of the G-7 countries in 1991 and 1992, it was essential to establish co-ordination to ensure that the available resources were allocated efficiently and that duplication of efforts would be kept to a minimum. The co-ordination of EBP activities was achieved from the early stages by including observers from the G-24 NUSAC, the EU, EBRD, European Investment Bank (EIB), OECD/NEA, WANO and the International RBMK Consortium (only for the RBMK SC) in SC meetings. The terms of reference for the SCs and their current composition can be seen in Annex 1.

Three meetings lasting three days were typically dedicated to each SC. The first part of each SCM was devoted to technical reports on the progress and problems registered by each user country and international organization and reports by the USA and Japan on their bilateral activities. This arrangement set the stage for frank discussions between all members and observers, a factor which proved to be crucial for the user countries involved and the IAEA. Initially, the WWER SC worked with the IAEA to define the detailed scope and methods for the reviews of each type of NPP and of the parallel activities. Another early IAEA and SC activity was the agreement on categorizing WWER safety issues. Subsequently, detailed work programmes were prepared annually by IAEA and cleared by the SCs. Progress was discussed at each SC meeting and recommendations were made for any proposed changes. EBP reports were issued regularly to SC members and the more important ones were discussed at SC meetings.

The SCs were able to influence ongoing and future EBP activities directly, inject new ideas and ensure that duplications with other programmes were prevented. The SCs had also a strong interest in some safety issues and actively encouraged EBP participation, in particular with respect to the safety of model 230 reactor pressure vessels and their resistance to pressurized thermal shock (PTS). Others were the leak before break capabilities of WWERs, site seismic characteristics and quality and usage of PSAs produced commercially for WWERs. After the decision to end the EBP in 1998, the SCs advised the IAEA on the successful completion of the programme.

By the end of 1998, the WWER SC had completed 24 meetings and the RBMK SC, 12 meetings.

2.3. BUDGET AND HUMAN RESOURCES

The implementation of the extrabudgetary part of the Programme depended solely on voluntary contributions from IAEA Member States. However, some activities were also funded through the Regular Budget and Technical Co-operation and were complemented by national and regional technical co-operation projects.

The IAEA organizational structure for the EBP was first formed at the end of 1990 and was gradually expanded with additional experts over the years, some of whom were provided on a cost free basis by Member States. In 1996, at the height of the EBP activities, a total of 23 IAEA staff (16 professionals and 7 support staff) were working full time on the Programme.

Cost-free experts from France, Germany, Hungary, Japan, Spain, Switzerland, USA and the EC were also assigned to work with the IAEA secretariat in Vienna. The participation of external experts was supported by the donor Member States, so that the limited resources of the EBP could be used more effectively to address technical activities. The combination of dedicated IAEA staff and external experts made efficient accomplishments possible in a relatively short time.

Between 1991 and 1998, total cash contributions amounted to nearly US $12 million. In addition to the EBP staff working at the IAEA some 17000 man-days of technical experts from various countries have participated in the Programme activities from 1990 to July 1998. This figure includes 3700 man-days (US $2.4 million) covering recruitment of experts under related TC projects between January 1995 and July 1998.

Annex 2 shows the cash and in-kind contributions of the Member States and international organizations between 1990 and 1998.

2.4. SAFETY REVIEW APPROACH

In the framework of the EBP, assistance was given to countries constructing and/or operating WWER and/or RBMK NPPs to identify relevant design and operational safety deficiencies, to judge their safety significance and rank them accordingly, and to review proposed safety improvement/modernization programmes.

To achieve the Programme objectives, the IAEA adopted a safety review approach which was applied to all EBP activities from the beginning. The approach was based on two elements:

1. Establishing the reference basis

The IAEA NUSS documents, which are the outcome of an international consensus, were considered to be the reference. The NUSS documents reflect both the current concepts on the safety of nuclear power plants with thermal neutron reactors and the specific attitudes towards the implementation of safety throughout the entire life of nuclear power plants.

Current Russian standards such as OPB-88 and PBYa-89 were included in the reference basis because the current Russian safety concept was found to be comparable with NUSS, despite some differences in approach [3].

Similarly, international good practices, including operational practices conveyed by experts from countries involved in the EBP activities, were considered in the reference basis for nuclear safety.

This reference basis was used to identify the deviations and deficiencies that were considered significant for plant safety; these were designated as ‘safety issues’ or ‘issues’ in short (Section 2.5).

2. Judging safety significance

The safety significance of the issues identified and their priority were categorized according to their impact on defence in depth [4]. Therefore, the impairment of defence in depth by a given issue involved a judgement of the effectiveness of the performance of the main safety functions: controlling the power, cooling the fuel and confining the radioactive material. This judgement also considered the principle that plant conditions with a relatively high frequency of occurrence would have only small consequences and plant conditions resulting in plant damage with high radioactive release would be of a low frequency of occurrence [5].

Defence in depth classification criteria used for the issues identified in design and operation are given in 2.5.1. Based on this approach, the impact of each issue on plant safety was considered individually. A safety issue and its ranking is considered a reference for plant specific safety improvements.

An assessment of the combined impact of all relevant issues on plant safety was found to be important, considering the main safety functions in different plant conditions. Fig. 1 presents the barriers which can be directly degraded by the safety issues. It also shows the main safety functions which might be affected by several issues during different operational and accident scenarios. Other systems and factors which can directly affect safety functions and the integrity of barriers are also shown.

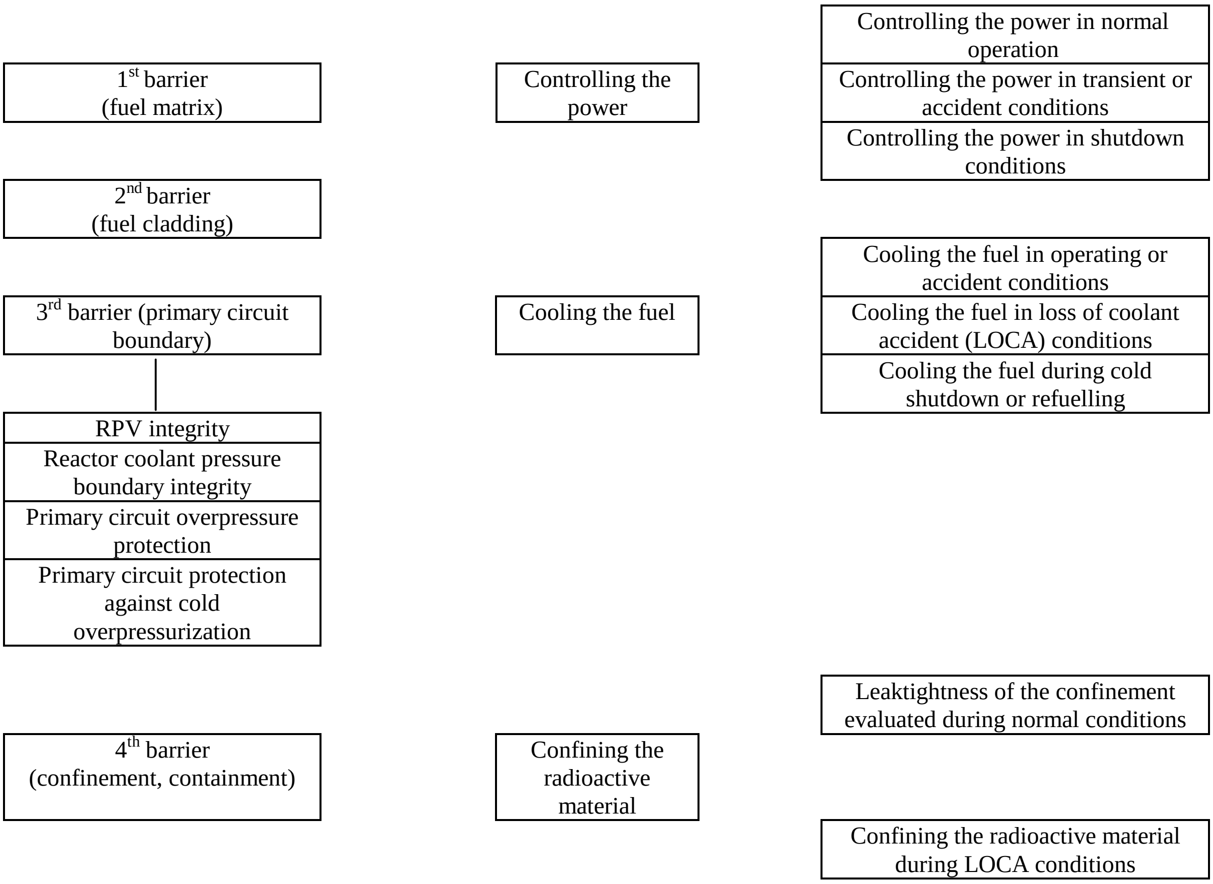

FIG. 1a. Barriers and main safety functions in different operational and accident conditions.

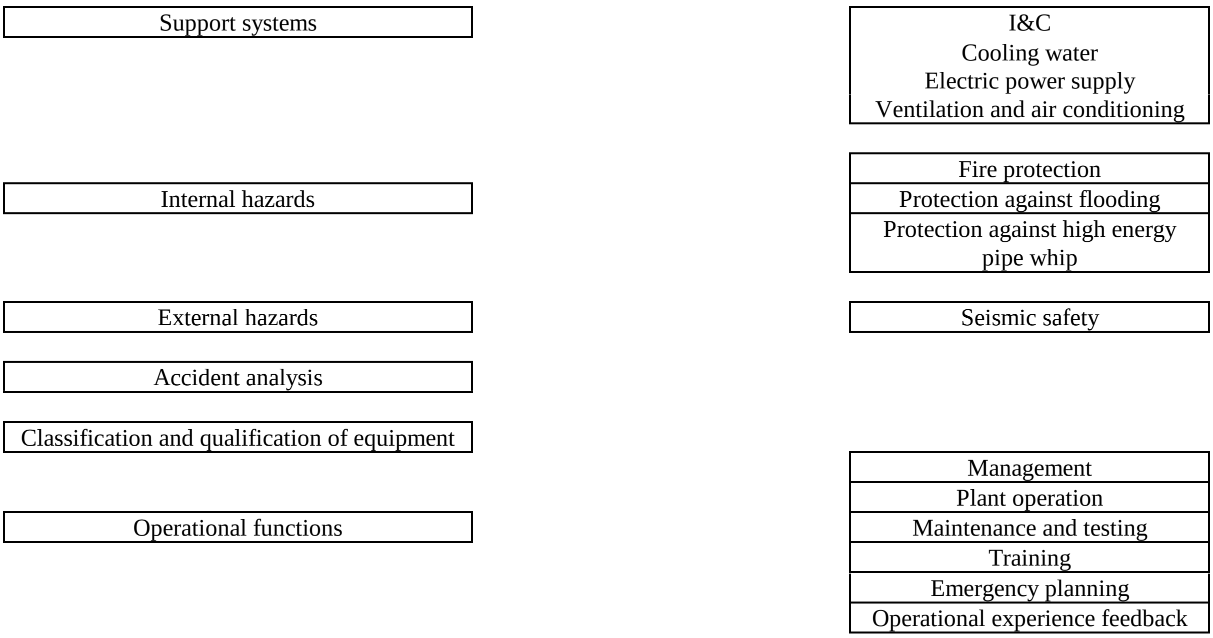

FIG. 1b. Other factors which can affect the barriers and the main safety functions

2.5. PROGRAMME ACTIVITIES

The programme implementation was based on the IAEA's broad expertise and well developed infrastructure for providing nuclear safety assistance. It also took full advantage of the results of safety evaluations completed and under way in the framework of many national, bilateral and international programmes.

Subsequent to the AGM of 1992, the scope of the extended Programme included the following major areas:

- Identification and ranking of generic safety issues
- Conceptual design and safety reviews
- Review of safety improvement/modernization programmes
- Studies of generic safety issues
- Preparation of guidance for safety analysis
- PSA reviews
- Training activities
- Establishment and maintenance of a database
- Assistance to G-24 NUSAC.

The greater part of the EBP efforts centred on the development of the Safety Issue Books for the different types of WWER and RBMK NPPs. Safety Issue Books are lists of safety issues generic to all units of a plant type with the associated safety significance by ranking and recommendations for safety improvements for the WWER-440/230, WWER-440/213, WWER-1000/320 and RBMK NPPs. The main sources for these Safety Issue Books were the Safety Review Missions carried out under the Programme, and the evaluation of the results of other missions by the IAEA safety services. The complex process of producing the Safety Issue Books ultimately ensured the consensus of all donor and recipient Member States involved in the EBP.

The Safety Issue Book for the WWER-440/230 NPPs was the first to be completed in 1992 [6]. The remaining Safety Issue Books were completed in 1995 and published in 1996 [7, 8, 9]. These documents were used as the benchmark for all subsequent expert missions to review plant specific safety improvement/modernization programmes.

Annex 3 shows the work plan implemented each year for the duration of the Programme.

2.5.1. Identification and ranking of generic safety issues

Each Safety Issue Book contains a consolidated list of safety issues, ranked according to their safety significance and the corrective measures proposed to improve them. The safety issues were identified by comparing the current plant state and operational experience with the reference basis established in the EBP. For practical reasons, the identification of issues encompassed the following areas: reactor core, component integrity, instrumentation and control (I&C), safety and support systems, internal and external hazards, accident analysis and operational safety.

The ranking of the individual safety issues based on four categories was first developed for WWER-440/230 plants and later applied to all other types of WWER plants.

Category I: These issues reflected a departure from recognized international practices. In certain cases, it was appropriate to address them as part of the actions to resolve higher priority issues.

Category II: Issues of safety concern. Defence in depth was degraded. Action was required.

Category III: Issues of high safety concern. Defence in depth was insufficient. Immediate corrective action was necessary. Interim measures were possibly also necessary.

Category IV: Issues of the highest safety concern. Defence in depth was unacceptable. Immediate action was required to overcome the issue. Compensatory measures were to be established until the safety problems were resolved.

Based on the experience in ranking the significance of safety issues for WWER plants, the IAEA has developed a common basis for judgement to evaluate the safety of operating plants built to earlier standards [4]. This approach based on three categories was applied first time to the RBMK plants.

High: Issues that reflected insufficient defence in depth and had a major impact on plant safety. Short term actions had to be initiated to improve safety as applicable to each specific NPP, until the issue was fully resolved.

Medium: Issues that reflected insufficient defence in depth and had a significant impact on plant safety. Short term actions were probably necessary to improve safety as applicable to each specific NPP, until the issue was fully resolved.

Low: Issues that reflected insufficient defence in depth and had a small impact on plant safety. Actions were desirable to improve defence in depth, if applicable and effective from a cost benefit point of view.

In general, issues detected on the basis of design analysis or operational experience such as shortcomings in the actual implementation of engineering design, material degradation due to improper specification or due to loads not specified in the design had been given higher safety importance than issues identified as a potential deviation from current safety standards. To the extent that information was made available to the IAEA, each Safety Issue Book included the country/plant specific status with respect to each safety issue.

The safety issues and their ranking for each reactor type are listed in Annex 4.

2.5.2. Conceptual design and safety reviews

As a first step for compiling generic safety issues, conceptual design reviews and site specific safety review missions were undertaken to identify and evaluate plant specific deficiencies in the areas of plant design and operation. Conceptual design review meetings for WWER-440/230 and RBMK NPPs were organized by the IAEA. The reviews brought Western specialists together with designers, operators and regulators from Eastern countries. International teams of some 15 experts participated in safety reviews at NPP sites which typically lasted three weeks. Follow-up safety review missions by groups of some 6 experts over a period of one week were later organized in order to assess the status of implementation of recommendations made by previous safety review missions.

Specific missions were conducted at plant sites to review seismic safety. These included reviewing the design basis, seismic structural design and construction.

Meetings were also held to review a broad range of safety aspects and safety upgrading measures generally applicable to most of the plants of a given type. Based on the review meetings, safety issues were categorized in accordance with their safety significance.

2.5.3. Review of safety improvement/modernization programmes

The Safety Issue Books formed the basis of the evaluations of safety improvement programmes for individual NPPs. Expert missions were conducted at the request of the Member States by relatively small international teams of some four experts and IAEA staff within a period of seven to ten days. The purpose of these missions was to review the safety aspects of the safety improvement programmes and to advise on the completeness and adequacy of the safety improvements proposed with respect to the recommendations made in the Safety Issue Books.

Other missions called Technical Visits and usually lasting five days were conducted, generally by IAEA staff. The information gathered from all missions was entered into the IAEA technical database established for WWER and RBMK NPPs (Section 2.5.8).

The list of missions is shown in the Appendix.

2.5.4. Studies of generic safety issues

Studies of generic safety issues with a higher safety concern were carried out to assess existing knowledge and to focus on actions for their resolution. They typically covered the topics indicated in the Appendix.

For some issues, the IAEA awarded technical contracts to the experts from the countries involved, and detailed reports were prepared on specific subjects.

Some of these studies included technical missions to the NPPs, e.g., the activities concerning bubbler condenser containment performance included analyses commissioned to the main design organization [10], and meetings at the IAEA, Expert Missions to review the bubbler condenser structural integrity at Mochovce and Bohunice V-2 NPPs in November 1995 [11], and at Dukovany NPP in May 1996 [12].

The IAEA also supported initiatives of Member States related to problems specific to WWER units, e.g., by collaborating with Finnish organizations in the International Seminar on Horizontal Steam Generators [13] or in the International Seminar on Transient and Accident Analysis and Code Validation for WWER-440 Reactors in Helsinki, November 1995 [14].

Generally, the preparation of topical reports for generic safety issues involved several meetings, with papers prepared by Eastern and Western experts, and a thorough discussion at joint experts sessions. The conclusions reached at the meetings were agreed between all the organizations involved and the regulatory bodies in the countries operating WWER units. As a results of the involvement of practically all the major organizations dealing with WWER safety, the EBP reports summarizing findings on topical safety issues were subsequently used as reference for further work, both in individual plants with WWER units and in other international programmes.

2.5.5. Preparation of guidance for safety analysis

The need for detailed guidance in the performance and review of accident analysis for WWER and RBMK nuclear power plants, identified as a priority in the EBP, led to the preparation of:

- Guidelines for Accident Analysis of WWER Nuclear Power Plants [15]
- Procedures for Analysis of Accidents in Shutdown Modes for WWER NPPs [16]
- Transient and Accident Analysis of RBMK NPPs [17]

These guidelines deal with the transient and accident analysis required at NPPs with WWER and RBMK type reactors to justify existing or newly proposed technical solutions, such as plant modifications for safety upgrading. The guidelines give advice on the selection and categorization of initiating events to be considered, on adequate specification of acceptance criteria, methods, computer codes, assumptions to be used and quality assurance procedures for accident analysis.

Guidance was also developed to assist Member States with respect to developing national approaches to cope with ATWS events and primary to secondary leaks within the design basis for WWER plants:

- ATWS for WWER Reactors [18]
- Treatment of Primary to Secondary Leaks in WWER Nuclear Power Plants [19]

Specific guidance to assist Member States in ensuring the integrity of physical barriers, i.e., the integrity of the RPV, the integrity of the primary piping and the containment structure of WWER NPPs, was also provided to countries operating WWER plants.

- Guidelines on the PTS Analysis for WWER NPPs [20]
- Guidance for Application of the Leak Before Break Concept [21]
- Methodology for Qualification of In-Service Inspection Systems for WWER NPPs [22]
- Evaluation Guidelines for Bubbler Condenser Metallic Structure in WWER-440/213 NPPs Containments [23]
- Guidelines for WWER-440/213 Containment Evaluation [24]

To address the topical issue of protection against fire hazards, a guideline was prepared on:

- Fire Hazard Analysis for WWER Nuclear Power Plants was developed applicable to all WWER type reactors [25]

These guidelines are being used by Member States for the preparation of Safety Analysis Reports (SAR).

2.5.6. PSA assistance

In the framework of the EBP and in co-operation with TC National and Regional Projects, the IAEA provided substantial assistance through workshops (WS) on PSA methods and to review PSA studies in various levels of completion, including:

In 1992:

- WS on PSA Methods (Moscow) [26]
- WS on Kola NPP Unit 1 PSA Review [27]
- Review of Top Level Risk Study for Kozloduy NPP Units 1-4 [28]

In 1993:

- WS on Data Collection and Processing (Ukraine Zaporozhe)
- WS on Reliability Data Exchange (Hungary) [29]
- Seminar on PSA in Regulatory Environment (Czech Republic)
- WS Methodology for Level 2 PSA (Russia)
- Safety Study Review: PSA Peer Review (Paks) [30]
  PSA Peer Review (Bohunice) [31]
  PSA Peer Review (Kola) [32]
- WS on definition of initiating events for WWER-type reactors (Russia) [33]
- Review of treatment of human reliability in PSA (Paks)
- Basic PSA seminar (Slovak Republic)
- WS on Basic PSA methodology (Ukraine)

In 1994:

- RBMK PSA Scope Review (Leningrad) [34]
- PSA Peer Review (Kozloduy) [35]
- PSA Peer Review (Bohunice) [36]
- PSA Peer Review (Paks) [37]
- WS on PSA Methodology [38]
- Seismic PSA Peer Review for Unit 5 (Kozloduy)

In 1995:

- Review of the Insights from WWER PSAs on the Safety Upgrading Programmes (Vienna) [39]
- WS on PSA Methodology for Fire, Flooding and Seismic Events (Moscow) [40]
- Peer Review of Level 1 PSA (Modelling) (Bohunice) [41]
- Peer Review of Level 1 PSA (Temelin) [42]
- Review of Characteristics of Level 1 PSA Codes [43]
- Peer Review of Seismic PSA Results (Kozloduy)

In 1996:

- WS on PSA Applications (Sofia) [44]
- WS on Harmonization of WWER PSA Model Assumptions and Data (Rez) [45]
- Peer Review of PSA Application to Determine Allowed Outage Times (Paks)
- Peer Review of Internal Fires and Floods, External Events and Level 2 PSA (Temelin) [46]

In 1997:

- WS on Reliability Data for PSA [47]
- PSA Peer Review (Kozloduy NPP Units 3 and 4) [48]

In 1998:

- WS on External Events PSA (Moscow) [49]

2.5.7. Training activities

A series of workshops were conducted within the framework of the EBP to facilitate and promote an exchange of information between regulatory bodies, technical support organizations and plant operators. In the early stages, these addressed a variety of topics such as safety inspections, risk-based optimization of operational tasks, preventive maintenance, emergency preparedness, accident management, and fire hazards. Examples of the areas addressed are:

- licensing of plant modifications [50, 51]
- use of databases on safety issues [52]
- international practices in RPV integrity assessment [53]
- RBMK fuel channel integrity [54]
- promotion of safety culture (RBMK) [55, 56]
- calculations for thermohydraulic code validation (RBMK)
- regulatory assessment of safety upgradings of operational NPPs (1996) [57].

2.5.8. Establishment and maintenance of a database

From the very beginning of the EBP, a technical database was developed and established on a computerized basis. The comprehensive database contains the main results of the IAEA EBP on the Safety of WWER and RBMK NPPs, including safety issues related to design and operation of WWER and RBMK plants, recommendations related to the safety issues and in particular, plant specific information on the current status of the upgrading measures implemented to improve plant safety.

The database aims at facilitating access to the information compiled and their exchange among the specialists particularly plant operators.

Therefore, the Safety Issue Books of all WWER and RBMK plants, information on safety review missions and their follow-ups, technical visits, reviews of safety improvements have been included in the database. Some relevant results of other programmes (e.g., EC-sponsored studies) were also taken into account.

The structure of the databases allows to register specific actions taken by individual plants to address safety issues. While the status of implementation of safety modifications varies from plant to plant, periodic visits to the plants have been conducted by the IAEA within the framework of the EBP to update information on the implementation of measures of relevant safety improvement/modernization programmes and to reflect the current plant status in the database.

The database has been distributed by the IAEA to Member States through liaison officers designated as the official contact persons in each country. Any future updates of the database will also be provided to the liaison officers who have full responsibility for the distribution of the database in their countries.

The databases are considered to be distributed only to organizations which are involved in the assistance programmes to countries operating WWER and RBMK plants.

The database are available in run time format on diskettes for use in IBM compatible personal computers. A user’s manual was also prepared [52] to facilitate the use of the database.

Activities related to maintaining and updating the database have been included in the IAEA Nuclear Safety Programme 1999-2000.

2.5.9. Assistance to G-24 NUSAC

In 1992, the IAEA was requested to assist the G-24 NUSAC Secretariat in developing a database containing information on nuclear safety assistance projects to countries of Eastern Europe and of the FSU. The objective of the database was to support the G-24 co-ordination mechanism.

From the very beginning, the IAEA assisted in the development of the database structure, provided information on its own programme activities and put forward suggestions to further improve the database.

The data of the IAEA technical database (2.5.8) are entered twice a year into the G-24 Project Data Bank in order to maintain updated information on IAEA activities.

As a result of the early involvement of the IAEA in the development of the G-24 Project Data Bank, it was possible to develop both the IAEA technical database and the G-24 Project Data Bank in such a way that they could be linked and jointly analysed. This link was achieved through common tables showing the areas covered by the issues or projects.

It thus became possible to run a joint analysis of the tasks conducted to produce indicators of the assistance activity, e.g., the number of projects addressing a particular safety issue or group of safety issues. This type of analysis can support overall or topical reviews but was not meant to be used as an absolute measure of the efficiency of the assistance provided. The joint analysis of the databases was an important tool in the evaluation of gaps and overlaps in the international assistance.

2.5.10. Publications

Some 200 publications have been issued covering safety review missions, safety guidance documents (2.5.5) and reports on generic safety issues (Table 2.2) related to plant systems and features.

Since the early days of the EBP, all documents produced were classified as reference documents (RD) or steering committee (SC) documents. The distribution of these documents was limited to the participants of the EBP but, on request, they were also made available to other interested parties. The reports on missions were restricted to the Member State concerned until the Member State made their comments. After that, reports were generally derestricted and made available to others. Documents which were considered to be of interest to a larger audience were published as IAEA TECDOCs. In 1995, a new EBP publication series was started with more stringent publication requirements. Most reports published in the EBP series were translated into Russian. The Safety Issue Books for the other WWER NPPs (excluding the WWER-440/230), the RBMK NPPs and the guideline documents were published as part of this new series.

All the documents developed were revised first by their authors. Then, they were independently reviewed by IAEA staff not directly involved in the preparation of the report. Finally, the reports were distributed to the SCs for their comments, prior to final publication. This redundant review process ensured the high technical quality of all of the reports published.

2.5.11. IAEA services

Beyond the SRMs carried out within the framework of the EBP to obtain plant specific information, the IAEA, within its regular programme, runs a number of safety services which operate in Member States at their request. In order to obtain pertinent safety related information, the EBP could avail itself of these services, tailoring them to its own needs. Moreover, it analysed in detail the results of past missions, especially ASSET and OSART missions, and used them in drawing up the list of safety issues. The principal safety services of the IAEA, along with a summary explanation of their purposes, are described below.

ASSET

The ASSET service reviews operational safety experience from the standpoint of events which have occurred. This includes investigating and identifying the immediate and root causes of incidents or accidents, the generic safety lessons learned and the appropriateness of corrective actions. The IAEA provides various types of ASSET services, including seminars and full reviews of operating experience.

OSART

The aim of this service is to assist Member States in enhancing the safe operation of nuclear power plants by reviewing operational safety practices and performance at individual plants, identifying strengths and weaknesses in the various programmes and providing advice for their improvement.

Assessment of Safety Culture in Organizations Teams (ASCOT)

In order to promote the ‘safety culture’ concept, the IAEA has developed the ASCOT service. ASCOT missions are intended to review the effectiveness of safety culture in an organization in a Member State on the basis of the principles and recommendations given by the International Nuclear Safety AG and issued in IAEA Safety Series No. 75-INSAG-4.

Engineering Safety Review Service (ESRS)

The ESRS provides advice on selected engineering safety aspects of nuclear power plants in construction or in operation. Areas of work include siting, design, fire safety, and the impact of ageing in nuclear power plant safety. Design includes aspects of reactor core, safety system, component integrity, instrumentation and control, support systems, external hazards and accident analysis.

Seismic safety missions

Seismic safety missions are performed within the framework of the ESRS. The purpose of seismic safety missions is to advise Member States on the safety of NPP sites and in particular on the seismic safety of the NPP structures, equipment and distribution systems. These services include a formal review and workshops aimed primarily at training.

International Peer Review Service (IPERS)

IPERS provides an independent review of the PSA methods applied, the completeness of the PSA, the appropriateness of the data used and the validity of the results.

International Regulatory Review Team (IRRT)

The purpose of the IRRT programme is to provide advice and assistance to Member States to strengthen and enhance the effectiveness of the nuclear regulatory body. The scope and overall purpose of IRRT missions are being further developed in close co-operation with the regulatory bodies of Member States and co-ordinated with the EC activities.

2.5.12. Co-ordination with Technical Co-operation projects

Since its inception, the EBP was implemented in close co-operation with national and regional TC projects. In this regard, the most important was a Regional Project (RER/9/035) which in 1995-1996 provided the means for carrying out safety missions and organizing technical meetings. In 1997 and 1998, three new Regional TC projects were initiated and focused on: support for safety assessment of NPPs (RER/9/046), capability for assessment of operational safety of NPPs (RER/9/047), and nuclear safety regulatory and legislative infrastructure (RER/9/052).

Strengthening the regulatory infrastructure

National assistance

For many years, and complementary to the work of the EBP, the IAEA has been assisting regulatory bodies in the CEEC and the Republics of the FSU through national, regional and interregional projects and training courses.

In the past, regulatory bodies in the countries of Eastern Europe and the FSU were in general underdeveloped in many areas. Moreover, due to the disintegration of the FSU, it was necessary to establish regulatory bodies in several newly independent states operating NPPs (Armenia, Lithuania, Ukraine).

For countries with the most urgent needs in the regulatory area (Armenia, Bulgaria, Lithuania, Slovakia, Ukraine), national technical co-operation projects were launched which had been tailored to the specific needs of each. In this connection, the results of the Model Project for the regulatory body of the Slovak Republic (1993-1998) are a prime example of the success of the IAEA in the area of support of regulatory bodies.

The Slovak Model Project adopted a global approach to strengthening the Nuclear Regulatory Authority (UJD) to the level of good international practice and was well co-ordinated with UJD's own efforts and other assistance projects of a more specific nature.

Forty expert missions were conducted during the period from January 1994 to December 1997. Some of these were single visits on a specific topic; in other cases follow-up visits on priority topics had been arranged to allow further progress to be made. Regular meetings between UJD and the IAEA were also held to evaluate progress and define subsequent stages of the project. Some of the priority topics were: inspection methodology and practice; quality assurance; emergency preparedness; training of regulatory body staff; and information to the public.

The large number of expert missions resulted in a significant number of recommendations and mission reports, many with overlapping recommendations and in some cases conflicting recommendations. UJD endorsed most recommendations, resolved conflicts, and established a comprehensive database in order to track issues to resolution. This database was kept up to date by senior managers of UJD and used as a basis for self-improvement. It was also reviewed on two occasions by an external expert who concluded that a large number of the recommendations had been satisfactorily addressed by UJD.

In addition to the expert missions, a series of well planned fellowships and scientific visits allowed a large number of UJD staff to gain a better understanding of international practices in regulation of nuclear facilities.

UJD have concluded that the Model Project was "well co-ordinated with other international assistance projects and has contributed to the level achieved by UJD in terms of its recognition by the national authorities of the Slovak Republic, the public and the utility, as well as internationally". The IRRT mission completed in March 1998 concluded that, since its formation in January 1993, UJD rapidly developed into a strong and effective regulatory body.

Regional assistance

Within the framework of national and regional TC projects (RER/9/023), training courses and workshops were organized which covered a large spectrum of regulatory activities. These included regulatory control of NPPs, the general approach to safety, information to the public, emergency planning and preparedness, decommissioning and safety culture. To date, more than 200 regulatory staff received training. Since the start of the project in April 1994, recipient countries, donor countries and G-24 NUSAC participated actively in the planning and implementation of the project activities.

Within the framework of the Regional TC project for the strengthening of regulatory bodies (RER/9/023 in 1994-1996, RER/9/052 in 1997-1998) the common needs of the regulatory bodies of the region were addressed. The assistance covered the whole spectrum of regulatory activities: legal and statutory framework, role, responsibilities and organization of the regulatory body, licensing process, regulatory inspection and enforcement, training and licensing of plant personnel, emergency planning and preparedness, and information to the public and media. The assistance was provided by means of training courses, workshops seminars, fellowships and scientific visits. Some of the countries of the region were also provided assistance through TC project RER/0/015, which addresses basic legislation.

The IAEA has been assisting in the re-establishment of the Co-operative Forum of WWER Regulators. This regional group of regulators meets annually and serves as a forum for exchange of experience in the implementation of regulatory policy and safety issues of common interest. Ad-hoc working groups have been established to discuss more detailed technical problems (radiation embrittlement of RPV, in-service inspection, licensing of dry spent fuel facilities).

Some of the countries of the region (e.g., Bulgaria, Romania, Slovakia, Ukraine) invited IRRT missions to perform peer reviews of the activities of their regulatory bodies. The aim of an IRRT mission is to identify good practices and make recommendations and suggestions which lead to a further strengthening of the regulatory body. The results of completed revisions are being used by the recipients to improve their effectiveness and identify where international assistance would be beneficial.

3. RESULTS CONCERNING THE WWER-440/230 NPPs

The WWER-440/230 NPPs were the first commercial pressurized water reactors developed by the USSR from prototypes commissioned at Novovoronezh in late 1963 (Novovoronezh NPP Unit 1) and late 1969 (Novovoronezh NPP Unit 2).

They were standardized 440 MWe twin units referred to as the WWER-440/230 plants. The first WWER-440/230 NPP was Novovoronezh NPP Unit 3 which began power operation in 1971. The list and location of the following WWER-440/230 units is shown in Table 3.1.

The two units at the Armenian site are a variation of the WWER-440/230 plants that contains special features intended to accommodate the effects of seismic events.

At Kozloduy NPP Units 3-4, which were the last WWER-440/230 plants to be built, safety-related systems have a higher degree of redundancy and segregation than the other WWER-440/230 plants.

WWER-440/230 plants were designed before the first Russian nuclear safety regulation (OPB-73) was published.

The WWER-440/230 plants were designed in the 1960s with the objective of producing electric power with high plant availability. This intent is evidenced in design features such as the low heat production with respect to the fuel weight; a reactor protection system that has three levels of response before the full scram; six primary coolant loops containing a hot and a cold leg isolation valve; horizontal steam generator (SG) designs with relatively large water inventories; two turbine generators for each unit permitting operation at reduced power; atmospheric steam dump valves operating with a 10-20 second response time; and the possibility to utilize unit tied systems for neighbouring units (system intermeshing). Plant parameters with large margins between operating conditions and safety limits provide great flexibility to cope with operational transients. The design features also offer a potential for on-line repairs or maintenance facilitated by measures (e.g., the primary coolant purification system) to keep the radiation fields low. Systems can be interconnected with each other and between units. High accessibility of equipment during operation and the attitude of management and personnel of giving priority to keeping the plants in operation favour high plant availability.

The safety concept of the WWER-440/230 NPPs was based on design rules and standards as they existed at the time of the design and construction of these plants in the USSR. The RPV is made of cylindrical forgings of ferritic steel. The primary circuit was manufactured using forged parts with the exception of the bodies of the cast austenitic pump and the main gate valve. The primary piping was made exclusively from austenitic stainless steel. On the basis of this provision, primary circuit failures which would result in severe core damage were not taken into consideration. Therefore, the plant design does not include any special provisions to protect against a large failure of the primary circuit boundary. The safety concept also required that essential primary circuit equipment and its auxiliary systems should have high reliability during their lifetime. Furthermore, it specified that protection against human errors be based, to a considerable extent, on organizational measures aimed primarily at the prevention of accident initiating events. These measures included in-service inspection and control of the quality of the manufacturing and installation of primary circuit equipment.

Therefore, the original bounding LOCA design basis accident (DBA) was the loss of integrity of the primary cooling circuit equivalent to a break of 32 mm diameter. Technical measures were applied only to cope with small leaks. However, the design of the corresponding safety systems did not consider common cause failures.

The primary circuit is located in a confinement system designed to an overpressure of 0.1 MPa which may occur in a DBA. The radioactive steam-air mixture generated must be localized in the enclosures. In addition, the confinement is also equipped with a spray system to reduce the pressure after the accident. In the case of accidents with breaks larger than 32 mm equivalent, the compartment pressure may exceed the design value and the steam-air mixture must be dumped via devices (flaps) into the atmosphere which close after the pressure decreases. This safety concept aims at the possibility of dumping relatively clean steam-air mixtures (with the normal radioactivity of the primary circuit) from the confinement into the atmosphere at the beginning of accidents slightly exceeding the design basis before any significant fuel damage occurs. However, the safety concept includes no provisions to cope with large breaks which could cause severe accidents with core melt or provisions against external hazards (e.g., seismic hazards).

Compared with current practice for most other plant types, the DBA analysis is very limited. Furthermore, there were no comprehensive safety analyses using appropriate tools to demonstrate to what extent the built-in safety margin, qualitatively known to exist, could cope with a wider spectrum of accidents typical of other PWRs.

In 1984, when the first results from operating experience of Loviisa plant showed an unexpected increase of embrittlement in RPV weld no. 4, the general designer OKB Gidropress developed a series of requirements for safety upgrading to allow continued operation of the RPVs.

In July 1989, the regulators of the WWER-440/230 operating countries jointly drew up a position paper with 16 requirements to ensure a safety level which they considered acceptable during the remaining design lifetimes of the units and a requirement for a special operating regime until backfitting was completed. Special operating regimes were introduced by operators, but considerable financial and technical problems had caused little progress generally on the backfitting requirements prior to the first AGM in September 1990.

TABLE 3.1. WWER-440/230 NPPs IN OPERATION

Country	Plant	Unit	Start of Operation
Slovakia	Bohunice NPP Units 1-2 (V-1)	1	1979 (full power 1978)
		2	1981 (full power 1980)
Bulgaria	Kozloduy	1	1974
		2	1975
		3	1981
		4	1982
Russia	Novovoronezh	3	1972
		4	1973
	Kola	1	1974

Country	Plant	Unit	Start of Operation
		2	1975
Armenia	Armenia	2	1980 Shutdown in March 1989 Restarted in November 1995

3.1. IDENTIFICATION AND RANKING OF SAFETY ISSUES

Following a recommendation of the first AGM, a Conceptual Design Review Meeting (CDRM) was organized in February 1991 in Vienna with the participation of the WWER-440/230 designers and operators and Western specialists. The objective of the meeting was to perform a safety evaluation of the original design concept of the WWER-440 model 230 NPPs.

The results of the meeting [58] served as a basis for SRMs, carried out at Bohunice, Kozloduy, Novovoronezh and Kola in 1991. These missions covered areas related both to design and operational safety and combined elements of Design Review and OSART [59, 60, 61, 62].

In parallel, ASSET missions were conducted at these NPPs. These missions provided further insights on the WWER-440/230 operational safety.

The results of the CDRM and of the four SRMs were then compiled and analysed during two project review meetings convened by the IAEA in Vienna, in August and November 1991, respectively. The objective of the meetings was to help the governments of Czechoslovakia, Bulgaria and the FSU to set priorities for the corrective measures required at their plants.

Close to 1300 individual safety items identified during the CDRM and the SRMs were grouped in broader categories representing some 100 issues of safety concern and then analysed. The safety significance of the issues was then evaluated by the international experts who participated in the meetings, applying the safety review approach described in Section 2.4. Issues were ranked according to their safety significance in four categories (Section 2.5.1). The task was greatly facilitated by the fact that the 1300 safety items and associated information had been stored in a computerized database.

Finally, the Safety Issue Book was produced in February 1992 [6]. It contains an overview of WWER-440/230 nuclear safety and a compilation of about 100 generic safety issues (see Annex 4). The report presents for each safety issue its description, the related safety items of the database, its ranking and related justification, and recommendations for safety improvement. The report also provides an overview of safety on the basis of the successive barriers for the confinement of radioactive material and the related main safety functions: controlling the power, cooling the fuel and confining the radioactive material within the barriers.

In addition to the Safety Issue Book, an overview of the major findings of the EBP on the safety of WWER-440/230s was published in May 1992 [2].

Early reviews of the design of WWER-440/230 plants revealed a number of inherent safety features as well as major safety concerns. The latter are summarized below.

The most important safety concern was the integrity of the primary circuit boundary (third barrier) and particularly, the RPV, due to the embrittlement phenomenon. There was incomplete information concerning the manufacturing data, the actual status of the vessels and the rate of RPV embrittlement under neutron flux.

The integrity of the primary circuit piping, part of the third barrier, was also a significant safety concern, because the DBA of the WWER-440/230 plants was a LOCA resulting from a small primary pipe break of a diameter equivalent to only 32 mm. Hence the importance of the implementation of an LBB concept and of efficient in-service inspection techniques.

The integrity of the confinement structures, the 4^{th} and last barrier, was also a safety concern due to the limited size of the design basis LOCA. In addition, confinement structures of the WWER-440/230 NPPs presented an unusually high leak rate (about 5000% per day at 70 kPa).

Other significant issues concerned the provisions to protect the integrity of barriers:

- Safety systems (except emergency feedwater) required to fulfil the main safety functions had limited capabilities because they were designed for a small LOCA.

- Safety systems and their support systems were not sufficiently protected against potential external and internal hazards. In some areas such as I&C or Electrical Power Supply, redundancy was poor. Past operational experience such as the big fires which occurred at Greifswald NPP Unit 1 in 1975, or at Armenia in 1982, had proven the seriousness of the issue, in spite of the fact that insufficiencies in other inherent safety features had been underlined during these accidents.

- Among the support systems, I&C reliability and performance were found to be particularly limited.

Insufficient accident analyses was also a major issue related to WWER-440/230 safety. The lack of any independent peer review to check and approve the SAR had led to a general inadequacy of accident analyses with respect to:

- the limited scope of the accident scenarios studied;

- partial studies of accident sequences only with analysis limited to a very short period of time;

- studies performed with too optimistic assumptions concerning the values of physical parameters, the number of operating components, etc. which did not take into account a single failure after the start of an accident; and

- analyses performed with computer codes which were either not validated or had modelling limitations.

3.2. REVIEW OF SAFETY IMPROVEMENTS

Following recommendations from the AGM and the Steering Committee, the second phase of the EBP started when the countries concerned began establishing safety improvement programmes, taking into account the results and recommendations of the Programme's first phase. The IAEA was then requested to advise on the safety improvements defined by the WWER-440/230 plant operators.

To accomplish this task, the IAEA organized from 1992 to 1994 a series of SRM follow-ups, also called Consultative Missions, to each of the operating WWER-440/230 NPPs. Their objective was to review the improvements performed or planned at each plant to address the safety issues with respect to the IAEA recommendations as described in the Safety Issue Book [6], and to give further advice when necessary. It was generally observed that most of the improvements were short term modifications [63, 65, 66, 67].

During this period, it became evident that it was necessary to compile the requirements and recommendations given by various organizations and the programmes established by the utilities of the WWER-440/230 plants in response to these recommendations. This would allow to clarify the differences between the recommendations provided by national or international organizations, verify that the IAEA recommendations had no significant gaps and compare the status and various improvement programmes of the plants. Such a compilation was initiated by the IAEA staff and reviewed by a group of international experts at a Consultants Meeting (CM) organized by the IAEA in January 1994. The review showed [68] that there were no significant gaps in the IAEA Safety Issue Book [6] and it presented the status of the plants at the beginning of 1994.

Among the requirements compiled during the review were the 81 measures established in 1991 by the Regulatory Body of the former Czechoslovakia relating to the safety upgrading of Bohunice NPP Units V-1. The IAEA missions confirmed the measures and recommended additional ones for implementation.

Two years after the first SRM, Bohunice NPP had established a preliminary version of its major safety upgrading programme addressing the long term safety issues. In April 1993, the Nuclear Regulatory Authority of Slovakia invited the IAEA to review this programme. Following this request, the IAEA invited nine international experts to a CM in Piestany, Slovakia on 12-16 July 1993. They reviewed the programme established by the Slovakian utility and gave advice concerning the various possible options to address safety issues. In particular, the experts gave detailed comments and advice on the definition of the new bounding DBA the plant should be able to cope with, after implementation of the major safety upgrading programme [69].

Following this meeting and the consultative mission to Novovoronezh (June 1993) which reviewed the long term safety improvement programme of this plant, it was felt that considerable progress had been made in the definition and implementation of safety improvements in the time elapsed since the Safety Issue Book had been published. Consequently, there was a need to review the strategies and safety impact of the proposals related to major safety upgrading and to reach consensus on the effectiveness of the safety improvements to reduce overall risk at the plants.

To address this need, the IAEA invited 15 international experts to a CM in Vienna in November 1993. The experts reviewed the main safety issues and provided comments on the solutions presented in the major safety upgrading programmes [70].

After the last SRM follow-up was performed (Kola, June 1994), the results of the first two years of the second phase of the EBP (1992-1994) were compiled and then discussed during a CM organized in Vienna by the IAEA from 26 to 30 September 1994. Representatives from the plants (Bohunice, Kozloduy and Novovoronezh) who attended the meeting updated the information collected on their sites during the last 2 years. In addition to plant representatives, nine consultants attended the meeting. For each of the main safety issues of the WWER-440/230 NPPs presented in [70], the objectives of the meeting were the following:

- to summarize each safety issue and the possible solutions to solve it ; and
- to describe the plant specific situation concerning each issue, including the remedial actions already implemented and remaining safety concerns.

The consultants used the IAEA database containing the information compiled for each issue at each plant. The database was updated and a new version was published [71]. The results of the meeting were published in [72]. They showed that many short term measures had been performed in each WWER-440/230 plant, thus increasing their level of safety. However, major issues were still pending in the framework of future major safety upgrading programmes. Therefore, IAEA assistance to the plants was pursued.

The report [72] also included an overview of the safety of Armenia NPP Unit 2 which was under preparation to be restarted. The overview was based on missions performed by the IAEA from 1992 to 1994 in the framework of TC project ARM/9/002 [64, 73, 74, 75]. Due to the similarities in design and historical background of the Armenia NPP as compared to the other WWER-440/230 plants, the Safety Issue Book [6] was used as a reference for the safety assessment of Armenia NPP Unit 2, starting with the first IAEA mission in 1992.

To provide further assistance to the plants, the IAEA organized technical visits to each of the WWER-440/230 plant sites between 1995 and 1998. The results of these missions [76, 77, 78, 79] were finally compiled and reviewed during a CM organized in Vienna by the IAEA in February 1998. The objective of the meeting was to update the information [72] compiled about three years earlier. Ten international experts including representatives from the Bohunice, Kozloduy, Novovoronezh, Kola and Armenia NPPs reviewed the progress achieved in nuclear safety by each of the operating WWER-440/230 NPPs until early 1998.

The results of the meeting, concerning the status of each plant, are summarized below from the meeting report [80].

3.3. PLANT SPECIFIC STATUS

The following is an overview of plant specific status of each of the WWER-440/230 plants compiled based on information provided by the NPPs.

3.3.1. Bohunice NPP Units V-1

On the most important issue, the RPV integrity, the RPV of both units was annealed and the annealing was supported by a comprehensive material property evaluation programme. Flux reduction measures were implemented and the flux to RVP was measured. Large samples taken from the outer surface of the RPV were examined for chemical and microstructural properties. The results were favourable, in particular concerning the chemical composition and the conservativeness of the assumptions regarding the transition temperature shift. It was concluded that no additional annealing of the RPV of both units will be necessary until after 30 years of operation.

In addition, the vessel degradation process will be monitored in the framework of a programme initiated by the IAEA. To reduce the potential loads on the vessel, fast closing main steam line isolation valves were installed and new interlocks were installed to prevent cold overpressure. The water temperature in the ECCS tanks was increased to 55°C.

Concerning integrity of the primary circuit, a programme to assess the LBB concept applicability has been fully implemented for the 500 mm and 200 mm pipes and the necessary plant modifications were performed. The process was reviewed by the IAEA. Three qualified independent leak detection systems are in place. Based on the LBB analysis results, seismic upgrading was carried out. A P-scan 3 ultrasonic test (UT) was used for ISI of LBB candidate piping. The LBB status was approved by the regulatory authority.

The pressurizer safety valves were replaced on the basis of a new design which takes into account common cause failures and new transients and accidents including ATWS and Feed and Bleed. A new relief valve which reduces the frequency of opening the safety valves and to be used for feed and bleed was also installed. The new design is capable of withstanding the new design basis earthquake and its layout prevents the main sources of common cause failure. The new valves are qualified to operate with steam, water and steam-water mixture.

The steam generator safety valves were replaced. To decrease the frequency of opening the steam generators safety valves and to make possible the decay heat removal through the feed and bleed mode on the secondary side, a new steam dump station to the atmosphere has been installed on each steam line. They are isolable from the corresponding main steam line by a motor driven isolation valve. They are qualified seismically and for steam-water mixture or water discharge.

ISI of main steam and feedwater lines has been improved and a leak detection system based on acoustic emission has been installed in main steam and feedwater lines.

In-core monitoring is provided by thermocouples and the axial neutron flux measurement system VOLNA was replaced by a new system of self powered detectors. An on-line core power distribution calculation system was installed.

The confinement leaktightness was improved by two orders of magnitude compared to the original status (about 50 % instead of about 5000 % per day at 70 kPa overpressure). Proper operability of the flaps was tested and each of the valves was provided with position sensors indicating in the main control room (MCR) the flap lifting. In the framework of the gradual reconstruction, a major confinement upgrade is being prepared including a spray system, based on new requirements from Slovak Regulatory Authority. In particular, the double-ended LOCA break of 500 mm will be taken into account using best estimate assumptions [69, 70].

On safety systems, recommended short term measures [72] were implemented. In addition, during the gradual reconstruction, the additional Emergency Feedwater System (also called “Super Emergency Feedwater System”) was modified to take into account potential common cause failures and to increase its autonomy (now 72 hours and possible connection to fire trucks). This system is now qualified seismically and as a safety grade system. Super emergency feedwater lines were replaced and connected to the main feedwater lines as close as possible to the SGs. Additional connections were installed to the SG blowdown. Concerning the ECCS, design studies were performed to define the future major modifications of the system which will enable it to cope with the new DBA defined by the Slovak Regulatory Authority [81]. The new systems will be installed in 1998 and 1999. New interlocks of HP SI pumps were installed, sumps of ECCS tanks were protected against any risk of clogging, flooding detection in the boron room have been installed and piping connection between the ECCS of the 2 units was installed. Steps to improve the reliability of the ECCS system are being taken.

The support systems were considerably improved as discussed below:

I&C classification and qualification were completed and reviewed according to the requirements of the Slovak Regulatory Authority. New qualified sensors were installed in the reactor protection system (RPS) and emergency safety features actuation system (ESFAS). Within the framework of the major safety upgrading, new fully redundant, segregated, seismic and environmentally qualified RPS and ESFAS were installed and are operated in OPEN LOOP mode. The validation and verification of the new system are ongoing. New reactor protection signals were installed and galvanic isolation of signals between reactor protection system and reactor power controller was implemented. Remote shutdown panels have been installed on both units. The capability of these panels will be increased to fulfil the functions of an emergency control room. The new process computer system has been installed and will be extended on the basis of two redundant channels. An accident monitoring instrumentation panel was also installed in each control room.

- Short term improvements were performed on the service water system. A new system, is to be installed in the framework of the gradual reconstruction.

- The sources of AC power supply of each unit have been reorganized in two independent trains each with an increased (double) output. This has been done by adding a new diesel generator, with twice the power of the generators replaced. In addition, power supply can now be provided via a connection which has been installed between Bohunice NPP Units V-1 and Bohunice NPP Units 3-4 (V-2) plants. A new connection to the nearby hydroplant with 10 MW power and 7-hour water capacity has been installed for every 6 kV emergency bus bar.

The reliability of DGs has been improved by replacement of the startup, control and excitation system. The emergency loading sequencer will be a part of the ESFAS system.

Concerning direct current (DC) and low voltage AC power supply and distribution, the following improvements were performed:

The three existing batteries for two units have been replaced by four new batteries organized in two trains (one battery per train) in each unit. The new batteries have a two hour discharge time and monitoring system capable of detecting galvanic interruption. They are seismically qualified.

New reversible motor generators were installed.

Concerning the reliability of the electrical equipment, most of the equipment having a low reliability, such as the control circuits of safety pumps and the diesel load sequencer, has been improved (e.g., the 6 kV and 0.4 kV breakers) or replaced. During the gradual reconstruction of the plant, the reversible motor generator sets were replaced by battery chargers and rectifiers in one train of each unit.

Ventilation and air cooling systems are being installed to protect the new I&C equipment.

Fire and flooding protection were improved to eliminate the concern about these two main sources of internal hazards.

The most important backfitting measures are:

- an extinguishing system in every safety cabinet;
- new fire detection system;
- cables coated with fire retardant;
- a system to drain the turbine oil tank;
- turbine roof cooling system;
- new fire doors;
- new extinguishing systems; and
- MCR overpressure.

Concerning seismic safety, the seismic design basis is being re-evaluated and a preliminary acceleration value of 0.25 g was calculated to be used in upgrading. There is a local seismic network to monitor the seismic activity around the site. On the basis of the preliminary value of the seismic input, the plant was reassessed and a number of equipment and structural upgrades was implemented.

Accident analysis has been developed in accordance with international practice and has provided the basis for defining and validating each step of the gradual reconstruction programme. A preliminary safety analysis report for the gradual reconstruction was submitted to the Slovak Regulatory Authority. A Level 1 PSA was prepared and is used for gradual reconstruction.

In conclusion, a considerable amount of safety improvements have been performed in all design areas at the Bohunice NPP Units V-1. Most of the remaining safety issues will be addressed when the capability of the plant is increased in order to cope with the new bounding DBA. Such improvement is included in the last part of the gradual reconstruction programme, which has already started. When the gradual reconstruction is completed, all IAEA recommendations for WWER-440/230 type reactors will be fulfilled at the Bohunice NPP Units V-1.

3.3.2. Kozloduy NPP Units 1-4

With respect to RPV integrity, Unit 1 was the subject of highest concern. The results of the extensive programme that was started in 1995 and funded mainly by the EC to evaluate Unit 1 RPV integrity were presented at a workshop organized by the IAEA in Sofia in May

1997 at the request of the Government of Bulgaria. The workshop concluded that the efforts carried out were of the ‘state of the art’ order and addressed in principle all the recommendations of the IAEA expert missions performed in 1994-1995. The results obtained indicated that, with respect to the safety factors of the applicable Russian standards, there was a sufficient margin between the actual material properties of vessel weld no. 4 in terms of critical brittle fracture temperature T_K resulting from the PTS analysis. The sensitivity analysis indicated that there is no immediate need for an urgent action regarding Unit 1 at present.

To protect against main steam line rupture, the worst transient for RPV integrity, fast closing isolation valves were installed on the main steam lines.

The inspection techniques were verified on Unit 1 during the 1996 outage.

In the framework of the long term safety upgrading programme, RPV integrity assessment and residual lifetime evaluation will be completed for each of the four units.

Concerning integrity of the primary circuit piping, the complete study of LBB concept applicability is under way and the necessary modifications are being performed.

One additional pressurizer safety valve, which has the capacity to perform ‘feed and bleed’ cooling, was installed in each unit.

Protection against cold overpressurization in shutdown conditions is ensured by a pressurizer safety valve and an automatic control system.

Confinement leaktightness was improved by about a factor of 10. The Bulgarian safety authorities and the plant are now defining objectives and acceptance criteria for the major upgrading of the confinement function.

The following improvements have been performed in the safety systems:

- For Units 3 and 4, an autonomous qualified Emergency Feedwater System (EFWS) was installed outside the turbine hall. This, in combination with new qualified steam generator safety valves (and qualified fire protection station-2) allows fulfilment of the core cooling function following any initiating event with the exception of LOCA. In addition, a feed and bleed procedure is now available on the primary circuit.

- Measures to improve ECCS protection against common cause failures were implemented, such as: physical separation of high pressure injection pumps, second draining pump to prevent flooding of the boron room, protective grids to prevent plugging of the ECCS sump, continuous monitoring of the containment spray heat exchanger tightness.

Support systems are also being improved:

- I&C equipment reliability is being analysed to check compliance with the single failure criterion and protection against common cause failures. Measures resulting from this review have been implemented. Units 3 and 4 include redundant control rooms. They will also receive a Safety Parameter Display System (SPDS).

- Units 1 and 2 were equipped with a special control panel so that basic reactor parameters can be followed if the main control is unavailable.

- The Service Water System (SWS) was improved to be better protected against common cause failures and to comply with the single failure criterion. Possibility of interconnections between the pairs of twin units has also improved the reliability of the system.

- Ventilation and air cooling systems are being developed to keep I&C and electrical equipment within required conditions. All compartments where radiation releases can occur are kept under negative pressure.

- Electric power supply systems were improved on Units 1 and 2 by complete separation in two trains including diesel generators and a staggered loading system (the original design of Units 3 and 4 is based on three independent trains).

On protection against internal hazards, the following steps have been taken:

- Based on fire risk analyses performed in the various areas of the plant, a wide programme of technical and organizational measures was implemented in order to prevent, detect and extinguish fires.

- Flooding risks were analysed in a systematic way and the corresponding measures are being implemented. During the reconstruction of the 14.7 m elevation of the intermediate building, new solutions for floor hydro-isolation will be applied in order to prevent failures in the electrical or I&C equipment located under this floor.

- To protect against the potential consequences of high energy pipe whip in the area at 14.7 m, the necessary supports are being installed. A methodology is under development for high energy pipe whip analysis in the containment.

In response to the assessment of the seismic risks of the site, the upgrading of safety systems is currently under way. However, structural upgrading work has not yet begun (see Section 3.4.4).

Within the framework of the short term safety improvement programme and the fulfilment of the IAEA recommendations, a broad spectrum of accident analyses is being performed to update and develop the SAR of the plant.

3.3.3. Novovoronezh NPP Units 3-4

Concerning RPV integrity, in the framework of the TACIS programme, templates were taken in 1995 from the vessel beltline (weld and base metal) of each unit. Based on the results of the analysis of these templates, OKB Gidropress performed RPV integrity assessments and concluded that the integrity of the RPV was demonstrated until the end of the design lifetime, i.e., up to 2001 and 2002 for Units 3 and 4, respectively.

To prevent cold overpressure or overcooling of the primary circuit, the following measures are planned:

- Fast closing valves are to be installed on the main steam lines of Unit 4 in 1998. This will be done later on Unit 3.

- Additional check valves are to be installed on main feedwater lines, close to the steam generators, to prevent fast drainage of the SG, following a feedwater line break.

- New pressurizer safety valves will be installed on the pressurizer of Unit 4 at the end of 1998, in the framework of the EBRD programme. This will be done later on Unit 3.

Concerning primary circuit integrity, the applicability of the LBB concept was demonstrated in 1997 under normal operating conditions for pipe diameters of 200 mm and above. In the framework of the EC's TACIS programme, finance was provided to perform studies for extension of the LBB concept to the whole range of conditions including the maximum design earthquake.

Confinement leaktightness has been improved by a factor of 1.5 to 2. For further improvements, fast acting isolation valves are to be installed in 1999 on all ventilation ducts connected to the confinement.

Major safety upgrading of the confinement is still planned, on the basis of a jet condenser, developed by All Russian Research Institute for Nuclear Power Plant Operation (VNIIAES), and a 200 m^3 tank installed in place of the hermetic compartment flap valves. The jet condenser would allow limiting the releases of radioactive material outside the confinement as well as maintaining the integrity of confinement structures in the event of a primary pipe break up to 200 mm in equivalent diameter.

The following improvements of safety systems are implemented:

- The steam dump valves to the atmosphere were changed at Units 3 and 4. The new ones ensure steam and water flow rate control under all conditions.

- Protection of ECCS components against common cause failures (including internal hazards) was improved.

The following improvements of safety systems are planned:

- An additional EFWS is to be installed in 1998 on Units 3 and 4, outside the turbine hall, capable of being connected to various sources of water.

- Unit 4 steam generator safety valves are to be replaced at the end of 1998, in the framework of the EBRD programme.

- A mobile diesel-pump station, to be connected to the EFWS, is planned for 1998-1999, to cope with beyond-the-design-basis accidents.

Support systems are also being improved:

- On I&C, the implementation of improvements is progressing, although slowly. An emergency control panel for post-accident monitoring of the main parameters of Units 3-4 is to be installed in the water treatment control room in 1998. A new information computer was installed on both units and primary sensors are being replaced.

- On the SWS, short term improvements to prevent common cause failures were implemented.

- Ventilation and air cooling were improved in the main control room and in the electrical equipment rooms so that temperature does not exceed 25°C. Smoke will be prevented from entering into the control room in case of fire in adjacent areas.

- In the area of electrical power supply, power cables from diesel generators to 6 kV switchgears were separated into two independent trains. Power cables from 6 kV switchgears to safety related components were replaced by new fire retardant cables with maximum possible separation in two trains. Emergency diesel generator startup time has been reduced to 30 sec. In the framework of the EBRD programme, the existing emergency batteries are being replaced by seismically qualified batteries of increased autonomy (2 hrs).

Pursuant to the major safety upgrading programme, all emergency power supply systems will be improved in each unit on the basis of the 'two separate trains' concept.

In the area of internal hazards:

- several fire protection improvement measures have been implemented, but measures resulting from the fire risk analysis performed remain to be implemented;

- protection of ECCS against flooding in the boron rooms was implemented; and

- pipe whip analysis remains to be implemented.

Concerning seismic safety:

- work is under way to verify the site seismic studies.

In the framework of the TACIS programme, accident analyses are being performed to support the licensing of the major safety upgrading measures.

3.3.4. Kola NPP Units 1-2

Kola NPP Units 1-2 RPVs were annealed based on the analysis of the templates taken in 1995 from the vessels of Novovoronezh NPP Units 3-4, the Kurchatov Institute confirmed that the critical brittle temperature will not exceed the maximum acceptable value at the end of Kola NPP Units 1-2 design lifetime (2003 and 2004, respectively). To reduce the loads on the vessel, fast-closing main steam isolation valves have been installed and ECCS hot leg injection put in place. Further, the boron acid tanks were equipped with a heating system. To reduce the neutron flux on the vessel wall, dummy fuel elements had been installed in 1985 and a low leakage load pattern was adopted in 1992. The pressurizer safety valves were changed four years ago and they are used to prevent cold overpressurization during shutdown periods. During other periods, protection against cold overpressure transients other than steam line breaks is provided only by operator actions based on procedures.

The IAEA has been informed that the LLB concept was implemented at Kola NPP in May 1998. Diagnostic systems are put in operation - vibroacoustic monitoring of reactor internal equipment, leakage diagnostic, monitoring of exterior subjects and modern NDT equipment for components and pipelines in the primary circuit. In the framework of EBRD programme, N16 primary to secondary circuit leakage detection system will be installed in 1998.

Confinement leaktightness has been improved by a factor of 5 to 6 and measures to increase it are still being implemented. Fast acting cut-off valves on confinement ventilation ducts were installed at Unit 2 and are to be installed at Unit 1.

The project involving major confinement improvements based on the jet condenser conducted at Novovoronezh, is also planned at Kola.

In the area of safety systems the following improvements were implemented:

- Improvements to prevent ECCS common cause failures were implemented such as: prevention of flooding in the boron room, prevention of ECCS sump clogging.

- The motors of the main coolant loop isolation valves were changed in 1995 for new motors qualified to operate under LOCA conditions.

The following improvements of safety systems are planned:

- In the framework of the EBRD programme, the steam generator safety valves are to be replaced in 1998 by new ones, qualified to operate in water steam or water flow;

- Installation of a new EFWS, outside the turbine hall, has been delayed (cancelled by EBRD);

- Emergency gas removing system from reactor and SG collectors;

- Emergency drainage system of primary circuit hot leg;

- Computer system replacement with the new system and SVTRK implementation;

- Reactor Protection System and flux monitoring system replacement is implemented at Unit 1, and is carried out at Unit 2 in 1998.

On the support systems, the following measures were implemented:

- The reactor protection system was replaced by a new one, based on the ‘two train concept’ on Unit 1 in 1996. This is to be done on Unit 2 in 1998. Control equipment of the confinement spray system is based on the two separate independent trains concept since 1996 on both units. In both units the initial process computer has been replaced by a more modern unit.

- Protection of the SWS against common cause failures has been improved. In addition, it is now possible to supply Units 1 and 2 from the system of Units 3 and 4.

- The emergency electric power supply systems are now organized in two independent trains. All electric components reaching the end of their life, or of insufficient reliability, have been replaced, namely: the batteries which now have a two-hour autonomy (two batteries on each train), the reversible motor generators, many switch boards (outfitted with 0.4 kV switches). A standby movable emergency diesel engine, qualified for arctic conditions, has been donated by the Norwegian Government in 1996. It can be connected rapidly to any electrical motor of the safety systems.

On the support systems, the following measures are planned:

- In I&C, post-accident monitoring instrumentation will be installed in 1998 in the framework of the EBRD programme.

- An SPDS is to be installed in the control room of Units 1 and 2 in 1998, in the framework of a Norwegian Government safety assistance programme.

- A system of air purification is to be installed in the control rooms in the framework of the EBRD programme.

In the area of internal hazards, a fire risk evaluation was conducted in the framework of TACIS. Some of the resulting measures have been implemented and others are planned; the real extent of implementation is not known.

Concerning accident analysis, many scenarios have been analysed in the framework of TACIS to prepare the major safety upgrading of the plant. Additional analyses will be required once the major safety upgrading programme is finalized.

3.3.5. Armenia NPP Unit 2

Concerning RPV integrity, the position of the plant is based on the conclusions of a meeting organized in Moscow by the Armenian Safety Authorities in July 1995, with participants from the IAEA and from the designer (OKB Gidropress).

The meeting concluded that the vessel of Armenia NPP Unit 2 can be operated until its end of design life (30 years) without any additional protection. However, OKB Gidropress recommended to take a sample on weld number 4 within 5 years in order to analyse its chemical composition. This would provide a verification of the values given by the original passport of the vessel. The plant will take this sample during the summer of 1999 outage in the framework of the TACIS 96 programme. However, the current position of OKB Gidropress is that sampling would not provide enough new information and therefore is no longer recommended.

To reduce the load on the vessel, a low leakage load pattern was adopted. In addition, the following measures are planned:

- fast closing valves, financed by the USDOE, will be installed on the main steam lines during the 1999 outage,

- the steam generator safety valves are to be changed during the outage of 1999, in the framework of the TACIS programme, and

- new pressurizer safety valves, financed by TACIS, are to be installed during the 1999 outage. They will also help solve the issue of cold overpressurization in shutdown periods. Such valves will also be qualified for ‘feed and bleed’ on the primary side.

Concerning integrity of the primary circuit piping, the plant is awaiting financing by TACIS of the studies which will determine the conditions of the applicability of the LBB concept to Armenia NPP Unit 2. When these conditions are known and fulfilled, the use of the existing leak detection system will be optimized and two additional systems based on different principles will be installed.

TACIS financing was requested for a primary to secondary leak detection system using N16 detection to be installed in 1999.

Confinement leaktightness is being improved continuously. A request has been sent to the EC for TACIS financing for a compressor which would greatly facilitate leak detection.

The following improvements are planned in the area of safety systems:

- The technological condenser of the low head EFWS in the boron room will be changed in 1999 in the framework of the TACIS programme.
- For the ultimate scenario of complete loss of power, a diesel pump will be installed in 1999 to feed the steam generators in the framework of the USDOE assistance programme.
- The motors of the main coolant loop isolation valves are to be changed in 1998 for new motors qualified to operate under LOCA conditions.
- To improve protection against common cause failures, the plant is planning to separate into two trains the ECCS and EFWS components located in the boron room. Various design possibilities will be studied in 1998.
- To face a common cause failure of ECCS in case of rupture on an ECCS header, the plant has planned to install check valves on some of the ECCS lines in 1998, in the framework of the TACIS programme.
- The installation of a special grid to prevent ECCS sump clogging is planned, but the date depends on VNIIAES, who are studying the design.

On the support systems, the following measures are planned:

- On I&C, the plant is looking for financing to change the old data processing computer. An SPDS will be installed in 1999, if US credits are available.
- The new seismically qualified Service Water System is still under construction. Equipment is financed by the USDOE. The end of construction is expected for summer 1999.
- In the area of emergency electric power, many components will be changed over the next two years to improve their reliability and performance: control and power cables, 0.4 kV switches, reversible-motor generator sets.

In the area of internal hazards, the following steps have been implemented or are planned:

- In fire protection, the replacement of the vinyl floor cover by a non-combustible floor cover is under way, with financing from the USDOE. USDOE will finance the replacement of the old system of fire detection in the cable ways (1998-1999) and of the fire doors in the cable tunnels and turbine hall with fire resistant doors with a built-in delay of 90 minutes.
- To prevent flooding in the boron room, an additional 80 m^3/h drainage pump has been installed.

- Nijni-Novgorod Atom Energo Project (NNAEP) has been requested by the plant to perform a high energy pipe whip analysis on the steam generator blowdown pipes in 1999, with Russian funding.

Concerning seismic safety, the following programme has been established by the plant (see Section 3.3.4):

- establishment of the list of buildings, structures and equipment necessary for a safe shutdown of the plant following a 0.35 g earthquake, by NNAEP and OKB Gidropress for end of 1998;

- structural dynamics calculations for the 0.35 g earthquake of structures, systems and components important to safety (see above list), by OKB Gidropress and NNAEP in 1998 with Russian financing;

- justification that the reactor can be kept in safe shutdown conditions during at least 72 h following a 0.35 g earthquake and complete loss of power, by OKB Gidropress and NNAEP with Russian financing; and

- following IAEA recommendations, further study planned for 1998-1999 of some aspects of seismic and volcanic hazards to the Armenia NPP site in the framework of the TACIS programme.

Accident analysis is included in some of the above safety upgrading measures. In addition, a list of transients needing further study has been established by the Safety Authorities and the plant.

3.4. SELECTED SAFETY ISSUES

In parallel with the activities earlier described in Sections 3.1 and 3.2, and following recommendations from the AG and the SC, the IAEA EBP has focused on selected safety issues for which assistance to the countries operating WWER-440/230 plants would be of particular interest.

The issues considered are those of high safety significance (ranked IV or III in the Safety Issue Book) and which are not addressed by existing documents in sufficient detail. Issues generic to all WWER plant types are presented in Section 6. Selected issues of high safety concern and specific to WWER-440/230 plants are discussed below.

3.4.1. Reactor pressure vessel integrity

The RPV integrity of WWER-440/230 plants has been recognized as of the highest safety significance due to relatively high impurity concentration (leading in combination with relatively high neutron flux to higher than anticipated embrittlement rate), lack of vessel specific material data, and in general due to deficiencies in the overall RPV integrity assessment. It should be noted that similar problems were also encountered at other old PWR plants throughout the world and led to the implementation of various corrective measures and even plant shutdown in some cases. Corrective measures were also proposed and implemented to some extent at WWER-440/230 plants. In addition to ‘common’ measures such as flux reduction and load reduction through hardware and operational regime modifications, annealing technology to restore material properties was developed in the FSU and applied to a number of WWER-440 plants. It should be noted that to date, annealing has successfully been applied only to WWER-440 nuclear power plants.

IAEA activities

a) Generic activities

In the framework of IAEA EBP, the WWER-440/230 RPV integrity received systematic attention since the Programme started. In order to provide a technical overview of the issue and of the work still needed, the IAEA prepared a status report dealing with this subject in 1992 [82]. The status report presents a comprehensive survey of technical information available to the IAEA at the time of publication and identifies those aspects requiring further investigation.

Further in-depth reviews of the status of this issue at individual plants and of the programmes completed, under way and planned in this area were conducted by the IAEA in the period 1993-1994. The information obtained was summarized in the report WWER-440/230 Reactor Pressure Vessel Embrittlement and Annealing, which also includes recommendations and conclusions on actions still needed [83].

It was recognized that the national, bilateral and international activities including those of the IAEA focused mainly on vessel material behaviour, whereas the issue of the RPV integrity is a much more complex problem. In addition to material behaviour, thermal-hydraulic analysis, structural analysis and fracture assessment have to be considered in the RPV integrity assessment. The assessment needs to be further supported by NDT for in-service inspection. The reliability of NDT, and its influence on the various elements of the analysis (feedback, operations, plant configuration), must be carefully weighed.

In order to shed light in these areas, the IAEA initiated several actions focused on material properties, on PTS analysis and on in-service inspection qualification.

The IAEA in 1995 compiled a consolidated register of aspects involved beyond NDT and consigned them in the report on WWER-440/230 Reactor Pressure Vessel Integrity [84]. Discussions covering all WWER plant types were held during the Workshop on International Practices for RPV Integrity Assessment, organized by the IAEA in 1996 [53].

As it was recognized that the problem was of a generic nature and was applicable to all WWER plant types, the issue was addressed generically basis (see also Section 6.2). The preparation of guidelines for PTS analysis and of methodology for NDT qualification applicable to all WWER plants was initiated. The respective reports on these topics have recently been finalized [20, 22].

The Round-Robin Exercise on WWER-440/230 RPV Weld Metal Irradiation, Embrittlement, Annealing and Re-embrittlement [85] was initiated in 1995 for the purposes of looking into material properties; this activity was subsequently transferred to the IAEA regular budget.

In order to promote the implementation of the Guidelines for PTS analysis [20] in countries operating WWER-440/230 plants, the IAEA initiated the Benchmark Exercise on PTS Analysis for WWER plants. The exercise, which consists of thermohydraulic and material behaviour analysis, will be completed by the end of 1998.

In addition to the above activities, the IAEA supported the development of a computer code to calculate fast neutron fluence at the RPV wall which is coupled with a fuel burnup code and enables to calculate neutron fluence for each fuel cycle. The code is to be used directly by plant staff and has been provided to the Novovoronezh and Kola plants, along with a related training.

b) Plant specific activities

In addition to the generic activities described above, the issue was addressed on a plant specific basis throughout the programme in the framework of expert missions, safety review missions, follow-up consultative missions and technical visits to individual plants [59, 60, 61, 62, 63, 65, 66, 67, 69, 70, 73, 74, 75, 76, 77, 78, 79, 80, 86, 87, 88].

For Kozloduy, several IAEA expert missions were organized at the request of the Government of Bulgaria to review the status of the issue. These missions focused in particular on Kozloduy NPP Unit 1 due to its outstanding concentration of embrittlement promoting impurities. A number of deficiencies, such as integrity assessment based on performance of NDT, differences in the assessment in the safety analysis and actual plant configuration, inadequate operating instructions and limitation of the analyses to the original design basis were identified through these missions [89, 90].

At the invitation of the Government of Bulgaria and considering the concerns expressed by the international community, in October 1995 the IAEA organized a meeting in Sofia to discuss the issue of the RPV integrity of the Kozloduy NPP Unit 1 prior to its restart for the next fuel cycle. The respective Bulgarian organizations considered the concerns expressed by the group of international nuclear safety experts participating in the meeting. The plant was restarted in autumn 1995 under special conditions, which the plant and regulatory authority of Bulgaria had agreed upon. These conditions included the preparation of a programme to verify RPV integrity, which was initiated in 1995 and carried out mainly in 1996 during an extended plant outage. Some activities of the programme were carried out in the framework of contracts directly with the Kozloduy NPP; the major part of the programme was realized through funding provided by the EC.

Based on the results, the Bulgarian organization decided to restart the Kozloduy NPP Unit 1 late in 1996 and invited the IAEA to organize a workshop to present the programme and its results to the international community [91]. The programme addressed in principle all recommendations of the IAEA expert missions [89, 90]. The results presented indicated that at present there is no immediate need for an urgent action regarding the RPV integrity of Kozloduy NPP Unit 1 at present.

Achievements

At all plants concerned, activities to address the issue are under way. The RPVs in which excessive embrittlement had been expected were annealed. At all plants, some steps to improve the knowledge of actual material properties were either taken, are under way or are planned. Templates have been taken from the inner surface of the RPVs of Kozloduy NPP Units 1-2, Novovoronezh NPP Units 3-4 and their mechanical properties have been evaluated. The re-irradiation of spare material is under way. Bulk samples have been taken from the outer surface of the RPV of the Bohunice NPP Units V-1 and used to verify chemical composition. Flux reduction measures and hardware modifications were also conducted. The integrity assessments were or are being re-evaluated.

Outlook

The implementation of the various measures developed is, however, plant or even unit specific and is described in detail in the reports of the various missions referred to (see Section 6.2). Further actions still required are discussed in Sections 6.2 and 9.

3.4.2. Primary piping integrity

The original design concept of the WWER-440/230 plant assured no loss of primary circuit integrity resulting in significant loss of coolant inventory. This was assumed to be achieved by a conservative design, the use of austenitic stainless steel for primary circuit piping, and the observance of high quality standards in manufacturing and assembly.

As a result, the ECCS is only able to cope with breaks of a limited scope and the confinement system is not designed for large diameter breaks. A large pipe break would therefore result in the loss of two main safety functions: cooling the fuel and confining the radioactive material.

The application of the LBB concept to the large diameter primary piping of WWER-440/230 plants has been recognized as a tool to restore some features of the original safety concept from the current point of view on maintaining primary circuit integrity. The application of the LBB concept to WWER-440/230 plants was initiated in 1998 in the former Czechoslovakia for Bohunice NPP Units V-1 in 1998 in order to address the above concerns, to verify the piping design and to provide a basis for seismic backfitting.

IAEA activities

a) Generic activities

From the outset, the IAEA Extrabudgetary Programme systematically turned its attention to application of the LBB concept. The IAEA first prepared a status report on the applicability of the LBB concept which provides an overview of the approaches adopted on this issue in countries operating nuclear power plants [92], in particular in Germany, Japan, USA, Russia and the former Czechoslovakia. This report draws conclusions on the role of the LBB concept in the safety reassessment of older plants, plant backfitting, the importance of leak detection system and in-service inspection. Additionally, the report integrates the concept into the plant’s operation (training, procedures, maintenance, etc.).

Subsequently, the IAEA prepared a report on guidance for the application of the LBB concept [21], described in further detail in Section 6.3.

b) Plant specific activities

The Guidance [21] was used by countries operating WWER-440/230 plants when applying the LBB concept as well as during the IAEA reviews of the LBB concept application at Bohunice NPP Units V-1 [93], and Kozloduy NPP Units1-4 [94, 95].

Achievements

The abovementioned IAEA assistance to Member States provided a basis for the development of the approaches to apply the LBB concept in the countries operating WWER-440/230 plants and for the reviews of the safety improvements achieved.

At all the plants concerned, activities to address the issue are under way or have been completed. The ISI approaches have been reviewed and modified as necessary. The application of the LBB concept was been completed fully or to a large extent at the Bohunice and Kozloduy plants and resulted in a number of plant modifications. At all units, acoustic emission leak detection systems were installed.

Recently, the IAEA has been informed that steps to apply the LBB concept at the Kola and Novovoronezh plants have been completed (see Section 6.3).

Outlook

Considering the multidisciplinary nature of the LBB concept application, a thorough review at individual plants would strengthen the confidence that the objectives of the efforts are being achieved.

3.4.3. Confinement

In the WWER-440/230 Safety Issue Book, the issues related to radioactive material confinement in case of LOCA were ranked as of high safety concern. The leaktightness of the hermetic zone was initially poor, corresponding to leak rates of several 1000% per day at design pressure, which makes impossible the effective prevention of radioactive releases in the event of an accident. Hence, the EBP experts agreed to designate the structure under consideration as a confinement boundary (confinement) rather than a containment.

The WWER-440/230 confinement has a relatively small volume (13,000 m^3) and low design pressure (0.2 MPa absolute) since the DBA is limited to a break equivalent to an orifice of 32 mm in diameter. For this size of break, the spray system was designed to limit the pressure buildup and to prevent uncontrolled releases to the environment.

To preserve the confinement structure in the case of larger breaks, until full rupture of one of the two pressurizer surge lines (200 mm each), large flaps in the confinement boundary were designed to open during the initial pressure peak after the accident and to release into the environment steam escaping from the broken reactor coolant system. Releases are limited by reclosing the flaps once the pressure has abated and subsequently turning on the confinement spray which condenses the steam; this causes the pressure to drop to a sub-atmospheric level for a short period of time, the length of which varies depending on the leaktightness of the confinement.

A break of the large pipe of a diameter of 500 mm, with double-ended blowdown, is estimated to result in an overpressure of 2 bar (compared to the 1 bar design value), even with all vent valves (flaps) functioning. The ability of the confinement to withstand this pressure is not known.

IAEA activities

a) Generic activities

In July 1993, the CM organized by the IAEA in Piestany at the request of the Slovak Nuclear Regulatory Authority and charged with reviewing the Major Safety Upgrading Programme of the Bohunice NPP Units V-1, devoted significant time to the improvement of the confinement. Among the recommendations given by the experts [69], the following were of particular interest:

- A LOCA with a break size equivalent to 200 mm should be considered when defining the DBA for the purpose of designing the upgraded confinement.

- The most important improvement in a confinement with a poor leaktightness can be achieved by maintaining the confinement at sub-atmospheric pressure as soon as possible after initiation of an accident for an extended period of time.

In November 1993, another group of international experts was invited to Vienna by the IAEA to review the major improvements for WWER-440/230 NPPs. The experts discussed extensively the DBA that the plant should cope with after the implementation of a major safety upgrading. Their recommendations on the confinement upgrading confirmed and completed those formulated in Piestany. In addition to suggesting the 200 mm double-ended break as the new bounding LOCA DBA, they recommended that the 500 mm pipe break be also considered, using realistic assumptions and best-estimate methodology [70].

In December 1993, the IAEA invited 15 international experts to a CM in Vienna which focused on containment and confinement performance in NPPs with WWER-440/213 and WWER-440/230 reactors [96]. Concerning the determination of the DBA for the improvement of WWER-440/230 confinements, the experts agreed with the conclusions of the prior two meetings. They developed these conclusions, complementing them with further details and justifications. In particular, they recommended that the LBB concept be applied to demonstrate the low probability of a large diameter piping break. The experts also reviewed and assessed seven pressure suppression devices with a condensing function and three proposals for filtered confinement venting. Finally, the experts gave recommendations on improvements for confinement leaktightness, for the confinement spray system and for confinement conditions under severe accident scenarios.

In July 1995, the same experts were invited by the IAEA to a CM in Vienna on confinement improvement options for NPPs with WWER-440/230 reactors. The purpose of the meeting was to review the actual status of the confinement improvement options and to evaluate their technical feasibility and possibility of timely implementation in the existing WWER-440/230 units. The group provided detailed recommendations on all aspects of WWER-440/230 confinement safety issues [81] including:

- improvement of the confinement leaktightness;

- improvement of confinement pressure relief valves;

- post-accident instrumentation;

- determination of the DBA (or upgrade basis accident) for the improvement of a WWER-440/230 confinement (consistent with the previous recommendations given in 1993);

- prevention of confinement failure due to pressure increase after an upgrade basis accident; and

- safety systems proposed for the Bohunice NPP Units V-1, Kozloduy and Kola NPPs.

Finally, the experts analysed the technical feasibility of major confinement improvements.

The problem of confinement leak rate tests needed additional guidance. The important factor in reducing leakages is the ability to detect existing leakages. This involves integral leak rate tests for the whole confinement at possible high overpressures, local leakage tests and methods for the interpretation of results. With the improvement of leaktightness, the leak rate testing pressure can be increased, which makes it possible to reveal new locations of leaks and eventually have them repaired. The methods used in the countries operating WWER reactors for confinement leak rate testing are based on old Russian regulations, on Western rules such as US ANSI or German KTA and on original work conducted individually by each country. At the request of the Slovak Nuclear Regulatory Authority, the IAEA organized in May 1995 in Vienna a CM on the subject within the framework of the EBP. The meeting fulfilled the following objectives [97]:

- to present the methods of leak rate measurements used in each country operating WWER NPPs; and

- to compare the advantages and deficiencies of each method and make proposals to further improve these methods.

b) Plant specific activities

IAEA activities in this area were essentially generic. However, they included review and assessment of proposals for specific plant confinement improvements.

Achievements

Concerning the confinement leaktightness improvements, the results achieved at Bohunice are impressive and have demonstrated that significant improvements are possible in this area.

Outlook

Major upgrading programmes exist for each plant in the area of confinement; however, all of them include new technological systems or equipment without the benefit of sufficient testing or accumulated experience. The cost of such improvements and the need for additional testing leave some doubts about the possibility of a timely implementation.

3.4.4. Seismic safety

The WWER-440/230 type plants, the oldest of their generation, generally lacked seismic consideration in their original design. The 1977 Vrancea earthquake, which caused some concern as to its effect on the Kozloduy NPP, was the reason behind the initiation of an upgrading programme there and at the Armenia NPP.

The seismic re-evaluation programme for an NPP generally involves the re-assessment of the seismic hazard, of the plant seismic capacity and, if required, the design and implementation of upgrades to components and structures. In general, terms of reference are needed for the implementation of a seismic upgrading programme.

IAEA Activities

Seismic safety review services were conducted at the Kozloduy NPP Units 1-4, Bohunice NPP Units V-1 and at the Armenia NPP. The breakdown of these services (reviews, follow up missions and workshops) at three plants is presented below for the period 1990-1997 in Table 3.2.

TABLE 3.2. SEVEN-YEAR SUMMARY OF IAEA SITE/SEISMIC SAFETY REVIEW SERVICES TO EASTERN EUROPEAN NPPs

		Number of services (1990-96)			
Country	Plant	W	S	SI	SC
Armenia	Armenia	3	-	5	5
Bulgaria	Kozloduy 1-4	1	2	5	5
Slovakia	Bohunice V-1	1	-	-	3

W: workshop, work plans and technical procedures
S: site safety review
SI: review of seismic input and tectonic stability
SC: review of seismic capacity

Terms of Reference (or Technical Guidelines) were prepared for the seismic re-evaluation and upgrading of these three NPPs. Plant managements and the regulatory authorities are following this guidance in the seismic upgrading programme. The following table shows the seismic safety status of these three plants.

TABLE 3.3. SEISMIC SAFETY STATUS OF SELECTED WWER NPPs IN EASTERN EUROPE

Plant	Original SDB	Reassessed SDB (RLE)	Capacity Check	Upgrades Easy fixes	Structural
Kozloduy 440	NED	0.2 g	Neg.	Yes	No
Bohunice V-1	NED	0.25 g?	Neg.	Some	Some
Armenia	0.1 g/0.2 g	0.35 g	No	Some	No

SDB: seismic design basis
NED: no explicit design
Neg.: mark indicates an ongoing activity with a preliminary indication of the reassessed SDB (RLE)
No: the activity has not started yet

The IAEA has been informed by the Slovak Regulatory Authority that for Bohunice Units V-1 all upgrades (easy fixes and structural) have been completed.

Major issues which have been identified in relation to WWER-440/230 NPPs can be summarized as follows:

1. The original seismic input value has been found to be much lower than what the site seismotectonic conditions require using IAEA current guidance (i.e., NUSS Safety Guide 50-SG-S1, Rev. 1). Proximity to suspected capable faults has complicated the problem further and caused the increase in the seismic input value. In one case (Armenia NPP), potential hazard from volcanoes had to be evaluated.

2. Seismic categorization of the plant was such that the turbine hall contained equipment belonging to the safe shutdown list. This meant that this structure would need to be seismically upgraded as well, a significant addition to the scope of upgrading.

3. Safety related structures (e.g., reactor building, diesel generator building, pump house) are generally designed as ordinary industrial structures with large spans of precast concrete frames having little lateral resistance and consequently poor performance under seismic loads. The part which is resistant (to some degree) to extreme loads of the reactor building is the ultimate pressure boundary (i.e., the confinement) which is built to surround the primary circuit. This results in a situation where the safety related items outside the confinement are vulnerable to all external loads and in particular those resulting from earthquakes.

4. The lack of documentation for the original design as well as any subsequent design changes has greatly delayed the understanding of the plant layout and the items to be considered within the scope of the safe shutdown list. Plant walkdowns have played a very important role to overcome this problem.

5. Although most of the safety related components and equipment can be considered to be seismically rugged, the way in which they are supported and anchored was found to be inadequate and generally of poor quality. A wide range of safety related components are in this category, e.g., diesel generator peripherals and support systems, electrical cabinets, control panels, batteries.

Achievements and Outlook

- Derivation of the new seismic input value and ground motion characteristics. Although the achievement in this area is very significant, this activity is still under way for the Bohunice NPP site. For the Kozloduy and Armenia NPP sites the response spectra have been established for re-evaluation and upgrading. However, the task related to the assessment of the so called “local earthquake”, i.e., the effects from a near field event, remains to be reviewed.

- Preparation of Terms of Reference (or Technical Guidelines) for the seismic re-evaluation and upgrading. This has been done for all three plants. However, the seismic input which was taken as the basis for Bohunice NPP was 0.25g tied to a site specific response spectrum. Following more detailed seismotectonic investigations, it appears that this will be modified. Consequently the Technical Guidelines will need to be changed accordingly. The first Terms of Reference (or Technical guidelines) document was prepared for the Kozloduy NPP Units 1-4 in 1993. The NPP management feels that there have been developments which need to be taken into consideration in a new version of this document. This revision is expected within the next two years.

- Easy fixes. This is a programme of upgrading which involves adding anchorage and support to components and equipment as well as electrical cabinets and control panels. It also encompasses the bracing of unreinforced masonry walls and other potential interaction problems. For the most part these fixes have been carried out in the three nuclear power plants, although the degree of completion and the quality of workmanship show some variation from one plant to the other.

- Structural upgrades. This is obviously the most important but also the most costly and time consuming effort for the nuclear power plant. Even before the IAEA involvement, the Bohunice NPP had undergone some structural upgrading, mainly involving cross-bracing and lateral supports to increase the stiffness of the frame structure of the reactor building. The effectiveness of these measures has been questioned by IAEA review missions. There is a need to re-evaluate these upgrades when the seismic input parameters are finalized for the Bohunice NPP site. The Armenia NPP was once upgraded (from the original design value of 0.1g to 0.2g) after the 1977 Vrancea earthquake, which at that time caused some rethinking as to the seismic safety of WWER type NPPs in the former Soviet Union. It is now necessary to check the “as-is” plant in relation to the site specific response spectrum anchored to 0.35g and implement further upgrades where necessary. For the Kozloduy NPP the structural upgrades have been designed in part, but the actual construction work has yet to start.

- Seismic qualification. Eventually, the equipment and components in these plants need to be demonstrated to be capable of withstanding the newly defined design basis seismic loads. Most of the time this requires a well defined laboratory testing programme. The tests which were made earlier by the Soviet laboratories considered much lower design basis values and adequate information on these tests is often lacking. Qualification on the basis of earthquake experience data which have been used extensively in the United States is also impractical for these plants because of the different types of equipment not included in the US database. One particular issue is relay chatter and consequently the seismic qualification of relays used in these NPPs.

- There has been some discussion in creating some kind of a WWER database (similar to the US-based seismic qualification utility group (SQUG) database. However, this has not yet materialized.

4. RESULTS CONCERNING THE WWER-440/213 NPPs

The WWER-440/213 NPPs were put into commercial operation between 1977 (Loviisa NPP Unit 1) and 1987 (Dukovany NPP Unit 4, Paks NPP Unit 4). The list and locations of the 16 operating units are shown in Table 4.1.

When the WWER-440/213 plants were designed, industrial norms and standards were no longer considered sufficient and standards specific to the nuclear industry started to be developed and implemented such as OPB-73 and PBYa-74. OPB-73 marked the beginning of a transition to the generally accepted international practice in nuclear safety (e.g., defence in depth, single failure criterion, etc.).

As a consequence, significant safety improvements were introduced in the design of this second generation of WWER type 440 NPPs. The characteristics and the layout of the main components remained the same. The WWER-440/213 NPPs were built with modules of two units, in a single reactor building, each unit with six primary coolant loops, isolation valves in each loop, horizontal steam generators and two 220 MWe steam turbines.

As concerns safety, the bounding LOCA DBA was defined as a double-ended guillotine break in the primary circuit piping. Redundancy of safety systems and their support systems was also increased, compared to the situation in the WWER-440/230 plants. The ECCS was designed in accordance with the new bounding LOCA DBA and covers the entire range of primary piping break sizes. It includes high head and low head pumps as well as accumulator tanks connected directly to the reactor vessel. The confinement function is fulfilled by a pressure-suppression containment with bubbler condenser. Other differences between WWER-440/230 plants and WWER-440/213 plants are detailed in an IAEA technical document [98].

However, due to the lack of specific nuclear safety standards in certain areas at the time the WWER-440/213 units were designed, some deficiencies still remained in the design of the WWER-440/213 plants, in particular, in areas such as: protection against potential common cause failures, high energy pipe breaks, fires or earthquakes, component classification and qualification, bubbler condenser integrity under DBA conditions.

Operational experience and detailed safety studies conducted at the request of the German Government (BMU) revealed many of the safety deficiencies of this type of reactors and steps to address these issues were taken by the countries in which WWER-440/213 plants are in operation.

TABLE 4.1. WWER-440/213 REACTORS IN OPERATION

Country	Plant	Unit	Start of Operation
Russia	Kola	3	1981
		4	1984
Ukraine	Rovno	1	1980
		2	1981
Hungary	Paks	1	1983
		2	1984
		3	1986

Country	Plant	Unit	Start of Operation
		4	1987
Slovakia	Bohunice V-2	1	1984
		2	1985
Czech Republic	Dukovany	1	1985
		2	1986
		3	1986
		4	1987
Finland	Loviisa*	1	1977
		2	1981

*With Siemens I&C and Westinghouse ice condenser containment

4.1. IDENTIFICATION AND RANKING OF SAFETY ISSUES

In 1993, following the recommendation of the AGM, the IAEA initiated a systematic compilation and ranking of safety issues of WWER-440/213 reactors. Preliminary fact finding missions, such as those performed to WWER-440/230 units, were not necessary due to the amount of information already available on the safety of WWER-440/213 NPPs.

A report listing safety issues and related proposed backfitting measures identified by the WWER-440/213 owners group was completed under contract with the IAEA in the middle of 1993 [99]. The content of this report was later presented at a WANO conference in Moscow in March 1994 [1]. The 33 safety upgrading measures covered by this report can be grouped under the following topics:

- protection of the RPV integrity;
- increase of the reliability of the Emergency Feedwater System;
- improved handling of large primary to secondary leakages;
- improvement of ECCS and other equipment to address LOCAs;
- improvement of the containment;
- fire protection;
- equipment for severe accidents;
- I&C reconstruction; and
- increase of seismic resistance (if needed).

Some of the upgradings above required additional accident analysis.

During the same period, some of the major findings of TC project RER/9/004 (initiated in 1991, following a request from the former Czechoslovakia) concerning the design basis and design features of WWER-440/213 reactors were summarized within [98] and also served as an important input to further work on safety issue ranking.

In April 1994, an IAEA CM was organized in Vienna on Backfittings and Safety Enhancement Measures in NPPs with WWER-440/213 Reactors. The objective of the meeting, which was attended by representatives from all of the operating WWER-440/213 plants, was to review and analyse safety issues based on the available information compiled by the IAEA Secretariat. This was done on the basis of comparison with national and international standards, and with practices used in currently operating Western plants designed in the 1970s [100].

Using the same methodology as that developed for the WWER-440/230 safety assessments (see Section 2.4), the experts identified the main safety issues. For each of them the actual status and the reasons for concern were described and a judgement on safety ranking made. In addition, measures proposed or already implemented to solve the issue in each of the WWER-440/213 NPPs were presented. Finally, comments and recommendations were given.

In response to a request from the Slovak Nuclear Regulatory Authority, a mission including nine international experts was conducted by the IAEA in May 1994 at the Mochovce NPP, where four WWER-440/213 units were under construction. The objective of the mission was to discuss the safety issues known to exist in WWER-440/213 reactors, the safety improvements already incorporated in the Mochovce design and the improvements proposed in the Safety Improvement Report (SIR) prepared by experts from EdF and Siemens in co-operation with Slovak organizations. The main safety issues remaining in Mochovce and highlighted by the mission report are the following [101]:

- Protection of ECCS against common cause failures should be improved, especially concerning the possibility of ECCS sump clogging by debris.

- Protection of EFWS against common modes should also be improved. EFWS lines should be rerouted outside the turbine hall.

- Protection against internal hazards should be improved, especially in the following areas:

  - fire protection in the turbine hall and in the reactor coolant pump rooms; and

  - protection against high energy pipe whip at the 14.7 m level in the intermediate building; however, before defining a design solution, a comprehensive analysis of the impact of such an event on safety equipment in the vicinity (pipes, valves, cables, I&C components) was to be performed. This pipe whip is also one of the potential sources of common cause failure of the EFWS (see above);

- Lifting-up of the primary collector cover in a steam generator was also considered an important case of LOCA for which the containment is bypassed. An accident of this type actually happened in Rovno in 1982.

- The bubbler condenser strength has to be re-evaluated for the case of the largest LOCA.

- Seismic input should be evaluated to determine the available margins.

A safety improvement programme was prepared for the Mochovce NPP based on the results of the safety review mission, the Safety Issue Book [7] and the Risk Audit Report No. 16. In October 1998, the IAEA will carry out a mission to advise the Slovak Regulatory Body on the Mochovce Safety Improvement Programme.

Following another request from Slovakia, the IAEA in September 1994 invited six international experts to a Design Safety Review Mission at Bohunice NPP V-2. This was the first comprehensive design review conducted by the IAEA in an operating WWER-440/213 plant. The objective of the mission was to review the plant design in the light of current international safety practices and to make recommendations to assist the plant management and safety authorities in taking decisions on how to achieve a higher level of safety. At the time of the mission, the experts observed that the situation concerning the RPV embrittlement was quite favourable (chemical composition of vessel materials, low neutron leakage schemes), which, together with a comprehensive vessel surveillance programme, represented a significant improvement over the WWER-440/230 plant situation. In addition, the plant had already made significant efforts to replace defective components (e.g., the DC power supply) and was preparing a replacement programme for I&C in order to increase its reliability. On the remaining safety issues, the experts made comments and recommendations which can be summarized as follows [102]:

- Concerning protection against potential common cause failures, it was recommended to improve physical separation of redundant safety equipment and to eliminate connection between safety and non-safety related equipment.

- Among the potential sources of common cause failures, fire hazard was still a significant issue deserving further study and improvements.

- Another source of common cause failures, earthquakes, was also to be considered, to make sure that safety related components are adequately protected.

- More generally, due to the lack of appropriate analysis and documentation related to the qualification of safety equipment and components with respect to seismic events or to the adverse environmental plant conditions after an accident, there was no assurance that the equipment and components would be capable of carrying out their intended functions.

- Upgrading of primary circuit in-service inspection programme was also among the main recommendations.

The IAEA has been informed that a new safety analysis report has been submitted to the regulatory authority as a result of periodic safety review after 10 years of Bohunice NPP V-2 operation. Based on the regulatory authority decision, the results of the IAEA Mission [102], the Safety Issue Book [7] and the Mochovce NPP safety improvement programme, a comprehensive safety improvement programme was prepared for Bohunice NPP V-2. All safety issues were addressed and implementation is under way.

On the basis of the results of the safety assessment missions to Mochovce in May 1994 and Bohunice NPP V-2 in September 1994, the results from TC project RER/9/004, the studies performed on specific issues such as I&C or the bubbler condenser performance and reports published by other organizations, the first draft of the Safety Issue Book established in April 1994 [100] was further improved and completed during two CMs on the Ranking of Safety Issues for WWER-440 model 213 NPPs organized by the IAEA in Vienna in October 1994 and in February-March 1995.

Representatives of Regulatory Authorities of all countries operating WWER-440 NPPs model 213 reactors participated in the last meeting. A draft report [103] was then issued and sent for review to all NPPs with 213-type reactors and to the concerned Safety Authorities. The contributions from all these sources were considered in the preparation of the final report, which also reflects the position taken by Western experts participating in the meetings.

The Safety Issue Book [7] was published in April 1996. Its objective was to present a consolidated list of generic safety concerns, called safety issues, ranked according to their safety significance and the corresponding corrective measures proposed to resolve them and improve safety. The Safety Issue Book was intended for use as a reference to facilitate the development of plant specific safety improvement programmes and to serve as a basis for reviewing their implementation.

The conclusions of the WWER-440/213 Safety Issue Book can be summarized as follows:

There are no category IV issues in the WWER-440/213 NPPs, which confirms that WWER-440/213 safety has been improved in all areas of major concern, compared to the WWER-440/230 plants.

The issues of high safety concern are the following category III issues:

- Insufficient qualification of equipment for anticipated ambient and seismic conditions for DBAs.

- Bubbler condenser behaviour at maximum pressure difference (double-ended guillotine break of the primary piping) is also of high safety concern. The regulations pertaining to strength calculations in force at the time of the bubbler condenser design did not correspond to Western practice and have been changed in Russia itself. New calculations from IAEA programmes show that the strength of some structural elements of the bubbler condenser is questionable.

- Non-destructive testing for reactor coolant system in the framework of in-service inspection (ISI) presents deficiencies and deviations from current standards. The ISI approach used so far is not adequate for a timely detection of degradation. As reliable ISI is a key element required to preserve the integrity of the third barrier (primary circuit boundary), this issue is also considered of high safety concern,

- In the area of systems, some issues already existing in the WWER-440/230 NPPs still remain in the WWER-440/213 NPPs. Two of them are of high safety concern:

  – The risk of ECCS sump screen blocking appeared higher than expected after an incident which happened in 1992 at a Swedish nuclear power plant. This could result in a common cause failure of the whole ECCS following a large break LOCA.

  – The layout of the EFWS, which is located in the turbine hall, is such that it might be exposed to common cause failure by fire, flooding, high energy pipe break or earthquake when it is needed to cool the core.

- In the area of internal hazards, two issues are considered of high safety concern:

  – Fire protection, which is all the more important as redundant safety related equipment is insufficiently separated in some areas such as the EFWS in the turbine hall, and power cables (or control cables) of redundant safety related components follow the same route or are located in the same compartments.

– High energy pipe breaks in the intermediate building at 14.7 m could result in multiple failures of safety related systems and, in some cases, to the loss of EFWS when it is needed.

- In the area of external hazards, seismic safety is also considered of high safety concern since the original seismic design basis is generally not in accordance with current international practice (see Section 4.3.2).

4.2. REVIEW OF SAFETY IMPROVEMENTS

Safety improvements of WWER-440/213 NPPs had already been started when the IAEA EBP was extended to this reactor type. The WWER-440/213 owners' group had established a list of safety issues and comprehensive safety improvement programmes were being developed such as those at Bohunice NPP Units 3-4 and at Paks NPP. Therefore, the compilation and ranking of WWER-440/213 plant issues included from the beginning the safety improvement measures which were proposed to resolve the issues.

The IAEA missions to review safety improvements at Mochovce and Bohunice NPPs were carried out before the final version of the Safety Issue Book was completed in 1995.

Later on two other missions conducted by the IAEA at the Dukovany NPP and at Paks NPP used the Safety Issue Book as a guidance for the IAEA reviewers.

Dukovany NPP Units 1-4 (October 1995)

At the Dukovany NPP, the utility, ČEZ, Inc., decided to establish a modernization programme, since the design did not entirely comply with current international safety practice. The safety goal of this programme was to achieve a safety level compatible with current international practice and to make it possible to extend the lifetime of the plant to 40 years, i.e., 10 years beyond the original design lifetime.

The plant started working on the programme at the end of 1993. Then the programme was audited by the external organization ENAC in the framework of a PHARE project. Finally, the IAEA was requested by the plant to review the programme in the framework of the EBP and of TC project RER/9/035 (WWER-SC-160).

A safety improvement review mission was organized by the IAEA with ten experts (including five IAEA staff members) in October 1995. The Dukovany NPP also invited experts from organizations which had contributed to the programme to participate in the review on the plant side. On the basis of the Safety Issue Book, the plant prepared a special report entitled 'Plant Specific Status at the Dukovany NPP', supported by a large amount of technical documents.

Both design and operational safety improvement measures were reviewed, using the Safety Issue Book as a reference basis. The main conclusions of the mission were the following:

- Each of the 74 safety issues identified by the IAEA in the design area of the WWER-440/213 reactors have been addressed by specific safety improvement measures.

- Concerning qualification of equipment, a programme addressing the issue was started in May 1995 with a deadline for completion at the end of 1997.

- The design safety issue related to the strengthening of the bubbler condenser structure should be significantly reconsidered to correspond to the intent of the IAEA recommendations.

- The issues in the area of component integrity are especially well addressed and the amount of work already done is very significant.

- In the area of systems, significant improvements are planned, such as:

  - the qualification of pressurizer and steam generator safety valves for steam and water flow;

  - the change in the layout of the emergency feedwater system so as to remove it from the turbine hall and fully protect it against both internal and external hazards;

  - the partial change of the existing insulation of the primary circuit in order to prevent ECCS sump clogging.

However, it was recommended that an additional analysis be performed on the service water system to detect possible sources of common cause failures (due for instance to internal or external hazards).

- In the area of I&C, further work was recommended on human engineering of control rooms and on full separation between the main and the emergency control rooms.

- The issues related to internal hazards (fire, flooding, pipe whip) have been addressed and the achievements of the plant concerning fire hazards are already significant.

- Accident analysis is being raised to the current level of international practice.

Paks NPP Units 1-4 (November 1996)

One year later, at the request of Paks NPP, an IAEA Safety Review Mission was conducted to review the safety upgrading programme of Paks Units 1-4 in November 1996. The mission, which included seven external experts and four IAEA staff members, was organized in the framework of the EBP and of TC project RER/9/035 (WWER-SC-197).

The original design of Paks NPP was in accordance with the Russian and Hungarian regulations in force at the time of the construction of Paks, but did not entirely comply with current safety standards. Since the startup of Paks 1 (1982), the knowledge about safety assessment of nuclear plants had been, worldwide, significantly improved. This knowledge had been well supplemented by the results obtained in the Hungarian research institutes.

In order to benefit from this progress, the Hungarian Atomic Energy Commission (HAEC) decided that a safety reassessment of Paks NPP was necessary and launched the Advanced General and New Evaluation of Safety (AGNES) Project in February 1992. Considered as a preparation for an up-to-date safety report, the AGNES Project included accident analyses of DBAs accidents and of severe accidents, complemented by probabilistic analyses. The resulting safety enhancement measures and their priorities were to be defined in the project report. AGNES was also closely linked to a decision of the HAEC taken in 1993 and prescribing the periodic renewal of the license of all Hungarian nuclear facilities.

The Paks NPP safety upgrading programme was developed on the basis of the results of the AGNES Project published in October 1984, also taking into account the operating experience of WWER-440/213 reactors and the exchange of experience with other users of WWER-440/213 NPPs.

The IAEA mission reviewed the Paks safety upgrading programme, using the Safety Issue Book as a reference. The main conclusions reached by the mission were the following:

- All the generic safety issues identified by the IAEA in the design area of the WWER-440/213 reactors have been addressed by the safety upgrading programme. In addition to these generic issues, the plant has also identified several plant specific issues.

- Concerning safety classification, it was recommended to reduce the number of systems included in safety classes, so that only systems necessary to fulfil a given safety function are included. This will reduce the extent of equipment qualification, which is a long and costly process.

- The issue of bubbler condenser mechanical strength has been addressed. Simplified calculations have confirmed the existence of weak points. Three-dimensional calculations are under way and may confirm the necessity of hardware improvements. The plant continuously improves the programme for periodical inspections, local leak rate tests of components of the hermetic boundary as well as of compartments. Equipment for leakage detection and measurement methods have been improved. In Unit 2, a set of 16 hydrogen recombiners has already been installed. Similar sets of recombiners are to be installed in the other units.

- In the area of component integrity, the RPV integrity assessment (including PTS analyses) has been performed in a systematic way; the follow-up actions are now to be performed. Planned non-destructive testing qualification is also very important and should be implemented.

- In the area of systems, many safety improvement measures have been implemented or are planned:

  - new sump strainers have been installed in order to prevent ECCS sump clogging;

  - the EFWS has been transferred from the turbine hall to the reactor building in order to protect it against common cause failures;

  - the replacement of pressurizer and steam generator safety valves by new valves qualified for two-phase water-steam flow and for seismic events is planned by 2001;

  - a N16 detection system has been installed to detect primary to secondary circuit leakages; and

  - emergency electric power will be improved within the next three years, due to several modifications such as replacement of the batteries by seismically qualified ones, reduction of the start up time for the diesel generators, etc.

  - In the area of I&C, the modernization of the reactor protection system is under way. Further work was recommended on human engineering of control rooms.

- In the area of internal hazards much work has been done:
  - concerning fire hazards, improvements have already been introduced, but a systematic fire hazard analysis was recommended by the mission and is also required by the Regulatory Body to be completed by the end of 1997
  - analyses of other internal hazards such as flooding or heavy load drops have been performed and analysis of high energy pipe break is under way
  - analyses of turbine missiles were to be started.
- In the area of external hazards, a large programme of seismic qualification of safety-related systems was under way, to be completed by 2002 (see Section 4.3.2).
- In accident analysis, the amount of work done by the plant (including the achievements in the framework of the AGNES Project) is substantial and of high quality. A few scenarios have to be investigated further, such as boron dilution in shutdown modes.

Rovno NPP Units 1-2 and Kola NPP Units 3-4

Safety improvement programmes have been established for Rovno NPP Units 1-2 and Kola NPP Units 3-4.

4.3. SELECTED SAFETY ISSUES

In addition to safety issues which are generic to all WWERs and are discussed in Section 6 of this report, the EBP focused on two issues of high safety concern specific to the WWER-440/213 NPPs: the strength of the bubbler condenser structure and seismic safety.

4.3.1. The strength of the bubbler condenser structure

The containment structure of the WWER-440 model 213 is composed of localization compartments surrounding selected primary system components (steam generators, inlet and outlet piping, pumps, isolation valves and the major portion of the reactor vessel) and having the same basic configuration, with an increased energy retention capacity, as that of the WWER-440 model 230. The major difference between the WWER-440/230 and WWER-440/213 is that the steam generator compartment of the WWER-440/213 plant is connected to a bubbler condenser tower. This additional building is connected to the reactor building by a rectangular tunnel.

The bubbler condenser containment system of the WWER-440/213 units was designed to prevent the escape of steam and fission products and to facilitate steam condensation, thereby reducing the pressure following the break of any single primary system pipeline, including the double-ended rupture of a 500 mm inner diameter pipe. The utilization of a bubbler condenser to condense steam and limit the maximum overpressure inside the containment is conceptually equivalent to the designs of PWR containments with ice condensers or BWR containments with pressure suppression pools.

The concept of bubbler condenser containment was based on the results of small scale experiments dealing with thermal and hydraulic phenomena which confirmed the thermal effectiveness of the condenser. Further thermal-hydraulic tests are planned in large scale facilities. However, these tests are meaningful only to the extent that the mechanical integrity of the structure is maintained under accident conditions. If the bubbler condenser fails at the start of an accident, then all the thermal-hydraulic analyses – on which the design of the bubbler condenser is predicated – become invalid. In this sense, the main issue for the confinement system of the WWER-440/213 is the strength of the structure of the bubbler condenser.

IAEA activities

a) Generic activities

A few years before the start of the EBP, analyses of the mechanical strength of the bubbler condenser structure were performed in Poland for the Zarnowiec NPP under an IAEA research contract as part of a broader re-assessment study of this plant’s containment system. The results of these studies confirmed large safety margins inherent to the thermal-hydraulic concept of bubbler condenser, but the evaluation of the mechanical strength under large break conditions (LBLOCA) revealed a number of weaknesses in the actual mechanical design of the bubbler condenser structure.

The Polish study was peer reviewed by the Ukrainian Energoproject Institute in the framework of the IAEA TC project RER/9/004. The review by Ukrainian specialists showed that in many cases the Polish approach was too conservative, but that the bubbler condenser structure was indeed too weak and had to be strengthened.

Additional input into this work was provided by a CM organized in the framework of the EBP by the IAEA in Vienna from 29 November to 3 December 1993 on Containment and Confinement Performance of NPPs with WWER-440/213 and WWER-440/230 Reactors. Based on the original experimental work performed by the designer of the bubbler condenser and made available within the framework of TC project RER/9/004 and on the numerous publications issued thereafter on this subject, the IAEA Secretariat had prepared a list of specific questions to the experts in general, and to the Russian designers in particular, so as to obtain answers to many concerns expressed over the years by various specialists. The meeting, which was attended by 15 external experts and four IAEA staff specialists, provided answers to these questions, which covered all aspects of the bubbler condenser design. The general conclusions of the meeting [96], based on available experimental and analytical evidence, were that the thermal-hydraulic behaviour of the bubbler condenser under DBA conditions is satisfactory to assure radiological safety, but the question of mechanical strength of the bubbler condenser structure continues to be open and requires further investigation.

It was also revealed that a more systematic approach, similar to that used currently in Western countries in the design of PWR and BWR containments, is needed in establishing the design basis for evaluation of the bubbler condenser performance.

In order to support further work in this direction, the IAEA started the preparation of guidelines for the evaluation of the metallic structure of the bubbler condenser. The CM of March 1994 on Evaluation Guidelines for Bubbler Condenser Metallic Structure in the Containment of WWER-440/213 NPPs prepared the first draft of the Guidelines [23]. The available information on bubbler condenser strength was initially evaluated and compiled by the IAEA staff in 1994. The compilation was based on the work performed in Poland and on the results reached within the EBP, and presented in IAEA-TECDOC-803.

An IAEA contract to evaluate the mechanical behaviour of bubbler condenser structure was subsequently given to the Russian specialists from the Design Office which had originally designed the bubbler condenser. The contract was completed in 1995 and the calculations identified several elements of the bubbler condenser which could be strengthened to assure the safety margins in the event of an LBLOCA. This work was subsequently used as a reference for plant specific reviews of bubbler condenser structure integrity within the framework of EBP expert missions.

As the problems with the bubbler condenser containment were not limited to the bubbler condenser structure behaviour, but included also the lack of a systematic demonstration of bubbler condenser containment ability to withstand accidents in the NPP, the work on IAEA guidelines in the area was continued during a CM organized by the IAEA in Vienna from 4 to 8 December 1995, a final document dealing with the metallic structure and the bubbler condenser containment in its entirety was reviewed and agreed on by the representatives of nuclear regulatory authorities in the countries operating WWER-440/213 NPPs [24].

b) Plant specific activities

The results of the generic studies described above were made available to all interested plants and organizations. At the request of nuclear regulatory authorities and plant operators, two expert missions were undertaken to carry out the following tasks:

- Review of bubbler condenser structural integrity in Mochovce and Bohunice NPPs [11] which was performed in November 1995.

- Review of bubbler condenser structural integrity in Dukovany NPP performed by an expert mission to the plant in May 1996 [12].

- A discussion of bubbler condenser integrity problems was also conducted during the Safety Improvement Review mission to Paks NPP in 1997.

As the documents prepared by the IAEA showed a lack of demonstrated safety margins in the strength behaviour of the bubbler condenser structure, some countries operating WWER-440/213 plants undertook advanced studies of the issue. In the Czech Republic the analyses were done using three-dimensional calculations involving plastic deformation theory. In Slovakia, an experimental stand was built and the mechanical integrity of the bubbler condenser structure was proved for transient loads equivalent to and even higher than those anticipated during accident conditions.

An experimental thermohydraulic qualification test programme of the Bubbler Condenser was planned in the OECD/NEA group. A separate PHARE/TACIS project funded by the EC is under way to verify strength characteristics under accident loads using Paks NPP as the reference plant. The results will be available by 2000.

Achievements

The IAEA-sponsored studies showed that the existing documents were insufficient to demonstrate the mechanical strength of the bubbler condenser structure under large break LOCA conditions. Walkdowns in the plants and documentation analyses confirmed the existence of weak points in the structure. Therefore intensive repair work was undertaken, both in individual countries and sponsored by international organizations.

Outlook

So far no general answer is available, but the results reached in individual plants suggest that the plastic deformations of the bubbler condenser structure will not lead to a loss of the bubbler condenser function even under LBLOCA conditions. Further work is under way.

The exact extent of the mechanical upgrading should be determined for each plant considering the actual status of the bubbler condenser in a given plant and the requirements of the national regulatory authorities. The loads acting on the partition walls inside the containment should also be checked, because the consequences of damages to these walls could lead to the failures of various safety or safety related systems in the plant [7].

4.3.2. Seismic safety

Seismic loads were not considered in the original design of WWER-440/213 type plants (except 0.06 g for the Mochovce NPP).

The seismic re-evaluation programme for an NPP generally involves the re-assessment of the seismic hazard, plant seismic capacity and if required, design and implementation of upgrades to components and structures. In general, a Terms of Reference document is needed for the implementation of a seismic upgrading programme.

IAEA Activities

Seismic safety review services were conducted at Paks NPP, Bohunice NPP V-2 and Mochovce NPP. The breakdown of these services (reviews, follow-up missions and workshops) for these three plants is presented in Table 4.2 for the period 1990-1997.

TABLE 4.2. SEVEN-YEAR OVERVIEW OF IAEA SITE/SEISMIC SAFETY REVIEW SERVICES TO EASTERN EUROPEAN NPPs

		Number of services (1990-96)			
Country	Plant	W	S	SI	SC
Hungary	Paks	-	-	7	5
Slovakia	Bohunice V-2	1	-	2	-
Slovakia	Mochovce	1	-	2	2

W: Workshop, work plans and technical procedures.
S: Site safety review.
SI: Review of seismic input and tectonic stability.
SC: Review of seismic capacity.

Terms of Reference (or Technical Guidelines) for the seismic re-evaluation and upgrading of all these three NPPs were prepared and both the plant management and the regulatory authority are following this guidance in the seismic upgrading programme. Table 4.3 shows the seismic safety status of these three plants.

TABLE 4.3. SEISMIC SAFETY STATUS OF SELECTED WWER NPPs IN EASTERN EUROPE

Plant	Original SDB	Reassessed SDB (RLE)	Capacity check	Upgrades	
				Easy fixes	Structural
Bohunice V-2	NED	0.25 g	Neg.	Some	No
Mochovce	0.06 g	0.1 g?	No	No	No
Paks	NED	0.25 g	Neg.	Yes	No

SDB: Seismic design basis.
NED: No explicit design.
Neg.: Inadequate seismic capacity for the reassessed SDB (RLE).
?: A question mark indicates an ongoing activity with a preliminary indication of the reassessed SDB (RLE).
No: The activity has not started yet.

The IAEA has been informed by the Slovak Regulatory Authority that the value of 0.1 g has been approved by them for Mochovce NPP RLE, the capacity check indicated the need for upgrades and both easy fixes and structural upgrades were implemented. It was also reported that for Bohunice V-2 easy fixes were completed and some structural upgrades were made.

The major issues identified for these plants are similar to those of WWER-440/230 type NPPs with some differences mainly due to site characteristics:

1. The original seismic input value has changed significantly both for Bohunice NPP V-2 and for the Paks NPPs. For Mochovce, the original design value was 0.06g and the revised value is indicated to be 0.1g although the work leading to this figure has not yet been reviewed by the IAEA.

2. The bubbler condenser poses a problem for the seismic categorization of these NPPs. There is still some question related to the structural integrity of this building under LOCA. Any combination of the LOCA loads with seismic loads has not been considered and it has not been decided that such a consideration is required.

3. The structural systems of these NPPs are similar to those of the WWER-440/230, and hence the issues are similar.

4. The issue of documentation is similar, although during the Co-ordinated Research Programme (CRP), ‘Benchmark Study for the Seismic Analysis and Testing of WWER type NPPs’ a great deal of work was performed to alleviate this problem. The Paks NPP was one of the prototypes for this CRP.

5. The issue of supports and anchorages is similar to that encountered in WWER-440/230 type NPPs.

Achievements and Outlook

- Derivation of the new seismic input value and ground motion characteristics. For Paks NPP this issue has been completely resolved. The new seismic input value has been established as 0.25g tied to a site specific response spectrum. For Bohunice NPP the situation was already summarized for the V-1 Units; the same is also true for the V-2 Units.

- For Mochovce, the new value is reported to be 0.1g (the minimum recommended by IAEA Safety Guide 50-SG-S1, Rev. 1); however, a review of this value is still pending.

- Preparation of Terms of Reference (or Technical Guidelines) for the seismic re-evaluation and upgrading. This has been done for all three plants.

However, the seismic input value will be revised for the Bohunice NPP and needs to be reviewed for Mochovce NPP.

- Easy fixes have been implemented to varying degrees of completion and workmanship quality; they are being considered for the Mochovce NPP.

- Structural upgrades. There is no structural upgrading programme for the Mochovce NPP. For both the Bohunice NPP Units 3-4 and the Paks NPP, structural upgrades have not yet been completely implemented.

- Seismic qualification is a common issue for all WWER type NPPs because most of the equipment and components for these NPPs has been manufactured and tested in facilities using similar codes and standards.

  Results of the CRP on Benchmark study for the seismic analysis and testing of WWER type NPPs. This CRP covered two types of WWER plants (WWER-440/213 and WWER-1000). The Paks NPP and Kozloduy Units 5-6 NPP had been the prototypes for these models. Many generic and specific tasks were performed under this CRP (by 25 international organizations) which have been and will be used by the plants. Some of these are listed below:

  – Safe shutdown systems identification/classification
  – Original design regulations, acceptance criteria, loading combinations
  – Comparative study of seismic design standards
  – Walkdown of Paks NPP
  – Dynamic analysis of Paks NPP structures
  – Full scale blast testing of Paks NPP
  – Shaking table testing of selected components from Paks NPP
  – On-site testing of components at Paks NPP
  – Compilation of previous component test data
  – Compilation of earthquake experience data
  – Paks NPP feedwater line analysis
  – Analysis of buried pipelines for Paks NPP
  – Comparison of 3-D and beam models for Paks NPP
  – Comparison of blast and vibrator tests for Paks NPP
  – Model shaking table tests of the ECCS tank in Paks NPP.

5. RESULTS CONCERNING THE WWER-1000 NPPs

The third generation of Soviet designed light water reactors, the 1000 MW(e) WWER-1000 nuclear power plants, are closer to Western PWR plants with respect to design philosophy, design features, and construction than the earlier generations. The WWER-440/213 and WWER-1000 plants are significantly improved in comparison with WWER-440/230 plants in terms of safety features.

However, in general, the thermal inertia and the safety margins of the WWER-1000 plants are smaller as compared with those of the WWER-440 plants, which results in higher safety requirements for components, and systems and operation.

WWER-1000 NPPs come in four different models. The designs of earlier models 187, 302 and 338 were started in 1972 and were completed in 1979. The design standard used was the OPB-73. These early models have historically been called the ‘small series’ because only five units have been constructed: Novovoronezh NPP Unit 5 (model 187), South Ukraine NPP Units 1 (model 302) and 2 (model 338) and Kalinin NPP Units 1 and 2 (model 338). The major design weakness of the ‘small series’ WWER-1000 plants is the lack of physical separation and functional isolation in safety systems and their support systems.

Because OPB-82 was in preparation prior to 1982, some of its new safety requirements have been reflected in the design of the WWER-1000 ‘standard series’ model 320. The concept of defence in depth is realized by general design criteria including the use of redundancy, diversity, independence and single failure criterion for safety systems. There are 15 units of the WWER-1000 model 320 in operation, 2 in Bulgaria , 4 in Russia and 9 in Ukraine, and an additional 7 units under construction in Russia (1), in the Ukraine (4), and in the Czech Republic (2). Table 5.1 depicts the WWER-1000 NPPs in operation.

TABLE 5.1. WWER-1000 REACTORS IN OPERATION

Country	Plant	Unit/model	Start of operation
Bulgaria	Kozloduy	5/320	1987
		6/320	1991
Russia	Balakovo	1/320	1986
		2/320	1988
		3/320	1989
		4/320	1993
	Kalinin	1/338	1986
		2/338	1987
	Novovoronezh	5/187	1981
Ukraine	Khmelnitsky	1/320	1987
	Rovno	3/320	1986
	South Ukraine	1/302	1982
		2/338	1985

Country	Plant	Unit/model	Start of operation
		3/320	1989
	Zaporozhe	1/320	1984
		2/320	1985
		3/320	1986
		4/320	1987
		5/320	1989
		6/320	1995

Operational experience has revealed some deficiencies regarding implementation of engineering design solutions, quality of manufacture and reliability of equipment used, and the consequential need for safety improvement. Other shortcomings reflect deviations from current safety standards which evolved over the last two decades since the original design of WWER-1000 plants.

Three main activities in the EBP were directed at WWER-1000 plants.

First, the EBP identified safety issues for standard model 320 WWER-1000 and ‘small series’ WWER-1000 plants, ranking them according to their safety significance (Section 5.1). The full list of generic safety issues for WWER-1000 plants is attached in Annex 4.

Second, at the request of Member States, the IAEA reviewed the safety aspects of modernization programmes developed in these States focusing on the issues identified and the recommendations made to resolve them (Section 5.2).

Third, the EBP dealt in depth with topical issues to contribute to the understanding of the safety concern and to provide international experience for their resolution. Section 5.3 is devoted to topical issues which are applicable to WWER-1000 plants only. Those which were found to be primarily important for WWER-1000 reactors, such as primary to secondary leaks (PRISE) and ATWS, were later made extensive in their relevance to all WWERs and therefore described in Section 6.

5.1. IDENTIFICATION AND RANKING OF SAFETY ISSUES

The IAEA review of the WWER-1000 reactors began in June 1992 with a CM in Vienna to compile information on safety aspects and studies carried out on the design differences between WWER-1000 plants and similar Western plants, and on safety reviews already performed for this reactor type. The results of this CM provided a technical basis at the IAEA to define, together with the Member States concerned, the scope and priority activities on the safety of WWER-1000 plants in the framework of EB Programme.

A major element of the IAEA Extrabudgetary Programme, as requested by Member States operating and constructing WWER-1000 plants, was the development of a comprehensive and internationally agreed list of safety concerns (safety issues) generic to all units of this reactor type to be used as a reference for plant specific improvements. Therefore, the activities on the safety of WWER-1000 NPPs were initially directed at identifying important safety issues, establishing their ranking based on safety significance and providing recommendations for resolving the safety issues. A CM held by the IAEA in December 1993 in Vienna clarified the safety concerns underlying the proposed measures and their improvements for activities already started in the Member States concerned [104]. These activities were complemented by the results of plant specific reviews carried out within the IAEA Programme. An SRM to Zaporozhe (Ukraine) in 1994 and various ASSET, OSART and Seismic Safety Review Missions in all Member States concerned laid the groundwork for the Safety Issue Book for WWER-1000/320 plants. Safety studies for Stendal [105] and Rovno [106] were also considered in developing the Safety Issue Book.

The results of IAEA topical meetings on generic safety issues of broader interest contributed further to a deeper understanding of the safety issues and their resolution by means of international experience. The first topical meeting on the safety of WWER-1000 reactors was convened by the IAEA in April 1994 to review safety aspects of the reactor core design such as control and protection system (CPS) design, limitation system design, use of in-core detectors for protection purposes, ATWS and beyond DBA requirements, and to make short and long term recommendations, considering Western practices [107]. Lessons learned from Three Mile Island accident were explained to Member States in order to strengthen defence in depth at plants by restoring 'critical safety functions' through the implementation of symptom-based emergency operating procedures (EOPs) supported by Safety Parameter Display Systems. Member States operating WWER-1000 plants were also assisted in optimizing their control of power distribution including xenon control to render the main safety function more reliable in normal operation and anticipated operational occurrences. For this purpose a specific CM on core control and protection strategy of WWER-1000 reactors was convened in February 1995 in Vienna [108]. Other topical meetings between 1993 and 1995 dealt with the control rod insertion issue (Section 5.3.1) and the issue of steam generator integrity (Section 5.3.2).

CMs were convened in October 1994 and February/March 1995 to agree on a consolidated list of safety issues, their ranking and on recommendations to assist Member States concerned in their national activities to improve the safety of WWER-1000/320 plants. The Safety Issue Book for WWER-1000/320 NPPs [9] was published in March 1996.

It was agreed at the AGM in December 1995 to consider the safety aspects of the 'small series' WWER-1000 units separate from those of the 'standard series' WWER-1000/320 units. Therefore a Safety Review Mission to the South Ukraine NPP was conducted to identify major deficiencies in design and operation of models 302 and 338 [109]. A Technical Visit was performed at VNIIAES and Novovoronezh NPP Unit 5 to identify safety issues with respect to design and operational features of the WWER-1000 model 187 [110]. The results of both IAEA missions were used to develop the Safety Issue Book for 'small series' WWER-1000 units [111] on a CM in September 1997 in Vienna which was completed in 1998. Nearly all generic safety issues identified for the standard model 320 of WWER-1000 reactors are applicable to the 'small series' WWER-1000 units. The clarification of some issues required a reformulation; some specific 'small series' issues were identified in the areas systems, I&C, containment and internal hazards mainly based on less restrictive requirements and applications of safety principles in earlier designs. The information on the scope and status of implementation of compensatory and corrective measures for the 'small series' units at South Ukraine and Novovoronezh have been reflected as plant specific in the respective Safety Issue Book [111].

All IAEA missions to plants and meetings in Vienna involved experts from plants, design organizations and regulatory bodies in Bulgaria, Czech Republic, Russia and Ukraine as well as Western experts to ensure the broad exchange of international experiences and practices.

The combined impact of major concerns is presented below based on the impact on the main safety functions. This overview is included in more detail in the Safety Issue Books for the ‘standard series’ [9] and ‘small series’ WWER-1000 NPPs [111] to assist the Member States in developing an integrated action plan for the implementation of corrective measures.

Controlling the power

Structural deformation of fuel assemblies affecting the reliable insertion of control rods and leading to water gap change between the fuel assemblies was identified as the most important safety issue which impacts controlling the power. Both effects were seen to as to reduce the safety margins under normal operating conditions due to higher local power density caused by water gaps and under transient and accident conditions because of delayed or reduced insertion of shutdown reactivity. The impact on safety caused by larger fuel assembly bow was urgently recommended for in-depth analysis and implementation of compensatory and interim measures (Section 5.3.1).

Another challenge to the performance of the main safety functions is related to the potential of different types of boron dilutions identified for WWER-1000 reactors. The direct service water cooling of the ECCS heat exchanger of the low pressure injection system (there is no closed loop intermediate cooling system except for model 338 units) makes their integrity more vulnerable than they were “cooled” by clean water. Boron dilution can occur if the heat exchanger integrity is lost and service water enters the safety injection system. If the primary circuit is depressurized under LOCA conditions a certain amount of unborated water can slowly or suddenly be introduced into the primary circuit, depending on the operating conditions of the low pressure injection system. There is also a potential of so-called “inherent” boron dilution in small break LOCAs (SBLOCAs), ATWS (Section 6.10) or PRISE (Section 6.5) events. Scenarios with rapid reactivity increase as well as those controlling power in low power and shutdown conditions (Section 6.1) have not been systematically analysed so far.

Cooling the fuel

The steam generators play a central role in cooling the core in operating and transient conditions, so that their integrity and feedwater supply must be ensured under all conditions. The integrity of SGs for core cooling at WWER reactors can be affected by supplying cold emergency water to the degraded SG collectors (Section 5.3.2) in case off-site power is lost since 320 model units (except Temelin NPP) do not have diesel backed auxiliary feedwater pumps. Further, feedwater supply to the SGs might be reduced in case the improperly protected emergency feedwater lines are affected by dynamic effects. The auxiliary feedwater system is also used for emergency situations at Novovoronezh NPP (‘small series’ WWER-1000/187 unit) because there is no emergency feedwater system.

Decay heat removal at cold shutdown state to the ultimate heat sink is carried out by means of the residual heat removal system. The heat exchangers of the residual heat removal system are cooled directly by the essential service water system.

Decay heat removal in case of LOCA might be affected for ‘small series’ WWER-1000 units, where all three redundant trains were found located in one room under the containment floor without physical separation (Section 5.3.3).

Two other concerns have been identified with respect to fuel cooling under LOCA conditions at all units of WWER-1000 NPPs. A major primary to secondary leak (Section 6.5) due to steam generator collector failure would quickly overfill the steam generator and the main steam line which has not been demonstrated to be qualified for hot water load. There is a potential of two scenarios which might lead to a bypass of the containment and losing the long term cooling due to loss of primary water to the environment. Either the steam line collapses before the fast acting isolation valves outside the containment or the BRU-A valve, not qualified for water flow, may fail to reclose and cannot be isolated for all units of model 320. Insufficient EOPs to cope with these scenarios would augment the risk of core damage and result in radioactive release to the environment. These beyond DBA scenarios and their consequences need to be analysed.

Other safety issues identified with the potential to aggravate decay heat removal under LOCA conditions are related to the potential of clogging the filters of the sump screens and /or the heat exchangers by insulation material, preventing cooling the fuel in the recirculation phase. Moreover, in the recirculation phase one sump entails the risk of losing an unacceptable amount of coolant from circulating or bypassing the containment if there is a passive failure either in the sump itself or in any of the three suction lines between the sump and the containment isolation valve. However, it is recognized that neither OPB-88/97 nor NUSS require the application of the single failure criterion for passive systems. For the 'small series' WWER-1000 units there are three sumps with one suction line each.

Another potential source for losing primary water and bypassing the containment are the damaged heat exchangers of the low pressure injection system. There could be an ingress of primary coolant into the essential service water system. A primary leak of the main circulation pump (MCP) seal could happen if the seal integrity in case of insufficient cooling cannot be demonstrated. Russian specialists have informed on the experimental justification of seal integrity by permissible leakages in short operating time; for long operating time the justification is being demonstrated. Provisions to remove residual heat in beyond DBA situations were found to be insufficient with respect to lack of RPV level indication, lack of accident monitoring instrumentation, and deficit of power supply to manage emergency situations. Moreover, the design of control rooms has been found deficient with respect to human engineering of MCR and ECR to support effectively the management of operators in emergency situations.

Confining the radioactive material

The reduction of the potential for the release of radioactive material during normal operation was found to be in accordance with the national release requirements. No shortcomings have been identified with respect to the fuel matrix and the integrity of fuel cladding as well as the provisions in design and operation to protect them within the design basis. However, the integrity of the third barrier revealed some deficiencies concerning both the susceptibility of components to damage and the means to protect the primary circuit integrity. Though some measures have been taken to prevent the risk of cold overpressurization, the PTS protection needs to be systematically improved to ensure that primary pressure is always below the permissible pressure for each value of the primary temperature during cold shutdown.

The RPV integrity of WWER-1000 plants has been recognized of high safety concern due to relatively high Ni concentration in the beltline area welds, which may lead to higher than anticipated embrittlement rate and yield the code prediction formulas invalid, deficiencies in the surveillance programme design (capsules located on the top of the core barrel in outlet water and a non-uniform and low neutron flux field), and in general due to deficiencies of the overall RPV integrity assessment. While the RPV integrity does not present an immediate safety concern at present, it may evolve into one towards the end of the plant life. In this connection it should be pointed out that the feasibility has not been established to use annealing for the WWER-1000 vessels due to the vessel geometry (small distance between core weld and nozzles) and material susceptibility to temper embrittlement.

The integrity of WWER-1000 steam generators is an important issue of high safety significance and further discussed for treatment in Section 5.3.2. The issue of in-service inspection is common to all WWERs and therefore discussed in Section 6.

Based on the results of IAEA missions and meetings to analyse the combined impact of issues on plant safety, the potential to bypass the containment has been found in several accident scenarios. All of them are caused by previously described deficiencies of the SG and system design. The IAEA has emphasized the need to systematically analyse LOCAs within the SAR with the potential to bypass the containment and bypass scenarios such as:

- A major primary to secondary leak caused by an SG collector break (Section 5.3.2) has the potential to bypass the containment by releasing the radioactive inventory of the primary coolant to the environment in the short term, if the BRU-A relief valve fails to close or the steam line cannot withstand hot water load.

- A similar situation may arise in the initial phase of a LOCA with damaged heat exchangers of the low pressure injection system. Then, radioactivity from the primary coolant will be discharged to the essential service water system, thus bypassing the containment.

- In the recirculation phase of a LOCA the coolant can be lost if there is a passive single failure either in the sump or in any of the three suction lines between the sump and the containment isolation valve. Such a failure would also lead to a containment bypass.

- A rupture of the heat exchanger of the closed autonomous circuit for the main circulation pumps could lead to a two phase flow discharge from the autonomous circuit to the intermediate closed cooling circuit, which is not designed for this pressure. Consequently, a rupture of the intermediate closed cooling circuit outside the containment cannot be excluded.

The IAEA, within its EBP activities, has also addressed the availability of provisions in design and operation to control severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents. The scenarios which may affect the containment integrity under severe accident conditions are related to rapid pressure increase due to hydrogen burning or long term overpressurization due to loss of the containment heat removal.

5.2. REVIEW OF SAFETY IMPROVEMENTS

At the request of Member States, the IAEA has provided assistance to review safety improvement programmes and modernization programmes between 1995 and 1997.

The purpose of these IAEA activities was to review the safety aspects of programmes to modernize WWER-1000 plants in operation and under construction. The Safety Issue Book for ‘standard series’ units of WWER-1000 reactors [9] was used as the main basis for the review activities. Advice was given on the completeness and adequacy of the safety improvements with respect to the respective IAEA recommendations. The plant specific reviews were performed by teams of seven to eight international experts selected by the IAEA both from its own staff and from countries involved in the assessment of WWER-1000 reactors.

IAEA activities

Review of plant specific modernization programmes for plants in operation:

Kozloduy NPP Units 5 and 6 (June/July 1995)

For Kozloduy NPP Units 5 and 6 [112] the individual measures addressed in the modernization programmes vary and most descriptions were found to be insufficient for an in-depth technical review. In spite of the compensatory measures taken to address the control rod insertion issue it was recommended that permanent measures for implementation be included in the modernization programme as soon as the root cause has been identified. Recommendations were given to systematically complete the integrity assessment of the RPV including PTS protection and to systematically study bypass scenarios of the containment and the environmental conditions of the compartments containing safety equipment for upgrading. While the intended programmes in the area of I&C and electrical power were found to address the issues, additional recommendations to further analyse the reliability of emergency power supply were made and high priority to fire hazards analysis in the modernization programme was requested. Though numerous operational safety improvements are being pursued, there is no co-ordinated plan or schedule for achieving needed improvements in parallel with the modernization programmes.

South Ukraine NPP Units 1 and 2 (July 1996)

The Safety Review Mission to South Ukraine NPP Units 1 and 2 [109] and the Technical Visit to VNIIAES and Novovoronezh NPP Unit 5 [110] have confirmed that nearly all safety issues identified for the standard model 320 already existed at ‘small series’ WWER-1000 plants. Most of them were found to be addressed or at least partly addressed in the Long Term Safety Improvement Programme and other safety improvement activities at South Ukraine NPP. It was recommended to urgently address the vulnerability of the ‘small series’ units against common cause failures due to lack of independence in safety systems and their support systems by both improving separation and preventing internal hazards to avoid challenges to the plant. The plant management was further encouraged to continue its efforts to strengthen defence in depth capabilities at plants by reducing the frequency of events, to prepare the new SAR as required by the regulatory body and to avail itself of the peer review for a PSA under development for complementing the deterministic analysis.

Review of plant specific modernization programmes for standard model 320 WWER-1000 plants under construction:

Rovno NPP Unit 4 (October 1995) and Khmelnitsky NPP Unit 2 (June 1996)

Benefit was taken from the conclusions and recommendations of the IAEA experts' mission to Rovno Unit 4 [113] in order to improve Revision 1 of the safety modernization programme at Khmelnitsky Unit 2 [114] submitted for IAEA review. The potential of higher feasibility and effectiveness for safety improvements of plants under construction was considered when the relevant Ukrainian programmes were reviewed. The IAEA team has emphasized the need to ensure the adequacy of margins for additional mechanical loads exerted on reactor core components in case of LOCA and/or seismic event to prevent further degradation of fuel assemblies. With respect to the issue of RPV embrittlement and its monitoring it was recommended that both the plant and the regulatory body carefully consider options for modifications of the surveillance sample programme before making final decisions to ensure an appropriate monitoring of RPV embrittlement rate which is not the case for the current design. The issues with respect to steam generator collector integrity and I&C were found to be well addressed. The plants were encouraged to continue efforts to properly address the issues of steam and feedwater piping integrity, containment bypass scenarios and particularly the issues in the area of accident analysis. Last but not least, the missions to both sites have revealed the need to improve several aspects of operational safety on site such as housekeeping, maintaining design intent and functionality of already installed systems and components.

Temelin NPP Units 1 and 2 (March 1996)

The Temelin NPP, originally designed according to standards of the FSU, is the first WWER-1000 reactor for which, after a series of reviews in the 1980s, a decision was taken by the Temelin NPP management to upgrade the design by foreign supply. The adoption of Westinghouse Electric Corporation (WEC) technology and practices for part of the scope of supply including fuel, I&C, radiological protection and accident analysis were found to properly address a large number of relevant issues identified for WWER-1000/320 plants. It was recognized during a mission in March 1996 [115], that all the safety issues identified by the IAEA have been addressed to meet the intent of the recommendations. The combination of Eastern and Western technology and practices and the potential compatibility problems seem to have been carefully considered at Temelin. Plant managers appear to be clearly committed to implementing operational programmes that are consistent with effective Western operational safety practices. Recommendations to Temelin management were to continue efforts to improve plant safety by providing the necessary resources, both financial and personnel, to complete the planned activities in a timely fashion. Temelin management should further continue its efforts to instil a strong safety culture among all station personnel, and should strengthen efforts to ensure that responsibility for improving safety is clearly placed, and accepted, down to the lowest levels possible. Management should foster an attitude that meeting the requirements of applicable laws and regulations is only one element of a strong safety culture. Implementation of symptom-based EOPs should contribute significantly to the plants ability to respond effectively to unexpected plant events. The need exists to locally override some automatic protective actions to carry out optimum response actions of the EOPs. This issue needs to be considered by the Czech Regulatory Body in the light of national regulations. Systems and components involving new design methodology or first of a kind equipment should be closely monitored during commissioning and initial operation to obtain performance data which could be used for confirmation of the safe design.

However, considerable effort remains to bring planned programmes to successful implementation.

MOHT-EdF Generic Reference Programme (March 1996)

At the request of the Ministry for Atomic Energy of the Russia, an IAEA CM was convened in March 1996 in Vienna to review the Generic Reference Programme for the Modernization of the WWER-1000 model 320 NPPs which was based on the original Russian Concept of Safety Upgrading for Operating WWER-1000 Power Units. About 20 Eastern and Western experts including IAEA staff attended the meeting. This reference programme was developed by experts of the Russian designers of the WWER-1000/320 reactors within the consortium MOHT together with experts of EdF, in order to address the evolution of the safety norms and regulations since the development of the original design, to take into account their operating experience feedback and to incorporate the results of PSAs and the results of international expertise on the safety of WWER reactors. It was intended as an input to prepare the specific modernization programmes for WWER-1000/320 plants in operation or under construction in Ukraine and Bulgaria. The MOHT-EdF programme is a set of prioritized measures addressing generic safety and availability issues both in the design and operation areas. The document is considered by its authors as a reference, i.e., a basic proposal to allow operators to establish specific and enforceable modernization programmes for WWER-1000/320 plants, taking into consideration the plant specific features, the requirements of national authorities and available funding.

Review of the MOHT-EdF reference programme showed that the IAEA and Riskaudit recommendations have been taken into account and that it addresses therefore most of the safety concerns identified. The Safety Issue Book for the WWER-1000/320 NPPs has had a major influence on the development of the programme. Implementation of the MOHT-EdF reference programme on a plant specific basis including some measures proposed in addition to the IAEA recommendations will make a major contribution to improve safety at these plants. Major safety issues such as control rod insertion/fuel assembly deformation, weaknesses in the areas of electrical power and I&C were found to be addressed by measures. The IAEA recommendation made during the review to modify the surveillance programme with respect to the positioning of containers for all plants under construction (e.g., the Temelin NPP) was addressed in the subsequent revision of the generic reference programme.

Achievements

Together with other sources, the Safety Issue Book for the standard model 320 WWER-1000 plants has been broadly used by Member States concerned to develop and update their national safety improvement programmes and modernization programmes. In addition, Western countries and other organizations have gained benefit from the Safety Issue Book to provide their assistance to Member States concerned. The general insight from all the IAEA activities to review the safety improvement programmes is that the overwhelming majority of safety issues in design and operational safety of the Safety Issue Book for WWER-1000/320 NPPs have been addressed by the proposed measures in the modernization programmes.

It was found that a certain number of measures need to be completed and some new measures need to be added to meet the intent of some of the recommendations of the IAEA.

It was broadly recognized during the reviews that the proposed improvements in the modernization programmes individually and in combination need to be examined to ensure that they do not cause adverse effects in an integrated action plan to improve overall safety.

While the development of the Safety Issue Book for ‘small series’ WWER-1000 plants was started in 1996, the respective units have been able to rely on the Safety Issue Book for the standard model 320 to improve plant safety reflected in the plant specific status [111] and to develop their modernization programmes.

IAEA activities derived from the Safety Issue Books for WWER-1000 NPPs and aimed at reviewing modernization programmes have contributed to developing the MOHT-EdF generic reference programme for the modernization of WWER-1000/320 plants in operation and under construction.

Outlook

In general, safety modernization programmes were found to be well developed and structured with respect to the design issues. Their implementation will make a major contribution to plant safety. However, the degree of detail of the individual measures addressed in the programmes and most descriptions needed further improvements. Considerable effort will be needed to bring planned programmes to successful completion.

The management at plant sites should also consider to merge parallel activities to improve safety into one single comprehensive modernization programme; such a move would facilitate transparency, agreement with regulators on the necessary improvement steps for license extension, and justification for financial support.

Experience from the resolution of issues at operating plants and progress made in root cause identification, e.g., the issue of control rod insertion reliability (Section 5.3.1), should carefully be followed up to revise the modernization programmes and allow feasible and effective implementation bearing in mind international assistance.

5.3. SELECTED SAFETY ISSUES

5.3.1. Control rod insertion reliability

Increased drop times of control rods beyond the design limit of four seconds and in some cases even sticking of rods in intermediate positions of the core have been observed.

These events have occurred in the early nineties in most Russian and Ukrainian WWER-1000 plants during the third year of operating fuel assemblies in the reactors. In 1994 an event involving the increase of control rod time also occurred in Bulgaria at Unit 6 of the Kozloduy NPP which was operated at the two years fuel cycle. The direct cause was attributed to increased friction between the rodlets and their guide thimbles due to bowing of the fuel assembly.

Unreliable insertion of control rods impacts the reactor shutdown capability. The change in water gaps also associated with fuel assembly bow will cause a higher local power density during normal power operation. In a reactivity initiated event where positive reactivity is inserted into the reactor core, a delay of the control rod insertion or a control rod jamming may not sufficiently depress the peak power level, may deteriorate the fuel cooling and may also lead to a transient of high peak primary pressure.

This issue is not applicable to Novovoronezh NPP Unit 5 which has a different fuel assembly design with an outside shroud providing higher stiffness against fuel assembly bow.

The issue is considered to be properly addressed by hardware measures taken and planned follow-up actions adopting WEC technology at Temelin NPP.

IAEA activities

A CM was convened in 1995 which specifically focused on control rod insertion reliability [108]. The objectives of the meeting were to exchange international experience and regulatory requirements, to review interim measures established in Member States to continue operation including modifications implemented or planned and to discuss the status of the root cause investigation of this issue. It was recommended to closely follow operational performance of fuel assemblies and to verify by at least three years of operating experience that excessive axial force is the main contributor to fuel bowing as supposed by the Russian designer. Russian design and research organizations were encouraged to continue investigating the underlying complex mechanism which lead to permanent fuel bowing after two to three years of operation in order to facilitate operation with existing fuel design under various conditions and to develop new designs.

The interim measures for continued plant operation applied at Ukrainian WWER-1000 reactors were reviewed at Zaporozhe NPP and supported by the expert team of a IAEA SRM [116] in 1994. Temelin NPP review with respect to core issues took into account that the fuel is being changed by WEC [115].

To monitor the progress made in resolving this issue since 1995 a Technical Meeting on the Control Rod Drop Issue for WWER-1000 NPPs was convened in November 1998 (report is under preparation). This meeting also reviewed the safety implications of water gaps due to fuel assembly bow and the measures taken.

Achievements

There is much experience at WWER-1000 reactors with implementation of compensatory measures to address the direct cause. This experience has demonstrated significant improvement in the delayed or incomplete insertion of control rods. Although these compensatory measures consequently improved plants' safety, some negative impacts mainly with respect to the fuel economy have been revealed. The main mechanism of the root cause leading to fuel assembly bow associated with delay of control rod insertion and water gaps was found to be irradiation induced creep or relaxation under excessive mechanical load caused by improper mounting of the upper intervals. The current experience with corrective measures associated with fuel and fuel assembly design changes to address the root cause is insufficient to know if the measures have totally corrected the problem.

Currently, the safety significance of this issue is considered to be low since remedial actions have been taken, long term corrective measures are underway, defence in depth provisions exist in the reactor design, and adequate safety margin remains. However, the industry should continue data collection on the problem and continue with the implementation of corrective measures.

Excessive water gaps associated with fuel assembly bow beyond that included in the design basis or methodology were found to be common to many WWER-1000 reactors and PWRs as well. A so-called "Generalized Methodology" is being developed by a number of Russian institutions headed by OKB Gidropress to address the water gap problem generically. It provides a generic licensing case by adding to the current design basis a large amount of plant data related to fuel bowing which has been collected for WWER-1000 reactors. This generic licensing case will avoid the necessity to perform cycle-specific reload safety justifications currently performed at each unit with excessive water gaps.

The present experience feedback indicates the possibility to reduce the safety implications of water gaps to an acceptable level from the plant operational point of view.

Outlook

To address both problems associated with fuel assembly bow, the delay of control rod insertion and excessive water gaps, the fuel assemblies and the whole core, respectively, need to be made firmer in the long term.

In the past few years similar events of delayed control rod insertion or even rod sticking have been observed at some PWRs in Sweden, USA, France, Spain and Belgium. The main mechanism of all these events including those identified at WWER-1000 plants are similar and considered as the consequence of insufficient verification of the safety of the existing and new fuel designs for advanced fuel performances. Fuel safety criteria and methods currently applied to existing and new fuel designs under advanced performances are being reviewed for Western PWRs under a task force in the CSNI WG2. This topic as related to WWER fuel is included in the framework of the IAEA's Nuclear Safety Programme for the years 1999-2000.

5.3.2. Steam generator integrity

Based on the accumulated operational experience and assessment of the consequences of failures, the WWER-1000 steam generator integrity and in particular steam generator collector integrity was recognized as an important issue of high safety concern. On 25 operating WWER-1000 steam generators cracks have developed in cold collectors during operation. In three cases, damage to the ligament integrity was found by increase of the steam generator water activity due to primary to secondary leakage. In all other cases the damage was detected in the course of in-service inspection during the annual scheduled outages. The first collector damage raised concerns regarding the safety of WWER-1000 NPPs. Consequently, substantial efforts were initiated to identify the root cause of the damage and to develop and implement compensatory and corrective measures.

Compensatory and corrective measures have been implemented on a plant specific basis at all steam generators of operating plants and have been effective except for one case when further cold collector damage was found at Balakovo NPP in 1995. This particular steam generator collector has been repaired and is in operation. It should be noted that, e.g., Kalinin Unit 1 is still operating with the original steam generators which have only some corrective measures implemented.

IAEA activities

In the framework of the IAEA Extrabudgetary Programme on the Safety of WWER-1000 NPPs in operation and under construction in Central and Eastern Europe, three meetings addressing the subject were convened in May 1993, November 1993 and September 1995. The report WWER-1000 Steam Generator Integrity [117], was prepared, integrating available information on the issue with emphasis on steam generator cold collector damage. The report concludes that steam generator collector cracking remains a matter of safety concern. Investigations of the damage provided a basis for development of compensatory measures, which have been effective to date to fulfil the safety related design basis. However, the mechanisms which lead to this damage still need to be quantified and design and operational aspects still need attention. It is also concluded that different approaches to address the issue were chosen by Russian and Czech experts, and exchange of information could provide a useful guidance for future activities.

The associated concerns were also addressed in the framework of the activities related to primary to secondary leakages, presented in Section 6.5 of this report.

In the framework of its Extrabudgetary Programme to assist China in evaluating safety of its NPPs (Lianyungang NPP), the IAEA in 1997 organized a Workshop on the Safety Improvements of WWER-1000 Steam Generators in China [118].

In addition to the generic activities described, the issue was also addressed on a plant specific basis throughout the programme in the course of safety review missions, missions to review safety improvement programmes and technical visits to individual plants (mission reports).

Achievements

The IAEA activities on this subject provided a forum for exchange of experience and assisted in establishing international consensus on actions still required. The corrective measures developed were taken into account in the design of the modified WWER-1000 steam generator. Five steam generators built using the modified design are in operation at present at South Ukraine Unit 2 since May 1991 (four SGs) and at South Ukraine Unit 1 since June 1991 (one SG). At other operating plants, the implementation of the corrective measures varies for each unit. The results obtained to date indicate that a catastrophic collector failure is unlikely to occur (‘leak before break’ collector behaviour).

Outlook

It should be pointed out that it is highly important to continue improving the in-service inspection, water chemistry control and monitoring, and develop and implement state-of-the-art condition monitoring and leak detection (primary to secondary) systems. Replacement of secondary system safety valves by new ones qualified for water and water/steam mixture flow may also need to be considered.

5.3.3. Vulnerability of safety and support systems for ‘small series’ WWER-1000 NPPs

For ‘small series’ WWER-1000 units the lack of physical separation and functional isolation of the most important components of the ECCS and its support systems was identified as the major concern. It may result in loss of control, i.e., loss of main safety functions in case of common cause failures (CCF):

- Fire or flooding in the ECCS compartment could result in loss of all equipment needed for normal shutdown and the complete ECCS would be at risk due to adverse ambient conditions in case of LOCA plus passive single failure to be assumed in the long term phase.

- Another concern is that the design of reactor protection system for the ‘small series’ WWER-1000 units is different from the WWER-1000/320 units. The technological part of reactor protection system consists only of one protection set, which meet the single failure criterion but can not be tested during reactor operation according to current national standards and international practice.

IAEA activities

A technical meeting on Physical and Functional Separation of Safety Systems for WWER-1000 Reactors was convened in August 1997 under the regional Europe TC project on Support for Safety Assessment of NPPs, RER/9/046, with the participation of experts from Eastern and Western countries.

A technical visit by IAEA experts was carried out at Kalinin NPP in November 1998.

Achievements

The IAEA meeting on the independence of safety systems and their support systems has provided guidance on the identification of safety implications caused by CCFs and measures to resolve them. The scope of the meeting included all WWER plant generations to the extent necessary [119].

A two-set system of reactor protection has been implemented at South Ukraine NPP Unit 1 and to be implemented at Unit 2 during the outage in 1998.

Outlook

The safety issue "Physical separation and functional isolation of the ECCS" (S16) has been ranked as Category III [111] because it is realized that the most important components of the ECCS are located in one compartment, but separated by fire walls. The risk of failure of the complete ECCS due to adverse ambient conditions in case of LOCA, assuming a passive single failure in the long-term phase, was estimated as remote based on the restricted flooding potential and the measures taken to improve the reliability of the ECCS.

While systematic analysis to establish long term upgradings of the plants are underway actions can be taken to improve the safety of their operating plants. Based on the experience in Western plants and other operating WWERs, some compensatory measures with obvious benefit can be implemented within a short time frame at a relatively low cost. It was strongly recommended by the IAEA to realize plans at KNPP for closing the holes of 2x3 m^2 in the fire walls by leaktight doors. These measures should be taken in parallel with the more extensive analysis (e.g., PSA work at Kalinin and Novovoronezh Unit 5 NPPs). Due to the limitations to the resources available for upgrading, the corrective measures selected for implementation should be those which offer the highest value in terms of the safety improvement benefit as compared with the cost of implementation.

It was also stated that for model 320 WWER-1000 NPPs, the lack of separation in the ventilation system deserves a higher attention than this issue has been given so far.

5.3.4. Seismic safety

Structurally, WWER-1000 NPPs are better designed than WWER-440 type NPPs. Furthermore, a minimum of seismic loads were considered in the original design.

The seismic re-evaluation programme for an NPP generally involves the re-assessment of the seismic hazard, plant seismic capacity and if required, design and implementation of upgrades to components and structures. In general, a Terms of Reference document is needed for the implementation of a seismic upgrading programme.

IAEA Activities

Seismic safety review services were conducted at Kozloduy NPP Units 5-6 and at the Temelin NPP. The breakdown of these services (reviews, follow up mission and workshops) for these two plants is presented below for the period 1990-1997.

TABLE 5.2. SEVEN-YEAR OVERVIEW OF IAEA SITE/SEISMIC SAFETY REVIEW SERVICES TO EASTERN EUROPEAN NPPs

		Number of services (1990-96)			
Country	Plant	W	S	SI	SC
Bulgaria	Kozloduy 5/6	-	-	1	2
Czech Republic	Temelin	2	4	-	-

W: Workshop, work plans and technical procedures.
S: Site safety review.
SI: Review of seismic input and tectonic stability.
SC: Review of seismic capacity.

Terms of Reference documents for these plants have not been prepared to date. The IAEA was involved with the seismic (and in general site related issues) safety of Temelin NPP until 1994. The original design of this plant was for 0.06g. The reassessed value is reported to be 0.1g (the minimum recommended value of the IAEA Safety Guide 50-SG-S1, Rev. 1). The reassessment of the seismic input or the seismic capacity of Temelin NPP was not reviewed by the IAEA. The seismic input for the Kozloduy NPP Units 5 and 6 was re-evaluated as 0.2g (same as Units 1-4) tied to a wide band site specific response spectrum. The original seismic design value for Units 5 and 6 was 0.1g. The seismic capacity of these units was reviewed within the scope of a seismic PSA. It should be noted however, that this PSA was not finished to the point of quantifying the core damage frequencies. Recently the Kozloduy NPP has requested the IAEA to develop terms of reference for Units 5-6. This is expected to be completed within two years.

Major issues which have been identified in relation to WWER-1000 NPPs can be summarized as follows:

1. The difference between the original seismic input and the reassessed values exists but to a much smaller extent when compared with WWER-440 NPPs. This is possibly due to the fact that these are comparatively newer plants and seismic issues have been considered more thoroughly.

2. The structural systems of WWER-1000 NPPs are much better than those of WWER-440 NPPs and in general do not present a major problem, although the use of precast concrete elements as well as the prestressing in the containment structure need to be checked.

3. The issue of documentation is similar, although during the CRP on Benchmark Study for the Seismic Analysis and Testing of WWER Type NPPs, a great deal of work was

performed to alleviate this problem. Kozloduy NPP Units 5-6 were among the prototypes for this CRP.

4. The issue of support and anchorages is similar to that found at the WWER-440 type NPPs.

Achievements and outlook

- Derivation of the new seismic input value and ground motion characteristics. For Kozloduy NPP, the new value was determined to be 0.2g tied to a wide band site specific response spectrum.

For Temelin, it is reported to be 0.1g. This value has not been reviewed by the IAEA.

- Aside from the review of a partially completed seismic PSA for Kozloduy NPP Units 5 and 6, there was no IAEA involvement in the review of the work performed (if any) in relation to other tasks (i.e., easy fixes, structural upgrading, equipment qualification) with the exception of tasks performed within the scope of the CRP on Benchmark study for the seismic analysis and testing of WWER type NPPs.

Kozloduy NPP has also recently requested a Terms of Reference document to be prepared from the IAEA.

- Tasks performed under the CRP on Benchmark study for the seismic analysis and testing of WWER type NPPs:
  - Safe shutdown systems identification/classification
  - Original design regulations, acceptance criteria, loading combinations
  - Comparative study of seismic design standards
  - Dynamic analysis of Kozloduy NPP Unit 5 and 6 structures
  - Full scale blast testing of Kozloduy NPP Unit 5
  - Shaking table testing of selected components from Kozloduy NPP Units 5 and 6
  - On-site testing of components at Kozloduy NPP Unit 5
  - Compilation of previous component test data
  - Compilation of earthquake experience data
  - Assessment of containment dome prestressing for Kozloduy NPP Units 5 and 6
  - Stress analysis of safety related piping of Kozloduy NPP
  - Analysis of buried pipelines for Kozloduy NPP Unit 5 and 6.

6. GENERIC SAFETY ISSUES COMMON TO WWER NPPs

The IAEA has identified safety concerns which are common to all types of WWER plants. CMs were convened to exchange information and provide guidance on the resolution of these generic safety issues in the framework of the EB Programme and some relevant TC projects.

6.1. CLASSIFICATION AND QUALIFICATION OF COMPONENTS AND SYSTEMS

In the existing WWER design, the components and systems are classified according to their functions as required by OPB-73 and OPB-82, i.e., normal operating system, protective actuation safety system, localization actuation safety system, protection safety system and support safety system. They were not classified according to the importance to safety, i.e., safety class, as required by OPB-88 and NUSS. Systems classification for Kola NPP Units 3-4 was carried out in 1990 in accordance with “General proposals for providing the safe operation of nuclear power plants” (OPB-88) of Main Designer (“Atomenergoproject”, St. Petersburg), and for Units 1-2 in 1994. Regarding quality classification and seismic classification, the PNAE-G-7-008-89 and PNAE-G-5-006-87 regulatory standards were published only in 1989 and 1987, respectively.

Many WWER-1000 units have added classification lists in the SARs, which include safety, quality and seismic classifications of components and systems important to safety. However, the qualification of these classified components and systems to seismic and environmental conditions at operating modes, including DBA, has not fully been demonstrated. As shown by safety reviews, this practice of qualification of components and equipment important to safety is either lacking or not evident.

An example of this deficiency is the qualification of electrical and I&C equipment including cable connections for LOCA conditions. The cable connections inside the containment of WWER plants are not able to withstand extreme environmental conditions and consequently, they have a high failure potential under LOCA conditions.

Likewise, the safety related systems or components such as ventilation systems, service water system, fire water supply pumps and indication and recording instrumentation, are not qualified for seismic loads. Their functional capability on demand in case of an earthquake would be questionable.

IAEA activities

Three Russian standards which define the safety classification [120], quality classification [121] and seismic classification [122] were compared with the IAEA NUSS standards through consultants' meetings in 1994 [3] and 1996 [123].

The EBP missions provided recommendations to the plants in the process of changing the classification of components and systems from the original Russian system to the IAEA NUSS standards. For example, in the Paks NPP the practice of classification was different from the guidelines of the IAEA; there, the systems were classified in accordance with the safety function to which they could contribute, whereas the IAEA guidelines state that safety class 1 or 2 includes the systems necessary to fulfil a certain safety function. This resulted in an excessive amount of systems and equipment classified in 1 and 2 safety class, with the corresponding large workload and long times necessary to check the qualification of NPP equipment. The recommendations of the EBP mission helped to clarify the situation so that the process of equipment qualification could be accelerated.

Achievements

The results of comparison have shown that although the concepts are the same, there are differences between Russian standards and IAEA NUSS standards. The definitions of safety classes in Russian standards are different from NUSS. Some safety-important elements, such as control rods, reactor internals, mechanical and electrical parts inside the components, shells of components made of non-metallic materials, supports and suspenders of piping, sealing gaskets etc. are not covered and defined in the Russian quality standards. For safety class electrical and I&C equipment, and safety class ventilation system, no quality standards are available in the regulatory document. In view of these preliminary results, in-depth comparison of Russian quality standards with internationally recognized codes could be considered in order to judge the quality level of components and equipment.

A retrospective review of the classification of components and systems either has been carried out or is planned at WWER plants in the framework of safety improvement programmes. The review covered mechanical as well as I&C and electrical equipment. After reviewing this classification, the necessary improvement measures will be developed. The maintenance, surveillance and in-service inspection procedures are to be modified to ensure compliance with the classification requirements. Many WWER plants plan to analyse equipment qualification, including seismic qualification in their safety improvement programme.

At operating WWER units, improvements have been made in specific systems and equipment. At Temelin NPP, classification of components was made according to the Decree of the regulatory body. The classification used is in compliance with the IAEA classification. A comprehensive project of equipment qualification has been started at Temelin. Rovno NPP Units 1 and 2 have upgraded the reactor protection system to a modern digital hardware system. The cable connections have been improved at Kozloduy NPP Unit 5 and are under way at other units. At the Loviisa NPP, safety classification was introduced by the Säteilyturvakeskus (STUK) of Finland in 1982. The I&C system has been qualified at the stage of Loviisa NPP construction and is being continuously supplemented as needs arise on the basis of current safety analyses of the plant. The early tests at Loviisa NPP showed many faulty elements, e.g., some hundreds of valves had to be replaced because they failed during the qualification process.

Outlook

After the re-classification of components and systems in WWER plants, their qualification in terms of quality, reliability, maintenance, surveillance and in-service inspection needs to be assessed and implemented.

Since many activities on the classification and qualification of equipment are under way at WWER plants, the IAEA will continue to act as a forum for exchange of experience and compile information on the progress made at individual plants.

6.2. REACTOR PRESSURE VESSEL INTEGRITY AND ASSESSMENT

The RPV is a specific component, the integrity of which has to be maintained throughout the life of the nuclear power plant, since there are no feasible provisions which would mitigate a catastrophic vessel failure. The RPV integrity is ensured by a margin between its load bearing capacity and acting loads which can occur during the plant operation. The RPV load bearing capacity diminishes during plant operation due to material ageing effects, in particular due to neutron irradiation induced embrittlement. The loads to be considered in the vessel integrity assessment are mainly related to events leading to a pressurized thermal shock.

IAEA activities

a) Generic activities

In the framework of the IAEA Extrabudgetary Programmes, the issue of the RPV integrity received systematic attention from the start of the Programme.

Based on the recommendations of the AG and the Steering Committee and in order to assist countries operating WWER plants, the IAEA initiated actions in several directions, focused on material properties, on pressurized thermal shock analyses and on in-service inspection qualification.

To assist in addressing the material properties aspects, the preparation of the Round-Robin Exercise on WWER-440/230 RPV Weld Metal Irradiation, Embrittlement, Annealing and Re-Embrittlement [85] was initiated in 1995 with the objective of developing a database which could be used as a basis for judging the reliability of the material data used in the RPV integrity assessment. This effort has been transferred to the regular budget of the IAEA and is under way at present as an IAEA Co-ordinated Research Programme. Organizations from Belgium, Bulgaria, the Czech Republic, Finland, France, Hungary, Russia, and Slovakia participate in the programme.

In order to provide a commonly agreed basis for the pressurized thermal shock (PTS) analysis, Guidelines [20] were developed on this topic, which deal with the RPV PTS assessment required to justify RPV integrity for nuclear power plants with WWER type reactors. The guidelines provide advice on the individual elements of the PTS assessment such as acceptance criteria, analysis methods, computer codes, assumptions to be used and quality assurance.

It should be pointed out that PTS assessment is a multidisciplinary effort and involves selection and categorization of initiating events to be considered, thermal-hydraulic analysis, structural analysis including fracture mechanics assessment, evaluation of material properties and neutron field calculations.

The objective of the guidelines is to establish a set of recommendations for RPV PTS assessment based on state-of-the-art approach reflecting practices, operational experience and results of research and development in Member States

The PTS assessment outlined in the guidelines covers transients and accidents to be considered in the reactor design. The purpose of the assessment is to provide a reasonably bounding demonstration of the RPV integrity by using realistic modelling methods for the individual elements of the analysis with conservative assumptions, initial and boundary conditions and appropriate safety factors in the assessment of the results. A deterministic approach is used in the guidelines by analysing limiting transients from each event group. Limiting in this sense is understood as limiting the RPV integrity.

In some cases, where the scope of the transients and accidents considered cannot be directly reduced, the material data involve large uncertainties or the RPV material embrittlement tends to be high, a probabilistic approach would result in important complementary information.

The demonstration of the RPV integrity is performed in terms of the safety margin between maximum allowable value of critical brittle fracture temperature and the actual RPV specific values.

In order to promote the implementation of the Guidelines for PTS Analysis [20], the IAEA initiated the Benchmark Exercise on PTS Analysis for WWER Plants. Further objectives of the exercise were:

- to verify the applicability of the Guidelines
- to provide a forum for exchange of experience and for comparison of calculation procedures for PTS analysis used in different countries operating WWER plants
- to establish consensus on technical aspects of practical implementation of the Guidelines [20], in particular on limitations of the methodology for individual elements of the analysis including impact of the limitations on the plants' safety assessment.

Based on the discussion with experts from Member States concerned, it was agreed to perform the benchmark exercise using a model WWER-440/213 plant and corresponding typical data. The plant data used do not, however, represent a particular WWER-440/213 unit in operation or under construction. The benchmark exercise will be carried out in four steps, covering thermal-hydraulic system analysis, mixing calculation, structure temperature/stress analysis, and fracture mechanics analysis. Organizations from Bulgaria, the Czech Republic, Finland, France, Germany, Hungary, Russia, Slovakia, Ukraine and USA are participating in the programme.

b) Plant specific activities

In addition to the generic activities described, the issue has also been addressed on a plant specific basis throughout the programme in the framework of expert missions, safety review missions, follow-up missions and technical visits to individual plants, and were focused on WWER-440/230 plants in principle, see also Section 3.4.1 [59, 60, 61, 62, 63, 65, 66, 67, 69, 70, 73, 74, 75, 76, 77, 78, 79, 80, 85, 86, 87, 88].

Achievements

The IAEA assistance to Member States either completed or still under way addresses almost all aspects of the concerns identified and recommendations given with respect to RPV integrity. IAEA tasks provided assistance and a forum for co-operation in establishing a balanced, technically justified and verified approach to RPV integrity assessment. In this sense, the activities are applicable to all WWER plants.

At a number of WWER plants, activities to address the RPV issue are under way. The integrity assessments have been or are being re-evaluated and where necessary, the implementation of corrective measures has been completed, is under way, is planned or being considered. The Guidelines for PTS analysis can provide a commonly agreed basis for reviews of the RPV integrity assessments.

Outlook

Upon completion of the round robin exercise and the PTS benchmark exercise, the outcome can serve as a tool for judging the reliability of the results of the plant assessments and promote the use of state of the art approaches. This will contribute to the establishment of a solid basis for decisions on plant modifications and continued operation. This is, however, a matter to be addressed in the framework of national or bilateral programmes. The IAEA will continue to provide a forum for exchange of experience and compile information on the progress made at individual plants.

6.3. GUIDELINES FOR THE APPLICATION OF THE LBB CONCEPT

The LBB concept has been developed and applied to a number of PWR plants in several countries with the objective to exclude specific considerations for dynamic effects associated with primary circuit large diameter breaks. The basis of the LBB concept is a demonstration that the piping concerned would leak significantly (reliably detectable leak) before a double-ended guillotine break can develop. This is achieved by quantifying and evaluating the process of loss of integrity and of accompanying leaks. The application of the LBB concept requires installation of leak detection systems or re-qualification of the existing ones. In addition to the operational aspects (shutdown requirements), provisions needs to be made to maintain the conditions of the concept throughout the operation (ISI, maintenance, surveillance). Primary circuit piping which meets the requirements can in principle be considered as to have a low probability of a large LOCA (less than 10^{-6} per reactor year).

The design concept of the WWER-440/230 plant required that there be no loss of primary circuit integrity resulting in significant deterioration of core cooling. A large pipe break would result in the loss of two main safety functions: cooling the fuel and confining the radioactive material.

The application of the LBB concept to WWER-440/230 plants large diameter primary piping has been recognized to be of major safety significance as a tool to restore some features of the original safety concept from the current point of view on maintaining primary circuit integrity. The application of the LBB concept to WWER-440/230 plants was initiated in the former Czechoslovakia for Bohunice NPP Units V-1 in 1988 for the purpose of addressing the above concerns as well as to verify the piping design and provide a basis for seismic backfitting.

For the more modern WWER plant types, the LBB concept has been considered, in addition to its usual use for PWRs, a tool for design verification, a basis for a plant’s seismic backfitting justification, for avoiding piping restraints and an additional line of defence.

IAEA activities

a) Generic activities

The issue of the application of the LBB concept received systematic attention as of the outset of the IAEA EBP. Following the publication of a status report Applicability of the LBB Concept [92], which provides an overview of the approaches adopted in this regard in countries operating nuclear power plants, it was recognized that there was a need for a more systematic guidance in this area.

The IAEA therefore prepared a report on Guidance for the Application of the LBB Concept [21]. The report recommended that LBB analysis be split into three programme levels:

- basic programme, including LBB methodology, fatigue damage analysis, and corrosion damage analysis

- supporting programme, including material database, static and seismic analysis, analysis of water hammer, stability of heavy components supports, leak rate calculations, leak diagnostics which provide the background data underlying the basic programme

- verification programme, including large scale experiments and leak rate testing which demonstrate the consistency and conservativeness of the assumptions used.

The format recommended for the safety case documentation is also described in the report. Annexes provide detailed information on the individual elements of the programmes, including the scope of the analysis and related recommendations. Attention is drawn to cases where plant specific aspects need to be taken into account especially and to cases where no international consensus exists or further information is required.

In 1995, the IAEA co-operated in the organization of the Seminar on Leak Before Break in Reactor Piping and Vessels in Lyon, France and supported the active participation of experts from countries operating WWER plants.

b) Plant specific activities

The Guidance for the Application of the LBB Concept [21] was used by countries operating WWER-440/230 plants when applying the LBB concept as well as during the IAEA reviews of the LBB concept application at Bohunice NPP Units V-1 [93], and Kozloduy NPP Units 1-4 [94, 95]. This guidance document was also used as a reference when reviewing the LBB concept application to Temelin NPP.

Achievements

The above described assistance of the IAEA to Member States provided a basis for the development of the approaches to apply the LBB concept in the countries operating WWER plants and for the reviews of the safety improvements achieved.

In addition to primary piping integrity, the results of PSAs and RPV PTS analyses for instance indicate the high safety importance of the secondary piping integrity.

Outlook

A balanced concept, consistent with the LBB as applied to the primary piping, still needs to be developed and implemented for the secondary circuit piping; the individual concepts and assessments should be combined in an integrated approach on maintaining the reactor coolant system integrity.

The IAEA will continue to provide a forum for the exchange of experience and compile information on the progress made at individual plants.

6.4. METHODOLOGY FOR QUALIFICATION OF ISI SYSTEMS

Radiation induced RPV embrittlement, application of LBB concept to primary circuit piping and components and steam generator integrity problems pose stringent requirements on both the capability and effectiveness of the in-service inspections performed at WWER plants. The capability and effectiveness of in-service inspection (ISI) have been identified as safety issues in the framework of the IAEA's Extrabudgetary Programme activities and ranked as of high safety significance (Category III) for WWER plants. Although efforts to improve in-service inspection are under way at many plants, a systematic demonstration of in-service inspection capabilities and limitations is actually lacking.

IAEA activities

Due to the high safety significance of WWER in-service inspection and bearing in mind the requests and suggestions from several countries operating WWER plants, the IAEA developed a report on Methodology for Qualification of In-service Inspection Systems for WWER NPPs [22].

The Methodology for qualification of in-service inspection systems for WWER NPPs incorporates the approaches and experience of several countries operating WWER plants, the USA (ASME/PDI), the EU and other Western European countries (ENIQ). It should be noted that these qualification approaches, methodologies and activities differ, in a number of aspects, as a result of the different industry and regulatory environments. In this respect, this qualification methodology is intended, in the short and medium terms, to be a pragmatic synthesis appropriate for the specific circumstances of the various countries operating WWER plants.

The objective of the report is to provide a methodology for qualification of ISI systems which might be used as a commonly accepted basis for further development of the necessary qualification-related infrastructures in countries operating WWER plants.

The report refers to any NDT method and defines how non-destructive in-service inspection systems (NDT procedures, equipment and personnel) should be assessed in order to demonstrate that a given inspection system is fit for its purpose. The following aspects are addressed: general qualification principles defining the administrative framework on which qualification processes should be carried out, the qualification approach, or how non-destructive in-service inspection systems should be qualified, the main steps of the qualification process specifying its minimum technical and documentation-related requirements, specific requirements regarding the NDT procedures, equipment, personnel and requirements applicable, the specimens to be used in practical trials, and the distribution of responsibilities based on international practice.

It must be pointed out that the methodology did not seek to provide criteria for definition of the extent of a qualification process (in terms of required inspection area(s), NDT method(s), the type(s) of flaws and required inspection effectiveness). These are matters to be agreed between the licensee and the regulatory body before any qualification process.

Achievements

The methodology developed represents a state-of-the-art basis for further activities in this area.

Implementation of the qualification methodology in all countries operating WWER plants will enable them to reach a common level of qualification related infrastructures, databases, experience and expertise.

A pilot study is being initiated to promote the application of the methodology developed. The study is part of a Regional Technical Co-operation Project (RER/9/020) on Advanced NDT for Primary Circuit Components of WWER NPPs.

Outlook

Qualification of in-service inspection systems is a complex and resource consuming task and hence, countries operating WWER plants are strongly encouraged to co-ordinate and optimize their qualification related initiatives and resources.

6.5. PRIMARY TO SECONDARY LEAKS

WWER design differs in many aspects from the standard Western PWR design; one of the most significant differences exists in the steam generator (SG) design. Horizontal SGs used in the WWER design have unique features if compared with vertical SGs, and especially, a larger amount of secondary coolant.

Since SGs play a central role in residual heat removal for many WWERs, the loss of SG integrity leading to interfacing LOCA was found to be of high significance for ensuring plant safety. An accident with subsequent SG cover lift-up occurred at Rovno NPP with WWER-440 reactors in 1982. Cracks have been revealed in the ligaments between tube holes in the primary SG collector of several WWER-1000 reactors in the early nineties which have the potential to propagate into larger cracks and to develop collector integrity failure if not treated in time (Section 5.3.2). Failure of the collector cover seal or collector rupture may cause large PRISE accidents which are unique to the WWER design.

The main safety concerns identified by the IAEA relate to loss of core cooling and radioactive releases due to long term loss of coolant bypassing the containment. Adequate automatic means do not exist to cope with large PRISE events because safety systems are typically designed to mitigate consequences of loss of primary or secondary coolant into the containment. Interventions of automatic safety features may even contradict those necessary to mitigate PRISE accident accidents. Operator actions supplemented by proper supporting measures are important to limit radioactive releases caused by PRISE accidents. PSA results for both WWER-440 and WWER-1000 plants show significant contribution to the risk of core damage arising from large PRISE accidents although different system capabilities exist for PRISE treatment in both designs. Large PRISE events must be further studied since some consequential failures have been identified such as possible steam line integrity failures caused by dynamic effects [7, 9] which cannot easily be addressed in the design. Although large PRISE accidents were not considered in the original WWER design as DBAs, and the probability of their occurrence has been significantly reduced by countermeasures that have been put in place, there is broad consensus on treating them in the SAR for WWERs.

IAEA activities

The IAEA realized the need for providing general guidance which could then be followed by Member States concerned. The issue was addressed for the first time in the EBP at a CM convened in June 1996 in Vienna in order to collect existing information contained in a comprehensive report issued in June 1996 [124] about PRISE events, on their specific hazards potential, the status of thermohydraulic analyses, preventive and mitigative measures taken and the radiological consequences. A second meeting, hosted by IVO, was convened in November 1997 in Helsinki to start developing a guidance document [19] which will be published by the end of 1998 as an EBP document.

The guidance document is intended to provide plant operators and regulators with practical guidance for the treatment of PRISE events, particularly large PRISE accidents, as DBAs for both WWER-440 and WWER-1000 NPPs. This document contains rules and practices related to PRISE events applied in Member States and the strategy for dealing with PRISE events. It discusses the conduct of conservative thermohydraulic analyses to identify strengths and vulnerabilities and to decide on the most effective improvements. This deterministic approach should be complemented by well qualified PSA studies to evaluate the risks involved, to select scenarios which need to be considered as DBA and to set priorities for improving defence in depth.

Achievements

Member States operating WWERs broadly agree on the need to strengthen plants' defence in depth capabilities to cope with PRISE events as DBAs.

The development of national approaches on treating PRISE events as DBAs varies considerably from country to country. While countries such as Finland (Loviisa NPP), the Czech Republic (Temelin NPP) and Hungary (Paks NPP) have almost completely established their national approach for PRISE treatment, others such as Ukraine have just started this process. The progress made in developing detailed emergency operating procedures differs remarkably among countries; in this context, co-operation between the various international groups developing symptom-based procedures for WWERs is very desirable. Thermohydraulic analyses performed so far indicate the need to optimize mitigation of the consequences of PRISE accidents through operator actions.

Outlook

The IAEA guidance document [19] will assist Member States in accelerating the development and implementation of national approaches treating a PRISE event as a DBA.

6.6. INSTRUMENTATION AND CONTROL

Instrumentation and control (I&C) has been recognized as an area which requires substantial improvements in WWER NPPs, particularly the WWER-440/230 and WWER-440/213 NPPs. The I&C equipment used at these plants was designed in the early 1960s and early 1970s, respectively. The specific criteria for achieving high functional performance and reliability of I&C, in accordance with its safety significance, were not included in the design, as these criteria were not available in these countries at that time.

As a result, the original I&C systems of WWER NPPs have deficiencies related to poor quality, deterioration with age, difficulty of maintenance, lack of self-monitoring capabilities, low level of automation, insufficient redundancy and insufficient protection against common mode failures. The I&C support to operation in the control room was generally insufficient for normal and abnormal emergency situations. In addition, man-machine interface problems, highlighted by the Three Miles Island accident, had not been addressed in a systematic way and this imposed high strains on the operators, especially in emergency situations.

IAEA activities

The situation of I&C in WWER units was highlighted at the beginning of the EBP during the missions to WWER-440/230 NPPs which resulted in several recommendations [6].

The AGM in December 1992 agreed that the safety issues related to I&C were generic and needed further analysis.

The IAEA commissioned with the Spanish Company Empresarios Agrupados a document proposing a technical basis for improvements related to the most significant aspects of I&C which included: I&C classification and related design criteria, control room habitability and remote shutdown panel, I&C support to operation and control room design, instrumentation setpoint margins and accident monitoring instrumentation.

Based on this report and on the results of a CM in May 1994, a technical report was published [125].

In addition to the aspects listed above, this report also addresses the topics of: reliability of I&C equipment; control and protection systems interactions; I&C redundancy, separation and independence; interlocking; I&C equipment qualification; I&C signal priority; and testability.

The guidance expressed in the report is based on IAEA/NUSS standards and on regulations in use in various Member States.

In September 1994, a CM reviewed the I&C safety issues specific to the WWER-440/213 NPPs [126].

Achievement and outlook

Since the beginning of the EBP, significant improvements have been made in many WWER NPPs in the area of I&C. However, large differences can be observed between the level of progress achieved in the different countries operating WWER NPPs.

The Bohunice NPP Units 1-2 and the Paks NPP are examples of WWER-440/230 and WWER-440/213 NPPs where significant improvements have been made and continue to be made in the area of I&C.

The IAEA will serve as a forum for an exchange of information on specific I&C issues and improvements and will provide, upon request, independent peer review services.

6.7. FIRE HAZARD

Concerns have been raised by fire incidents at some WWER nuclear power plants: Greifswald Unit 1 in 1975; Armenia Unit 1 in 1982; Balakovo in October 1992; Kozloduy in September 1992 and Zaporozhe in May 1993.

The Russian fire protection standard VSN-01-87 was published in 1987, after the completion of the WWER design. Therefore, many deviations of the design from VSN 01-87 were indicated in the deviation list of the TOB at WWER plants.

Safety reviews of WWER plants have identified weaknesses in the fire protection area which in many cases are deviations from the current safety standards [7, 9, 111]. The following are examples of deficiencies identified:

- lack of qualified fire doors in fire barriers
- redundant cable trains run too close to each other
- lack of qualification of penetrations
- lack of fire resistance of overlayers covering the cables
- lack of segregation of cables in the cable spreading room, and
- inadequate protection against oil fire.

IAEA activities

a) Generic activities

The IAEA has contracted INITEC of Spain to provide a simplified probability-based fire hazard evaluation methodology and to illustrate its benefits and drawbacks through the application to Bohunice NPP Units V-1. The study was completed in May 1993 with the co-operation of RELKO, Ltd., a Slovak company [127].

The simplified fire hazard analysis methodology was further refined and reviewed through a CM in September 1993 [25].

In the framework of TC project RER/9/004, detailed plant specific analysis concerning fire protection in WWER-440/213 NPPs was included as one of the specific issues [98].

b) Plant specific activities

At the request of the Ukrainian regulatory body, a workshop on Fire Protection was conducted at the Zaporozhe NPP in August 1993 with the participation of relevant Ukrainian organizations to assist plant operators in developing their capability to improve fire safety. The workshop included the transfer of fire protection techniques, a plant walkdown inspection and the development of specific recommendations.

Achievements

Systematic guidelines are available to evaluate the response of the plant to fires and to identify remedies to protect the plant from the fire initiated damage.

Typical examples of backfittings reported and observed are: replacement of unqualified fire doors, protection of turbine roof against fires, prevention of fire from lubrication oil of reactor coolant pumps, covering of cable trains with fire resistant materials, installation of fireproof sealing of cable penetrations and fireproof belts in cable boxes, installation of fireproof partition walls, anti-smoke measures, and improvement of fire detection and extinguishing, etc.

Outlook

The improvements in the fire safety at WWER plants are expected to proceed for a long period of time. The IAEA will continue to act as a forum for exchange of experience and compile information on the progress made at individual plants.

6.8. ACCIDENT ANALYSIS AND SAFETY ANALYSIS REPORTS

The original SARs called Technical Justification for Nuclear Safety (TOB in Russian terminology) for WWER units were updated in Russia and some other countries. However, the format and contents of the existing TOBs are based on the old Russian regulatory documents TS TOB RU-87 (Typical Content of Technical Justification of Reactor Facility Safety) and TS TOB AS-85 (Typical Content of Technical Justification of NPPs) which are still valid for existing plants in Russia.

Concerns regarding accident analysis in the existing TOBs remain with respect to the accident spectrum, the assumptions used, the acceptance criteria, the quality of analysis and computer code validation. Accidents not considered in the TOBs so far include overcooling transients related to pressurized thermal shock, ATWS, boron dilution accidents, accidents during shutdown conditions and severe accidents. Every accident analysis needs a plant model or a detailed model of a specific part which must be constructed on the basis of reliable data. The experience of WWER plant owners shows that it is sometimes very difficult to obtain reliable and verifiable data on the plant construction.

According to the IAEA NUSS code on design, accident analysis shall be performed to ensure that the overall plant design is capable of meeting prescribed and acceptable limits for radiation doses and releases set by the regulatory body for each plant condition category. In addition, the operating organization needs additional analyses for protection and signal settings, for personnel training to cope with accidents, and for the preparation of emergency operating procedures. The accident analyses in the SARs available at some WWER plants do not meet the above given needs.

IAEA activities

a) Generic activities

The need for detailed guidance in the performance and review of accident analysis for WWER nuclear power plants has been recognized as a priority in order to solve these issues. The Guidelines for Accident Analysis of WWER Nuclear Power Plants[15] were developed through the organization of three CMs in 1994 and 1995.

The guidelines deal with the transient and accident analysis required to justify original designs and existing or newly proposed technical solutions such as plant modifications for safety upgrading at nuclear power plants with WWER type reactors. The guidelines give advice on selection and categorization of initiating events to be considered, on adequate specification of acceptance criteria, methods, computer codes, and assumptions to be used as well as on quality assurance procedures for accident analysis.

To facilitate the use of the guidelines and the resolution of the safety issue Computer Code and Plant Model Validation, a Workshop on Advanced Codes Validation and Uncertainty Evaluation was held in Slovakia in October 1996 with Western lecturers and participants from WWER operators, regulators and supporting institutions.

b) Plant specific activities

In addition to the generic activities described above, the issues in the accident analysis area have been addressed on a plant specific basis in the framework of safety review missions and technical visits to individual plants.

Nearly all the WWER plants have plans or ongoing activities to update the accident analysis in their safety improvement programmes, also utilizing bilateral co-operation programmes with other countries. The section on accident analysis of both the Bohunice NPP V-2 SAR after 10 years of operation and the Mochove preliminary SAR is based on the IAEA Guidelines for Accident Analysis of WWER NPPs [15].

c) Other IAEA activities

The IAEA has completed its TC Regional Programme (1985 to 1990) on activities related to the accident analysis of WWER-440/213 reactors with the participation of Bulgaria, the former Czech and Slovak Federal Republic, Hungary, Poland and the Ukraine. The programme was moved forward to the evaluation of the safety aspects of the WWER-440/213 under the IAEA TC project RER/9/004 (1991 to 1994) [98, 128].

From 1993 to 1994, the IAEA carried out its TC Regional Project RER/9/020 on the accident analysis of WWER-1000/320 reactors. The project activities included the development of databases for each of the WWER-1000/320 units separately (Kozloduy NPP, Temelin NPP and Zaporozhe NPP) for the nuclear steam supply system (NSSS) and containment systems. The participating countries were: Bulgaria, Czech Republic, Poland (technical support), Russia (technical and scientific support), and the Ukraine.

Achievements

The Guidelines for Accident Analysis of WWER Nuclear Power Plants have been widely used in the countries operating WWER NPPs. Revised accident analyses have been performed or are being performed in Bulgaria, Czech Republic, Finland, Hungary, Russia, Slovakia and the Ukraine using a systematic approach to the scope and methodology that is consistent with international practice. These countries have issued new regulatory documents or requirements for improving and updating the SARs and accident analyses for their WWER plants. At all WWER units, activities to upgrade the SARs are under way.

Outlook

The IAEA will continue to provide a forum via its activities for exchange of experience and compile information on the progress made at individual units. The guidance for a proper approach to the best estimate calculation for WWER nuclear power plants will be considered in a document in the framework of the IAEA Safety Standards Series Programme.

It is expected that the IAEA, under its TC programmes, will be called on for assistance in providing guidance on upgrading SARs for WWER plants, and that this will be a major IAEA activity.

6.9. LOW POWER AND SHUTDOWN OPERATION

A number of events at nuclear power plants worldwide, as well as results from Probabilistic Safety Assessment studies for NPPs, have indicated that the events occurring during shutdown modes may contribute significantly to the overall risk associated with the NPP operation. During low power and shutdown (LPS) conditions of NPPs, the safety functions are weakened as a result of the inoperability of some safety systems, the broad spectrum of extensive maintenance work performed, the bypass of a number of protection and interlock signals, insufficient and/or unreliable measurements, the increased role of human factors, etc.

In spite of the importance of the issue, safety rules and guidelines applicable in most countries do not contain the detailed requirements concerning the safety analyses specific for shutdown conditions. Only recently have systematic analyses of the accidents during shutdown conditions been undertaken. Operational procedures for safe shutdown from all operating conditions exist at NPPs in Russia and other countries. However, the current operating procedures used to cope with shutdown accidents are insufficiently detailed and not supported by sufficient analytical results.

IAEA activities

The IAEA convened a CM on Accidents During Shutdown Conditions for WWER NPPs in November 1995 [129] and a second meeting in October 1996 to develop Procedures for Analysis of Accidents During Shutdown Modes of WWER NPPs [16]. The two meetings emphasized a deterministic approach.

In December 1997, IAEA convened a technical meeting on Safety Analysis of NPPs During Low Power and Shutdown Conditions in the framework of the TC project RER/9/046 [130]. The main objectives were to exchange information on the studies performed and countermeasures taken for coping with accidents during LPS conditions. The scope of the meeting covered: safety analysis of LPS conditions: lessons learned and measures taken for coping with accidents at LPS conditions; status of technical specifications and emergency operating procedures for LPS conditions. Probabilistic aspects with LPS conditions were also addressed.

Achievements

The IAEA technical report Procedures for Analysis of Accidents in Shutdown Modes for WWER NPPs [16] published in 1997 provides important guidance to address the issue. The procedures describe the vulnerability of key safety functions during shutdown conditions, define operational states and conditions, list the events to be considered, and establish a set of criteria for performing deterministic analysis of accidents initiated by events during shutdown conditions. The guidance applies to all WWER plants which at present are in operation and under construction.

Work is under way on resolving this issue at WWER units.

Outlook

In the field of deterministic accident analysis during shutdown conditions, efforts should be concentrated on the validation of available computer codes and applicability of their relevant thermohydraulic correlations for shutdown conditions, on the development of initiating events and criteria. In the field of plant operation, aspects covering administrative control; limits and conditions; emergency operating procedures; hardware modification resulting from analyses and experience feedback; and training of personnel before outages should be developed and strengthened.

Due to complexity of the issue and the need for effective utilization of limited resources, further co-ordinated activities and follow-up on world experience will be made available to Member States in the framework of IAEA services.

6.10. ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

Current international practice requires that operators of PWRs demonstrate their capabilities to cope with ATWS by a systematic evaluation of plants' defence in depth. Countries operating PWR plants require design consideration of ATWS events on a deterministic basis. The regulatory requirements may concern either specific mitigating systems or acceptable plant performance during these events. The prevailing international practice for performing system transient analysis of ATWS for licensing is the best estimate approach. Available system transient analyses of ATWS events indicate that WWER-1000 reactors, like PWRs, have the tendency to shut themselves down if the moderator temperature coefficient is sufficiently negative. In addition, various control and limitation functions of the WWER-1000 plants also provide a degree of defence against ATWS.

However, at the CM on Control Rod Insertion Reliability for WWER-1000 NPPs held in February 1995 it was reported [108] that the plant behaviour in case of loss of feedwater without scram relies not only on a strong negative moderator coefficient feedback but also on certain system functions such as: qualification of pressurizer safety valves for liquid discharge, availability of auxiliary feedwater, availability of BRU-K,-A for secondary side pressure control, boron injection to reach long term subcriticality in order to exclude severe consequences. Some of these were identified as safety issues [9]. For the time being, only limited studies are available indicating high peak pressure in the primary circuit which may reach the design limits, so that mitigative measures seem to be necessary. Moreover, a stuck open pressurizer safety valve can lead to a LOCA scenario, as indicated above.

A specific feature of the WWER-440 reactors is the presence of loop seals in the hot legs. Their impact on single or two-phase natural circulations and the potential for boron dilution effects are being investigated. The emergency gas removing system from reactor and SG collectors and the emergency drainage system of the primary circuit hot leg are implemented based on the research results. The boron dilution analysis was not carried out.

IAEA activities

Due to the situation described above and the fact that ATWS events are being analysed within the framework of modernization programmes to establish preventive or mitigative measures, the IAEA convened a CM on ATWS for WWER-1000 Reactors in August 1996 in Vienna [18]. The IAEA developed a guidance document to provide assistance for assessing

WWER-1000 reactors capabilities to cope with ATWS events, i.e., to assist in judging the adequacy of the reactor protection system and the value of additional defences should it fail.

The IAEA Report [18] gives special guidance on performing ATWS analyses of system transients for licensing purposes, identifies initiating events for WWER-1000 reactors and recommends the best estimate approach to the transient analysis of ATWS events. Further guidance is directed to reliability assessment of I&C related to ATWS including its common cause failure potential and the environmental qualification of systems and components necessary to mitigate ATWS.

The report also provides guidance for ATWS events in WWER-440 reactors, taking into consideration the characteristics of this reactor type and its I&C.

Achievements

In all countries operating WWERs the need for ATWS investigations is recognized and reflected in the safety improvement programmes. ATWS analysis for WWERs is not required for the licensing process in Bulgaria, Czech Republic exclusive of Temelin NPP, and Russia. Design consideration of ATWS is required if expert assessments or PSA results show that the ATWS can contribute essentially to the probability of core damage or to off-site radioactive releases (Russia) or whenever only one fast acting shutdown system exists for which the regulatory body stipulates that failure must be assumed.

Outlook

The development of a national policy on the ATWS issue in line with international practices is expected for national safety authorities. This policy should consider the systems necessary to mitigate consequences and/or requirements on acceptable plant performance in an ATWS event.

6.11. SEVERE ACCIDENT ANALYSIS AND ACCIDENT MANAGEMENT

Analyses of severe accidents have not been systematically performed at WWER units [7, 9]. The results of these analyses are needed to prepare accident management measures for prevention and mitigation of beyond design basis or severe accidents.

The FSU recommended in 1991 that ten events should be analysed for consideration of beyond DBAs [111]. The calculations have been done for Russian and Ukrainian plants. Based on the results obtained, guidance on the accident management for specific plants were developed. However, because of the absence of a systematic approach and required quality and due to their lack of scope, the results and guidance are not used in many plants.

IAEA activities

The issue has also been addressed on a plant specific basis in the framework of safety review missions and technical visits to individual plants.

Within the framework of the IAEA TC Regional Programme on Computer Aided Safety Analysis from 1985 to 1990, the methodology for severe accident studies was developed and later further developed and implemented under the IAEA TC project RER/9/004 (1991 to 1994) with Bohunice NPP V-2 as the reference plant. The computer codes STCP and MELCOR were adopted to WWER-440 design features and used for scenarios with core melt. The RELAP5 code was used for evaluation of preventive accident management measures.

Achievements

Severe accident analyses are at different developmental stages for WWER units. Lists of initiating events resulting in beyond DBAs and severe accidents are being prepared, planned or completed at WWER units.

Results of severe accident analysis, developed mostly by national/international efforts with some IAEA participation, are available which can already be applied to safety improvements on existing NPPs.

For the Paks NPP, the AGNES Project covered four kinds of severe accidents: in vessel phase, containment phase, radioactivity releases, and accident management. The Dukovany NPP, Mochovce NPP, and Rovno NPP Units 1-2 can generally use the results from Bohunice NPP V-2 (reference plant for IAEA RER/9/004 project) because of the high degree of design standardization for WWER-440/213 type plants. Severe accident studies, contemplating all the important accident scenarios and directed at the three earlier mentioned plants and the Loviisa plant, were conducted using the MAAP code and verified with the new code versions (MELCOR, SCDAP/RELAP).[7].

Accident management procedures are under development based on the available analysis results. As a part of this effort, symptom based emergency operating procedures (also covering beyond design basis conditions) have already been developed and implemented in some WWER-440/230 and WWER-440/213 units.

For the WWER-1000/320 NPPs, a set of initiating events have been evaluated, and guidelines have been established. For Temelin, selected sequences of severe accidents have been analysed. The first Temelin plant analyses, including LOCA-sequences and some transients, were performed with the aid of the MARCH3-M, CORCON-mod2 and WECHSL codes. Databases for Temelin NSSS and containment are now ready for the extensive severe accident analyses.

Outlook

The IAEA will continue to support methodological development of severe accident analysis and serve as a forum for the exchange of experience among individual plants.

6.12. PROBABILISTIC SAFETY ASSESSMENT

IAEA activities

In the past several years, national programmes for safety re-assessment have been strengthened in the countries constructing and operating these plants and several bilateral and multilateral assistance projects have been initiated to improve their safety.

These safety assessments were mostly based on deterministic reviews and engineering judgement. The results obtained have formed the basis for the programmes of safety upgradings (PSUs) and the priority actions established in the countries operating WWER and RBMK NPPs.

More recently, PSAs have been initiated for these NPPs. To date, most WWERs and some of the RBMKs have PSA studies at various levels of completion.

The IAEA is reviewing the impact of probabilistic safety assessment results on the programmes of safety upgradings [39]. It seems that most of the safety concerns have been identified by the deterministic assessments which were earlier carried out. Therefore, the role of PSA will be to prioritize actions based on their quantitative impact on plant safety.

The review deals with three types of WWERs, namely: WWER-440/230 (Bohunice NPP Units V-1, Novovoronezh NPP Units 3-4), WWER-440/213 (Bohunice NPP Unit 3, Dukovany NPP Unit 1, Paks NPP Unit 3) and WWER-1000 (Balakovo NPP Unit 4, Kozloduy NPP Units 5-6, Temelin NPP Unit 1).

The scope and the technical quality of PSAs reviewed by the IAEA vary considerably. The treatment of common cause failure and human reliability analysis is weak or not addressed at all in some studies. At those plants where it has been included, important operator actions identified by the PSA studies are not part of current plant operating procedures or training programmes. The availability of plant specific data and the overall quality of plant documentation varies substantially from plant to plant. Due to these limitations, generic Western data is used extensively in some studies. Therefore, plant specific data collection campaigns have been initiated with IAEA assistance in some plants. Another major shortcoming, particularly in the initial phase of the studies, was the lack of best estimate accident analysis for the definition of success criteria.

PSA-based risk management tools for day to day optimization of plant operation and risk monitoring are also being developed at several plants. The successful utilization of these tools will depend highly on the completeness of the PSA and adequacy of the models developed thereafter.

Achievements

In general, PSA results are being used both to define new programmes of safety upgradings and to complement existing ones. However, the differences in the scope and technical quality of PSAs performed to date limit a wider sharing of insights among plants, even among those of the same type. PSA measures and priorities defined on the basis of some PSA insights need careful consideration before they can be adopted.

Major differences can be generally related to modelling assumptions, the scope of human reliability analysis, data robustness (particularly grouping and generic frequencies of initiating events which in some cases differ up to two orders of magnitude) and design differences. It is also noted that there are some very conservative assumptions in some PSAs, and PSA results are often dominated by these assumptions. Furthermore, failure to model operator actions in the WWER PSAs have a severe impact on risk profiles. Therefore the priority of PSU measures defined on the basis of these PSA insights need careful consideration.

PSA results have also identified several PSU measures with a negligible impact to reduce the frequency of core damage. Another group of PSU measures is not modelled in PSAs either due to the limited scope of some PSAs or because the measures cannot be qualitatively considered in PSAs. Priorities for those PSU measures can only be assessed based on deterministic considerations and engineering judgement.

The reviews carried out revealed differences in the interpretation and use of generic data and modelling assumptions. Moreover, results of some PSAs are dominated by a few conservative assumptions and as a consequence the risk profile is not fully representative of the priority safety upgradings.

The IAEA organized in co-operation with the US Department of Energy (DOE) a series of PSA workshops to provide a forum for PSA practitioners, NPP operators, regulators and other safety specialists, to discuss a common approach for dealing with PSA areas which require harmonization. The first workshop was organized at the Nuclear Research Institute Rez in the Czech Republic in 1996 to address the following topics:

- Initiating Events Definition and Frequency for Primary and Secondary Pipe Breaks.
- Initiating Events Definition and Frequency for SG Pipes and Collector.
- WWER Accident Analysis including: success criteria and accident analysis results, grouping of LOCAs according to break size for each WWER design type, feed-and-bleed modelling and ATWS modelling.
- WWER PSA Modelling Assumptions including: sump filter clogging, and reactor coolant pump seal leakage.
- Turbine Generator (TG) Hall Effect for WWER-440/213 NPPs.

The results of this workshop are described in the meeting report [45].

A second workshop was held in Bratislava, Slovakia in 1997. Focus was on the development of PSA data in Eastern and Central Europe [47]. The workshop was the initiation of a project sponsored by the USDOE to establish a Reliability Database for PSAs for Soviet-Designed Reactors.

Outlook

In the framework of ongoing TC projects, the IAEA continues to assist in the peer reviews of PSAs and to serve as a forum for the exchange of information between PSA practitioners, regulators and NPP operators.

Efforts to improve the quality of PSA models and data are most important and should be pursued to allow for a wider use of PSA insights. In this context, the involvement of plant operating staff in the development of the PSA is essential to ensure that it reflects the actual plant configuration and operation.

The ultimate objective should be to establish a realistic (living) PSA model which can be used as a management tool to optimize plant operation and to support decisions on plant modifications.

7. RESULTS CONCERNING RBMKs

In 1954, a demonstration 5-MWe RBMK-type reactor for electricity generation began operation in Obninsk. Subsequently a series of RBMKs were developed representing distinct generations of reactors having significant differences with respect to their safety design features.

The RBMK reactor is a heterogeneous, pressure-tube type thermal neutron power reactor with a graphite moderator and boiling light water as the coolant with on-line refuelling capability. The heat flow diagram is a typical of one for a single circuit boiling water reactor. Refuelling at full power is accomplished by means of the refuelling machine. Under normal operation and nominal reactor power, generally two fuel channels are refuelled per day.

The RBMK design philosophy resulted on the one hand in inherent safety features such as:

- Intense natural circulation ensures adequate heat removal from the core in case of accidents involving equipment failures. Furthermore, there is no need for emergency water injection in the circuit in such events because water inventory in the circulation circuit and drum separators is sufficient to provide residual heat removal during up to 60 min., even with the loss of all water supply sources.

- The main circulation circuit consists of two loops, which mitigates LOCA accidents and lowers demands on emergency core cooling systems.

- High heat accumulation capacity of graphite stack provides for slow core heating in case of hypothetical severe accidents involving the loss of all heat removal systems.

On the other hand, it resulted in significant deviations from current safety standards and practices:

- The main circulation circuit contains a lot of pipework with many welds, bends, headers, valves, which significantly increases the scope of in-service inspection and maintenance.

- A part of the main circulation circuit is located outside the confinement area, which may result in unacceptably high off-site doses in case of ruptures of the reactor coolant circuit in these compartments.

- Due to the high initial temperature of graphite and high heat accumulation capacity of graphite stack, cooldown of an RBMK reactor takes considerable time.

- There is a lack of a full containment.

All operating RBMKs were connected to electric power grids during the period 1973 (Leningrad NPP Unit 1) to 1990 (Smolensk NPP Unit 3). Today, there are 14 RBMK power reactors in operation in three States: 11 units in Russia, one in Ukraine, and two in Lithuania. The gross electric power rating of all but two RBMKs is 1000 MWe; the exceptions are the two units at Ignalina NPP in Lithuania which are rated at 4200 MW (originally 4800 MW) thermal power, corresponding to about 1300 MWe (originally 1500 MWe) gross.

The grouping of the RBMK NPPs in generations is generally made based on differences in systems of core control emergency core cooling, reliable power supply and accident localization.

Six plants (Leningrad NPP Units 1-2, Kursk NPP Units 1-2 and Chernobyl NPP Units 1-2) are generally considered as the first generation units because they were designed before 1973 when the first standards on the design, construction and operation of Nuclear Power Plants (OPB-73) were introduced in the FSU. This generation of RBMK units was designed, constructed and operated mostly in accordance with general industrial standards and rules. Special standards were used with respect to radiation protection and utilization of radioactive materials.

Second generation RBMK units (Leningrad NPP Units 3-4; Kursk NPP Units 3-4; Ignalina NPP Units 1-2; Chernobyl NPP Units 3-4; and Smolensk NPP Units 1-2) were designed in accordance with the OPB-73 standards. These standards introduced well-known safety principles such as multiple barrier protection and single failure criteria. They required accident analysis to be performed for engineered features and organizational measures to ensure NPP safety based on examination of accident initiating events. The ultimate DBA, which was considered at that time, was an instantaneous break of a pipeline of the main circulation circuit with coincidence of a single failure of safety systems. Special systems dedicated to perform safety functions were required by these standards.

The Ignalina NPP is included in the second generation group, however, it contains safety features which are different in their technical solution from those of the other second generation units.

In 1982, revised safety standards were introduced into practice in the FSU. The basic safety requirements which were contained in OPB-73 remained practically unchanged. In addition to the previous standards, OPB-82 required that the technical means and organizational measures stipulated by the design of the NPP must guarantee safety for any event, taking into account the design, together with a single failure (independent of the initial event) at an active or passive element (containing mechanical moving parts) of the safety systems. Accidents for which technical safety measures were not provided by the design were considered to be hypothetical. For these hypothetical accidents, it was required that plans for protection of personnel and the public be developed and implemented in accordance with the requirements imposed by other regulatory documents, e.g., on health regulation.

Second generation RBMK units in many aspects meet the OPB-82 requirements.

In 1988 new safety regulation OPB-88 was put into practice in the FSU. This regulation took into account not only the experience accumulated in the FSU but the international experience as well.

One RBMK NPP unit (Smolensk NPP Unit 3), which is considered as belonging to the third generation, has been built taking into account these standards. Additional design changes are now being incorporated in the construction of Kursk NPP Unit 5, which is considered as belonging to the fourth generation of RBMK NPPs.

The list and location of RBMK NPPs in operation is shown in Table 7.1.

TABLE 7.1. RBMK REACTORS IN OPERATION

Country	Plant	Unit/Model	Start of Operation
Lithuania	Ignalina	1/RBMK 1300	1983
		2/RBMK 1300	1987
Russia	Kursk	1/RBMK 1000	1977
		2/RBMK 1000	1979
		3/RBMK 1000	1983
		4/RBMK 1000	1986
	Leningrad	1/RBMK 1000	1974
		2/RBMK 1000	1976
		3/RBMK 1000	1980
		4/RBMK 1000	1981
	Smolensk	1/RBMK 1000	1983
		2/RBMK 1000	1985
		3/RBMK 1000	1990
Ukraine	Chernobyl	3/RBMK 1000	1981

7.1. IDENTIFICATION AND RANKING OF SAFETY ISSUES

The IAEA's RBMK safety programme aimed to consolidate results of various national, bilateral, and multilateral activities and to establish international consensus on required safety improvements and related priorities. It assisted both regulatory and operating organizations and provides a basis for technical and financial decisions. A wide range of activities were covered, and since 1992, a number of reviews and assessments have been conducted. Smolensk NPP Unit 3, Ignalina NPP Unit 2 and Leningrad NPP Unit 2 have served as RBMK reference plants for the programme.

The IAEA conducted a first review of safety improvements proposed for RBMKs in October 1992 [131]. Initial activities were aimed at the understanding of the safety concerns underlying the proposed safety improvements. In June 1993, a safety assessment of design solutions and proposed safety improvements of Smolensk NPP Unit 3 [132] was organized. It was conducted by an international group of experts and IAEA staff over a period of two weeks at the plant site. Smolensk NPP Unit 3 is the most advanced of the operating RBMK plants and its design incorporates safety improvements identified from analyses of the Chernobyl accident and other studies. A similar review was performed for the Ignalina NPP units in October 1994 [133].

Additionally, the IAEA's ASSET reviewed the plant specific operational experience at all RBMK sites. An OSART mission was also conducted at Ignalina in September 1995 [134].

Activities of the IAEA's RBMK safety programme were co-ordinated with those of an international consortium on the Safety of Design Solutions and Operation of Nuclear Power Plants with RBMK Reactors established under the auspices of the EU. With the completion in 1994 of the safety reviews of Smolensk NPP Unit 3 and Ignalina NPP Unit 2, these programmes reached important milestones. To make results available to the international technical community, the IAEA convened a technical meeting in May-June 1995 [135]. Results of both the IAEA and EC programmes were presented, thereby reflecting the great number of tasks accomplished by the international experts and Russian and Lithuanian organizations to review the safety of RBMK nuclear power plants.

Both projects produced a large number of recommendations for enhancing the safety of RBMK plants. Most of them correlate with the measures already included in national programmes for RBMK units which are under way in Russia, Lithuania, and Ukraine.

Based on the initial stage of its RBMK programme, the IAEA prepared a consolidated list of design and operational safety issues for RBMKs “RBMK Nuclear Power Plants: Generic Safety Issues” [8]. The report was prepared on the basis of the safety reviews of the Smolensk NPP Unit 3 and Ignalina NPP. Therefore, it is primarily applicable to RBMK NPPs of this vintage. Whenever applicable, reference was also made to RBMKs of other generations.

In 1997 this work was extended to include the design aspects of RBMKs of the first and second generations. Relevant information on the safety improvements considered and/or implemented for RBMKs of this vintage was obtained by the IAEA from a technical visit at Leningrad NPP Unit 2. Also, during this later stage of the IAEA programme use was made of the results of other international activities, particularly those sponsored by the EC for first and second generation RBMKs.

A total of 59 safety issues related to RBMK design and operation were identified. Most of them are of a generic nature and therefore relevant to all RBMK generations. Those safety issues related to design aspects were ranked according to their impact on defence in depth at the plants [8, 136], see also Annex 4.

All findings and recommendations from the various technical meetings, safety reviews of Smolensk and Ignalina NPPs, ASSET reports and the results of the EC work were collected in the IAEA-EBP database. Also included was plant specific safety information provided by the main design institute for RBMK reactors in Moscow and by the RBMK NPPs.

Achievements

As a result of the above-mentioned activities, different aspects of RBMK NPP design and operation were subject to international review based on the IAEA NUSS publications, Russian regulations and national and international practices.

International consensus was reached on a total of 59 safety issues in the following seven topical areas: core design and core monitoring; instrumentation and control; pressure boundary integrity; accident analysis; support and safety systems; fire protection; and operational safety.

It was recognized that exactly how and when the identified safety issues were addressed was a matter to be resolved between the operating organization and the regulatory body. Therefore, these identified safety issues did not replace a comprehensive, plant specific safety assessment which needed to be performed in the framework of the national licensing process.

Each design safety issue was ranked using the definitions specified in Section 2.5.1 of this report, taking into account the safety improvements already under way or implemented.

Over the past decade, a considerable amount of work has been carried out by designers and operators to improve the safety of RBMK reactors and to eliminate the causes that led to the Chernobyl accident. As a result, major design and operational modifications have been implemented. However, certain safety concerns remain, particularly regarding the first-generation units.

There was common agreement that implementation of measures included in the national upgrading programmes and those recommendations made in the framework of the IAEA EBP can bring important safety enhancement of RBMK NPPs.

Outlook

Further activities are required at the national, bilateral and multilateral levels to realize the safety upgradings and those recommendations made in the framework of the IAEA and other international projects; the necessary funding is, however, required.

In-depth safety analyses (ISA) are now being performed for some units, based on current safety norms and rules and best international practice, to identify backfitting or compensatory safety measures. An SAR was recently completed and reviewed for Ignalina NPP, and similar work is being started for Kursk NPP.

Further international co-operation, including IAEA participation, will be required on plant specific evaluation: safety assessment, precursor analysis and operational experience. It will bring openness and will help concentrate the efforts of all parties involved on resolving safety issues.

7.2. REVIEW OF SAFETY IMPROVEMENTS

7.2.1. Safety assessment of design solutions and proposed improvements to Smolensk NPP Unit 3 RBMK (June 1993)

In the framework of the IAEA Extrabudgetary Programme on the Safety of RBMK Nuclear Power Plants, a progress review meeting was convened by the IAEA at the Smolensk Training Centre at Desnogorsk, Russia, from 7 to 18 June 1993. The Centre is some 4 km from the site of Smolensk NPP Unit 3, the most advanced (third) generation of RBMK plants which started operation in mid-1990. A number of safety improvements identified on the basis of the analysis of the Chernobyl accident and other studies have already been incorporated into the unit.

The objective of the meeting was to further discuss the findings and recommendations as documented in the technical document on "Safety Assessment of Proposed Improvements to RBMK Nuclear Power Plants" [131], and in particular their applicability to Smolensk NPP Unit 3. Smolensk NPP Unit 3 and Ignalina NPP Unit 2 are the reference plants for the IAEA programme, because these units are the RBMK flagships (3rd generation and 2nd generation with some modifications). The meeting covered three of the areas reviewed at the IAEA CM held in Vienna from 27 October to 5 November 1992, namely: core monitoring and control, components integrity and accident mitigation. In addition, the following areas were also reviewed: support and safety systems, instrumentation and control, seismic safety, fire protection and operational safety.

Twenty-nine safety experts from Canada, Finland, France, Germany, Italy, Japan, Spain, Sweden, Switzerland, United Kingdom and USA and ten from the IAEA secretariat participated in the review. Most of the participants were involved in RBMK safety reviews in the framework of bilateral and multilateral assistance programmes and were therefore familiar with the topical areas assigned to them for review. Altogether sixty RBMK specialists were present, including 57 from Russia, two from Ukraine and one from Lithuania, along with 14 interpreters.

The original Smolensk NPP Unit 3 TOB was used as the basic document for the IAEA review. The TOB information was complemented by results of specific safety analyses and other documentation related to plant design and conduct of operation. No specific additional documentation was prepared for the meeting. Members of the IAEA review team also conducted plant walkdowns specific to their respective areas of review.

Core monitoring and control

The Smolensk NPP Unit 3 startup was delayed following the Chernobyl accident in order that safety modifications required from the analysis of that accident could be incorporated. Some of these improvements are still under way and further measures to improve the control and monitoring system and to reduce the safety significance of the operating reactivity margin (ORM) have been proposed.

Pressure boundary integrity

The review of the main circulation circuit included fuel channels, fuel channel inspection, refuelling operation, design strength criteria, primary piping inspection, application of the leak before break concept and primary circuit valves design. The review of the control and shutdown circuit was focused on the design and inspection of the special channels, the cooling system of the special channels and the drive mechanism for the control and shutdown rods.

Only small progress was reported on the implementation of the recommendations of IAEA-TECDOC-694 mainly due to the limited availability of resources. It was concluded that these recommendations were still valid unless stated differently and should be implemented as soon as possible. At present, work is being carried out mainly on testing of automated in-service inspection equipment development, on installation of advanced leak detection system, on fracture mechanics analysis, and on leak before break analysis.

Accident mitigation

A principal focus of the review was to investigate the possibility of multiple channel tube ruptures during accident sequences. This was highlighted as a need based on IAEA-TECDOC-694 owing to the fact that the design basis for the reactor cavity pressure protection system is a single pressure tube rupture. The actual capacity installed at Smolensk NPP Unit 3 is to handle nine simultaneous pressure tube failures based on conservative analysis. But the reviewers were concerned as to whether there were other sequences which could lead to a larger number of pressure tube failures, such as those involving a group distribution header (GDH) which feeds 42 fuel channels or even involving breaks postulated in the pump discharge header. Such sequences were discussed in some detail, both for full pipe break and for partial pipe break size where stagnation flow in the channels may be particularly hazardous.

Support and safety systems

In the area of safety systems and their support systems, Smolensk NPP Unit 3 is characterized by:

- a good level of redundancy (generally 3 times 50%) of all safety systems
- the same level of redundancy for the safety support systems

- a good level of redundancy also of the safety related systems
- a physical segregation between redundant trains of equipment.

Instrumentation and control

The RBMK instrumentation and control (I&C) systems are extensive, measuring and recording nearly all plant parameters, there is also an extensive plant protection system. The review has enabled the generation of a description of some of the main systems of the plant and allowed their functions to be established.

The main functions of the I&C systems are interconnected and use common systems and equipment in shared locations.

Some parts of the system are well segregated; the ECCS system is a good example of this. In other cases there is unfortunately common cable routing or use of a common unit that would appear to invite common cause failures.

The flux based automatic protection system is comprehensive provided the ex-core detectors can see events at the core centre. For the Smolensk NPP Unit 3 nine zone local automatic control system, the local power increase due to a withdrawal of one of 40 potentially “dangerous” manual control rods will not be detected. Hence, fuel damage cannot be excluded.

A sample of the logic was examined in detail, and the implementation of the logic is a good design practice with the appropriate use of normally open and normally closed relay logic to ensure fail safety.

The SKALA system for data collection, processing, recording and display is comprehensive, the equipment is however showing its vintage.

Fire protection

In the area of fire protection the primary concern of the review was to ensure that a fire will not prevent achieving safe shutdown or maintaining core cooling. A secondary concern was to minimize the impact of fire on operation of the plant, with particular emphasis on associated impact on safe shutdown.

Seismic safety

The review of the seismic safety of Smolensk NPP Unit 3 included the following subjects: evaluation of seismic input and soil effects, response and capacity of structures, response and capacity of equipment and piping. For the seismic safety review, this was the first meeting and the objective was to reach an understanding of the database, criteria and methods used to evaluate and improve the seismic safety of Smolensk NPP Unit 3 and in general RBMK type NPPs. The meetings in Moscow and Desnogorsk with the specialists of VNIPIET, RDIPE and NNAEP as well as the one day walkdown of the plant were sufficient to attain this objective.

Smolensk NPP Unit 3 is located in a region where destructive earthquakes are uncommon. Effects of large but very distant earthquakes as well as small local earthquakes contribute to the design basis.

The structures and systems of Smolensk NPP Unit 3 are qualified for 0.05g and 0.1g for distant and local seismic effects respectively.

Based on a plant walkdown during the mission, no major problems were identified for seismic capacity of structures or equipment at this low seismic level. Since 1993, there have been more detailed efforts to assess the seismic threat to RBMK plants including the independent review of Ignalina NPP SAR.

The Russian specialists have already performed a significant amount of high quality work in the analysis and testing of safety related structures and components and, in particular, the core assembly.

Operational safety

Evaluation of operational safety has been found by many NPPs to be a valuable contribution to their ongoing programme of improvement. The evaluation at Smolensk NPP, principally directed to Unit 3, was intended to identify the strengths of the plant's programmes and to point out areas where improvements could be made.

In the areas reviewed it has been established that the management and operational safety programmes have many of those elements necessary to ensure satisfactory performance and safe operation.

There are, however, a number of areas where operational safety can be improved to achieve a better alignment with good international practices. The overriding concern is that the highly prescriptive approach, with many of the practices being dictated by rules prepared by organizations external to the plant (as is normal in Russia), may have resulted in a passive attitude, which did not question practices. A self-appraising, critical culture, for example, would identify where there are gaps in the standards, where technical instructions do not adequately apply to the abnormal situation, or where long-accepted practices could be improved.

Development and implementation of symptom-based emergency procedures going on in the framework of the bilateral co-operation between Russia and USA are of great importance and should be continued.

7.2.2. Safety assessment of proposed modifications for the Ignalina NPP (October 1994)

At the request of the Government of Lithuania, the International Atomic Energy Agency organized a CM on the Safety Assessment of Proposed Modifications at the Ignalina NPP.

The objective of the meeting was to further discuss findings and recommendations presented in the two technical documents [131, 132] and their application to the particular situation of the Ignalina NPP. Since design information and a series of proposed modifications for Ignalina NPP had been prepared by the main RBMK designer, Research and Development Institute for Power Engineering (RDIPE), it was considered appropriate to conduct the meeting in two parts. The first, on 17-22 October 1994 at the RDIPE headquarters in Moscow and the second, on 24-28 October 1994 at the plant site in Lithuania.

Twelve international experts and IAEA staff participated in the meetings, together with a large group of RDIPE specialists and plant staff. Six Ignalina NPP staff members participated in the discussions at RDIPE and 11 RDIPE specialists participated in the discussions at Ignalina NPP.

Specific documentation detailing the topics to be reviewed was prepared by RDIPE at the request of the IAEA.

Information from a level 1 PSA carried out for Ignalina NPP in the framework of a Swedish-Russian-Lithuanian project (Barselina) was also available.

Core monitoring and control

The core monitoring and control system of Ignalina NPP Unit 2 was reviewed on the basis of information provided by the reactor designer prior to and during the meeting in Moscow and supported by the discussions with the Ignalina NPP staff at the site of the plant. The review focused on the main design differences between Ignalina NPP Unit 2 and Smolensk NPP Unit 3 [132].

Basically, the measures taken in the Ignalina NPP to improve core safety after the Chernobyl accident were the same as in Smolensk NPP and the rest of the RBMK-1000s. Slightly different measures were taken to reduce the void reactivity coefficient.

Pressure boundary integrity

The pressure boundary integrity review covered the fuel channels, the main circulation systems; the reliability of the components and the application of the LBB concept. Special attention was paid during the review at the plant to both the fuel channels and main circulation system in-service inspection programmes, techniques, procedures, equipment and results.

At the time of the IAEA review, there was not enough data from the fuel channel post irradiation tests to accurately predict the remaining lifetime of the fuel channel tubes of Ignalina NPP Units 1-2.

Seismic qualifications of Ignalina NPP fuel channel, fast shutdown rods and refuelling machine had not been performed at the time of the IAEA review.

Accident mitigation

The accident mitigation review covered the design basis for the Ignalina NPP (i.e., the regulations applied to the safety system design, and to the assessments of these systems), the analysis capability (i.e., the analysis methods, assumptions, and experimental support used for the safety assessments of RBMK plants), the results of some DBA analyses, and the application of these results in emergency operation procedures.

Safety and support systems

The review of safety and support system was focused on a few systems in which previous reviews had identified safety deficiencies. The review of safety systems covered the ECCS and accident localization system (ALS). Support systems included service water, intermediate cooling and electric power supply.

Instrumentation and control

The instrumentation and control group reviewed a total of 22 different proposals for upgrading or improving various aspects of the I&C system. The results of the review supported several proposals including: changes to the shutdown system mechanics, provision of a second set of protection equipment; additional trip parameters and the replacement of the TITAN computer system.

The priority of the modifications was discussed among the parties and a preliminary assessment made on the basis of the available information.

Summary of the 1994 Consultants Meeting

Until 1991, standards from the FSU were in force for the Ignalina NPP. Since then a national system of regulations has been under development, but these are not yet fully established. In 1997, the Russian regulations OPB-88 and PBYa-89 were replaced by Lithuanian regulations. These are based on the original Russian regulations and IAEA standards.

The monitoring of the fuel channel condition, particularly with respect to the closure of the graphite gap and the hydrogen content in the tube, was considered essential in defining the lifetime of the reactor components. This may have consequences for the programme of safety modifications. It was recommended that an extra effort be made to determine the present conditions of the fuel channels and to periodically monitor the relevant safety parameters. The review indicated that a decision about replacement of fuel channels needed to be made soon.

The reviewers of the 1994 consultants meeting concluded that an integrated approach was needed for addressing all the issues associated with the upgrade of the control and safety systems, including a review of the existing arrangements.

The new Ignalina NPP SAR and its review

Work is currently under way to resolve the safety issues identified during the IAEA 1994 consultants meeting on Ignalina NPP; important achievements have been made since that time.

In the years 1994-1997 an in-depth safety assessment of the Ignalina NPP was carried out as a condition of the Grant Agreement between the Lithuanian Government, Ignalina NPP and the EBRD to fund a project of safety upgrades. An SAR was prepared by the plant with the aid of Western engineering organizations and the reactor chief designer RDIPE. A thorough on-line review of the SAR was carried out by a team of Eastern and Western technical safety organizations. The project was unique because it was the first undertaking to produce a plant specific Western-style SAR, with a common review by Eastern and Western safety experts [137, 138].

The Ignalina NPP has responded to the SAR and the review results within the framework of a “Safety Improvement Programme 2”. Major safety issues have to be resolved before a regular license will be granted to the Ignalina NPP Units. For Ignalina NPP Unit 1 the licensing process is expected to be concluded by mid 1999.

7.2.3. Technical visit to the Leningrad NPP Unit 2 (May 1997)

To collect information about plant specific safety upgrading programmes and their correlation with the above mentioned report a technical visit to Leningrad NPP in Russia was organized on 12-16 May 1997. The objectives of the visit were to review the applicability of the generic safety issues identified for Smolensk NPP Unit 3 and Ignalina NPP Unit 2, to identify the plant specific status with respect to the generic safety issues, and to obtain the information on the scope and degree of implementation of the safety upgrading programmes.

As had been stated at the exit meeting with the plant management, the IAEA team noted with satisfaction the scope of work performed and planned. However, it was stressed that whether or not the improvements will bring the Leningrad NPP to an acceptable safety level is a national decision to be taken by the Russian Regulatory Authority (GAN).

Core monitoring and control

At the time of the visit (May 1997), Unit 2 had introduced 200 new fuel elements containing burnable poison (erbium) dispersed in the fuel matrix. According to information provided to the IAEA by June 1998, 633 new fuel elements had been installed. The measure is a relevant safety improvement as it will lead to a decrease in the reactivity void effect even if the number of additional absorbers are reduced to zero. It will also decrease the safety significance of the operational reactivity margin (ORM).

I&C

Major improvements have been achieved by separating the reactor control and protection into three distinct trains. The core control and monitoring system now uses both in-core and ex-core neutronic sensors.

Components integrity

All fuel channels of Units 1 and 2 have been replaced. Analytical predictions of tube/graphite gap have shown good agreement with the actual situation encountered during retubing. In-service inspection is being improved and LBB concept application is progressing with the help of acoustic equipment provided by Japan; other leak detection systems are being developed and installed. Nevertheless, implementation of the analytical and experimental elements of the LBB concept is progressing at a slower pace.

Systems

Important improvements have been introduced in the first phase of reconstruction of the emergency core cooling system. These include replacement of the GDH and installation of check valves at each GDH; installation of three ECCS headers for each half of the reactor. The water capacity of the drum separators has been increased by a factor of 2.5. Additional venting capacity was introduced to cope with the break of up to nine pressure tubes.

Two new buildings, one for Unit 1 and one for Unit 2, are being constructed to segregate the ECCS in three trains and to install accumulators and new diesel generators. The buildings will also contain the ALS occupying five floors. A confinement protection was constructed around the reactor building of Unit 1 to provide leaktightness. The IAEA was informed that since the visit, the work was also realized in Unit 2.

Fire protection

Important improvements in fire prevention and mitigation have been implemented. Measures have been taken to reduce unnecessary fire loads. The feedwater pumps in the turbine hall have been protected and water guns to cool the structures have been installed. A number of additional measures are planned. Assistance in this area has been provided by Finland.

A plant walkdown revealed good housekeeping conditions.

Probabilistic and deterministic safety analyses are being performed by RDIPE and the Leningrad NPP for its Unit 2 with assistance from Sweden, UK and the US. An on-line independent review of the analyses is being performed by GAN SEC with international participation by STUK, Finland and GRS, Germany. In addition, an in-depth safety analysis is being planned, similar to the Ignalina NPP study, in co-operation with Sweden, UK and the US.

7.3. SELECTED SAFETY ISSUES

7.3.1. Reactivity control and shutdown systems

As demonstrated during the Chernobyl NPP Unit 4 accident in 1986, the RBMK design had some serious deficiencies related to its reactivity control and shutdown system. Some of the more significant concerns were: the positive scram effect of the control rods in case of their moving down from the topmost position, a large positive void coefficient and slow insertion of negative reactivity.

Since the accident, a number of modifications have been made to all RBMKs to improve the ability to control the power and rapidly shut down the reactor. Additional changes and refinements are still being considered, the most significant being an additional shutdown system.

IAEA activities

a) Generic activities

The initial IAEA activities under the EBP focused on the evaluation of the objectives and basis for all of the modifications made since the Chernobyl accident and to determine the status of the modifications at each of the RBMK reactors. In order to do this it was first necessary to more fully understand the basic core design of the RBMKs. Reviews in this area focused on the basic core design and the analytical tools used during the design and to determine the adequacy of the various modifications.

One of the more significant issues was the large positive void coefficient and how it was reduced following the Chernobyl accident. A specific concern was the relationship between the positive void coefficient and the ORM (a measure of the worth of the control rods inserted in the reactor). Also reviewed was the relationship of the positive void coefficient relative to the number of additional absorbers and also with respect to the number of fuel assemblies with increased fuel enrichment. At a review meeting organized by the IAEA in 1995, consensus was reached on the appropriateness of codes and methods in use to determine void reactivity feedback [139]. There are still some outstanding questions relative to the value of the positive void coefficient during low power operation. A further study was commissioned to the Kurchatov Institute to quantify the void and other reactivity effects at low power conditions [140].

Another area of concern is the fact that voiding the special channels associated with the CPS would result in an increase of reactivity which cannot be coped with by the CPS. There have been some modifications to reduce the voiding effect. These safety improvements include the step-by-step introduction of control rods of a new skirt design and a split of the control and protection cooling system in two halves. However, the splitting of the CPS cooling system into two halves envisaged in 1995 has not yet been realized in any RBMK.

Russian specialists pointed out that: in case of voiding of the cooling channels associated with CPS, an increase of reactivity could be possible only if the shutdown system fails to be actuated by one of three independent signals related to water level/pressure decrease. Moreover, during CPS channels voiding time (120s which in ten times exceeds total rod insertion time 12s) two more independent signals related to period and power increase are generated. So in this case five independent signals for actuating shutdown system are generated.

Based on this, the accident concerning the CPS channels voiding under conditions of criticality is not taken into consideration. At the same time existence of large (~4.5 ß) positive reactivity coefficient requires measures on splitting of CPS into two halves and on replacing old design absorption rods (2091-01) with the new ones (2477-01 or KRO). A complex of these measures will eliminate the effect of CPS channels voiding.

Because of the large size of the RBMK core, there were concerns relative to the ability to adequately monitor and control local areas within the core. The reviews focused on the type and location of the core monitoring instrumentation and the process used for controlling the reactor during normal operation.

Depending on the core design, the core is divided into 7-12 regions for the accommodation of the reactor control and protection system (RCPS) local automatic control (LAC) subsystem and a local emergency protection (LEP) subsystem. These systems are designed to automatically stabilize the main harmonics of the radial-azimuthal power density distribution and to protect the reactor from local power peaks exceeding given operational or safety limits of critical core parameters such as linear heat rate, critical power ratio and potentially leading to fuel damage. The subsystems are driven by signals of the in-core instrumentation. The axial neutron fields are controlled by shortened absorber rods which are inserted into the core from the bottom of the reactor (32-40 rods).

During these reviews the various types of control rods and the speed with which the control rods can be inserted have also been investigated. One of the concerns that came out of these reviews is the lack of a fully independent second shutdown system for the RBMKs. Consensus was reached on the need for a fully independent additional system. Topical meetings to discuss the attributes and needs of a second shutdown system for RBMKs were held by the IAEA. Various design options were reviewed against safety requirements on shutdown effectiveness and means, independence, diversity, rate of shutdown, reliability and testability. Design work is continuing but little progress has been made on the actual implementation of a system, because the technical feasibility study is not yet complete.

In the meantime, the need to have a second, fast acting, independent and fully diverse shutdown system in place in the RBMK reactors was demonstrated by ATWS analyses performed in the course of the production of the Ignalina NPP SAR and its independent review [137, 138].

To consolidate and review results of the studies conducted on the causes of the Chernobyl accident, a meeting was organized by the IAEA in Moscow in 1996. Conclusions of this review have been presented at the "International Forum, One Decade After Chernobyl" in 1996 [141].

b) Plant specific activities

Since all the RBMKs have basically the same core design characteristics, very little remained beyond the generic concerns that were reviewed at each site. The focus of the reviews at the sites was to obtain the specific status of the modifications, and to determine their plans for additional modifications. This also allowed the IAEA experts to obtain general information relative to the capabilities of the plant staff to review the basis for the various modifications and specific analysis.

One area of specific interest that was addressed during the IAEA review at Leningrad NPP was the use of burnable poison (erbium) dispersed in the fuel to eliminate or reduce the need to control the positive void coefficient by the ORM concept (see Section 7.2.3). The most important feature of this modification is that ORM would no longer play a significant role in the value of the void coefficient. At the Ignalina NPP the use of "erbium" fuel was started in 1995, and by mid-1998 some 500 and 900 new fuel assemblies were loaded in Units 1 and 2, respectively.

Achievements

Consensus was reached that the understanding of the causes of the Chernobyl accident is sufficient to ensure the adequacy of measures taken to eliminate its reoccurrence.

There is now general agreement that a fully independent additional shutdown system is required for RBMK plants of all generations. The requirements to be fulfilled by the new system should be those of the IAEA safety standards.

Because of the high safety significance of the additional shutdown system and of the considerable time lag necessary for the development, assessment, testing and implementation of such a system, the independent reviewers of the new Ignalina NPP SAR have required that compensatory measures be implemented in Ignalina NPP until the additional system is in place. Ignalina NPP has responded to this requirement by the development of a compensatory system which has been installed in Unit 1 during the outage in the summer of 1998.

Deficiencies in the analytical tools used for analysing core behaviour have been identified and efforts are under way on the development of 3D complex neutronic and thermohydraulics computer codes.

Efforts have been directed at the development of 3D tools for analysing neutron fields, coolant density and temperature distribution of fuel and graphite.

In this context it has to be mentioned that some achievements were registered in the meantime. For example, coupled 3D physics/thermal-hydraulics codes are now in use for RBMK (as well as for WWER), some of which were developed in the framework of bilateral scientific-technical co-operation between Russia and Germany. These activities started in the early 1990s by adaptation, validation and application of German LWR codes and significant progress has been made to date. Work in this area is being continued.

There was general consensus among the experts that for future investigations, if any, the availability of a consistent set of input data and initial conditions are mandatory and a standard benchmark calculation based on a consistent database is desirable. However, it was noted that care must be taken in interpreting existing instrument readings and records because of the time delays and measurement errors involved.

The procedures used at RBMK sites to measure the void reactivity coefficient were found adequate in general. However, recommendations for improvements have been identified. These included the use of 3D codes to represent each fuel channel.

Plant specific safety analyses are required. The SAR and the review work carried out for the Ignalina NPP are a good example of the analyses required. However, the scope of the work was substantially reduced as compared with a typical Western SAR.

Outlook

In general, core issues are well understood but much work is still required to implement the agreed safety modifications at each RBMK plant.

The coupled neutronic and thermal-hydraulic methods in use for void reactivity coefficient (VRC) predictions need further review aiming to eliminate reliance on the measured VRC in the future. A comprehensive assessment of the uncertainty of the measured VRC should be undertaken and should include random and systematic components. It is believed that the ongoing development of the erbium poisoned fuel cycle will also contribute to reducing or eliminating the safety significance of the ORM concept.

Further effort is required to reduce the reactivity effect caused by a loss of coolant in the CPS channels. By an introduction of control rods of a new design characterized by a dysprosium poisoned skirt around the tip of the control rod this effect will be halved. Control rods of the new design have been irradiated in several RBMKs.

Development of a fully independent and diverse additional shutdown system is currently under way for all RBMK reactors. By June 1998, the technical feasibility study had not yet been completed. Therefore, the actual implementation of the system has not been started. An independent assessment of the additional shutdown system's capability should be made before its installation. The EC has been providing assistance and continuation of this assistance is under consideration.

Further effort is required to develop 3D computer codes with adequate thermal-hydraulic feedback to properly take into account the spatial interaction of the neutron fields with the fields of temperatures and water density in the core. It was recommended that the methodology in use to generate cross-section libraries for the 3D codes be improved and validated and that the 3D code development and validation efforts which are currently under way be continued to provide improved calculation tools in the area of core design.

Additional validation of the computer codes in use is still necessary and uncertainty analyses should be performed to determine the expected range of key parameters as a result of uncertainties in plant data and calculation methods.

7.3.2. Fuel cooling in emergency conditions

The lack of an adequate emergency cooling system is one of the most significant deficiencies of the first generation RBMKs. The design basis for these early generations is limited to a small break in selected locations of the reactor coolant piping. The design basis also assumes that normal electric power is available.

There are concerns relative to the ability of the ECCS of all generations to provide essential cooling water to those portions of the core where it is needed the most. The ability of the protection system to identify the need for ECCS and automatically respond in a timely manner to a wide range of LOCAs and interface LOCAs has also been questioned.

IAEA activities

a) Generic activities

During the first review meeting in Vienna in 1992 there was considerable discussion relative to the design and operation of the ECCS for all generations of RBMKs. Focus was on the planned modifications to install a full capacity ECCS for the first generation NPPs. During these discussions several concerns were raised relative to the design basis for the ECCS, the qualification of the equipment to operate under accident conditions, and most significantly, concerns relative to the separation and diversification of the ECCS.

Further reviews revealed shortcomings in the provision of long term heat removal. Concerns were also raised regarding the reliability of the isolation equipment of pipes connected to the primary coolant system.

b) Plant specific activities

The reviews at the Smolensk and Ignalina NPPs focused on options to improve the design and operation of the ECCS. A technical review by IAEA experts who visited Leningrad NPP Units 1-2 confirmed that important safety improvements are being implemented. These include the reconstruction of the safety systems and construction of new buildings, one for each unit, to segregate the emergency core cooling system and an ALS capable of mitigating the consequences of a rupture of several pressure tubes. Of particular concern during this visit was the recognition that even though some modifications had already been installed, the full service ECCS would not be complete until 2001 or so.

Achievements

The IAEA reviews provided an international forum to review the design and function of the ECCS against modern international standards. As a result of these reviews, the main concerns and required safety improvements have been agreed upon and design and operational modifications are being implemented. For Leningrad NPP Units 1-2 these include the replacement of the GDH and installation of check valves at each GDH, and the installation of three emergency core cooling headers for each half of the reactor.

Outlook

The key outstanding concern in this area is the completion of the ECCS upgrades at the first generation plants, Leningrad NPP Units 1-2 and Kursk NPP Units 1-2. Furthermore, there are additional recommendations for modifications and upgrades at the second and third generation plants to improve the separation and diversification of the ECCS and its actuation logic [8].

7.3.3. Pressure boundary integrity

Main safety concerns in the area of pressure boundary integrity are related to the fuel channel integrity, break of critical components (including LBB concept application, seismic design, ageing assessment) and in-service inspection.

7.3.3.1. Fuel channel integrity

The reviews performed within the framework of IAEA Technical Co-operation and Extrabudgetary Programmes have found that fuel channel integrity in the RBMK NPPs is an issue of high safety concern [8]. To date, three single fuel channel ruptures have occurred due to water flow blockage or power imbalance at the operating RBMK plants. Fuel channel rupture results in release of radioactivity to the reactor cavity and may lead to a release of radioactivity to the environment if the confinement safety system does not function properly. A multiple fuel channel rupture exceeding the venting capacity of the reactor cavity overpressure protection system poses a major threat to plant safety since it may lead to reactor cavity overpressurization and consequently develop into a severe accident. Probabilistic evaluations by Russian specialists indicate that the probability of independent multiple pressure tube ruptures is smaller than 10^{-8}/reactor-year [142].

Further, due to uncertainty in the design stage, the gas gap between the fuel channel pressure tube and the graphite blocks closes after approximately 17 years of plant operation, according to the experience gained at Leningrad NPP Units 1-2. The duration of the closure process also depends on the material conditions during manufacturing. Unit specific measurements are required to adequately predict the time point of gas gap closure. There is no safety justification available for continuing to operate the plant with gas gaps closed because such condition is not considered allowable by the designer. The reactors are being retubed to avoid operation in this out-of-design condition.

Russia has a special licensing procedure for continuation of plant operation to the next scheduled outage.

The approach used is different from the predictive defect follow approach adopted for ISI elsewhere, which would allow timely detection of degradation should it occur.

IAEA activities

The IAEA activities first addressed the problem of multiple fuel channel rupture, which was recognized to be of high safety concern. The meeting organized on the subject included discussion of sequences potentially leading to such accidents, related analysis tools and discussion on fuel channel integrity (both with respect to the pressure tube and graphite stack) [143].

Further, the IAEA organized in co-operation with the Government of Lithuania a Workshop on fuel channel integrity, which provided a forum for detailed exchange of experience on the subject.

The outcome of the Workshop allowed to focus the discussion on aspects related to the behaviour of the fuel channel pressure tubes, the behaviour of the graphite stack, the interaction of the fuel channel pressure tubes and the graphite, and the behaviour of the reactor structure under normal operating conditions as well as under accident conditions. Fuel channel duct integrity was also addressed. The results of the discussion were consolidated in a report [144], which also covers the main concerns identified: degradation of the materials due to ageing, the gas gap closure, the potential for failure propagation, status of the available safety justification.

Achievements

Consensus was reached on the necessity to improve in-service inspection, including destructive and non-destructive testing.

It has been observed that limited local gap closure occurs at the time of pressure tube replacement. The safety justification for short term operation in this condition has been documented neither for normal operation nor for accidental conditions. The work to formulate the safety justification is reportedly under way.

No straightforward indication was found that the impact of a small percentage of channels operating in a closed gas gap condition for a “short” period of time will increase the risk of an unstable fuel channel tube fracture under normal operation conditions [144].

It was agreed that for normal operation and with the retubing approach adopted in Russia (complete reactor retubing at a power production level at which local gas gap closure occurs), no fuel channel material degradation mechanisms have been identified that could influence the fuel channel integrity.

Consensus was reached that propagation of single tube rupture is unlikely; however there is no published analysis to support it. The IAEA has been informed that such an analysis was recently completed in the framework of PNNL-RDIPE co-operation and will be published by the end of 1998.

Outlook

In-depth analyses to assess the probability of fuel channel failure propagation under condition of open gas gaps and to evaluate the consequences of transients during conditions of closed gas gaps are under way in the framework of international co-operation (bilateral programme, Lithuania and USA and RDIPE in Russia).

The Russian designer developed and implemented a retubing strategy to avoid closure of the gas gap between the fuel channel pressure tube and the graphite blocks. With this approach and for normal operation no fuel channel degradation mechanisms have been identified which could influence the fuel channel integrity. There is a general agreement that there is no safety justification for operating the RBMK reactors with closed gas gaps available at present.

The development of acceptance criteria pertinent to fuel channel integrity for transient and accident conditions should be continued. Peer review of the acceptance criteria developed should be carried out.

Should other potential scenarios be identified which, although they are of a low probability, conceptually have the possibility of leading to multiple tube rupture, they should be analysed in detail for review before closing the multiple channel tube rupture issue.

There is still a need to establish a comprehensive database containing all available data pertinent to fuel channel integrity (both fuel channel and graphite).

The in-service inspection, both non-destructive and destructive, and improvements need to be continued with high priority. The number of measurement samples should be increased, and advanced measurement techniques should be used.

7.3.3.2. Break of critical components

The primary circuit of the RBMK is a complex structure with respect to both layout and materials used. Some primary components and piping are outside the confinement. The design basis of the first generation plants does not include breaks of 800 mm diameter piping and the safety systems are not designed to cope with such failures. Moreover, 800 mm dia. piping breaks can lead to a failure of civil structures in the first generation plants. The design of later RBMK plants considers the 800 mm dia. breaks with respect to the ECCS and ALS capacity (except for Leningrad NPP Units 3-4 where the ALS is not designed to cope with 800 mm breaks). Moreover, there are no pipe whip restraints nor jet shields and the components are separated to a limited extent only. Consequently, the dependent failures in the main circulation system could occur due to double-ended guillotine break (DEGB) of large diameter components (this expert judgement is not confirmed by any RBMK specific analysis). Successful application of the LBB concept is therefore considered of a high priority since it can provide complex assessment of all relevant features of component design, assembly and operation, sufficiently demonstrate the low probability of large diameter high energy piping breaks and provide the operator with early warning before a major break can develop.

The LBB concept has been recognized as a viable methodology to demonstrate the integrity of large diameter high energy piping. A successful application of the LBB concept enables to exclude specific considerations for dynamic effects associated with primary circuit large diameter piping breaks for a number of PWRs. In cases where intergranular stress corrosion cracking (IGSCC) cannot be ruled out – and this is also the case for RBMKs (325 mm piping – downcomers, GDH feeders, ECCS piping, purification and heat removal system piping) – the LBB concept has not been applied and is not approved for BWR piping to date.

In January 1997, during routine in-service inspections by radiography, cracks were found in welds of 325 mm diameter austenitic piping in Leningrad NPP Unit 3. Out of 974 welds inspected by radiography, 35% showed indications. None of these cracks were through – wall cracks. The cracked areas were cut away to be investigated and intergranular cracks were found in the heat affected zone.

Inspection of a sample of welds on this type of piping carried out later at Kursk NPP Unit 1 revealed defects in five welds out of 80 inspected. A 100% inspection was subsequently performed on all the piping by radiography and later by UT. A similar degradation was found later on the 325 mm diameter piping of other RBMK NPP units. The results of metallographic tests on the removed cracked material through fractographic analysis of flaw surfaces confirmed the degradation mechanism. These results also demonstrated that the radiographic method of inspection was not effective for this type of degradation. For a more precise evaluation which took account of the specifics of design and structure of austenitic welds, new UT procedure was developed, tested and approved for use at RBMK plants in Russia in the framework of the international assistance programme. The implementation of this new procedure is under way at RBMK plants and includes both manual and automatic methods. In parallel, repair techniques were developed and implemented as necessary.

Russia has a special licensing procedure for continuation of plant operation to the next scheduled outage. The approach used is different from the predictive-defect-follow approach adopted for ISI elsewhere.

IAEA activities

a) Generic activities

In 1995, the IAEA organized a Topical Meeting on the Application of the LBB concept to the RBMK NPPs. The objective of the meeting was to review the activities carried out to date on the application of the LBB concept to the main circulation circuit of RBMK NPPs and to discuss plans for further work in this area. The focus of the review and discussion was on the contents, scope, and adequacy of the analyses and approach to validation used. The results of the discussion were consolidated in a report [145], which covers aspects related to the applicability of the LBB concept, the LBB methodology and the application at the plants.

At a request of the Government of Ukraine and considering the safety concerns associated with the recent intergranular stress corrosion cracking findings of in-service inspections (ISI) at RBMK NPPs, the IAEA organized a regional workshop on the subject.

The objective of the workshop was to exchange experience on environmentally assisted cracking of NPP austenitic piping accumulated at the BWR vessel type plants and RBMK plants. Presentations and discussion during the workshop addressed aspects related to:

- operational experience (ISI techniques used and results obtained, etc.);
- root cause analysis (metallurgical investigations, defect assessment practices, etc.);
- repair techniques and procedures including required qualification;
- safety concerns and safety analyses performed.

Forty-one experts from the Czech Republic, Germany, Lithuania, Russia, Spain, Switzerland, UK, Ukraine, USA and the IAEA participated in the workshop. The workshop was organized in the framework of the IAEA TC Project RER/9/052 and hosted by the Government of Ukraine at the Chernobyl Nuclear Safety Centre at Slavutych, Ukraine, on 22-26 June 1998.

The Workshop provided an excellent forum for detailed technical discussions [146]. It was concluded that IGSCC of austenitic stainless steel piping is a generic safety issue for reactors operating with BWR type water chemistry. This type of degradation is known from the early 1970s for non-stabilized stainless steel piping (USA, Japan, etc.) and from the beginning of the 1990s, for Ti-stabilized stainless steel (Germany), and at present appears to be already well under control for vessel type BWRs. For RBMK NPPs this phenomenon is rather new, but can be addressed by a similar set of measures which are effective for the improvement of safety by previous vessel type BWR operational experience (UT for ISI, leak detection, repair technologies, stress improvement in welds, water chemistry improvement, etc.).

b) Plant specific activities

In the framework of the activities of Lithuanian organizations to improve the safety of Ignalina NPP, the preparation of a report providing guidance for the application of the LBB concept to the Ignalina NPP has been initiated. Upon its completion and at the request of the Government of Lithuania, the IAEA organized a workshop to assist VATESI experts in reviewing the report. The workshop was hosted by the Swedish International Project in Stockholm, Sweden on 16-18 February 1998. Eighteen experts from the Czech Republic, Germany, Lithuania, Russia, Sweden, USA and the IAEA participated in the workshop and provided a number of comments and recommendations [147] to the guidance developed. VATESI’s guidance on the application of the LBB concept to the Ignalina NPP appears to be a good and useful document. It provides an excellent basis for the development of a regulatory document covering the main principles related to the application of the LBB concept application to the Ignalina NPP.

Achievements

The LBB concept has been recognized as a viable methodology to demonstrate the integrity of large diameter high energy piping at RBMK NPPs. It was agreed that application of the LBB concept at RBMK will enable to exclude specific considerations for dynamic effects associated with primary circuit large diameter piping breaks for piping which meets all the necessary requirements, e.g., no IGSCC.

In the framework of the bilateral co-operation with Japan, the development of an acoustic noise-based leak detection system was initiated for the Leningrad NPP initiated. Analyses to apply the LBB concept to the Smolensk NPP are being carried out in the framework of international assistance (TACIS). NDT equipment is to be delivered to Smolensk within the TACIS programme. The results obtained to date (Phase I) demonstrate reportedly the applicability of the LBB concept to the 800 mm dia. piping (pressure and suction headers). The work on Phase II still continues. In the framework of the bilateral co-operation with Sweden, regulatory requirements for the application of the LBB concept to Ignalina NPP have been developed and the steps to initiate practical implementation are under consideration at present.

Actions to address the IGSCC issue for RBMK plants have been initiated for all operating reactors but are still to be completed. In developing these actions, the RBMK operating countries utilized and benefited from the experience accumulated in vessel type BWRs. International co-operation to address the issue is of high importance and a matter of urgency.

Protection against failure of austenitic piping due to IGSCC should be provided by multiple layers of defence such as water chemistry, inspection, evaluation, mitigation, repair, leak monitoring, emergency core cooling systems and containment. Further international co-operation, in particular in the areas of ISI and leak monitoring, can provide beneficial results.

Outlook

In order to achieve the expected safety benefits, the activities to apply the LBB concept have to be completed on a plant specific basis and address in a comprehensive way all aspects involved, e.g., analyses, hardware modifications, installation/qualification of leak detection systems, in-service inspection and other operational aspects.

In applying the LBB concept, particular attention needs to be given to the corrosion damage (IGSCC) observed recently at RBMK plants.

It was concluded that IGSCC of austenitic stainless steel piping is a generic safety issue for reactors operating with BWR type water chemistry. This type of degradation appears to be already well under control for vessel-type BWRs, while it is a new safety issue for RBMK plants, which needs further attention.

7.3.4. Confinement system

The second and third generation RBMKs have a more or less leaktight confinement system with reinforced concrete structures and a pressure suppression system, referred to as ALS. This system only encloses part of the main circulation system and consists of several “leaktight compartments”. This system excludes a significant portion of the main circulation system, the steam-water lines, the steam drum separators and all the steam lines, and parts of the downcomers from the steam drum separators to the suction headers. There is no ALS for the first generation RBMKs.

The reactor cavity venting (RCV) system is a confinement system which is used at the RBMKs to relieve pressure associated with a leaky or ruptured fuel channel. This system is significant because if too much pressure builds up in the reactor cavity (e.g., rupture of several tubes), the lid or top of the cavity could be lifted off of the reactor cavity compartment, thus rupturing all fuel channels with consequential large radiological releases (see also Section 7.3.3.1).

IAEA activities

a) Generic activities

The safety reviews conducted by the IAEA provided a forum to examine the design and operation of the ALS and the RCV system.

b) Plant specific activities

Accident analysis results presented by the designers for Smolensk NPP Unit 3 indicate, under conservative assumptions, that the capacity of the overpressure protection system is adequate to cope with a simultaneous rupture of up to 10 pressure tubes loaded with high-power fuel assemblies.

The review of Ignalina NPP revealed that the defence in depth concept is not satisfied with respect to confining all parts of the reactor coolant circuit, e.g., upper sections of the pressure tubes, steam-water lines, drum separators upper sections of downcomer lines, communication lines between drum separators. In the event of pipe ruptures in these lines steam is discharged directly to the atmosphere. This also occurs at all other RBMKs.

It has been reported that according to the Ignalina NPP SAR it was demonstrated in SAR that all LOCAs, where significant fuel failures can occur if the fuel overheats, are located within confinement, and implementation of accepted design modifications will ensure that steam line breaks do not lead to fuel failures, and that doses to the public will remain within accepted limits for breaks which vent directly to the atmosphere.

The modifications to increase the relief capacity of the reactor cavity up to a simultaneous rupture of ten fuel channels have been fully supported by reviewers and have been completed for Leningrad Units 1-3, Kursk Unit 1, and Ignalina Units 1-2 NPPs.

The review conducted at the Leningrad NPP indicates that the activity confinement systems, the SOVA buildings, that are being constructed for Units 1 and 2 are comparable to the ALS for the second and third generation RBMKs. A primary concern is that this modification is not expected to be complete until 2001 [148].

Achievements

The need for upgrading plants of the first generation has been recognized by countries operating RBMKs; however, the time frame for completing these much needed modifications is much too long due to the lack of funds.

Outlook

There are still no plans for upgrading the confinement function at any of the RBMKs to include those portions of the main circulation circuit not currently covered by the ALS.

7.3.5. Support system functions

In order to ensure proper operation of the various safety systems, the various systems supporting those safety systems must also be highly reliable and capable of operating under potentially severe conditions. Most significant problems that affect defence in depth are those related to insufficient separation between different trains or buses of equipment; insufficient separation between operational and safety functions; insufficient diversification of components and systems; poor reliability and poor maintenance; and lack of equipment qualification.

IAEA activities

a) Generic activities

Because it was not possible to review all of the various systems and components at the RBMK plants, reviews conducted focused on the design and operation of a few of the more relevant support systems including: I&C; service water and ultimate heat sink; and electrical power supply, including the diesel generators.

During the review at Smolensk NPP in June 1993, a particular concern was the fact that the I&C systems had the same instrumentation and components performing both safety and control functions.

It was found that the high redundancy demonstrated in several of the front line safety systems is not present to the same extent in the supporting systems such as the service water and intermediate cooling systems. Moreover, the high level of redundancy in the safety systems cannot always be given full credit due to potential common cause failures. It has also been found that the differences between the plants are so important that each recommendation has to be evaluated on a plant specific basis.

The IAEA has also assisted the International Electrotechnical Commission (IEC) in a joint project to identify measures that can be implemented to improve RBMK safety through enhancement of I&C systems [149].

b) Plant specific activities

In general the reviews of Smolensk NPP Unit 3 and Ignalina NPP confirmed that there was poor separation and essentially no diversification of the subsystems and components. There was little or no separation or isolation between the safety and non-safety functions of the service water systems. These reviews also revealed problems with maintenance of some of the equipment and equipment reliability.

While the specific design of the various support systems are different at each plant, the issues and concerns that were identified are of a general nature and not specific to any one site or plant.

Achievements

Results and information obtained from the IAEA reviews have been of utmost importance for in-depth studies carried out in the framework of bilateral and other multilateral programmes. International agreement was reached on the relevant safety issues and the safety upgrades required.

Outlook

Activities are continuing on a plant specific basis to improve the RBMK support systems. However, significant efforts will be required to implement the plant specific modifications required.

7.3.6. Accident analysis

The analysis of postulated accidents available in the TOB for Smolensk NPP Unit 3 and Ignalina NPPs has been found to be limited and in general does not provide a clear description of the assumptions used in performing the analysis. While these technical justifications were prepared in accordance with the national regulations effective at the time the individual plants were put into service, they do not meet current international practices. The computer codes used at the time of RBMK design had limited modelling capability and there was little experimental data available against which to validate these computer codes.

IAEA activities

a) Generic activities

Since the first IAEA reviews, concerns and questions were raised relative to the adequacy of the accident analysis and technical basis for the various safety functions. During subsequent reviews at the plant sites, more time and discussions were devoted to understanding the nature of the accident analysis that has been performed to support the safety of RBMKs.

In addition, the IAEA sponsored several meetings specifically focused on improving the accident analysis capabilities for RBMK plants.

To address the problem of code validation, and in particular to validate the tools required to investigate accident scenarios which could lead to multiple pressure tube ruptures, a matrix for code validation was established and an international exercise was initiated in 1995 under IAEA co-ordination. The exercise was based on experimental results provided to the IAEA by the Government of Japan [150]. Experts from Germany, Russia, Sweden, Switzerland and USA participated in this task. Russian experts presented results obtained in the framework of the project Severe Transient Analysis sponsored by the EC.

The specific objective of this exercise was to validate a set of thermal-hydraulic computer codes for selected phenomena relevant to RBMK safety analysis, in particular to the analysis of the multiple tube rupture scenario.

Events leading to multiple pressure tube rupture have received considerable attention in the safety evaluation of RBMKs, since they might develop into severe accidents. The partial break of a group distribution header was identified as a potential candidate early in the evaluation process, since analyses of the event based on RELAP5, ATHLET and CATHARE had shown that for a certain break size, flow stagnation accompanied by flow fluctuations might lead to excessive heat-up of the pressure tubes.

The nature of these flow fluctuations was not fully understood and the recommendation was made to experimentally investigate the problem or to validate the thermal-hydraulic codes used in the analysis on the basis of available experimental data.

Out of the series of tests performed by PNC, a group of international experts selected a test matrix addressing the phenomena of interest in a topical meeting organized by the IAEA and hosted by PNC in 1994. Based on this material, in 1995 the IAEA launched an international code validation programme aimed at validating thermal-hydraulic computer codes for selected phenomena relevant to RBMK safety analysis.

These phenomena include flow stagnation at near zero delta p, dry-out, post dry-out heat transfer under upward and downward flow conditions, rewetting, critical flow, counter current flow (CCFL and CCFL breakdown), natural circulation flow, density wave oscillations at low power, and flow pattern induced oscillations in horizontal piping.

The tests selected for the validation exercise cover flow fluctuations induced by high power transients, dry-out channel power under low flow, flow fluctuations in two parallel channels without net coolant flow (two cases: water level in the downcomer 1 m and 10 m, respectively), and degraded heat transfer in single channel induced by an inlet pipe rupture.

It was confirmed by the participants that the experimental data provided by PNC present important information related to the phenomena involved in RBMK accident analysis and are sufficient to provide the required experimental database for the intended code validation exercise. In particular, it was agreed that the need for an additional dedicated test previously recommended for the investigation of flow stagnation had diminished.

The code validation exercise contributed to an improved understanding of potential causes, nature and behaviour of flow oscillations in channel type reactors. Flow oscillations, accompanying flow stagnation as observed in previous analyses of the partial break of a GDH might be real. Consequently, improved channel dry-out behaviour and heat transfer should be given credit for, if adequately simulated by the computer codes in use.

The exercise has shown that ATHLET if properly modelled is able to predict both post dry-out density wave oscillations and flow pattern-induced two phase flow oscillations in parallel channel flow. The amplitudes of the calculated flow oscillations are under-predicted in the ATHLET analysis; this needs further investigation.

Channel dry-out for low upward flow is in good agreement with the experiment. For low downward flow, channel dry-out is over-predicted. At high flow rates (>3.75 kg/s) it is under-predicted regardless of the flow direction.

ATHLET also calculates channel behaviour under stagnant flow conditions in remarkable agreement with the experiment. Stagnant flow was achieved in the experiment by the break of an inlet pipe in a two channel test rig. Due to the specific setup of the test, the trajectories of power and flow did not enter the unstable region of the stability map, no flow oscillations were observed in the experiment. However, channel heat-up, counter current flow, steam cooling, re-wetting and flow recovery were adequately predicted by the code during the stagnation period and thereafter.

RELAP5, although in general in acceptable agreement with the experiments, did not show flow oscillations or channel dry-out when applied to the post dry-out density wave oscillations experiment. Extensive sensitivity and modelling studies are still under way to further investigate this problem.

A topical meeting on 3D computer codes for core and systems analysis was organized by the IAEA in 1996 [151]. The meeting reviewed the state of development of 3D computer codes used for core and system analysis of RBMK NPPs, assessed the status of the code validation, and identified code requirements. A valuable contribution was presented from the EC-sponsored project with the University of Bremen (Germany) and the Kurchatov Institute (Russia).

In April 1997, the IAEA initiated the development of guidelines for RBMK accident analysis. The objective of the preparation of the guidelines was to establish a minimum set of requirements dealing with deterministic accident analysis reflecting current approaches adopted both in Eastern or Western countries involved in accident analysis for RBMK reactors. International experience and practices, consistent with state-of-the-art methodology, and conservative approaches to deterministic safety and accident analysis were used in preparation of the guidelines.

The document prepared [17] deals with the transient and accident analysis required to be included as an integral part of SARs and to be developed by the power plant operating and/or the reactor vendor organization, as specified by the respective national nuclear safety authority. This analysis can also be used to justify existing or newly proposed technical improvements and plant modifications for safety upgrading at nuclear power plants with RBMK type reactors. The document does not address analysis requirements for beyond design and severe accidents.

The IAEA also provided assistance in the area of PSA. In 1994, the IAEA reviewed early PSA work carried out in Russia for the Leningrad NPP [34]. The Pilot Risk Study reviewed provides elements which can be used in the efforts currently being conducted to develop a level 1 PSA for Leningrad NPP.

In 1996, the IAEA convened a meeting to review the characterization of core damage states for channel reactors. Approaches used for Ignalina NPP, CANDU and the US Navy reactor at Oak Ridge were discussed [152]. Different levels of core damage have been agreed as a function of the severity and number of channels affected.

In the framework of a trilateral Swedish-Russian-Lithuanian programme, a PSA level (1+) was performed at the Ignalina NPP in 1994 (Barselina Project). A level 2 PSA is foreseen.

In February 1996, a meeting was organized by the IAEA [153] in co-operation with the RDIPE in Moscow to review a paper entitled “Chernobyl Accident Causes: Overview of Studies Over the Decade” prepared by Russian experts and to be presented at the International Forum One Decade after Chernobyl: Nuclear Safety Aspects, in Vienna in April 1996.

b) Plant specific activities

IAEA activities relative to accident analysis were mostly of a generic rather than a plant specific nature.

Results of accident analysis discussed during the IAEA reviews at Smolensk NPP Unit 3 and Ignalina NPP confirmed the need to substantially improve the quality and completeness of the analysis, consistent with state-of-the art methodology.

The Ignalina NPP has just recently completed a Western-type safety analysis report (SAR). This report was prepared in co-operation with Eastern and Western industrial partners as part of the EBRD Grant Agreement and has been reviewed by an international consortium of experts from technical safety organizations. The SAR production and its review were supervised by an international panel of senior safety experts.

The new SAR for Ignalina NPP did not cover all the aspects required for a full Western SAR to qualify completely for a license application. However, the independent reviewers concluded that the Ignalina NPP SAR has dealt with the majority of essential issues. The IAEA was not involved in this review process.

Achievements

There is general agreement of plant operators and the general RBMK designer that the accident analysis and technical justifications need to be updated to concur with plant upgradings and to meet best international practices.

Guidelines developed in the framework of the EBP [17] establish a minimum set of requirements dealing with deterministic accident analysis and reflect current approaches adopted both in Eastern or Western countries involved in accident analysis for RBMK reactors.

There are still differences among specialists in the interpretation of the causes of the Chernobyl accident.

Consensus was reached that the present understanding of the potential causes of the accident is sufficient to judge whether the improvements of the core design and shutdown system as well as the procedures implemented in all RBMK type plants are adequate to prevent another accident of the kind which took place at Chernobyl.

Outlook

It is desirable that plant specific Western style SARs be prepared for each of the RBMKs (except for Chernobyl NPP Unit 3 because of its early shutdown). As part of Russia's agreement with the EBRD, a SAR is being developed for the first generation units at Kursk. The effort is supported in the framework of the Russia-USA bilateral co-operation.

Further and more detailed investigation of fuel and core damage progression may be required to gain improved confidence in damage state definitions.

7.3.7. Fire protection

There have been many fires at the various Soviet-designed reactors. One of the most damaging occurred at Chernobyl NPP Unit 2 on 11 October 1991 when a fire started in the turbine-generator. Before the fire was brought under control, the turbine generator had been destroyed, and portions of the turbine building roof were damaged to the point that it collapsed and then disabled part of the ECCS (fortunately, the normal feedwater pumps and water lines were not affected).

In general, fire risks were not adequately considered during the design of the RBMK reactors. The poor degree of separation of the safety and safety support systems increase the potential for relatively minor fires to disable entire safety systems.

IAEA activities

a) Generic activities

A team of fire safety experts participated in the IAEA review at Smolensk NPP [132]. The types of issues and concerns identified during this review had already been recognized by the NPPs and actions were already under way at some of the plants to improve the situation. It was recognized that the concerns raised were not specific to Smolensk NPP but were valid for all the RBMKs.

These problems have already been tackled to some extent in some RBMK units. In other cases this has been remedied by a newer design, by national upgrading programmes, or by bilateral/international assistance programmes.

Throughout the plants, all areas with flammable material should be provided with fire detection equipment connected to a proper alarm system. The existing systems need both extension and quality upgrading. Additionally, improvements with respect to quality assurance, inspection and maintenance are necessary. Some plants have started these measures as a part of their own upgrading programme, and some in connection with bilaterally or internationally financed programmes.

Manual fire suppression capability is generally very strong at nuclear power plants in the FSU. This applies to the number and the training of fire brigade personnel. Deficiencies, however, exist in personal protective equipment, communication equipment, and fire fighting equipment, such as fire extinguishers, hoses, and nozzles. Automatic fire suppression is mainly realized through fixed water sprinkler and deluge extinguishing systems. Local carbon dioxide or foam extinguishing systems also exist. The reliability and coverage of the existing systems needs detailed analysis and assessment. Stationary automatic water extinguishing systems should be added to some compartments which so far have not been fully protected. There is also a need for improvements in quality assurance as well as in the inspection and maintenance activities.

The reliable supply of water assures the proper availability and operation of both manual and automatic fire suppression capability. However, differences between sites and the different generations of RBMKs are extensive and measures of different magnitude are needed.

b) Plant specific activities

In June 1992, at the request of the Government of Ukraine, the IAEA conducted an ASSET mission to review the root cause investigation of the causes of the Chernobyl NPP Unit 2 fire and the actions taken to avoid its reoccurrence [154].

Achievements and outlook

Much has already been done at some of the plants to reduce the risk associated with fire. Many plants are receiving assistance from multilateral or bilateral programmes. The work that is being done has been focused on: reducing the fire loads (reduction of flammable materials as far as possible), improving fire detection capabilities, and improving fire suppression capabilities. Where possible, structural fire protection (separation of redundant safety related equipment, quality of fire barrier elements) should also be improved.

Work in this area needs to be pursued as a matter of high priority.

7.3.8. Seismic safety

IAEA activities

The review of Smolensk NPP was made within the framework of a wider scope review of the design of this type of NPP. This was the only plant specific activity related to RBMKs.

It should be noted that structurally speaking, RBMK type NPPs are very similar to WWER-440 type NPPs and their vulnerabilities to external events in general (seismic included) are the same.

Bearing these characteristics in mind, a CRP on The Safety of RBMK type NPPs in Relation to External Events was initiated. The first meeting was held in Moscow in April 1998 with nine contract/agreement holders and five observing organizations from Russia. There was great interest in the CRP by all the organizations concerned in Russia (e.g., RDIPE, NIKIET, VNIPIET, NNAEP, CKTI, VNIIAM). The work programme will be further discussed and modified as required.

Achievement and Outlook

Mainly due to the fact that most RBMK type NPPs are located in low seismic hazard areas, little was done in relation to seismic safety of these plants in the previous years. However, it is recognized that these plants have structural vulnerabilities for extreme external loads, such as blast and impact. A CRP was initiated in 1997 on the Safety of RBMK type NPPs in relation to external events. The programme also covers seismic issues but has a much broader scope including blast loads and aircraft impact. A work plan was prepared during the first RCM in April 1998 in Moscow. According to this plan, as an initial activity, the Leningrad NPP will be analysed by different working groups for external dynamic loads (vibration, blast and impact).

8. OPERATIONAL SAFETY OF WWERs AND RBMKs

At the request of the Member States, the IAEA in 1988 started providing operational safety services to Soviet-designed reactor plants. Initial OSART missions were at Paks, Rovno Unit 3, Dukovany, Kozloduy Unit 5 and Loviisa NPPs in 1989-90, with an ASSET mission at the Ignalina NPP in 1989.

In 1990, assistance was strengthened and in addition to OSARTs and ASSETs, operational aspects were investigated in the framework of safety review missions. The latter combine design and operational aspects in a joint review.

8.1. WWER-440/230 NPPs

Four plants were concerned: Novovoronezh Units 3-4, Kola Units 1-2, Kozloduy Units 1-4 and Bohunice Units 1-2. Armenia NPP Units 1-2 had been shut down since the 7 December 1988 earthquake, and because no foreseeable restart had been planned for the early 1990s, this plant was not included in the EBP reviews.

ASSETs were performed at all plants in operation in 1990 and 1991. They identified the following main issues, with various levels of severity depending on the plants:

- plants had generally an industrial culture strongly oriented towards energy generation; sometimes plant management failed to engage in the openness and communication needed to develop safety culture;
- the reliability of plant safety systems, personnel and procedures was not sufficiently monitored;
- surveillance programmes were not comprehensive and did not include acceptance criteria, and therefore it was unable to detect safety deviations and latent weaknesses;
- management was generally technically qualified and senior personnel knowledgeable but the proficiency of maintenance staff needed improvement;
- despite satisfactory licensing event reporting criteria, operating experience feedback was not fully effective: safety event reporting and root cause analysis were found insufficient and corrective actions were not always appropriate, some grave event reports were kept restricted;
- fire prevention, material conditions and housekeeping needed improvement.

In 1991, SRMs were performed at each of the four plants in operation. The weak points found in operational safety areas were common to all of these plants at that time, and included:

- a lack of safety culture at all levels, the production function being predominant over safety considerations; the absence of exchanges with the international nuclear community had contributed to stagnation in safety
- insufficient maintenance of equipment, material conditions of safety related items considered unacceptable in some plants and often poor housekeeping

- non-existing or unsuitable emergency operating procedures, which was a logical consequence of missing accident analysis

- training 'on the job' or on limited-scope simulators, and therefore inappropriate for many accident sequences

- a complex organization which favoured the dissipation of responsibilities and very little delegation, with small problems being taken to the managerial level

- insufficient dissemination of information.

These weaknesses were aggravated by the absence of equipment for copying documents, thereby making document control difficult. As a consequence, there was a strong oral tradition which was detrimental to written documents and therefore posed problems for experience feedback or quality assurance. These conditions existed in the nuclear industry in Western Europe at the start of the 1960s.

IAEA activities

To help plant operators overcome their deficiencies and follow their progress in the area of operations, the IAEA consolidated SRM recommendations and ranked them in accordance with their safety significance [6].

Four deficiencies ranked at the highest safety significance (category IV) affected operational safety functions. The recommendations made focused on:

- greater involvement of plant management in all areas of operational safety

- development of a safety culture within the upper management and throughout the plant staff

- development of user-friendly emergency operating procedures covering all the potential accident scenarios

- maintaining safety related equipment in a condition which will guarantee that they are as reliable as possible.

In addition, there were fifteen deficiencies of high safety significance (category III) concerning different aspects of management, concerning operating procedures, personnel training and emergency planning.

Further advice and consultancy were provided to the plants via the organization of consultative missions in 1993 and 1994 [65, 66, 67] and technical visits from 1995 to 1997 [77, 78, 79].

On the major issue of safety culture, which is behind most of the operational deficiencies, the IAEA developed a new service – ASCOT – became available in 1993. The main objective of ASCOT was to promote safety culture based on the principles and recommendations contained in INSAG-4. Since then, ASCOT activities have developed successfully either in combination with OSART services or in the form of seminars. As regards operational experience feedback, ASSET seminars were held soon after the SRMs to help them in the development of their self-assessment programmes.

For the Armenia NPP Unit 2 which restarted in November 1995, specific missions on operating procedures (April 1995), emergency planning, operational experience feedback (June 1996) and a short technical visit which included operational safety (November 1996) were organized in the framework of TC project ARM/9/003. This project had been initiated in 1994 to provide support and advice to the newly created Armenian Safety Authority. An ASSET seminar was also organized in Yerevan in May 1997.

After this first strong impulse at the improvement of the WWER-440/230 plants, the IAEA continued to offer operational safety services such as OSART missions at Bohunice (1996) and Kozloduy (1998), ASSET seminars and missions at the WWER-440/230 plants, and ASCOTs at Kozloduy (1993) and Bohunice (in 1994 in Bratislava and in 1997 at the plant).

Achievements and outlook

Since the beginning of the EBP, a wide exchange of operating practices between Eastern and Western plants has taken place. In the process, management abilities as well as knowledge of the operating staff of the WWER-440/230 NPPs have improved. More notably the following developments were set in motion:

- new organizational structures favouring clear responsibilities have been established on all the sites
- the safety culture has improved, in particular through the cleanliness and good material conditions of the plants
- quality assurance programmes have been developed and are being implemented on all the sites
- new procedures have been or are being established in the areas of plant operation, surveillance testing and maintenance
- predictive maintenance programmes have been developed
- a systematic approach to the training process has been adopted and computer-assisted training developed
- emergency planning is progressing on all the sites.

In short, progress has been observed on the operational safety issues. Nonetheless, efforts have to be pursued on several sites to upgrade the procedures, especially emergency operating procedures, prepare plant specific SARs, and improve training in the maintenance area. It would also be worthwhile to develop the use of computers for assistance to the operator in case of emergency; to centralize the storage and distribution of documents; to improve the recording and feedback of operating experience and to address remaining problems in emergency planning.

Regarding the Armenia NPP, as it was only partially reviewed, conclusions on its status are premature at this stage. Most likely, it will need more time and efforts to catch up with the other operating WWER-440/230s because of its past isolation.

8.2. WWER-440/213 AND WWER-1000 NPPs

The scope of the EBP was extended in 1992 to include the WWER-440/213, WWER-1000 and RBMK plants in operation and under construction. However operational safety was not included in the scope of the IAEA review missions, except for the mission to Zaporozhe. The experts involved in these tasks could rely on the results of several previous IAEA activities, such as technical co-operation projects, OSART and ASSET missions. In addition, results of external reviews in this area were made available to the IAEA. For the identification of the operational issues, the following material served as a basis.

- For the WWER-440/213 plants:

1. Analysis of OSARTs to Paks, Dukovany, Loviisa, Mochovce, Bohunice NPPs
2. ASSETs to Paks, Dukovany, Rovno NPPs
3. Operating experience of the WWER-440 model 230
4. Report on Rovno NPP [106]
5. Safety Reassessment of Paks NPP
6. Core damage frequency for Dukovany NPP

- For the WWER-1000 plants:

1. SRM to Zaporozhe NPP
2. OSARTs to Rovno, Temelin, Kozloduy, Zaporozhe, Khmelnitsky NPPs
3. ASSETs to Khmelnitsky, Balakovo, Kalinin, Kozloduy, South Ukraine NPPs
4. Report on Rovno NPP Unit 3 [106]

In the presentation which follows, both WWER-440/213s and WWER-1000s are considered together because, in the operational safety and management areas, issues are obviously more homogenous per country and culture rather than per type of reactor.

From all the reviews performed at NPPs operating these models, there appeared to be a great diversity in the operational practices and management according to the countries and little difference within the same country. However, it was possible to identify several points that are characteristic to the plants visited.

- All plants evidenced motivated management teams supported by well educated and experienced staff; a professional conduct of operations and sound maintenance programme; staff cumulative dose and radioactive waste at low levels; some operating experience feedback process.

- A great diversity between these plants was noted in the overall plant performance and areas such as housekeeping and material conditions; training; emergency planning and preparedness.

- Some weaknesses were observed at almost all plants, especially a complex plant organization with little communication; a reward and reprisal system not favouring individual and collective performance enhancement; a weak QA organization; normal and emergency operating procedures, including the ‘limits and conditions of safe operation', need development and revision; operating experience feedback is insufficiently structured to be fully effective.

- For some countries it also appeared that the organizational relationship and responsibilities between the governmental bodies (for energy, as the utility, for safety, as the regulator, and some others) and the plant management are complex and unclear.

IAEA activities

The IAEA made detailed recommendations in [7, 9] based on the reviews conducted. The measures recommended intend to stimulate the operating organizations and plant management to correct the deficiencies identified and to achieve a better alignment with international practices. Among the proposals the most important are:

Operating procedures are key elements of plant safety both for the normal and emergency modes of operation. Improving the format and content of normal operating procedures and elaboration of state or symptom oriented emergency operating procedures are considered to be very important. But senior plant management should also evaluate the current use of procedures and modify a plant's practice as necessary.

The justification of limits and conditions of safe operation needs to be developed systematically on the basis of reliability and accident analyses and operational experience and included in the Technical Specifications.

Many of the elements of the safety culture are established at these plants. The principles of safety culture should be incorporated into the daily activities and incident prevention through training and qualification programmes.

In spite of the wide variety of approaches used to gain feedback from the safety related operational experience, several improvements are recommended in this area, including reporting criteria, the application of a root cause analysis methodology, setting up multi-disciplinary engineering support groups and improving co-operation between WWER operators.

The importance of quality assurance is generally recognized by plant management. Review and improvement of the QA programmes are recommended in order to determine department responsibilities and to maintain an independent system for verification and approval of all procedures prior to implementation and to monitor that procedures are followed.

The management of the plants should devote great attention to the improvement of maintenance procedures and maintenance programmes. Errors in maintenance and testing can result in erroneous functioning of safety systems or degradation of defence in depth.

The configuration management is crucial for safety to ensure the plant is operated in accordance with the design basis. In this respect, the records and data related to different plant activities (e.g., maintenance, surveillance tests, backfitting) should be stored so that they are easily accessible and retrievable, preferably with the use of electronic data. Moreover, the control of plant modifications should be strict and systematic at every step of the process. This is especially important in units which are undergoing far reaching changes in the course of safety improvement programmes.

The surveillance programmes of the plants need to be reviewed and improved to detect degradation or hidden failures taking into account equipment history and operating experience feedback and to identify procedural deficiencies. Test intervals should be considered carefully in order to ensure the functionality of equipment and to avoid unnecessary tests which could result in decreasing equipment availability.

Operational and maintenance staff needs to be trained to develop, maintain and improve their abilities for their activities and, in particular, to diagnose and manage plant events.

Adequately equipped and organized emergency centres are essential to co-ordinate and carry out accident management measures expected to protect the personnel, the public and the environment. Therefore, it is recommended to construct and equip the emergency centres, including procedures and documentation, and carry out the necessary drills and exercises.

Generally, radiation protection practices at WWER plants are satisfactory, the collective doses of the personnel are kept low. However, the radiation monitoring instrumentation originally designed and supplied needs upgrading to cover the whole range of parameters including accidental conditions.

Most of these NPPs were covered by an ASCOT seminar, either directly at one of the plant sites or collectively in the capital of the country, which addressed safety culture and provided a methodology for its implementation.

Achievements and outlook

Measures identical to those adopted for WWER-440/230 plants were taken and it was possible to appreciate the evolution of operational safety practices and management at these plants through visits, meetings and in particular, the follow-up visits of previous reviews.

Generally, the conclusions are similar to those of WWER-440/230 plants but, because of the larger number of plants reviewed and the longer period, it was possible to observe some other aspects. In certain countries, the effect of the very difficult political transition and economical situation has slowed down the expected progress. The missions found a management still behaving in a traditional Soviet style, prescriptive and authoritarian, practising weak monitoring and overseeing a personnel with a passive attitude; the concept of safety culture was often known but not yet really alive in the management. However, such situations are becoming increasingly rare.

8.3. THE RBMK NPPs

The first RBMK NPP to receive a review was the Ignalina NPP with an ASSET mission in 1989. Then, in the framework of the EBP, all the plants underwent ASSET missions since 1992-93, a review was performed in 1993 at Smolensk NPP [132] and a limited scope review of operational aspects at Chernobyl NPP in 1994 [155]. Ignalina NPP was the object of an OSART in 1995 [134].

The Ignalina ASSET mission in 1989 was the first IAEA review in a RBMK plant. Besides the positive findings, the team identified several areas that needed attention, namely:

- the lack of a surveillance policy and programme

- an excessively high threshold for event reporting and analysis, no methodology to analyse the weaknesses detected, corrective actions not always appropriate
- no systematic requalification test of equipment after work or before return to operation
- no safety and reliability indicators.

Although the two other reviews performed at RBMK plants were short and limited, they offered opportunities for improvement to the plant managers.

The review team at the Smolensk NPP identified:

- a strong management, based on rules, with a complex plant organization, and no clear reporting nor monitoring,
- adequate human and material resources, a staff with many years of experience due to the absence of turnover, but with a passive and non-critical attitude.

At Chernobyl, the team reviewed the same areas and highlighted radiation protection as one of the areas in need of urgent improvement.

The OSART mission to Ignalina NPP in September 1995 encountered:

- well educated staff
- senior managers present in the work areas
- self-assessment methods for maintenance activities
- generally improving material conditions and housekeeping.

The following areas needed improvement:

- insufficient overall funding to ensure all safety matters were budgeted accordingly
- management objectives limited to compliance with regulations
- organizational structure that did not allow sufficient operation and surveillance of important equipment
- unsatisfactory level of radiation doses and contamination
- emergency planning insufficient for radiological release limitation and emergency response personnel training
- nuclear safety regulations which did not focus sufficiently on management and safety programme effectiveness.

An OSART follow-up visit was carried out in June 1997 [156].

IAEA activities

The IAEA issued a document following these reviews [8] which consolidates operational issues and offers some recommendations for improvement. Essentially they relate to the following actions:

- Review the organizational structure of NPP management, including responsibilities and accountabilities at all levels and carry out periodic independent reviews of management.

- Develop an overall QA programme with independent assessment of its effectiveness. Suppliers of equipment and companies providing engineering services for reconstruction and modernization should comply strictly with the QA programme.

- Establish a relationship with personnel based on trust and openness to implement principles of safety culture: qualification improvement, self-evaluation, self-critical attitude.

- Check the documentation regularly, upgrade it as necessary and keep in good condition.

- Attention should be given to equipment labelling, material condition and housekeeping, lighting of premises and good access for operation and maintenance.

- Review training programme, facilities and material. Introduce the concept of continuous training. Ensure regular training for operation staff on full scope simulator.

- Establish a guideline to ensure quality and completeness of procedures, then enhance normal operating procedures.

- Continue development of emergency operating procedures in co-operation with other plants and with international support.

- Develop an effective experience feedback and event investigation system, based on the ASSET, or another, methodology.

- Update a maintenance programme to achieve fully effective preventive, predictive and corrective maintenance.

- Establish a procedure for effective control of the temporary and permanent modification processes.

- Based the surveillance test periodicity on equipment reliability data and operating experience; surveillance tests should be carried out so that the intended function of the system is confirmed; surveillance procedures should encompass detailed instructions and acceptance criteria.

- Establish an ALARA programme.

Two broad issues addressing quality assurance (QA) and regulatory interface have not been specifically attributed to any particular topical area, but were recognized as affecting all of them.

The main concern relative to QA relates to ensuring that the design bases for the various analyses, safety reviews and safety upgrading actions stem from the actual plant status and configuration. Another aspect of this issue is ensuring that the relevant design documentation is updated as the plant configuration is modified and upgraded. It is therefore of utmost importance that the organizational structure promotes the raising of safety concerns, responds quickly in evaluating these safety concerns and implements timely corrective actions if they are warranted.

ASSET missions have been conducted at all RBMK plants to improve their capability of analysis of operating experience.

To promote safety culture, ASCOT seminars were held at Chernobyl and in Kiev. In addition, two workshops were organized by the IAEA in Sweden and Lithuania in co-operation with the Swedish International Project on Nuclear Safety and the US Department of Energy [55, 56].

Achievements and outlook

The Ignalina OSART mission in 1995 and its follow-up visit in 1997 provided a good opportunity to observe the important progress achieved and the areas still in need of further effort. Most of the issues were solved or evidenced good progress. Examples of the main developments were: creation of a plant safety committee; a set of measurable goals with associated indicators; a complete QA programme; managerial training for plant managers and supervisors; improvement of plant housekeeping and material conditions; strengthening of the operational experience feedback system; an ALARA programme; and effective improvement of the emergency planning and preparedness.

The activities still needing attention are: the individual performance appraisal system; continuing training programme; surveillance programme and procedures; the plant modification process.

9. OUTLOOK

9.1. GENERAL

The EBP has singled out the safety issues for each type of reactor and has listed them in the individual Issue Books. In the course of the EBP, these safety issues have been addressed in the safety improvement or modernization programmes at the relevant plants. Currently, as the EPB is being wound up, work on these issues is in various stages of completion: projects to resolve these issues are under way, have started or have been planned. This Section discusses the tasks still to be carried out according to the information available to the IAEA on the basis of (a) the WWER and RBMK database (see Sections 2.5.8–9) and (b) the status of the ongoing bilateral and multilateral assistance programmes. The discussion summarizes the information contained in the foregoing ‘Achievements and Outlook’ subsections of the present report relative to each reactor type and to operational safety (see Sections 3–8).

The description of the work which remains to be done is structured according to safety topics under the main headings WWER design, RBMK design, operational safety and safety assessment.

The assistance provided by the IAEA is based on the “Integrated Strategy for Assisting Member States in Establishing/Strengthening their Nuclear Safety”. The central element of the strategy is the development of country specific nuclear safety profiles and action plans tailored to the specific areas where assistance is needed.

This section also reviews the planned or proposed major activities of the IAEA for the budget cycle 1999-2000, which are funded either from the regular budget or from Technical Co-operation funds. The specific activities of the Technical Co-operation projects are defined on a yearly basis in response to the requests for assistance from the relevant Member States; they are the result of detailed discussions with the countries receiving assistance. The IAEA programme also takes into account other international and/or bilateral assistance programmes, as far as these are known to the IAEA, and is subject to the natural IAEA constraints on budget and manpower.

9.2. WWER DESIGN

Classification and qualification of components and systems

Comparisons between Russian standards and IAEA NUSS standards have indicated the need for a reclassification of safety-related components and systems. Therefore, the reliability, maintenance, surveillance and in-service inspection procedures need to be reviewed and improved in order to ensure compliance with classification requirements.

The upgrading and replacement of the I&C equipment of WWER NPPs, which will be required as a result of the reclassification, will be a major effort for operating plants. Electrical equipment, including cable connections, also need to be qualified to meet the safety-graded requirements.

The IAEA will provide, upon request, independent peer review services.

Instrumentation and control

I&C has been recognized as an area which requires substantial improvements in WWER NPPs, particularly the WWER-440/230 and WWER-440/213 NPPs. The I&C equipment used at these plants was designed in the early 1960s and early 1970s, respectively. The specific criteria for achieving high functional performance and reliability of I&C, in accordance with its safety significance, were not included in the design, as these criteria were not available in these countries at that time.

A report [125] was published by the IAEA providing a technical basis for improvements related to the most significant aspects of I&C.

Since the beginning of the EBP, significant improvements have been made in many WWER NPPs in the area of I&C. However, large differences can be observed between the level of progress achieved in the different countries operating WWER NPPs.

The IAEA will serve as a forum for an exchange of information on specific I&C issues and improvements and will provide, upon request, independent peer review services.

Control rod insertion reliability

It has been reported that the implementation of measures proposed by the designer ensure that the design limits for the control rod drop time are fulfilled. While the understanding of the root cause of this issue has increased, the quantification of all contributors is still pending. Since the delay of control rod insertion is directly caused by fuel bowing, the associated appearance of water gaps between fuel assemblies is a safety concern which is being addressed.

The review of fuel safety criteria and methods applied to existing and new fuel designs under advanced performances are being performed for PWRs (Task Force in the CSNI WG2) and addressed in the IAEA Nuclear Safety Programme (1999-2000) for WWER fuel.

Reactor coolant system integrity

Reactor pressure vessel integrity related activities have been initiated in all countries operating WWER plants through national and international programmes, but in most cases still need to be completed. In particular, activities related to the material behaviour of the first WWER-440 vessels (both the 230 and the 213 model plants) and the WWER-1000 vessels with high Ni content, including surveillance programmes and updating of integrity assessment have to be completed with high priority. IAEA assistance is being provided in the framework of the round-robin and PTS exercises (see Section 6.2).

Operational experience with WWER horizontal steam generators has shown the high importance of the timely implementation of preventive, compensatory and corrective measures, such as adequate inspection, water chemistry monitoring and control, leak detection, repair and replacement. Continuous attention to these areas is required, including the establishment of approaches on the treatment of large primary to secondary leaks. IAEA assistance will be focused on the application of the guidelines which have been developed [19].

The application of the LBB concept to the large diameter primary piping of WWER-440/230 NPPs has been recognized as a tool to restore some features of the original safety concept from the current point of view of maintaining primary circuit integrity. The application of the LBB concept to WWER-440/230 plants has been initiated in the former Czechoslovakia for Bohunice NPP Units 1-2 in 1988, with the objective of addressing the above indicated concerns, to verify the piping design and provide a basis for seismic backfitting.

The IAEA has developed guidelines for the LBB application [21] and provided peer review services for Bohunice NPP Units 1-2. The IAEA will provide, upon request, peer review services to evaluate the correctness of the application of the LBB concept to primary and secondary piping.

To maintain RCS integrity, improvements of ISI systems are necessary and have to be achieved through the use of more modern methods, techniques and equipment as well as through enhanced reliability of results by qualification of the entire ISI system.

IAEA programmatic activities are conducted in the framework of the International Working Group on Life Management of NPPs. A Regional TC project on Advanced Non-Destructive Testing for Primary Circuit Components of WWER NPPs will be continued beyond 1998. It includes a pilot study for the application of the Guidelines for In-Service Inspection Qualification developed in the framework of the EBP [22]. IAEA assistance is also provided in the framework of national TC projects.

Confinement/Containment integrity

WWER-440/230 plants

Confinement leaktightness improvements results have demonstrated that significant improvements are possible in this area. Leaktightness improvements need to be completed at all NPPs.

Concerning upgradings of the confinement, plans exist for each plant as part of the major upgrading programmes. However, they all include new technological systems or equipment for which sufficient testing or experience is not available yet. Additional testing required to qualify the new technological systems needs to be pursued to allow for timely implementation.

The IAEA can play a major role in organizing technical meetings to review national approaches for upgrading confinements, taking into account new bounding LOCA design basis accidents.

WWER-440/213 plants

The studies sponsored by the IAEA have shown that the existing documents were insufficient to demonstrate the mechanical strength of the bubbler condenser structure under large break LOCA conditions.

The exact extent of the mechanical upgrading should be determined for each plant considering the actual status of the bubbler condenser in a given plant and the requirements of the national regulatory authorities.

An experimental thermohydraulic qualification test programme for bubbler condensers was planned in the OECD/NEA group. A separate PHARE/TACIS project, funded by the EC, is under way to verify strength characteristics under accident loads using Paks NPP as the reference plant. The results will be available by 2000.

Physical and functional separation of safety systems

Deviations from current national standards and international practice with respect to basic design principles have been identified, especially at the first generation WWER plants i.e., at WWER-440/230 and ‘small series’ WWER-1000 plants.

The original design and layout of the WWER intermediate building, where secondary piping and respective valves are located, is such that failure propagation cannot be excluded. Activities addressing this concern are still required at the majority of WWER NPPs.

Deficiencies in the independence of safety systems and the consistent application of the single failure criterion are of high safety significance and have not been appropriately addressed in the safety improvement/modernization programmes. Safety issues such as the insufficient independence of the ECCS trains in the ‘small series’ WWER-1000 plants have been addressed by compensatory measures while systematic analyses are performed to establish long term upgradings.

Fire safety

This issue is relevant to both WWER and RBMK NPPs.

Despite improvements in the areas of fire prevention and mitigation, systematic fire hazards analyses still need to be carried out at most of the plants. The results of these analyses will indicate the additional improvements required. International assistance will be required to implement these modifications.

IAEA assistance will be provided in the framework of its engineering safety services.

Seismic safety

The seismic safety of most WWER-440 NPPs (models 230 and 213) has been assessed. This assessment included tectonic stability of the site, design basis seismic ground motion and the seismic capacity of structures, systems and components. In all cases where the assessment was completed, it was found that the present capacity of the plants was insufficient for the newly established demand (i.e., seismic input). Therefore, seismic upgrading programmes have started and progressed to different degrees of completion in each plant. Obviously, the most difficult and costly upgrades (such as major structural work) are still not completed in most plants.

The CRP on Benchmark study for the seismic analysis and testing of WWER type NPPs was concluded at the end of 1997. This programme addressed WWER 1000 and WWER-440/213 type NPPs. The results of this study are being used extensively by the NPPs which are in the process of seismic upgrading.

In the future, the emphasis will shift from assessment to efficient ways and methods of upgrading as well as requalification of piping systems and equipment. IAEA assistance to review the work performed will continue in the framework of the Technical Co-operation projects.

Low power and shutdown operation

The safety of WWER plants at low power and shutdown operation needs to be analysed and the necessary improvements in hardware and procedures implemented.

In the field of deterministic accident analysis during shutdown conditions, the efforts should be concentrated on the validation of the available computer codes, the applicability of their relevant thermohydraulic correlations for shutdown conditions and on the development of initiating events and criteria. In the field of plant operation, administrative control, limits and conditions, emergency operating procedures, hardware modification resulting from analyses and experience feedback, and training of personnel before outages should be developed and strengthened.

The IAEA plans to establish a benchmark exercise to promote the application of the guidelines [16] that it has developed.

ATWS protection

Although the requirements of ATWS analysis for the licensing process differ among countries operating WWERs, the need for ATWS analysis is recognized in all countries and addressed in their safety improvement/modernization programmes. The IAEA has developed a guidance document on performing ATWS analyses of system transients for licensing purposes.

National policies on the ATWS issue in line with international practice, considering the systems necessary to mitigate consequences and/or requirements on acceptable plant performances in an ATWS event, need to be developed under the auspices of the safety authorities.

The IAEA plans to establish a benchmark exercise to promote the application of the guidelines [18] that it has developed.

Severe accident analysis and accident management

Analyses of severe accidents have not been systematically performed at WWER units. The results of these analyses are needed to prepare accident management measures for prevention and mitigation of beyond design basis or severe accidents.

A new IAEA safety services structure which falls under the regular budget and technical co-operation programmes is being offered in the area of accident management to Member States with NPPs in operation. The Accident Management Development and Implementation Services (AMS) will assist in developing plant specific AM programmes, review the analytical methodology application and relevant parts of accident analysis, and assist in the implementation of AM programmes. The assistance in the development of AM guidelines will be an integral part of the new services.

9.3. RBMK DESIGN

Since the Chernobyl accident, much work has been implemented to avoid its recurrence. Analyses carried out jointly by RBMK designers and operators and OECD experts have identified the need for additional safety improvements, particularly for RBMKs of the early generation. The need for implementing such improvements is generally agreed; however, the actual work is in many cases (e.g., Leningrad NPP Units 1-2) delayed due to financial problems.

The main areas which need to be pursued include:

Validation of computer codes

Much work has already been carried out to validate computer codes used for RBMK design verification and accident analysis. The ability of Western computer codes (e.g., RELAP, ATHLET, CATHARE) to simulate scenarios which could potentially lead to multiple pressure tube ruptures has received considerable attention.

The international exercise co-ordinated by the IAEA using experimental data provided by the Government of Japan has contributed to an improved understanding of the potential causes, nature and behaviour of flow oscillations in channel-type reactors.

Further work is still required to clarify remaining questions on the capability of the codes. International assistance is needed to complete this work.

Accident analysis

The IAEA has developed guidelines for accident analysis of RBMKs. To promote the application of accident analysis, an exercise is being initiated using the Kursk installation as a reference plant. Russian organizations are participating in this effort together with Western experts. Completion of this exercise should provide good example of the application of state-of-the-art methodology for the accident analysis of RBMKs and for inclusion in the ISAs. This project should be completed in 1999.

Core design improvements

The new fuel will lead to a decrease in the reactivity void effect even if the number of additional absorbers is reduced to zero. This new measure also decreases the safety significance of the operating reactivity margin. Work in this area is to be continued in the framework of national activities.

The use of advanced coupled 3D neutronical thermohydraulic codes is of utmost importance. Ongoing work to develop and adopt existing codes in the framework of bilateral and multilateral programmes is most important and should continue.

Shutdown system improvements

Work has begun to develop a fully independent and diverse additional shutdown system for RBMKs. The feasibility study has not yet been completed. There is general agreement that this work is of high priority. Therefore, design work needs to be completed and a safety evaluation performed to provide the safety justification for the implementation.

International assistance is of great importance to accelerate the development of the system for installation at all RBMKs.

A technical meeting was organized by the IAEA in 1995 to review various conceptual designs against IAEA NUSS requirements. A follow-up meeting will be organized by the IAEA at the end of the feasibility study.

Main circulation circuit integrity

Fuel channel integrity analysis needs to continue with emphasis on the safety assessment of reactor operation with a limited number of fuel channels locked in graphite as well as on ISI improvements (gas gap evaluation and destructive post-reactor testing).

Activities to apply the LBB concept to specific sections of the main circulation circuit (MCC, the reactor coolant system) have been initiated some time ago in Russia and are now well under way in the framework of international assistance programmes for a pilot plant (Smolensk NPP Unit 3). The application of the LBB concept should still be completed on a unit specific basis to all operating RBMKs and should include the outfitting of required leak detection systems.

The IAEA has assisted the Ignalina NPP to review the application of LBB. Similar assistance will be provided to other RBMK NPPs at the request of the Member States concerned.

Actions to address IGSCC in MCC 325 mm dia. austenitic stainless steel piping, which is a new and generic issue for RBMKs, has been initiated but still needs to be completed. Guidance based on comprehensive analysis of the problem as well as international experience accumulated to date has to be developed, addressing inspection, assessment, mitigation, repair and leak monitoring. A comprehensive practical programme to address this issue, based on guidelines and recommendations, needs to be implemented as a matter of urgency.

The IAEA is reviewing together with RBMK operators and regulators the results of the workshop held in Ukraine in June 1998 [146] to define what further assistance is required.

To maintain MCC integrity, improvements of ISI systems are necessary and have to be achieved through the use of up-to-date methods, techniques and equipment as well as through improvements of reliability of results by qualification of the whole ISI system.

9.4. OPERATIONAL SAFETY

At the request of the Member States concerned, the IAEA started providing operational safety services for WWER and RBMK nuclear power plants in 1988. In 1990, after the initiation of the EBP, this assistance was strengthened. The more recent OSART and ASSET missions have noted several improvements. Self-assessment practices are becoming increasingly important. However, there is still scope for improvement. The most important areas where further work needs to be done are: general management and safety culture, QA and document management, training and the development of emergency operating procedures.

In the ongoing USDOE assistance pursuant to the “Lisbon Initiative”, some working groups are still in the process of establishing a set of symptom-based EOPs. The Russian operating organization VNIIAES is developing beyond design basis accident management procedures for the WWER-1000 NPPs.

In the area of emergency planning and preparedness, bilateral assistance activities are under way within the TACIS and PHARE programmes to improve emergency management and for further operational improvements.

IAEA assistance will continue via its operational safety services (OSART, ASSET, ASCOT). Focus is on promotion and review of self-assessment.

9.5. SAFETY ASSESSMENT

Guidelines for in-depth safety assessment of WWER and RBMK plants have been approved by Gosatomnadzor, Russia, in September 1997. Actual work to prepare ISAs is under way in the framework of USDOE and EBRD sponsored projects. It is of utmost importance that this work be continued and ultimately completed at all NPPs.

Other countries operating WWER plants have issued new regulatory documents or requirements for safety analysis reports. The accident analysis part of the safety analysis report also needs to be upgraded commensurately with international practice.

It is expected that the IAEA, under its TC programmes, will play a major role in providing guidance on upgrading SARs for WWER plants.

In general, PSA results are being used to define new programmes of safety upgradings and to complement existing ones. However, the differences in the scope and technical quality of PSAs performed to date limit a wider sharing of insights among plants, even those of the same type. PSA measures and priorities defined on the basis of some PSA insights need careful consideration before they can be adopted.

Efforts to improve the quality of PSA models and data are most important and should be pursued to allow for a wider use of PSA insights. In this context, the involvement of plant operating staff in the development of the PSA is essential to ensure that it reflects the actual plant configuration and operation.

The ultimate objective should be to establish a realistic (living) PSA model which can be used as a management tool to optimize plant operation and to support decisions on plant modifications.

IAEA assistance will be continued to provide guidance on the elaboration of PSAs and IPERS.

9.6. IAEA ACTIVITIES PLANNED FOR THE BUDGET CYCLE 1999-2000

The IAEA will continue to provide nuclear safety assistance to its Member States operating these NPPs both in the framework of its nuclear safety programme (regular budget) and technical co-operation projects. A specific project on WWER and RBMK safety has been included in the IAEA Nuclear Safety Programme for 1999-2000 (H105). The project includes the following tasks:

- Technical visits to NPP sites to determine the status of implementation of safety improvements with respect to earlier IAEA recommendations;

- The updating of databases and their distribution to Member States;
- The evaluation of database information and preparation of technical reports on the solutions implemented to address safety issues of WWER and RBMK NPPs;
- The preparation of annual overview reports on the safety of WWER and RBMKs in Member States;
- Workshops to provide guidance on the preparation and review of selected sections of the SARs;
- The CRP on Round-Robin Exercises on WWER-440 RPV (water-cooled and moderated) weld metal irradiation embrittlement and annealing.

In the area of technical co-operation, three regional TC projects in Europe are being included in the 1999-2000 budget cycle. The specific activities of each project are being prepared by the IAEA Secretariat in consultation with the Member States concerned.

a) Support for Safety Assessment of Nuclear Power Plants

The objective of the project is to support and strengthen the existing capabilities of the operating and technical support organizations within Member States of the region to carry out safety analyses for their respective NPPs.

Within this project, the following activities will be undertaken: (1) provision of guidance material for the preparation of SARs including assistance for reviewing the impact of external and internal hazards; (2) assistance in the review of safety improvement programmes; (3) assistance in the review of PSAs; (4) assistance in the preparation of emergency and accident management; (5) assistance in reviewing human factor and human-machine interface issues; (6) assistance in reviewing performance and adequacy of plant simulators and analysers, etc.

The above topics will be addressed through expert review services, technical meetings, regional workshops, training courses, fellowships and scientific visits.

b) Capability for Assessment of Operational Safety of Nuclear Power Plants

The goal for the next years is to achieve region-wide excellence in operational safety management commensurate with internationally accepted levels.

The aims of this project are to assist Member States of the region in carrying out effective self-assessments of operational safety performance, perform peer reviews and provide the means and forum for the exchange of operating experience. Additionally, assistance will be directed at the deficiencies and safety gaps identified during the implementation of the integrated safety strategy for countries in the region.

The IAEA will provide training courses, regional workshops, ASSET and OSART missions to peer review self assessments, ASCOT seminars, assistance visits on topics identified during peer review missions and fellowships. The IAEA will act as the focal point for the information exchange and shall make certain its activity does not overlap with the activities of other international organizations.

c) Nuclear Safety Regulatory and Legislative Infrastructure

The goal of the project in the next years is to further strengthen the regulatory authorities in the region on the basis of IAEA recommendations and good international regulatory practices.

Assistance will focused on the legal framework, organization, review and assessment, preparation of regulatory requirements, licensing of operating personnel and emergency planning and preparedness.

The IAEA will provide training courses, regional workshops, IRRT missions, fellowships and scientific visits. For selected countries, the assistance to regulatory authorities will be through national TC projects in accordance with the specific needs of the country concerned.

10. CONCLUSIONS

There is general agreement among countries providing and receiving assistance that the specific objectives of the EBP have been fully met. These objectives were:

- to identify safety shortcomings in design and operation of WWER and RBMK NPPs;
- to evaluate the safety significance with respect to their impact on defence in depth;
- to establish international consensus on priorities for safety improvements;
- to provide assistance in the review of the completeness and adequacy of safety improvement programmes bearing in mind IAEA recommendations; and
- to undertake specific studies of unresolved topical safety issues.

The Programme results and related publications are being widely used as a technical basis for the development of safety upgradings for these plants and to prioritize national, bilateral and other international programmes.

The makeup of the EBP allowed great flexibility and a capacity to respond quickly to new requests and needs which emerged during the implementation of the programme. In this context, the role of the Programme’s Advisory Group and Steering Committees was particularly important to advise on priority actions and programme changes.

As a result of national and multilateral projects, many steps have been taken to enhance the safety of WWER and RBMK NPPs. However, there is a significant variation in the present safety status of individual WWER and RBMK NPPs, even among those of the same generations. Differences are mainly due to the following factors:

▪ Design modifications have been continuously incorporated in response to new safety requirements.

▪ While some plants were fully designed and built by former Soviet organizations, other plants were designed and constructed with a high degree of domestic participation.

▪ Different national approaches of plant owners/operators and regulators resulted in significant differences in plant safety upgradings since the startup of the plants.

▪ Safety improvement programmes have been dependent on available financing, which in turn is linked to the economic situation of individual countries and varies significantly from full national financing to a large dependence on international sources.

Although plans for safety upgrading exist in all countries, some countries have implemented only interim compensatory measures, while other countries are already in an advanced stage of major safety upgrading.

Despite the improvements in safety already achieved, much remains to be done at individual NPPs, particularly at those WWERs and RBMKs of earlier design. The work to be carried out at these plants will be particularly important if they are not decommissioned in the near future.

Of utmost importance is to ensure that for each NPP a safety demonstration based on plant specific safety analysis is developed by the operating organizations, reviewed and approved by the national regulatory authorities. This will allow an assessment of the overall safety impact of plant modifications. This process – which has been initiated, but not finalized in several plants – needs to be completed as a matter of highest priority.

Bilateral and multilateral assistance programmes will remain important to complement national efforts.

Upon completion of the EBP, the IAEA will continue to provide nuclear safety assistance to its Member States both in the framework of its nuclear safety programme and of technical co-operation projects. A specific project on WWER and RBMK safety has already been included in the IAEA Nuclear Safety Programme for 1999-2000. Three ongoing regional Technical Co-operation projects are also being extended until the year 2000.

An important element of this assistance is to strengthen the national regulatory authorities in the region on the basis of IAEA recommendations and good international regulatory practices.

An International Conference is being organized by the IAEA in co-operation with the EC and NEA in June 1999 to review the results of national, bilateral and other international programmes to enhance WWER and RBMK safety. Countries operating these NPPs are expected to provide a comprehensive overview of the actual achievements of all efforts to improve safety, and to identify those areas which require further work. The conference results should be of critical importance to focus future international assistance.

Safety upgrading projects have shown that substantial improvements in safety are feasible.

According to the Convention on Nuclear Safety, each Contracting Party “shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shutdown the nuclear installation as soon as practically possible. The timing of the shutdown may take into account the whole energy context and possible alternatives, as well as the social, environmental and economic impact.”

ABBREVIATIONS

AG	Advisory Group
AGM	Advisory Group Meeting
AGNES	Advanced General and New Evaluation of Safety, a project of Paks NPP
ALS	accident locations system (for RBMK reactors)
ASCOT	Assessment of Safety Culture in Organizations Teams
ASME	American Society of Mechanical Engineers (USA)
ASSET	Assessment of Safety Significant Events Teams (IAEA)
ATWS	anticipated transients without scram
BMBF	Research Ministry (Germany)
BMU	Ministry for the Environment (Germany)
BRU-A	steam dump valve to atmosphere
BRU-K	steam dump valve to turbine condenser
BWR	boiling water reactor
CCF	common cause failure
CDRM	Conceptual Design Review Meeting
CEEC	Central and Eastern European Countries
CKTI	Central Design and Engineering Institute
CM	Consultants Meeting
CNRA	Committee on Nuclear Energy Regulatory Activities (NEA)
CPS	control and protection system
CRP	Co-ordinated Research Programme (IAEA)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
DBA	design basis accident
DC	direct current
DEGB	double ended guillotine break
EBP	Extrabudgetary Programme

EBRD	European Bank for Reconstruction and Development
EC	European Commission
ECCS	emergency core cooling system
EdF	Electricité de France (French utility)
EFWS	emergency feedwater system
EIB	European Investment Bank
ENIQ	European Network for Inspection and Qualification (EU)
EOP	emergency operating procedure(s)
ESFAS	emergency safety features actuation system
ESRS	Engineering Safety Review Service
EU	European Union
FSU	Former Soviet Union
GAN	Russian Nuclear Energy Authority (Gosatomnadzor)
GDH	group distribution header
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH (Germany)
HAEC	Hungarian Atomic Energy Commission
HP	high pressure
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical Commission
IGSCC	intergranular stress corrosion cracking
INITEC	National Institute of Industry and Technology (Spain)
IPERS	International Peer Review Service (IAEA)
IRRT	International Regulatory Review Team (IAEA)
ISA	in-depth safety analysis
ISI	in service inspection
JAERI	Japan Atomic Energy Research Institute
LAC	local automatic control

LBB	leak before break
LEP	local energy protection
LBLOCA	large break loss of coolant accident
LOCA	loss of coolant accident
LPS	low power and shutdown
MCP	main circulation pump
MCR	main control room
MITI	Ministry of International Trade and Industry (Japan)
MS	Member State(s)
NDT	non-destructive testing
NEA	Nuclear Energy Agency
NFIR	number of fully inserted rod
NIS	new independent states
NNAEP	Nijni-Novgorod Atom Energo Project
NPP	nuclear power plant
NSA	nuclear safety account
NSSS	nuclear steam supply system
NUSAC	Nuclear Safety Co-ordination
NUSS	Nuclear Safety Standards of the IAEA
OECD	Organization for Economic Co-operation and Development
ORM	operational reactivity margin
OSART	Operational Safety Review Teams (IAEA)
PHARE	Poland and Hungary Assistance for Reconstruction of Economy
PNC	Power Reactor and Nuclear Fuel Development Corporation (Japan)
PNNL	Pacific Northwest National Laboratory
PRISE	primary to secondary leakage
PSA	probabilistic safety assessment
PSU	programmes of safety upgradings

PTS	pressurized thermal shock
PWR	pressurized water reactor
RBMK	boiling water cooled graphite moderated pressure tube type reactor
RCPS	reactor control and protection system
RCV	reactor cavity venting
RDIPE	Research and Development Institute of Power Engineering (formerly NIKIET)
RPV	reactor pressure vessel
SAR	safety analysis report
SBLOCA	small break loss of coolant accident
SC	Steering Committee
SCM	Steering Committee Meeting
SG	steam generator
SI	safety injection
SIP	Swedish International Project
SIR	safety improvement report
SKI	Statens Kärnkraftinspektion (Sweden, Swedish Nuclear Power Inspectorate)
SPDS	safety parameter display system
SQUG	seismic qualification utility group (USA)
SRM	safety review mission
STUK	Säteilyturvakeskus (Finland, Finnish Centre for Radiation and Nuclear Safety)
SWS	service water system
TACIS	Technical Assistance to the Community of Independent States
TC	Technical Co-operation (IAEA)
TECDOC	technical document
TOB	Technical Justification for Nuclear Safety (Russian terminology)
UJD	Nuclear Regulatory Authority of the Slovak Republic
UK	United Kingdom of Great Britain and Northern Ireland

US/USA	United States of America
USDOE	United States Department of Energy
UT	ultrasonic test
VATESI	State Regulatory Authorities for Nuclear Energy Safety
VNIIAES	All Russian Research Institute for Nuclear Power Plant Operation
VNIIAM	All Russian Scientific and Research Institute of Atomic Machinery Building
VNIPIET	All Russian Scientific Research and Design Institute
VRC	void reactivity coefficient
WANO	World Association of Nuclear Operators
WEC	Westinghouse Electric Corporation
WS	workshop
WWER	Water cooled, water moderated energy reactor

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ANNEX 1: GENERAL TERMS OF REFERENCE FOR STEERING COMMITTEES ON WWER AND RBMK SAFETY PROGRAMMES[1]

1. To advise the IAEA on the major programme activities including the preparation of missions[2] to specific NPP sites and topical review meetings.

2. To evaluate the results of the programme activities and to propose high priority near term actions (both related to hardware and software) for improving plant safety.

3. To identify unresolved safety issues and propose required safety studies.

4. To monitor progress of project implementation.

5. To be a forum for the exchange of information on work going on and/or planned in the frame of IAEA and CEC programmes and related activities of OECD/NEA, WANO and financial organizations (The World Bank, EIB and EBRD). To give advice on matters requiring co-ordination.

6. To advise the IAEA on matters requiring co-ordination between IAEA's Extrabudgetary Programmes and TC activities.

7. To advise the IAEA on matters requiring co-ordination with the G-24 programme and the RBMK consortium and to consider what technical support the IAEA should provide to the G-24 co-ordinated activities.

[1] Member States nominate members of the Steering Committees and the IAEA designates them.
[2] Safety review missions, OSARTs, ASSETs, Seismic safety missions, ASCOTs, and related follow-up missions.

WWER Steering Committee Members

COUNTRY		PERIOD
ARMENIA	Mr. V. Nersesyan	1998,1996
	Mr. A. Martirosyan	1997
	Mr. A. Avakian	1994,1996
AUSTRIA	Mr. G. Weimann	1993-1998
BULGARIA	Mr. G. Kastchiev	1997-1998
	Mr. L.K. Kostov	1996
	Mr. Y. Yanev	1991-1996
CZECH REPUBLIC	Mr. J. Stuller	1993-1996
	Mr. M. Sváb (representing Mr Stuller)	1994,1997-98
	Mr. M. Hrehor (representing Mr. Stuller)	1992-1993, 1996
CZECH AND SLOVAK FEDERAL REPUBLIC	Mr. Z. Kriz	1991-1992
FINLAND	Mr. H. Väyrynen	1993-1998
FRANCE	Mr. Y. Cornille	1991-1998
GERMANY	Mr. P. Kelm	1994-1998
	Mr. F.W. Heuser	1991-1994
HUNGARY	Mr. L. Vöröss	1996-1998
	Mr. G. Vajda	1995
	Mr. J. Vigassy	1993-1994
JAPAN	Mr. T. Haga	1992-1998
	Mr. Y. Anoda	1996,1998
	Mr. A. Kohsaka	1992-1996
	Mr. T. Shinkawa	1996
	Mr. M. Kato	1996-1998
	Mr. M. Hirano	1997
RUSSIAN FEDERATION	Mr. A.M. Kirichenko	1991-1998
	Mr. A. Abagian	1991-1992, 1995-1996
SLOVAK REPUBLIC	Mr. M. Lipar	1998
	Mr. J. Misak	1993-1997
SPAIN	Mr. A. Alonso	1991-1998

COUNTRY		PERIOD
SWITZERLAND	Mr. A. Voumard	1991-1998
UKRAINE	Mr. L. Benkovskii	1994-1998
	Mr. V. Kim	1994-1996
UNITED KINGDOM	Mr. R. Bye	1994-1998
	Mr. D. Goodison (Chairman)	1991-1998
UNITED STATES OF AMERICA	Mr. W. Pasedag	1994-1998
	Mr. G. Chipman	1991-1993
OBSERVERS		
EBRD	Mr. D. Rousseau	1993-1998
	Mr. O. Herbelot	1992
EUROPEAN COMMISSION	Mr. L. Lopez Arcos (representing Mr. Jousten)	1993-1998
	Mr. N. Jousten	1993-1998
	Mr. R. Timans	1993
	Mr. R. Cibrian	1992-1993
European Investment Bank	Mr. A. Boioli	1993-1994
G-24 NUSAC Secretariat	Mr. A. Lacroix	1997-1998
	Mr. Ch. Gronemeyer	1994-1996
	Mr. H.W. Kalfsbeek	1993-1994
OECD Nuclear Energy Agency	Mr. H. Rosinger (representing Mr. Frescura)	1996-1997
	Mr. N. Maki (representing Mr. Frescura)	1994,1996
	Mr. G.M. Frescura	1993-1997
WANO	Mr. J.P. Baret	1991-1998
	Mr. P. Blanc	1997-1998
The World Bank	Mr. A.G. Adamantiades	1991-1992

RBMK STEERING COMMITTEE MEMBERS

COUNTRY		PERIOD
CANADA	Mr. A. Brown	1993-1995
FRANCE	Mr. M.D. Bastien	1993-1998
GERMANY	Mr. E. Kersting	1997-1998
	Mr. D. Reichenbach	1995-1996
JAPAN	Mr. S. Shibuya	1998
	Mr. H. Mochizuki	1997
	Mr. Y. Hayamizu	1993-1996
	Mr. Y. Kasai	1993
LITHUANIA	Mr. S. Kutas	1997-1998
	Mr. P. Vaishnys	1994-1996
	Mr. S. Brasas	1993
RUSSIAN FEDERATION	Mr. B.A. Gabaraev	1995-1998
	Mr. E.O. Adamov	1993-1994
SPAIN	Mr. A. Alonso	1993-1994, 1998
SWEDEN	Mr. J.H. Nistad	1993-1998
SWITZERLAND	Mr. S. Chakraborty (Chairman)	1993-1998
UKRAINE	Mr. V.V. Gryshchenko	1993-1998
	Mr. G. Kopchinsky	1993
UNITED STATES OF AMERICA	Mr. G.L. Smith	1994-1998
	Mr. M. Zentner	1994
	Mr. D. Giessing	1993

OBSERVERS

EBRD	Mr. G. Grabia	1993-1994, 1997-1998
	Mr. F. Maltini	1995-1996
	Mr. F. Démarcq	1993
European Commission	Mr. L.V. Bril	1993-1998
	Mr. R. Timans	1994
	Mr. F.W. Kalfsbeek	1993
G-24 NUSAC Secretariat	Mr. J-P. Lacroix	1997-1998
	Mr. A. Lapicoré	1993-1996
International RBMK Consortium	Mr. M. Hayns	1993-1994

ANNEX 2: BUDGET AND MANPOWER

INTERNATIONAL ATOMIC ENERGY AGENCY

EXTRABUDGETARY PROGRAMME ON THE SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS
STATUS OF INCOME FROM CONTRIBUTIONS AS AT DECEMBER 1998 (IN US $)

DONOR	*Contributions to WWER 1991-94*	*Contributions to RBMK 1991-94*	*1995*	*1996*	*1997*	*1998*	*TOTAL INCOME*
Austria	75,502.79	-	18,181.82	-	32,554.75	-	126,239.36
Belgium	-	-	30,000.00	30,000.00	-	15,000.00	75,000.00
Canada	19,841.27	19,841.27	-	-	-	-	39,682.54
Finland	-	-	18,519.70	-	-	-	18,519.70
Germany	307,080.52	-	-	-	-	-	307,080.52
Japan	1,408,747.00 **b/**	1,683,785.00 **a/**	1,485,811.00	1,260,805.00	1,504,257.00 **d/**	1,535,691.00 **d/**	8,879,096.00
Netherlands	175,438.60	314,465.41	-	-	-	-	489,904.01
Norway	30,793.40	-	15,552.10	-	-	-	46,345.50
Sweden	30,000.00	-	-	-	-	-	30,000.00
Spain	300,000.00	79,103.09	122,043.79	1,261.19	-	-	502,408.07
Switzerland	278,280.59	205,025.62	530,197.18	290,977.96	207,854.41	68,493.00	1,580,828.76
United Kingdom	100,000.00	-	-	-	-	-	100,000.00
USA	500,000.00 c/	-	500,000.00	-	320,000.00	100,000.00	1,420,000.00
TOTAL INCOME	**3,225,684.17**	**2,302,220.39**	**2,720,305.59**	**1,583,044.15**	**2,064,666.16**	1,719,184.00	13,615,104.46

a/ also for WWER-440/213, WWER-1000
b/ for WWER-440/230
c/ for assistance to Bulgaria
d/ includes the contribution to the project on the Safety of Nuclear Installations in Asia

COST-FREE EXPERTS PARTICIPATING IN WWER-RBMK EXTRABUDGETARY ACTIVITIES

COUNTRY	*1991*	*1992*	*1993*	*1994*	*1995*	*1996*	*1997*	*1998*
France	1	1	1	2	2	1	1	1
Germany				1	1	1	1	1
Hungary*		1	1	1	1			
Spain					1	1	1	
Switzerland				1	1			
USA		1	1	1				
European Commission			1	1				

* partially paid by the IAEA

EXPERTS PARTICIPATING IN WWER-RBMK EXTRABUDGETARY ACTIVITIES (IN MAN-DAYS)*

COUNTRY	***1990***	***1991***	***1992***	***1993***	***1994***	***1995***	***1996***	***1997***	***1998***
Argentina		6	17	9	5				
Armenia				3	14	8		7	10
Austria	24		9	6	47	37	19		7
Belgium	12	20	34	37	56	14	12	11	2
Brazil				55				3	
Bulgaria			94	106	107	69		11	22
Canada		19	66	102	56	19	25	20	
China					10	5	5		5
Croatia						5			
CSFR			77						
Czech Republic				40	256	92	19	26	36
EBRD			8	19	26	39	12	9	10
EC	24	24	96	72	35	19	25	10	11
EIB			3	9	5	5	2		
ENAC					5				
Finland		82	18	126	122	62	62	12	17
France	12	83	91	233	224	151	93	32	26
Germany	12	123	124	138	246	138	90	66	41
Hungary	24	24	12	66	69	55	24	15	14
India					12				
International RBMK Consortium				18	2	56			
Italy	12	20	48	58	17	26	13		4
Japan		40	119	148	144	92	81	15	19
Korea, Republic of					12				
Lithuania			38	29	108	82	17	18	9
Netherlands		20	3	36	3	2	5		
Norway			3		3	4	5		
NUSAC				44	33	48	15	4	7
OECD/NEA				9	14	14	6	4	2
Pakistan				19					
Poland					14		16		
Republic of Macedonia				5					
Romania	24								
Russian Federation	24	20	217	439	481	333	189	161	67
Slovak Republic				76	93	121	35	15	35
Slovenia			5	19	3			5	
South Africa		19		4	12	10	5		
Spain		97	89	137	159	71	79	12	16
Sweden	12		51	54	56	93	39	13	6
Switzerland		39	67	80	57	38	63	21	15
Ukraine			18	43	99	135	42	70	17
United Kingdom	24	109	90	218	183	64	57	16	16
United States of America	12	179	323	392	330	81	103	50	36
WANO			33	14	16	22	8	8	8
World Bank			68	3					
Yugoslavia		19							
Total	216	943	1821	2866	3134	2010	1166	634	458

*excludes experts recruited under TC projects.

ANNEX 3: WORK PROGRAMME FOR 1990 - 1998

ANNEX 4: SAFETY ISSUES FOR WWER 440/230 NUCLEAR POWER PLANTS

Area	Issue title	Ranking
Reactor core		
	In-core monitoring	II
	Core design margin evaluation	II
	Fuel examination	II
	Reloading procedures and test programme	II
Systems		
	Confinement - Leaktightness	III
	Confinement - Severe accident conditions	III
	Decay heat removal - SG inventory	IV
	Decay heat removal - Heat removal path	II
	Decay heat removal - Service water system	III
	Decay heat removal - Component reliability	III
	ECCS - full LOCA spectrum capability and long term cooling	III
	ECCS - redundancy and separation	IV
	ECCS - primary break isolation options	II
	Main steamline isolation	IV
	Primary circuit pressure relief	III
	Secondary circuit pressure relief	II
	Reliable isolation	II
	Ventilation/cooling capability	III
	Dynamic loads due to piping failures	II
Components integrity		
	Embrittlement - Baseline information	IV
	Embrittlement - Validation of annealing	IV
	Embrittlement - Flux reduction	IV
	Embrittlement - Cold pressurization	IV
	Vessel ISI - Inspection techniques	III
	Vessel ISI - Corrosion monitoring	II
	Vessel stress analysis	III
	Leak before break applicability	IV
	Primary circuit in-service inspection	III
	Primary circuit stress analysis	III
	Vessel support integrity	III
	Secondary circuit in-service inspection	II
Instrumentation and control		
	Accident monitoring instrumentation	II
	Reliability of I&C equipment	III
	Control and protection systems interaction	II
	I&C redundancy, separation and independence	IV
	I&C support to operation and control room design	III
	Interlocking	II
	I&C/Electrical equipment qualification	III
	I&C/Electrical equipment classification	III
	I&C signal priority	III

Area	Issue title	Ranking
	Testability of I&C equipment	III
	Control room habitability/Shutdown panel	III
	Instrumentation setpoint margins	II
Electrical power supply		
	Electrical redundancy, separation and independence	IV
	Reliability of electrical equipment	III
	Diesel generator loading	IV
	Battery discharge time	III
	Connection to offsite power supplies	II
Accident analysis		
	Confinement analysis	III
	Emergency protection signals	III
	Severe accident analysis	II
	Accident during shutdown or refuelling	II
	Qualification of safety analysis	II
	Scope of accident and transient analysis	III
	Loss of coolant accidents	III
	Radiological consequences	II
	Evaluation of modifications	III
Fire protection		
	Fire protection - Analysis	III
	Fire protection - Equipment	III
	Fire protection - Inspection	III
Management		
	Management involvement	IV
	Management development	III
	Safety culture	IV
	Housekeeping	II
	Organization	III
	Modification control	III
	Document management	I
	Configuration management	III
	Experience feedback	III
	Quality assurance	III
	Radiation protection practices	II
	Industrial safety practices	II
	Computer utilization	I
Operating procedures		
	Procedures - Programme	II
	Emergency operating procedures	IV
	Limits and conditions	III
	Procedures - Operation	II
Plant operations		
	Surveillance programme	II
	Procedures - Surveillance	III
	Work control	III

Area	Issue title	Ranking
	Organization of shifts	III
	Labels and operation aids	II
	Chemistry	I
Maintenance		
	Maintenance programme	II
	Procedures - Maintenance	II
	Equipment material conditions	IV
	Warehouses	I
Training		
	Training programme	III
	Training of plant operators	III
	Training - Facilities	II
	Training material	III
	Training records	I
Emergency planning		
	Emergency response programme	III
	Emergency response - Procedures	III
	Emergency response - Facilities	III
	Emergency response - Training	II
	Post accident sampling	II

SAFETY ISSUES FOR WWER 440/213 NUCLEAR POWER PLANTS

Area	Issue title	Ranking
General		
	Classification of components	II
	Qualification of equipment	III
	Reliability analysis of safety class 1 and 2 systems	II
Reactor core		
	Prevention of uncontrolled boron dilution	II
Components integrity		
	Reactor pressure vessel integrity	II
	Non-destructive testing	III
	Primary pipe whip restraints	II
	Steam generator collector integrity	II
	Steam generator tube integrity	II
	Steam generator feedwater distribution pipe	I
Systems		
	Primary circuit cold overpressure protect ion.	II
	Mitigation of a steam generator primary collector break	II
	Reactor coolant pump seal cooling system	II
	Pressurizer safety and relief valves qualification for water flow	II
	ECCS sump screen blocking risk	III
	ECCS suction line integrity	II
	ECCS heat exchanger integrity	II
	Power operated valves on the ECCS injection lines	I
	Steam generator safety and relief valves qualification for water flow	II
	Steam generator safety and relief valves performance at low pressure	II
	Steam generator level control valves	I
	Emergency feedwater make-up procedures	I
	Feedwater supply vulnerability	III
	Main control room ventilation system	II
	Hydrogen removal system	II
	Primary circuit venting under accident conditions	II
	Essential service water system	II
Instrumentation and control		
	I&C Reliability	II
	Safety system actuation design	I
	Review of reactor scram initiating signals	II
	Human engineering of control rooms	II
	Physical and functional separation between the main and emergency control	II
	Condition monitoring for the mechanical equipment	I
	Primary circuit diagnostic system	II
	Reactor vessel head leak monitoring system	II
	Accident monitoring instrumentation	II
	Technical support centre	II

Area	Issue title	Ranking
	Water chemistry control & monitoring equipment (primary & secondary)	I
Electrical power supply		
	Start up logic for the emergency diesels	I
	Diesel generator reliability	I
	Protection signals for emergency diesel generators	I
	On-site power supply for incident and accident management	II
	Emergency batteries discharge time	II
Containment		
	Bubbler condenser strength behaviour at maximum pressure difference possible under LOCA	III
	Bubbler condenser thermodynamic behaviour	II
	Containment leak rates	II
	Maximum pressure differences on walls between compartments of hermetic boxes	II
	Peak pressure in containment and activation of sub-atmospheric pressure after blowdown	I
Internal hazards		
	Systematic fire hazards analysis	II
	Fire prevention	III
	Fire detection and extinguishing	II
	Mitigation of fire effects	II
	Systematic flooding analysis	I
	Turbine Missiles	I
	Internal hazards due to high energy pipe breaks	III
	Heavy load drop	I
External hazards		
	Seismic design	III
	Extreme cold	I
	Man induced external events	II
Accident analysis		
	Scope and methodology of accident analysis	II
	QA of plant data used in accident analysis	I
	Computer code and plant model validation	II
	Availability of accident analysis results for supporting	I
	Main steam line break accident analysis	I
	Overcooling transients related to pressurized thermal	II
	Steam generator collector rupture analysis	II
	Accidents under low power and shutdown conditions	II
	Severe accidents	I
	Probabilistic safety assessment (PSA)	I
	Boron dilution accidents	I
	Spent fuel cask drop accidents	I
	Anticipated transient without scram (ATWS)	I
	Total loss of electrical power	I
	Total loss of heat sink	I

Area	Issue title	Ranking
Operation		
	Procedures for normal operation	
Emergency operating procedures		
	Limits and conditions	
	Need for safety culture improvements	
	Experience feedback	
	Quality assurance programme	
	Data and document management	
	Philosophy on use of procedures	
	Surveillance programme	
	Communication system	
	Radiation protection and monitoring	
	Training programmes	
	Emergency control centre	

SAFETY ISSUES FOR WWER 1000/320 NUCLEAR POWER PLANTS

Area	Issue title	Ranking
General		
	Classification of components	II
	Qualification of equipment	III
	Reliability analysis of safety class 1 and 2 systems	II
Reactor core		
	Prevention of inadvertent boron dilution	II
	Control rod insertion reliability/Fuel assembly deformation	III
	Subcriticality monitoring during reactor shutdown conditions	II
Component integrity		
	RPV embrittlement and its monitoring	III
	Non-destructive testing	III
	Primary pipe whip restraints	II
	Steam generator collector integrity	III
	Steam generator tube integrity	II
	Steam and feedwater piping integrity	III
Systems		
	Primary circuit cold overpressure protection	II
	Mitigation of a steam generator primary collector break	II
	Reactor coolant pump seal cooling system	II
	Pressurizer safety and relief valves qualification for water flow	II
	ECCS sump screen blocking	II
	ECCS water storage tank and suction line integrity	II
	ECCS heat exchanger integrity	II
	Power operated valves on the ECCS injection lines	I
	Steam generator safety and relief valves qualification for water flow	III
	Steam generator safety valves performance at low pressure	II
	Steam generator level control valves	I
	Emergency feedwater makeup procedures	I
	Cold emergency feedwater supply to SG	I
	Ventilation system of control rooms	II
	Hydrogen removal system	II
Instrumentation and control		
	I&C reliability	II
	Safety system actuation design	I
	Automatic reactor protection for power distribution and DNB	I
	Human engineering of control rooms	II
	Control and monitoring of power distributions in load follow mode	II
	Condition monitoring for the mechanical equipment	I
	Primary circuit diagnostic systems	II
	Reactor vessel head leak monitoring system	III
	Accident monitoring instrumentation	II
	Technical support centre	II
	Water chemistry control and monitoring equipment (primary and	

Area	Issue title	Ranking
	secondary)	I
Electrical power supply		
	Off-site power supply via startup transformers	I
	Diesel generator reliability	I
	Protection signals for emergency diesel generators	I
	On-site power supply for incident and accident management	II
	Emergency battery discharge time	III
Containment		
	Containment bypass	II
Internal hazards		
	Systematic fire hazards analysis	II
	Fire prevention	III
	Fire detection and extinguishing	II
	Mitigation of fire effects	II
	Systematic flooding analysis	I
	Flood protection for emergency electric power distribution boards	II
	Dynamic effects of main steam and feedwater line breaks	II
	Polar crane interlocking	II
External hazards		
	Seismic design	II
	Analyses of plant specific natural external conditions	I
	Man-induced external events	II
Accident analysis		
	Scope and methodology of accident analysis	II
	QA of plant data used in accident analysis	I
	Computer codes and plant model validation	I
	Availability of accident analysis results for supporting plant operation	I
	Main steam line break analysis	I
	Overcooling transients related to pressurized thermal shock	II
	Steam generator collector rupture analysis	II
	Accidents under low power and shutdown (LPS) conditions	II
	Severe accidents	I
	Probabilistic safety assessment (PSA)	I
	Boron dilution accidents	I
	Spent fuel cask drop accidents	I
	ATWS	II
	Total loss of electrical power	II
	Total loss of heat sink	II
Operating procedures		
	Procedures for normal operation	
	Emergency operating procedures	
	Limits and conditions	
Management		
	Need for safety culture improvements	

Area	Issue title	Ranking
	Experience feedback	
	Quality assurance programme	
	Data and document management	
Plant operation		
	Philosophy on use of procedures	
	Surveillance programme	
	Communication system	
Radiation protection		
	Radiation protection and monitoring	
Training		
	Training Programmes	
Emergency planning		
	Emergency centre	

SAFETY ISSUES FOR 'SMALL SERIES' WWER 1000 NUCLEAR POWER PLANTS

Area	Issue title	Ranking
General		
	Classification of components	II
	Qualification of equipment	III
	Reliability analysis of safety class 1 and 2 systems	II
Reactor core		
	Prevention of inadvertent boron dilution	II
	Control rod insertion reliability/Fuel assembly deformation	II
	Subcriticality monitoring during reactor shutdown conditions	I
Component integrity		
	RPV embrittlement and its monitoring	III
	Non-destructive testing	III
	Primary pipe whip restraints	II
	Steam generator collector integrity	III
	Steam generator tube integrity	II
	Steam and feedwater piping integrity	III
	Structural integrity related monitoring	II
Systems		
	Primary circuit cold overpressure protection	II
	Mitigation of a steam generator primary collector break	II
	Reactor coolant pump seal cooling system	I
	Pressurizer safety and relief valves qualification for water flow	II
	ECCS sump screen blocking	III
	ECCS suction line integrity	I
	ECCS heat exchanger integrity	II
	Power operated valves on the ECCS injection lines	I
	Steam generator safety and relief valves qualification for water flow	II
	Steam generator safety valves performance at low pressure	I
	Steam generator level control valves	I
	Ventilation system of control rooms	II
	Hydrogen removal system	II
	Boron injection system capability (Novovoronezh NPP)	III
	Boron injection system capability (others NPPs)	I
	Feedwater supply vulnerability	III
	Physical separation and functional isolation of the ECCS	III
	Limited boron acid storage for HP injection	II
Instrumentation and control		
	I&C reliability	II
	Safety system actuation design	I
	Automatic reactor protection for power distribution and DNB	I
	Human engineering of control rooms	II
	Reactor protection system redundancy	III

Area	Issue title	Ranking
	Condition monitoring for the mechanical equipment	I
	Primary circuit diagnostic systems	II
	Accident monitoring instrumentation	II
	Technical support centre	II
	Water chemistry control and monitoring equipment (primary and secondary)	I
	Separation of the primary circuit instrumentation taps to I&C detectors	II
	Nuclear instrumentation system range overlap	I
Electrical power supply		
	Diesel generator reliability	I
	Protection signals for emergency diesel generators	I
	On-site power supply for incident and accident management	II
	Emergency battery discharge time	III
	Ground faults in DC circuits	II
Containment		
	Containment bypass	I
Internal hazards		
	Systematic fire hazards analysis	II
	Fire prevention	III
	Fire detection and extinguishing	II
	Mitigation of fire effects	II
	Systematic flooding analysis	I
	Protection against flood for emergency electric power distribution boards	II
	Protection against the dynamic effects of main steam and feedwater line breaks	II
	Polar crane interlocking	II
External hazards		
	Seismic design	II
	Analyses of plant specific natural external conditions	I
	Man-induced external events	II
Accident analysis		
	Scope and methodology of accident analysis	II
	QA of plant data used in accident analysis	I
	Computer code and plant model validation	I
	Availability of accident analysis results for supporting plant operation	I
	Main steam line break analysis	I
	Overcooling transients related to pressurized thermal shock	II
	Steam generator collector rupture analysis	II
	Accidents under low power and shutdown (LPS) conditions	II
	Severe accidents	I
	Probabilistic safety assessment (PSA)	I
	Boron dilution accidents	I
	Anticipated transients without scram (ATWS)	II
	Total loss of electrical power	II

Area	Issue title	Ranking
	Total loss of heat sink	II
Operating procedures		
	Procedures for normal operation	
	Emergency operating procedures	
	Limits and conditions	
	Need for safety culture improvements	
	Experience feedback	
	Quality assurance programme	
	Data and document management	
	Philosophy on use of procedures	
	Surveillance programme	
	Communication system	
	Radiation protection and monitoring	
	Training programmes	
	Emergency centre	

SAFETY ISSUES FOR RBMK NUCLEAR POWER PLANTS

Area	Issue title	Ranking
Core design and core monitoring		
	Core design and core design methods	High
	Core design void reactivity coefficient of the primary and CPS circuit	High
	Spatial power control and protection	Medium
	Operational reactivity margin (ORM)	High
	Additional shutdown system	High
	Subcriticality margins in first generation RBMKs	High
Instrumentation and control		
	Diversity and segregation of I&C systems	High
	Initiation of ECCS and other safety systems	High
	I&C system maintenance and periodic testing	Medium
	Reliability of I&C systems	Low
	Replacement of NPP main computer	Medium
	I&C equipment upgrades	Medium
	Operator support	Medium
Pressure boundary integrity		
	Fulfillment of inspection requirements	High
	In-service inspection	High
	Break of critical components	High
	Fuel channel and tract integrity	High
	Special channel integrity	Low
	Fuel handling	Medium
	Seismic and ageing assessment	Medium
Accident analysis		
	Scope and methodology for accident analysis	Medium
	LOCA analysis	High
	Cavity overpressure protection	High
	Steam line break analysis	Medium
	Pipe whip analysis	Medium
	Loss of power (Black out)	Medium
	Radiological consequence analysis	Medium
	Performance and utilization of PSA	Medium
	Anticipated transient without scram (ATWS)	High
	External hazards	Medium
Safety and support systems		
	ECCS capability and performance	High
	Long term cooling and water make up	Medium
	ECCS reliability improvement	High
	Reactor trip and ECCS - actuation signals	High
	Interfacing system LOCA	Medium
	Adequacy of confinement function	High
	Habitability and accessibility of essential control areas	High
	Reliability of ultimate heat sink	Medium

Area	Issue title	Ranking
	Reliability of electrical system	Medium
	Electrical equipment qualification	Medium
	Diesel generator reliability	Medium
Fire protection		
	Passive fire protection	High
	Automatic fire detection	Medium
	Manual fire suppression capability	Medium
	Automatic fire suppression capability	Medium
	Fire water supply	Low
Operational safety		
	Organization and staffing	
	Quality assurance	
	Safety culture	
	Management of documents	
	Material condition	
	Training programmes and materials	
	Operating procedures for normal operation	
	Emergency operating procedures	
	Experience feedback and event investigation	
	Maintenance programme	
	Modification control	
	Surveillance test programme	
	Radiation protection programmes	

APPENDIX: LIST OF SAFETY MISSIONS

In 1991:	Plant type(s)	Reference/Report No.
Bohunice NPP Units 1-2	(WWER-440/230)	WWER-RD-022
Kozloduy NPP Units 1-4	(WWER-440/230)	WWER-RD-033
Novovoronezh NPP Units 3-4	(WWER-440/230)	WWER-RD-034
Kola NPP Units 1-2	(WWER-440/230)	WWER-RD-035
In 1992:		
Bohunice NPP Units 1-2	(WWER-440/230)	WWER-SC-038
In 1993:		
Kozloduy NPP Units 1-4	(WWER-440/230)	WWER-RD-049
Novovoronezh NPP Units 3-4	(WWER-440/230)	WWER-RD-050
Smolensk NPP Unit 3	(RBMK)	IAEA-TECDOC-722
In 1994:		
Kola NPP Units 1-2	(WWER-440/230)	WWER-RD-068
Mochovce NPP	(WWER-440/213)	WWER-RD-074
Bohunice NPP Units 3-4	(WWER-440/213)	WWER-RD-078
Zaporozhe NPP Units 1-6	(WWER-1000/320)	WWER-RD-064
Chernobyl NPP Units 1-3	(RBMK)	WWER-RD-004
Ignalina NPP Unit 2	(RBMK)	IAEA-EBP-RBMK-03
In 1995:		
Kozloduy NPP Units 5-6	(WWER-1000/320)	WWER-SC-143
Rovno NPP Unit 4	(WWER-1000/320)	WWER-SC-151
Dukovany NPP	(WWER-440/213)	WWER-SC-160
Novovoronezh NPP Units 3-4	(WWER-440/230)	WWER-SC-161
In 1996:		
Temelin NPP Units 1-2	(WWER-1000/320)	WWER-SC-171
Khmelnitsky NPP Unit 2	(WWER-1000/320)	WWER-SC-178
South Ukraine NPP Units 1-2	(WWER-1000/302,338)	WWER-SC-182
Paks NPP Units 1-4	(WWER-440/213)	WWER-SC-197
Kozloduy NPP Units 1-4	(WWER-440/230)	WWER-SC-164
Bohunice NPP Units 1-2	(WWER-440/230)	WWER-SC-180
In 1997:		
Novovoronezh NPP Unit 5	(WWER-1000/187)	WWER-SC-199
Kola NPP Units 1-2	(WWER-440/230)	WWER-SC-107
Leningrad NPP Unit 2	(RBMK)	RBMK-SC-049

GENERIC SAFETY ISSUES STUDIED

	Plant type(s)	Reference/Report No.
Technical basis for I&C design improvements in WWER-440/230 and 440/213 NPPs	(WWER-440/230 and WWER-440/213)	IAEA-EBP-WWER02, WWER-SC-040, WWER-SC-099, WWER-SC-106
Review of the methods used for leak rate measurements for WWER-440/230 confinements and WWER-440/213 containments	(WWER-440/230 and WWER-440/213)	IAEA-EBP-WWER-10, WWER-SC-085, WWER-SC-101, WWER-SC-135, WWER-SC-139, WWER-SC-149, WWER-SC-170
Bubbler condenser structural integrity	(WWER-440/213)	WWER-SC-095, WWER-SC-142, WWER-SC-159, WWER-SC-166, WWER-SC-170, WWER-SC-183
Core control and protection strategy	(WWER-1000/320)	WWER-RD-071
Control rod insertion reliability	(WWER-1000/320)	WWER-SC-121
WWER-1000 steam generator integrity	(WWER-1000/320)	IAEA-EBP-WWER-07, WWER-RD-057, WWER-SC-176, WWER-SC-115
Anticipated transients without scram (ATWS) for WWER-1000 reactors	(WWER-1000/320)	WWER-SC-186
ATWS for WWER reactors	(WWER)	WWER-SC-186
WWER RPV integrity	(WWER)	IAEA-EBP-WWER-06, IAEA-EBP-WWER-08, IAEA-TECDOC-659, WWER-SC-126, WWER-SC-144, WER-SC-192, WWER-SC-193, WWER-SC-200, WWER-SC-205
Methodology of fire hazard reviews	(WWER)	IAEA-TECDOC-774, WWER-RD-051
Physical separation and functional isolation of safety systems for WWER reactors	(WWER)	WWER-SC-209
Primary to secondary cooling circuit leakages for WWER-1000 and WWER-440 NPPs	(WWER)	WWER-SC-179
Treatment of primary to secondary leaks in WWER nuclear power plants	(WWER)	WWER-SC-206
Application of the leak before break concept	(WWER, RBMK)	IAEA-TECDOC-710, IAEA-TECDOC-774, WWER-RD-046, WWER-RD-055, WWER-RD-056, WWER-SC-113, WWER-SC-125, WWER-SC-158, WWER-SC-173, RBMK-SC-034

	Plant type(s)	Reference/Report No.
Code Validation for RBMK LOCA Analysis	(RBMK)	RBMK-SC-026, RBMK-SC-047, RBMK-SC-056
Multiple pressure tube rupture	(RBMK)	IAEA-EBP-RBMK-02
RBMK fuel channel integrity	(RBMK)	RBMK-SC-038, RBMK-SC-050
RBMK shutdown systems	(RBMK)	IAEA-EBP-RBMK-01, RBMK-SC-032
Void reactivity feedback in RBMK NPPs	(RBMK)	RBMK-SC-025, RBMK-SC-042, RBMK-SC-054
3-D computer codes for core and system analysis	(RBMK)	RBMK-SC-048

REPORT TITLES

IAEA-EBP-RBMK-01	RBMK Shutdown systems (available also in Russian)
IAEA-EBP-RBMK-02	Multiple pressure tube rupture (available also in Russian)
IAEA-EBP-RBMK-03	Safety Assessment of proposed modifications for Ignalina NPP (available also in Russian)
IAEA-EBP-WWER-02	Technical basis for I&C design improvements in WWER 440/230 NPPs
IAEA-EBP-WWER-06	WWER-440/230 Reactor Pressure Vessel Integrity
IAEA-EBP-WWER-07	WWER-1000 Steam Generator Integrity (available also in Russian)
IAEA-EBP-WWER-08	Guidelines on the PTS Analysis for WWER NPPs (available also in Russian)
IAEA-EBP-WWER-10	Review of the Methods Used for Leak Rate Measurements for WWER-440/230 Confinements and WWER-440/213 Containments
IAEA-TECDOC-659	Reactor pressure vessel embrittlement
IAEA-TECDOC-710	Applicability of the leak before break concept
IAEA-TECDOC-722	Safety assessment of design solutions and proposed improvements to Smolensk Unit 3 RBMK nuclear power plant
IAEA-TECDOC-774	Guidance for the application of the leak before break concept
RBMK-SC-025	Working Material, The analysis of the experimental procedure used at RBMK NPPs to measure the void reactivity coefficient, prepared by Kurchatov Institute for the IAEA
RBMK-SC-026	Working Material, Code validation data base for RBMK LOCA analysis, prepared by O-Arai Engineering Center Japan
RBMK-SC-032	Report of a Consultants Meeting on Design Options for RBMK Shutdown Systems
RBMK-SC-034	Report of the Topical Meeting on the Leak Before Break Concept Application to the RBMK NPPs, St. Petersburg, Russia
RBMK-SC-038	Draft Report of a Workshop on RBMK Fuel Channel Integrity, Kaunas, Lithuania
RBMK-SC-042	Report of a Consultants Meeting on Void Reactivity Feedback in RBMK NPPs

RBMK-SC-047	Final Report of a Consultants Meeting on Code Validation for RBMK LOCA Analysis, Munich, Germany
RBMK-SC-048	Report of a Topical Meeting on 3D Computer Codes for RBMK Core and System Analysis, Munich, Germany
RBMK-SC-049	Report on the Technical Visit to the Leningrad NPP Unit 2, Sosnovy Bor, Russian Federation
RBMK-SC-050	Fuel Channel Integrity
RBMK-SC-054	Working Material - Low Power Void Reactivity Feedback in RBMK Nuclear Power Plants
RBMK-SC-056	Report of the Consultants Meeting on Code Validation for RBMK LOCA Analysis
WWER-RD-004	Team’s Analyses of Soviet Design WWERs DOE
WWER-RD-022	Final Report of a Safety Review Mission organized by the IAEA to Bohunice Units 1&2 Nuclear Power Plant, CSFR
WWER-RD-033	Final Report of a Safety Review Mission organized by the IAEA to Kozloduy Nuclear Power Plant Units I-IV, Kozloduy, Bulgaria
WWER-RD-034	Final Report of the Safety Review Mission organized by the IAEA to the Novovoronezh, USSR
WWER-RD-035	Final report of a safety review mission organized by the IAEA to Poliarne Zori, USSR, Kola Units 1 and 2
WWER-RD-046	Review of the leak before break concept application to the Temelin WWER-1000 nuclear power plant
WWER-RD-049	Kozloduy Safety Review Mission Follow up. Final Report of a Safety Review Mission Follow-up organized by the IAEA to Kozloduy NPP Units 1-4, Bulgaria
WWER-RD-050	Novovoronezh Safety Review Mission Follow-up. Final Report of a Safety Review Mission Follow-up organized by the IAEA to Novovoronezh Nuclear Power Plant Units 3-4, Russian Federation
WWER-RD-051	Report of the Fire Protection Workshop at Zaporozhe NPP
WWER-RD-055	Final Report of 2nd Review of LBB Concept Application of the Temelin WWR-1000 Nuclear Power Plant
WWER-RD-056	Report of the Consultants Meeting on the Leak Before Break Concept Application to the Bohunice WWER-440/230 Nuclear Power Plant

WWER-RD-057	Report of a Consultants Meeting on Steam Generator Collector Integrity of WWER-1000 Reactors
WWER-RD-064	Final Report on a Safety Review Mission organized by the IAEA to Zaporozhe, Ukraine
WWER-RD-068	Kola Consultative Mission . Final report of a safety review mission follow-up organized by the IAEA to Kola NPP Units 1 and 2.
WWER-RD-071	CM on core control and protection Strategy of WWER-1000
WWER-RD-074	Mochovce Safety Improvement Review: Report of consultants meeting in Mochovce (Slovak Republic)
WWER-RD-078	Bohunice V2 safety review mission. Final report of a safety review mission organized by the IAEA to Bohunice-V2, Slovak Republic
WWER-SC-038	Final Report of the Safety Review Mission Follow-up organized by the IAEA to Bohunice Nuclear Power Plant
WWER-SC-040	IAEA Extrabudgetary Programme on the Safety of WWER-440/230 Nuclear Power Plants. Status Report on the Safety Issue: Instrumentation and Control.
WWER-SC-085	Report of a consultants meeting on containment and confinement performance in NPPs with WWER 440/213 and 440/230 reactors
WWER-SC-095	Report of a consultants meeting on Evaluation Guidelines for bubbler condenser metallic structure in WWER/440/213 NPPs containments
WWER-SC-099	Technical basis for I&C Design Improvements in WWER 440/230 NPPs
WWER-SC-101	Report of a consultants on confinement improvement options for NPPs with WWER 440/230 reactors
WWER-SC-106	Draft report of a consultants meeting on safety issues and safety improvement measures connected with instrumentation and control in NPPs with WWER-440/213 reactors
WWER-SC-107	Report of a consultants meeting on safety improvements of WWER-440/230 NPPs
WWER-SC-113	Report of a consultants meeting on 3rd review of the LBB concept application to the Temelin WWER 1000 NPP

WWER-SC-115	Report on 3rd seminar on horizontal steam generator concentration, Lappenranta, Finland
WWER-SC-121	Report on the consultants meeting on control rod insertion reliability for WWER 1000 nuclear power plants
WWER-SC-125	Report of the experts' mission to review the LBB concept application to Kozloduy NPP 1-4, Sofia, Bulgaria
WWER-SC-126	Report of the experts' mission to review the thermal hydraulic stress and fracture mechanics analysis for PTS assessment of Kozloduy Nuclear Power Plant Units 1-4 reactor pressure vessels, Sofia
WWER-SC-135	Report of a Consultants Meeting on a review of the methods used for leak rate measurements for WWER 440/230 confinements and WWER 440/213 containments
WWER-SC-139	Report of a Consultants Meeting to Initiate the Preparatory Work on Guidelines for WWER 440/213 Containment Evaluation, Vienna, Austria, 13-17 March 1995
WWER-SC-142	Report of a Consultants Meeting on the Review of Bubbler Condenser Structural Integrity Calculations
WWER-SC-143	Report of an Experts Mission to Review the Modernization Programme of the Kozloduy NPP Unit 5 & 6, Kozloduy, Bulgaria
WWER-SC-144	Report of an Experts Mission to Review the Irradiation Embrittlement Surveillance Programme for Kozloduy NPP Unit 5 and 6 Reactor Pressure Vessels, Sofia, Bulgaria
WWER-SC-149	Report of a Consultants Meeting on a Review of the Methods Used for Leakrate Measurements for WWER 440/230 Confinements and WWER 440/213 Containments
WWER-SC-151	Report of an Experts Mission to Review the Modernization Programme of Rovno Nuclear Power Plant Unit 4
WWER-SC-158	Report of a Consultants Meeting on Final Review of the Leak Before Break Concept Application to the Temelin WWER-1000 Nuclear Power Plant
WWER-SC-159	Report of an Experts Mission to Review the Bubbler Condenser Structural Integrity in Mochovce and Bohunice V-2 Nuclear Power Plants
WWER-SC-160	Report of an Experts Mission to Review the Safety Improvement Programme in Dukovany Nuclear Power Plant, Czech Republic

WWER-SC-161	Report of a Technical Visit organized by the IAEA to Novovoronezh Nuclear Power Plant, Units 3 and 4, Russian Federation
WWER-SC-164	Final Report of a Technical Visit Organized by the IAEA to Kozloduy Nuclear Power Plant, Units 1-4, Bulgaria
WWER-SC-166	Travel Report from IAEA Consultants Meeting on Review of Bubbler Condenser Structural Integrity in Mochovce and Bohunice NPPs, Mochovce, Slovak Republic
WWER-SC-170	Report of a Consultants Meeting on Guidelines for WWER 440/213 Containment Evaluation
WWER-SC-171	Report of the Review of WWER-1000 Safety Issues - Resolution at Temelin Nuclear Power Plant
WWER-SC-173	Report of an Experts Mission to Review the Leak Before Break Concept Application to Kozloduy Nuclear Power Plant Units1-4
WWER-SC-176	WWER-1000 Steam Generator Integrity, Japan
WWER-SC-178	Report of an Experts Mission to Review Modernization Programme of Khmelnitsky Nuclear Power Plant Unit 2
WWER-SC-179	Report of a CSM on Primary to Secondary Cooling Circuit Leakages for WWER-1000 and WWER-440 NPPs
WWER-SC-180	Final Report of a Technical Visit Organized by the IAEA to Bohunice Nuclear Power Plant, Units 1 & 2, Slovak Republic
WWER-SC-182	Report of the Review of the Safety Improvement Programme for South Ukraine Nuclear Power Plant Units 1&2 and to Identify the Safety Issues of "Small Series" WWER-1000 Nuclear Power Plants, South Ukraine, Yuzhnoukrainsk
WWER-SC-183	Report of an Experts Mission to Review the Bubbler Condenser Structural Integrity in Dukovany Nuclear Power Plant
WWER-SC-186	Report of a Consultants Meeting on Anticipated Transients without Scram for WWER-1000 Reactors
WWER-SC-192	Co-ordinated Research Programme: Round-Robin Exercise on WWER-440 RPV Weld Metal Irradiation Embrittlement, Annealing and Re-embrittlement; Terms of Reference
WWER-SC-193	Workshop on International Practices for Reactor Pressure Vessel Integrity Assessment, Rez, Czech Republic
WWER-SC-197	Review the Safety Upgrading Programme of Paks NPP Units 1-4, Paks, Hungary

WWER-SC-199	Report of a Technical Visit to Novovoronezh Nuclear Power Plant Unit 5
WWER-SC-200	Report of a Workshop on Kozloduy Unit 1 Reactor Pressure Vessel Integrity
WWER-SC-205	Pressurized Thermal Shock Analysis Benchmark Exercise: Terms of Reference
WWER-SC-206	Treatment of Primary to Secondary Leaks in WWER NPPs
WWER-SC-209	Report of a Technical Visit to Kola NPP Units 1 and 2