WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION

Nuclear safety in

EU candidate countries

October 2000

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FOREWORD

The Western European Nuclear Regulators' Association (WENRA) is the association of the Heads of nuclear regulatory authorities of Western European countries with nuclear power plants, namely Belgium, Finland, France, Germany, Italy, the Netherlands, Spain, Sweden, Switzerland [(*)] and the United Kingdom. The association has the following objectives:

- To develop a common approach to nuclear safety and regulation, in particular within the European Union,
- To provide the European Union with an independent capability to examine nuclear safety and regulation in candidate countries,
- To evaluate and achieve a common approach to nuclear safety and regulatory issues which arise.

Nuclear safety in the candidate countries to the European Union is a major issue that needs to be addressed in the framework of the enlargement process. Therefore WENRA members considered it was their duty to offer their technical assistance to their Governments and the European Union Institutions. They decided to express their collective opinion on nuclear safety in those candidate countries having at least one nuclear power plant: Bulgaria, the Czech Republic, Hungary, Lithuania, Romania, Slovakia and Slovenia.

The report is structured as follows:

- A foreword including background information, structure of the report and the methodology used,
- General conclusions of WENRA members reflecting their collective opinion,
- For each candidate country, an executive summary, a chapter on the status of the regulatory regime and regulatory body, and a chapter on the nuclear power plant safety status.

Two annexes are added to address the generic safety characteristics and safety issues for RBMK and VVER plants. The report does not cover radiation protection and decommissioning issues, while safety aspects of spent fuel and radioactive waste management are only covered as regards on-site provisions.

In order to produce this report, WENRA used different means:

- For the chapters on the regulatory regimes and regulatory bodies, experts from WENRA did the work,
- For the chapters on nuclear power plant safety status, experts from WENRA and from French and German technical support organisations did the work,
- Taking into account the contents of these chapters, WENRA has formulated its general conclusions in this report.

WENRA's methodology for reaching the collective opinion expressed in the general conclusions

[(*)] The Swiss member of WENRA did not take part in establishing the present report has been to compare the current situation in the candidate countries to that in Western European countries using a common format which is reflected in the structure of the chapters. All major safety issues identified in past international co-operation have been considered. For each candidate country, a comparison was made with the current Western European practices and, whenever appropriate, discrepancies or deficiencies were clearly identified.

WENRA has not made a detailed safety assessment of the different nuclear power plants. Nuclear safety is a national responsibility and it belongs to the regulatory body of the various candidate countries to regulate the safety of all nuclear installations on their national territory, in line with the national legislative and regulatory framework.

WENRA's collective opinion on the regulatory systems is based on generic preconditions for an independent and strong regulatory regime such as a comprehensive nuclear legislation, the existence of an adequate licensing system, appropriate resources and technical support. WENRA's collective opinion on nuclear power plant safety is based on widely applied standards in Western European countries for the defence-in-depth and associated barriers. Quantitative comparisons of probabilistic safety assessments have not been used as the available results are of different depth and quality.

A first version of this report was issued in March 1999. It was solely based on the direct evidence WENRA had gathered through the different activities of its members (participation in multilateral assistance programmes, and in particular the Phare programmes and the IAEA extra-budgetary programme, and in bilateral contacts). In particular, information necessary to formulate an opinion on the regulatory regimes and the regulatory bodies were in many cases derived from the regulatory assistance projects of the RAMG implemented under the Phare programme. With regards to the safety status of nuclear power plants, WENRA had to recognise that in some cases the direct information was not sufficient to formulate an opinion.

For the present version, WENRA took the appropriate steps to collect the necessary information. In addition to the direct evidence already available, supplementary information was gathered through meetings with the candidate countries' regulatory bodies and plant operators. In particular, an ad-hoc Task Force was established to gather and evaluate additional information on VVER-440/230 reactors.

GENERAL CONCLUSIONS OF WENRA

ON NUCLEAR SAFETY IN CANDIDATE COUNTRIES TO THE EUROPEAN UNION

We, Heads of the Nuclear Regulatory Authorities assembled in WENRA, considering the status achieved on nuclear safety in the candidate countries to the European Union and taking into account the results of the investigations of experts from WENRA and from French and German technical support organisations, come to the following conclusions:

BULGARIA

Status of the regulatory regime and regulatory body

At present, the regulatory regime is not in line with Western European practice because it does not provide sufficient independence to the regulatory body. The resources of the regulatory body are also insufficient to allow it to carry out its responsibilities.

Nuclear power plant safety status

<u>Kozloduy units 1-4 (VVER-440/230)</u>
Although improvements have been made, the Kozloduy 1-4 units have not reached an acceptable level of safety. Among others, a concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. Even if a solution could be found to this issue, significant time and effort would be required to achieve the necessary improvements to bring them up to equivalent Western European reactor standards. The Bulgarian Government has announced its decision to close down Kozloduy units 1-2 before 2003.

<u>Kozloduy units 5-6 (VVER-1000/320)</u>
If their modernisation programmes are carried out properly, the Kozloduy 5-6 units should reach a level of safety comparable to that of Western European reactors of the same vintage.

CZECH REPUBLIC

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in the Czech Republic are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

<u>Dukovany units 1-4 (VVER-440/213)</u>
Already in the early years of operation, improvements were implemented to remove safety deficiencies of the original design. An extensive modernisation programme has been established and it will allow Dukovany units 1-4 to reach a safety level comparable to that of Western European reactors of the same vintage. All issues, except the modernisation of the Instrumentation and Control systems, will be completed by 2004.

Temelin units 1-2 (VVER-1000/320)
The safety improvement programme for Temelin units 1-2 is the most comprehensive one ever applied to a VVER-1000 reactor. Standard Western practices were used to integrate Eastern and Western technologies and to deliver the corresponding authorisations. The on-going commissioning process has to confirm the integration of the different technologies. A few safety issues still need to be resolved. If these are resolved, Temelin units 1-2 should reach a safety level comparable to that of currently operating Western European reactors.

HUNGARY

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Hungary are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

Paks units 1-4 (VVER-440/213)
A major safety improvement programme has been implemented at Paks units 1-4, bringing these units to a safety level that is comparable to that of Western European reactors of the same vintage. An extensive modernisation of the Instrumentation and Control system is underway for further enhancement of safety.

LITHUANIA

Status of the regulatory regime and regulatory body

The legal and regulatory system has substantially developed over the past years. A licensing system is in place. However, further efforts are needed to reach a level comparable to Western European practice. In particular, the legal status of the plant need to be changed in such a way that operating organisation is given full responsibility and authority for the safety of the plant. The resources and technical support of the regulatory body need to be strengthened and its independence need to be maintained in the ongoing reorganisation of governmental institutions.

Nuclear power plant safety status

Ignalina units 1-2 (RBMK 1500)
The Ignalina units 1-2, although they have been much improved, cannot realistically reach a safety level comparable to that of Western European reactors of the same vintage. A decision has already been taken to shutdown unit 1 before 2005. The current financial situation of the plant needs to be improved in order not to delay the ongoing safety improvement programme.

ROMANIA

Status of the regulatory regime and regulatory body

Romania is taking the appropriate steps to establish a regulatory regime and regulatory body comparable with Western European practice. Further efforts are needed to ensure the necessary safety assessment capabilities, to develop the emergency response organisation within the regulatory body and to revise the pyramid of regulatory documents.

Nuclear power plant safety status

Cernavoda unit 1 (Candu 6)
The Candu 6 reactor of Cernavoda is similar to those in operation at Gentilly 2 and Point-Lepreau in Canada. The main concern is with the financial situation of the plant: under the current situation, the plant management may have serious difficulties in ensuring and maintaining an adequate level of safety.

SLOVAKIA

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Slovakia are comparable with Western European practice. However, the human and financial resources of the regulatory body need to be further improved in order to provide reasonable work conditions for the staff.

Nuclear power plant safety status

Bohunice V1 (VVER-440/230)
A major upgrade programme is nearing completion, which has made significant improvements to reactor safety. A concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. If a solution can be found to this issue, the plant should reach a safety level comparable to that of Western European reactors of the same vintage. The Slovak Government has announced its decision to close down these units in 2006 and 2008.

Bohunice V2 (VVER-440/213)
Since 1990, significant improvements have been implemented at Bohunice V2 (units 3-4). Once the on-going upgrading measures have been implemented, i.e. around 2002, the safety level of these units is expected to be comparable to that of Western European reactors of the same vintage.

Mochovce units 1-2 (VVER-440/213)
Compared to earlier reactors of the same type (VVER 440-213), the Mochovce units 1-2 included several modifications already at the design stage. Although some residual work is still needed to confirm all parts of the safety analysis, the safety level of the Mochovce units 1-2 is comparable to that of nuclear power plants being operated in Western Europe.

SLOVENIA

Status of the regulatory regime and regulatory body

In order to be fully comparable with Western practice, the nuclear legislation needs to be revised, addressing the identified deficiencies. The regulatory body has evolved and operates in general accordance with Western practice and methodologies, however the budget and financial situation need to be improved in order to increase its independent safety assessment capability.

Nuclear power plant safety status

Krško (Western PWR)
The Krško plant is a Western design pressurised water reactor and its safety level is comparable with that of nuclear power plants in operation in Western European countries. A large modernisation programme has been recently completed. The safety implications of the long-term

plant ownership need to be assessed. In addition, the evaluation of a few technical issues needs to be finalised.

J.P. SAMAIN Director General Federal Agency for Nuclear Control (FANC/AFCN) Belgium	J. LAAKSONEN Director General Radiation and Nuclear Safety Authority (STUK) Finland
A.C. LACOSTE Director Nuclear Installation Safety Directorate (DSIN) France	W. RENNEBERG Director General for Nuclear Safety Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU) Germany
R. MEZZANOTTE Director, Department of Nuclear Safety and Radiation Protection National Agency for Environment Protection (ANPA) Italy	R.J. VAN SANTEN Director Nuclear Safety Department (KFD) Ministry of Housing, Spatial Planning and Environment The Netherlands
J.M. KINDELAN Chairman Nuclear Safety Council (CSN) Spain	J. MELIN Director General Swedish Nuclear Power Inspectorate (SKI) Sweden
L. WILLIAMS HM Chief Inspector Nuclear Installations (HSE) United Kingdom	

REPORT

Executive summaries

BULGARIA

Status of the regulatory regime and regulatory body

Since the early 1990s, there have been significant improvements in the legislative basis and in the capabilities of the nuclear regulatory body (the Committee on the Use of Atomic Energy for Peaceful Purposes - CUAEPP).

However, much remains to be done to bring the regulatory regime up to Western European standards. The Bulgarian Governments needs to enact legislation that will make explicit the independence of the CUAEPP from bodies concerned with the promotion of nuclear power. The government needs to provide adequate funding to the CUAEPP to enable the recruitment and retention of adequate numbers of qualified staff. Funding is also needed to enable the development new technical support facilities for the CUAEPP. Resources need to be committed to the drafting and introduction of necessary new and revised legislation.

Nuclear power plant safety status

There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased. However, the lack of Safety Analysis Report of any Bulgarian nuclear power plant is a serious shortcoming for judging the safety. Therefore to confirm the improvements implemented a consistent safety case has to be established and it has to be reviewed by the CUAEPP.

<u>Kozloduy 1-4</u>
Despite the significant safety improvements already achieved considering the present safety status of the plant, there are still some major safety issues which are closely linked to the original basic design of the VVER-440/230 reactors and which are difficult to be removed, such as the limited confinement function and capability and the vulnerability against common cause failures. The work at Kozloduy is at least three years behind that at Bohunice and consequently safety improvement is not as far advanced. At present in view of the large amount of work required to be carried out it is difficult to have a final judgement on the adequacy and feasibility of all measures foreseen. It seems, however that financial provisions for continued safety improvements are inadequate. For Kozloduy 1 and 2 the implementation of relevant measures cannot be expected taking into account the announced closure dates.

<u>Kozloduy 5-6</u>
An extensive programme for further upgrading of these units with assured financing has been reviewed by Western TSOs and is at an early stage of implementation. Safety assessments done by Western TSOs for similar plants indicate that with the completion of the planned safety upgrades, it could be possible to achieve a level of safety for units 5 and 6 which is in line with international recognised safety practices.

CZECH REPUBLIC

Status of the regulatory regime and regulatory body

The nuclear legislative framework in the Czech Republic is comparable with Western European practice. It is considered that the SÚJB has a status comparable to that of Western European regulatory bodies. The SÚJB has developed a series of regulatory practices, including a well-defined licensing process, which compare favourably with those of Western European nuclear regulators.

Further improvements could arise from the following suggestions. It is recommended that the Government of the Czech Republic consider giving high priority to the implementation of the new Act on emergency preparedness and planning. The SÚJB should be asked to make proposals in view of removing too detailed requirements from the high level documents of the regulatory pyramid. Also, the contracting rules of the SÚJB need to be adapted so that it can obtain, when appropriate, the necessary high quality technical support on a long term basis.

Nuclear power plant safety status

Dukovany NPP

In the early years of operation, modifications were carried out to remove safety deficiencies in the original design. An extensive modernisation programme, called MORAVA, will be implemented by 2004 with the exception of I&C replacement.

The safety culture appears to be adequate. Safety assessments and verification documents, e.g. periodic safety reviews, are conducted in a way which is comparable to Western practice.

After full implementation of the modernisation programme it is expected that Dukovany NPP will achieve a safety level comparable to that of NPPs of the same vintage operating in Western Europe.

Temelin NPP

The safety improvement programme for Temelin NPP is the most comprehensive which has been applied to a VVER-1000/320 plant.

International co-operation has had a considerable influence on the plant’s safety improvements (design, operation, safety approvals), and on the development of safety culture.

The combination of Eastern and Western technologies was successfully managed. Interfaces between the different technologies were considered throughout the modernisation programme and a standard Western practice was used to combine Eastern and Western technologies. The commissioning process will need to confirm the integration of the different technologies.

Some safety issues still need further clarification but if these issues are resolved, Temelin NPP will achieve a safety level that is comparable to that of operating Western PWRs.

HUNGARY

Status of the regulatory regime and regulatory body

The Hungarian approach to the licensing, regulation and control of nuclear facilities has developed strongly in the last ten years. A proper licensing process is in place, legislation and regulations are up-to-date, and the Hungarian regulatory practices are comparable with those of Western European countries.

However, there are some issues that need further consideration by the Hungarian Government. These are:

- The fact that the Minister of Energy Affairs is also the HAEC President creates an apparent conflict of interest, even though the formal mandate of HAEC President precludes this,
- The number of different authorities with direct responsibilities in the regulation of nuclear facilities increases the risk that important issues may be overlooked and reduces the efficiency of the regulatory work.

The NSD needs to continue its efforts to develop the inspection approach towards process oriented comprehensive team inspections.

Nuclear power plant safety status

The basic technical structure of Paks NPP is good from the safety point of view and the key safety systems are comparable to Western plants of the same vintage. No major shortcomings in the present safety systems have been identified in any of the several, independent, in-depth assessments done so far. Also the performance of the bubbler condenser containment in the case of large break LOCA has been verified in full-scope tests. There is still need for detailed analysis of the experimental results and for complementary tests of other design basis accidents (steam line break and small LOCAs). Paks containment structures provide adequate protection against design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU. However their leak rates are somewhat higher than those that are typical of Western European reactor containments.

Operational safety aspects are generally comparable to Western plants of the same vintage. However, management changes related to the political changes in the Government cause some concern. Periodic safety reviews are conducted in line with Western practices and have already led to an increase in safety.

It is expected that after the implementation of safety improvements already scheduled, the plant will reach a level of safety that compares favourably with plants of the same vintage in Western Europe.

LITHUANIA

Status of the regulatory regime and regulatory body

The legal and regulatory system has developed substantially over the last years. A licensing system is in place and the regulatory body VATESI has developed its approaches to safety assessment and inspection. Further efforts are needed, however, in order to be comparable with Western European practice.

The Lithuanian government needs to consider the legal status of Ignalina NPP, in order to give the operating organisation the full responsibility and authority to handle all financial and other management issues and thus to make the organisation able to take the full responsibility for safety. The legal obligation of VATESI to formally license suppliers needs to be changed, given a reasonable transition period. The imposed reduction of resources to VATESI, in terms of budget and staff, needs to be compensated as soon as possible and the resources successively strengthened in order for VATESI to handle all normal regulatory tasks and to contract the necessary technical support. In the reorganisation under way, of governmental institutions reporting directly to the Prime Minister, special attention needs to be given the independence of VATESI.

VATESI needs to give high priority to the development of the internal Quality Management system and take the final steps in separating the roles of the regulatory body and the operator in all supervisory activities.

Nuclear power plant safety status

The two units of Ignalina NPP (INPP) belong to the more advanced and improved design generation of RBMK reactors. In addition, the original design has been considerably improved through different safety improvement programmes. Most of the generic safety concerns with RBMK reactors have been satisfactorily addressed. More measures will be implemented, for instance the installation of a new diversified and independent shut down system at unit 2. However, weaknesses remain with respect to the last barrier for protection of the environment, especially in case of a severe accident. The weaknesses have to do with a less robust design of the confinement of the INPP reactors as compared with Western light water reactors. It is not realistic to make the INPP confinement system comparable. Consequently, regarding mitigation of accidents, a safety level comparable to light water reactors of the same vintage in operation in Western Europe will not be reached at Ignalina NPP. Therefore special attention needs to be given the prevention of accidents during the remaining operating time, including the need to ensure a high level of operational safety.

The financial situation of INPP needs to be much improved in order to cover all operational expenses as well as implementing the safety improvement measures considered necessary for the remaining operating time. Issues relating to safety culture need a stronger implementation. The symptom based emergency operating procedures need to be finalised and implemented without further delay. Due to the decision on decommissioning of unit 1, special attention needs to be given to keep a sufficient number of technical specialists, as well as maintaining the motivation of the staff, for the remaining operating time of both reactors.

ROMANIA

Status of the regulatory regime and regulatory body

Romania is taking appropriate steps to establish a regulatory regime and a regulatory body comparable with Western European practice. Roles, duties and responsibilities of organisations involved in nuclear safety are in line with those assigned to similar organisations in Western Europe. The independence of the regulatory body from the organisations involved in the use and promotion of nuclear energy is fully established by the law and is sufficiently reflected in the practice. The regulatory regime and the regulatory body have both improved during the licensing process of Cernavoda NPP.

However some improvements are necessary to reach a situation comparable with the practice in Western European countries:

- The independent assessment capability, the inspection practice and the technical support of CNCAN need to be strengthened. The salaries at CNCAN need to be further improved to preserve suitably qualified staff. Adequate resources need to be assigned to set up and implement a training programme for new staff. Existing agreement with the Canadian nuclear safety authority needs to be more effectively used for training purposes and for seeking advice on regulatory issues specific of the CANDU technology. A strategic plan could support the assignment of existing limited resources to higher priority needs,
- National organisations that would have a role in a nuclear emergency need to make their emergency procedures and lines of communication more effective. In addition, CNCAN needs to further develop its competence and staff numbers in this area and establish an emergency response centre,
- The responsibility for auditing and approving vendors and suppliers should rest with the operating organisation and not with the regulatory body.

Nuclear power plant safety status

Romania has only one NPP into operation. It is a CANDU 6 reactor similar to those in operation at Gentilly 2 and Point Lepreau in Canada. The plant was constructed and commissioned under the responsibility of a Western Consortium (AECL, Ansaldo). The Cernavoda plant managers and operators have a professional attitude and have assimilated a western safety approach and culture.

It is important that the Romanian Government ensures that the current financial problems of the utility do not affect the ability of the management to maintain an adequate level of safety at the plant. Western support, especially from Canadian experts, should be made available when it is needed in the future.

Based on available information it is apparent that additional assessments are needed to confirm design safety margins against seismic events and the adequacy of fire protection. Also, the resolution of specific safety issues for similar plants that have been addressed or are currently under discussion in Canada need to be noted and incorporated where necessary into an improvement programme. The current high level of qualification and safety culture of the plant managers needs to be preserved in the longer tem. The plant management safety culture should be extended to all plant personnel and to the necessary service and support interfaces existing in the country. There is finally a need for improvement in some areas of plant operation such as training, emergency preparedness and accident management.

SLOVAKIA

Status of the regulatory regime and regulatory body

The nuclear legislative framework in Slovakia is in line with Western European practice. The ÚJD has made significant progress over the recent years and has taken the appropriate steps to develop a series of regulatory practices comparable with those of Western European nuclear regulators. It is considered that, in general, the ÚJD status is comparable to that of regulatory bodies in Western European countries. On-going developments will improve its effectiveness.

It is recommended that the government of Slovakia consider the following suggestions. The ÚJD financial resources need to be further increased, in particular but not only, to maintain the independent assessment capability which was initiated under Swiss assistance. In order to retain highly qualified staff, the salaries at the ÚJD need to be made comparable with those of the operator's staff. It is suggested that the government give a high priority to the adoption of the national emergency plan. Also, the Atomic Act should be amended to remove some duties of the ÚJD that are not directly dealing with nuclear safety.

Finally, it is recommended that the ÚJD pay particular attention to ensure a clear separation between the technical support it receives and that provided to an operator.

Nuclear power plant safety status

The safety of Slovakian nuclear power plants has been improved since the early 1990´s in a determined manner with a strong national commitment, and significant investments have been made in technical upgrades. Guidance received from the IAEA has been used efficiently. Operational practices at all Slovakian nuclear power plants are consistent with those in Western Europe.

The following conclusions can be made:

<u>Bohunice V1 (units 1-2)</u>
The revised design requirements provide a coherent target for safety improvement of the plant. The utility has made significant progress towards establishing a new design base and implementing the relevant measures. Some work remains to be done but no technical obstacles in completing it are foreseen. It will be completed in 2000.

If a solution can be found to the concern related to the confinement ability to cope with the double ended guillotine break LOCA, the safety level of these units is expected to be comparable with that of units of the same vintage in Western European Countries.

<u>Bohunice V2 (units 3-4)</u>
Since 1990, significant improvements have been implemented at Bohunice V2. However, in order to achieve adequate reliability of safety systems in all operating situations, an extensive modernisation programme is planned for implementation between 1999-2006, with the major upgrades relating to safety being completed by 2002.

The safety of Bohunice V2 units seems generally adequate. Once the ongoing safety upgrades have been implemented (by about year 2002), the safety level of these units is expected to be comparable with that of units of the same vintage in Western European countries.

Mochovce (units 1-2)

Compared to their VVER-440/213 predecessors, units 1 and 2 of Mochovce included several modifications during the design phase. The most important of these are the use of higher quality equipment and the improvement of systems used in accident situations. However, some design weaknesses remained, and a dedicated nuclear safety improvement programme was developed for the Mochovce NPP in 1995. This programme, which is almost complete, was reviewed by Western European Technical Safety Organisations.

Although some residual work (e.g. bubbler condenser qualification, Mochovce site seismicity characterisation) is still needed to confirm all parts of safety analysis, the safety level of Mochovce units is comparable to that of the nuclear power plants being operated in Western Europe.

SLOVENIA

Status of the regulatory regime and regulatory body

The Slovenian Nuclear Safety Administration (SNSA) operates, in general, according to Western practice and methodologies. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, there are some issues that need to be addressed.

It is recommended that the Government of the Republic of Slovenia addresses the fact that the existing legislation on nuclear and radiation safety is not fully in line with current Western European practice, and its review needs to be completed. In addition, the lack of a final resolution of issues related to shared ownership of Krško NPP may affect the plant's long term financial situation, and have an impact on safety. Furthermore, the legal and financial situation of SNSA needs to be improved in order to increase its independent safety assessment capability. Finally, the national response to nuclear and radiological emergencies needs to be improved by implementing an integrated national emergency plan, paying special attention to the interface with the Croatian authorities. The SNSA, on its side, needs to develop further its own technical capabilities in order to be able to make better independent decisions, and needs to continue defining its regulatory requirements to allow it to make the licensing decisions.

Nuclear power plant safety status

Slovenia has one nuclear power plant located in Krško. The design of the Krško NPP is similar to other Westinghouse PWRs of the same type operating in the USA, Belgium, Switzerland, Korea and Brazil. The safety of the Krško NPP is comparable to that of nuclear power plants of the same vintage into operation in Western Europe. The NPP has had a continuous backfitting and upgrading programme and a large modernisation programme, including the replacement of steam generators and a full scope simulator. The site organisation and the operational safety practice are similar to those in Western Europe.

For the future the following issues need to be addressed. The implications on safety of the ownership for the long term and the upcoming privatisation process of the energy sector need to be carefully assessed. In addition, efforts to strengthen the engineering capability of the utility need to be continued, including resources to ensure the necessary technical support from foreign organisations. Closer contacts with Western European utilities would also be beneficial. Finally, the evaluation of a few issues, like the seismic characterisation of the site and the onsite storage of spent fuel like need to be finalised and further attention is deemed necessary to the performance of a periodic safety review.

Detailed chapters

BULGARIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. The primary legislation for nuclear safety, the Act on the Use of Atomic Energy for Peaceful Purposes, was enacted in 1985 and amended in 1995 and 1998. Enforcing regulations, which give interpretation and meaning to the early primary legislation came into force in 1985 but, due to shortage of resources, there has been slow progress in making revisions to reflect the 1995 and 1998 amendments. The Act gives responsibility for licensing and regulation to the Committee on the Safe Use of Atomic Energy for Peaceful Purposes (CUAEPP).

2. A Programme on the Development of a Comprehensive Legislative Framework on Safety of Spent Nuclear Fuel and Radioactive Waste Management was adopted by the Council of Ministers in December 1999.

3. A three-year programme for the development and revision of much of the present regulatory documentation was agreed in 1998 by the nuclear regulator and relevant Ministries. This will lead eventually to greater consistency and facilitate the adoption of a less prescriptive regulatory approach. However, the high workload of the regulatory body means that this programme is already behind schedule and is likely to be delayed even further. In 1999 a new Act setting out further ambitious amendments to the earlier Act was developed by the CUAEPP in co-operation with other Ministries. However, this was rejected by the Council of Ministers in February 2000, which will delay improvements in the legislative basis by at least one year. The proposed amendments aim to bring about the harmonisation of the Bulgarian and West European legislation on the safety of nuclear facilities and in the field of accounting and control of nuclear material leading to less prescriptive Bulgarian nuclear legislation. A new Act on the safety in the use of Nuclear Energy is under development by a joint working group under the leadership of the CUAEPP. The new Act will cover in detail the management of radioactive waste, spent nuclear fuel and decommissioning of nuclear facilities. It will also establish a stable legal mechanism for the financing of the regulatory body.

4. Existing legislation adequately defines the legal obligations of the Operator (Kozloduy NPP) giving it responsibility for the safe control of the plant and for civil liabilities under the Vienna Convention. The primary legislation also requires the operating company to make payments into funds for dealing with radioactive waste and for decommissioning. The regulations to implement this came into force only in 1999 and the first contributions to this fund were made in the same year. Kozloduy NPP is wholly owned by the State.

5. All the key international conventions related to nuclear safety have been ratified and incorporated into national legislation.

Status of the regulatory body and technical support infrastructure

6. The current legislation places a dual role on the CUAEPP. First as a <u>State Body</u> with membership from organisations concerned with the promotion of nuclear power and the

operation of the power stations, and secondly as a legal entity charged with regulation of safety. This implies a lack of independence of the CUAEPP as a safety regulator. The Council of Ministers decided in April 1999 that the CUAEPP should be replaced by an Agency of the Government, the State Atomic Energy Agency (SAEA). This would ensure regulatory independence from those organisations promoting nuclear energy and would give SAEA sole responsibility for regulating the nuclear facilities and the storage and transport of nuclear material. However, the legislation to implement this was rejected in February 2000.

7. Funding for the CUAEPP comes from the State Budget and is controlled by the Ministry of Finance. It is currently inadequate. Budget restrictions imposed in 1996 reduced the CUAEPP staff by about 25% to 77. Of these, 50 posts were allocated to the Inspectorate on the Safe Use of Atomic Energy (ISUAE), the enforcement and inspection division of the CUAEPP. CUAEPP salaries are still approximately 20% of those in the nuclear industry, and this makes recruitment and retention of well qualified staff difficult. Currently, there are not enough staff to adequately carry out all necessary safety assessment and site inspection duties and the low salaries make the CUAEPP vulnerable to the loss of more experienced personnel. The training for inspectors and succession planning will need to be improved when resources are available. A Council of Ministers decision in April 1999 approved step by step increase of the CUAEPP personnel to 88 in 1999, 102 (currently 80) in 2000 and a figure of 110 is under negotiation for 2001. Proposed legislation would fix the regulator's salaries at a minimum 80% of the equivalent industry level but this is not yet in place. Consequently the CUAEPP continues to lose staff.

8. In the past, frequent changes in senior management positions put additional strain on the CUAEPP. However, it is now benefiting from greater managerial stability, particularly with respect to the position of the Chairman. This is assisting the CUAEPP to plan and implement the improvement process.

9. Existing legislation gives the CUAEPP some enforcement powers. Penalties for contravention of regulations are defined in the primary legislation. Enforcement relies heavily on fines. The CUAEPP also has the responsibility to authorise suppliers of equipment to the licensees.

10. The CUAEPP currently has limited resources to perform technical evaluations and relies to a considerable extent on external support. In the last few years local technical support organisations have developed in number and expertise. However, there are still only a limited number in the country, which means they are sometimes contracted to work for both regulator and the utility. Following the enactment of the proposed new national legislation, the CUAEPP intends to have a permanent body dedicated to the technical support of the regulatory authority. There is currently a need for continuing external assistance and this need for support is unlikely to change in the near future.

11. The funds available to the CUAEPP for nuclear safety research and support are provided by charges levied on the licensees. Compared with equivalent funds typically available to Western regulators, this is at a fairly low level.

12. The status of the regulatory body is not yet comparable with its Western European counterparts. However, if political commitment would be achieved and resources made available, well-prepared plans exist to transform the CUAEPP.

Status of regulatory activities

13. An internally generated improvement plan issued by the CUAEPP in 1998 set out an ambitious programme for codifying the CUAEPP's nuclear safety and licensing requirements. This will gradually replace the prescriptive legacy of the former Soviet Union and bring the Bulgarian regulator in line with a Western European approach. The aim is to create a strong and independent regulatory body with sufficient funding to carry out the full range of regulatory activities, including the making of regulations, site inspection, assessment and enforcing the utility to implementation of periodic safety reviews and draw up safety analysis reports as a basis for licensing. Some good progress has already been made, but unfortunately, because the CUAEPP has insufficient resources, implementation of the plan is already falling behind schedule. On the positive side, the CUAEPP is making much better use of its site inspectors by adopting a Western approach in which the licensee carries out routine, qualification inspections of pressure parts and lifting equipment, under a general supervision by the CUAEPP. The CUAEPP management has recognised the importance of introducing an internal quality management system and it is making progress in the development of a manual and documents to support the regulatory work.

14. In general, the CUAEPP is staffed by technically competent personnel. However, since 1992, due to the limited resources, the CUAEPP has had to rely on external independent technical assessments in carrying out the licensing of plant modifications and improvements. Much of this has been provided under assistance projects funded by the EC, the IAEA and bilateral programmes. In the absence of significant increases in CUAEPP resources, such assistance will continue to be needed, at least until the completion of the modernisation of units 5-6.

15. Regulatory decisions on the future upgrading of the Kozloduy units 1-4 need to be taken against a clear licensing plan which has yet to be fully established and implemented. This will require the development of the relationship between the utility and a competent and fully recognised regulator. At present the safety justification is expected to be completed in 2001. The CUAEPP needs to give clear guidance on the compliance targets with consistency between initiating event classification, analysis assumptions and acceptance criteria.

Emergency preparedness on governmental side

16. In the past there were too many regulatory documents, produced over a number of years, to provide for an effective, consistent emergency response. A new Regulation for Emergency Planning and Preparedness for Actions in Case of a Radiation Accident was approved by a Decision of the Council of Ministers in March 1999. This regulation defines the responsibilities for planning, advising and decision making in the case of a nuclear emergency. In October 1999 the CUAEPP started to up-date the National Emergency Plan for Action in Case of the Nuclear Accident at Kozloduy NPP. This is scheduled for completion at the end of 2000. The role of the CUAEPP will continue to be to monitor the situation and to give advice to the central committee which takes the decisions. The CUAEPP has received international assistance in the development of its own Emergency Preparedness Manual. The CUAEPP has completed the modernisation of its Emergency Response Centre (ERC) with new offices, a diesel-generator for emergency power supply and new telephone and computer systems. Bulgaria is a participant in the IAEA Regional Assistance Project to promote harmonisation of emergency planning in Central and Eastern Europe.

17. Currently, there are annual communications-only exercises involving all relevant national authorities, and there is a site emergency exercise each year. But the new emergency planning regulation requires a full national nuclear emergency exercise every 5 years. A full national emergency exercise will be organised following the bringing into operation of the upgraded ERC. Bulgaria has participated in the last three INEX-2 international exercises organised by the OECD.

Conclusions

18. There have been significant improvements in legislation, organisation and operation of the CUAEPP. However, many of the weaknesses identified previously still remain. Lack of progress in several areas is no doubt partly due to the severe economic situation facing the country. Low wages, combined with a high workload and poor working conditions, have a negative effect on staff morale and the loss of further valuable staff is likely.

19. In order to reach Western European standards, it is recommended that the Government of Bulgaria considers the following issues, several of which were addressed in the draft Act amending the CUAEPP that was rejected by the Council of Ministers in February 2000:

- There is no substitute for a strong, independent and competent regulator. Over recent years, the technical competence, strength and continuity of the Bulgarian regulator has been strongly supported by Western experts. Major efforts are still needed to ensure that the regulatory authority achieves a status that is comparable with that considered acceptable in Western European countries. The independence of the CUAEPP from bodies concerned with the promotion and supply of nuclear power needs to be made explicit,
- Managerial and organisational stability of the regulatory body should be maintained,
- Budget and salaries for the CUAEPP need to be increased to allow it to recruit and retain sufficient numbers of competent staff even though agreement has been given to increase staffing levels. In addition, the CUAEPP needs funds to obtain independent technical support when required and to support independent nuclear safety research,
- There remains a shortage of competent technical support organisations within the country. The technical capabilities available to the regulator need to be enhanced and should be independent from those used by the licensees.

20. In addition to the progress that has already been made, the CUAEPP needs to:

- Ensure that sufficient resources are committed to continue the drafting and introduction of new and revised legislation identified in the CUAEPP Improvement Plan, and to the programme for codifying CUAEPP's basic nuclear safety and licensing requirements.

Chapter 2: Nuclear power plant safety status

Data

1. On the site at Kozloduy, Bulgaria has in operation six nuclear power plants operated by the state owned company Kozloduy NPP. The Bulgarian Government has announced the closure of Kozloduy units 1 and 2 not later than 2003.

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Kozloduy 1	VVER-440/230	1970	1974	2004
Kozloduy 2	VVER-440/230	1970	1975	2005
Kozloduy 3	VVER-440/230	1973	1980	2010
Kozloduy 4	VVER-440/230	1973	1982	2012
Kozloduy 5	VVER-1000/320	1980	1987	2017
Kozloduy 6	VVER-1000/320	1984	1991	2021

2. On the Belene site, construction of two VVER-1000/320 units was started in the 1980's but the work was frozen in 1990.

(i) Kozloduy units 1-4

3. The main parts of the information summarised in this chapter are based on knowledge and experience acquired by the Technical Safety Organisations (TSOs) during the Kozloduy short term upgrading programmes in the early nineties and during the TSOs assistance to the regulatory body within the framework of EBRD's Nuclear Safety Account (NSA) programme.

   Information on more recent issues was acquired during the WENRA-Task Force Mission in October 1999.

Basic technical characteristics

Design basis aspects

4. The first four units on the Kozloduy site are VVER-440/230 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2. Units 3 and 4 are more advanced type VVER-440/230 reactors having some of the design improvements of the later VVER-440/213. These include three-way redundancy and better segregation of safety systems, an Emergency Control Room and a low pressure core cooling system. During the early 1990's the utility implemented reconstruction programmes on all units based on IAEA recommendations and TSO advice. This involved installation of additional safety systems with the objective of eliminating or diminishing major safety shortcomings. The aim of these programmes was:

- To establish the Reactor Pressure Vessel status,
- To improve the behaviour of the confinement system,
- To improve the plant behaviour in respect of internal and external hazards,
- To improve systems and equipment reliability,
- To improve organisation and operational safety.

Two further major items are being implemented on a longer-term basis up to 2002:

- To demonstrate the ability of the plant to cope with Loss of Coolant Accidents larger than the current (100 mm) Design Basis using conservative analysis; the licensing analyses of the new design basis accident (200 mm) have been performed and are currently under review at the CUAEPP,
- To demonstrate the ability of the plant to cope with the complete rupture of a main primary coolant pipe (as Beyond Design Basis Accident) using best estimate analysis; analyses are expected to be available by the end of the year 2000.

Reactor pressure vessel and primary pressure boundary

5. The current condition and the inspection programme of the reactor pressure vessels (RPV) appear adequate. The RPVs of units 1 and 3 were annealed in 1989 and the RPV of unit 2 in 1992. Measurements of impurity concentrations in the weld near the core, and recent experimental results on irradiated samples taken from units 1 and 2 RPV, indicate that under the current design basis with postulated 100 mm break LOCA further annealing of RPV 1 and 2 would not be needed. However, investigations of additional samples seem to be necessary to confirm the re-embrittlement behaviour. Whilst internal cladding prevents direct sampling of RPV 3, it can be established from the original test coupons that chemical composition of the key weld of RPV 3 is bounded by those for unit 2. For unit 4, lower impurity contents in the affected weld mean that RPV embrittlement will not be a problem during its operational life. As part of the revision of the design basis to a 200-mm break, an extended pressurised thermal shock analysis of the RPV is necessary.

6. The utility operating Kozloduy 1-4 has implemented measures to reduce the probability of a large primary circuit break. The present design basis covers pipe ruptures up to 100 mm including primary to secondary leakages in the steam generators (SG). Pipework above 100 mm, i.e. 200-mm pressurizer surge lines and 500-mm main coolant circuit pipework, together with major primary circuit components such as main coolant pumps and valve bodies are covered by a state-of-the-art leak-before-break case (LBB). The LBB case has been performed by a Western industrial company under a Phare contract and has been accepted by the CUAEPP. The proposed extension of the DBA to cover ruptures up to 200 mm, if implemented, will provide some overlap between the prevention and mitigation measures. The calculations required to demonstrate fulfilment of LBB criteria for the 500-mm and the 200-mm primary circuit pipework have been carried out. The LBB case is underwritten by an in-service inspection programme and by two suitable instrumentation systems to detect incipient leaks. The LBB criteria commonly used in the Western countries require that three independent reliable and fast leak detection systems be used. A third independent system is currently being considered to replace another less sensitive one. The risk of a large primary to secondary leak caused by a steam generator collector head lift has been reduced by the use of flow limiters and by specific maintenance including in-service inspection. With the above-mentioned installation of the third leak detection system it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

Confinement

7. Despite recent efforts that have led to significant reductions (by a factor of 10), the confinement system leak rate is still excessive, and effort is required to further reduce it. A necessary confinement improvement, the jet vortex condenser discharging through a water pool, is planned to ensure the confinement's structural integrity in case of large break LOCA accidents up to 500-mm breaks. Implementation on units 3 and 4 is planned for

installation before 2002. This design solution is completely different from that already installed at Bohunice. However, the jet vortex condenser still requires confirmation of the claimed performance and a proof of the absence of unwanted side effects under the whole spectrum of conditions. Therefore, for Kozloduy 1 and 2, implementation cannot be expected taking into account the declared shutdown dates of these units.

Safety systems and hazards

8. By 1997, with assistance from the EU (Phare) and Nuclear Safety Account, substantial short term safety improvements have been implemented on all four units, e.g. improvements to reactivity control and additional reactor protection signals, measures to ensure the integrity of pressurised components, measures for improving protection against hazards in general (e.g. fire protection), and improvements to emergency power supply. In order to provide reliable cooling of the reactor circuit a new emergency steam generator feed water system (2x200%) has been implemented for units 3 and 4. Extension of this system to units 1 and 2 was provided in the year 2000. This system allows the cooling down of the corresponding unit and maintaining it in the cold shutdown state. A primary bleed and feed capability is available for all units. The steam lines are fixed and protected against multiple breaks in the non-isolatable part as well as inside the turbine hall downstream the isolating valves. Upgrading for protection against earthquakes is going on to achieve new seismic requirements of 0.2 g.

9. Units 3 and 4 are already equipped with a low-pressure core cooling system that facilitates Design Basis Accident extension. Compared to the VVER-440/213 type design, however, accumulators are absent and ECCS pumps and confinement spray pumps are located in the common boron compartment room.

I&C systems and emergency power supply

10. Replacement of safety related I&C (Reactor Protection System) will be necessary if it cannot be demonstrated that the reliability of the old relay based system complies with current international standards. It has to be noted that this replacement will be impractical for units 1 and 2 due to their limited residual lifetimes. The emergency power supply system fulfils international requirements, e.g. IAEA Safety Guides. For each unit the system is redundant and single-failure proof, and the equipment is qualified for accidental conditions.

Beyond design basis accidents and severe accidents

11. A series of safety improvements have been introduced in recent years in order to cope with some BDBA conditions such as the installation of a new emergency feed water system, emergency feed water supply by mobile pumps, implementation of equipment and procedures for primary bleed and feed. Consideration should also be given to a mitigative severe accident management strategy when the prevention-related work is reasonably complete.

Safety assessments and programmes for further improvements

12. In the early 1990's a consortium of Western TSOs assessed the safety status of units 1-2 and units 3-4 separately, and reviewed the corresponding modernisation programmes designed for safety during short-term operation. The TSO consortium gave recommendations for short-term safety upgrading measures under the condition of limited operational time, which were additional to those already identified by the utility.

    In 1997 the utility proposed a more extensive safety-upgrading programme for units 1-4.

Several modifications have already been introduced, with the aim of operating these units up to the end of their design life. This programme has undergone several updates, but has not been reviewed systematically by the regulatory authority. For internal review by the utility, a plant modification procedure exists in the frame of the NPP QA programme.

Safety assessment and documentation

13. The lack of Safety Analysis Reports (SAR) to Western standards for units 1-4 is a significant shortcoming, even though many different analyses were performed in the past. In the early nineties, international ad-hoc teams or foreign expert organisations rather than Bulgarian experts carried out a major part of the safety assessment used as a basis for safety upgrades.

14. A limited number of (mainly) generic safety analyses are available for units 3-4. In support of the CUAEPP, Western TSOs in collaboration with Bulgarian institutions have recently developed the requirements for a detailed Safety Report for units 3-4. This so-called Safety Substantiation Report (SSR) has to be provided to the regulator on completion of the extended modernisation programme (expected in 2002). The NPP has recently submitted a first revision of the Safety Substantiation Report for units 1-4 to the regulatory body. At present this report is undergoing a second revision.

Probabilistic safety assessment

15. Level-1 PSAs of varying levels of complexity have been carried out, considering the plant design status after short term upgrading and covering initiating events at full power. Separate PSAs for units 1-2 and 3-4 were performed by Bulgarian institutions, in collaboration with those from Spain and Russia, partly based on generic data from Russian NPPs and also data from the IAEA. In the PSA for units 3-4 seismic effects and internal fires were also considered. At present they are in the process of verification after IPERS missions. A PSA level-1 for shutdown states is currently underway. In-depth review is still outstanding.

Decommissioning

16. Bulgarian regulations, based on rules inherited from the former USSR, require the utility to provide documentation for decommissioning at least five years before the planned shutdown of a reactor. At present the preparation of technical proposals for units 1-2 decommissioning is underway in the frame of a Phare project with completion planned in 2000.

Operational safety

17. There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased.

Organisation, procedures, operation and maintenance

18. A number of upgrading measures have been fully implemented in Kozloduy units 1-4, or are well advanced:

- The training of existing and new personnel is now based on a systematic approach; management training has been introduced; operating personnel now have access to a modern multi-functional simulator of the Kozloduy training centre with well trained instructors,
- The technical specifications for operation have been significantly upgraded and are now unit specific,

- Symptom-oriented accident procedures for units 1-4 are under development in the frame of an international programme for VVER-440/230 reactors and are planned to be implemented by the end of the year 2000.

<u>Safety culture and management, quality assurance</u>

19. Since 1992, with Western assistance to the utility and the safety authority, plant management has pursued the objective of improving operational safety. The main goal of the management is to motivate personnel to continue the gradual increase in the safety and reliability of operation in order to reach a level comparable to Western practices. Results of the OSART mission of the IAEA to Kozloduy units 1-4 in January 1999 show that the status of operational safety has significantly improved. OSART gave a series of recommendations and encouraged NPP management to continue these improvements. A follow up OSART mission was agreed for the end of 2000.

20. There is competent staff at the plant dedicated to the continuous safety upgrading process. The management structure has been reorganised, the responsibilities clearly defined, and a Quality Assurance (QA) programme established. In the past the utility and the plant management have made significant progress in the implementation of a modern safety management system but improvements are still needed. In early 2000 the structure of previous NPP management (EP-1 and EP-2) was reorganised and now the units 1-4 (VVER-440) and units 5-6 (VVER-1000) have a common management.

21. The announcement of closure dates for units 1 and 2 present a new challenge for the utility and the plant management. Appropriate measures will be needed to ensure that motivation of staff for safe operation remains adequate during the remaining period of operation.

<u>Operational experience</u>

22. A systematic analysis of operational experience feedback (from Kozloduy and from other PWRs) has been ongoing since the early nineties.

<u>Emergency preparedness</u>

23. There is an on-site emergency plan in place. However, the national approach to emergency planning as a whole is currently under review.

(ii) Kozloduy units 5-6

24. The statements presented in this chapter regarding the safety of Kozloduy units 5-6 are based on the knowledge gained through active TSO involvement in the plant modernisation, IAEA mission records, and the information received through the VVER regulators forum.

Basic technical characteristics

25. The units 5-6 on the Kozloduy site are VVER-1000/320 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2.

26. In principle, the main safety features of units 5-6 are similar to the design of Western PWRs of the 1970’s. In the early years the units suffered from frequent disturbances mainly due to the low quality of some equipment. With the replacement of some control valves, such as Feed Water control valves and a number of items of I&C and electrical equipment, as well as modification of the Steam Generators, a reasonable performance has now been achieved. This is important for safety because the frequency of disturbances that might

initiate an accident has been reduced.

Safety assessments and programmes for further improvements

27. A plant specific safety assessment is not yet available, although insights gained from TSO assessments of similar plants (e.g. Rovno 3) may be applicable to units 5-6. In developing the extended modernisation programmes for the VVER-1000 reactors, the utility has performed some plant specific safety analyses based on both deterministic and probabilistic approaches. But it has also used IAEA recommendations and operational experience at similar plants.

28. The PSA for Kozloduy units 5-6 is the first one performed in Bulgaria by its own experts. It is a level-1 study covering initiating plant events at full power, and also including fire and seismic events. It has undergone an IPERS mission review and a review by Western TSOs. In the frame of the modernisation programme, the operating organisation intends to adapt the PSA to the new plant status taking into account the TSO recommendations.

29. A programme for further upgrading of the units 5-6 is at an early stage and has been reviewed by Western TSOs. The main safety improvements relate to fuel and control rod optimisation, long term cooling including measures for prevention of sump filter clogging, electrical systems, instrumentation and control, containment integrity and radiation monitoring. The programme involves major Western and Russian partners and is planned for completion in stages over the next few years. Safety assessments by Western TSOs for similar plants in Ukraine and the Russian Federation have indicated that, after safety upgrading, it should be possible to achieve a level of safety in line with international recognised safety practices. However, to confirm this, a consistent safety case needs to be established and an adequate safety analysis needs to be made. Both will need to be reviewed by the CUAEPP.

Operational safety

30. Information and conclusions presented above for the Kozloduy 1-4 units are also generally applicable for units 5-6.

National industry infrastructure for technical support

31. In Bulgaria there are only limited resources of independent technical support organisations in support to Kozloduy NPP. These include Energoproject Sofia, several institutes of the Academy of Science, Riskengineering, ENPRO consult and BEQE.

On-site spent fuel and waste management

32. Spent fuel of the VVER-440 reactors is stored in an on-site fuel store erected in the 1980’s. For the VER-440 units there is currently an agreement with Russia which permits transport of this spent fuel back to the Russian Federation. Presently the spent fuel store is being modified to accept VVER-1000 fuel from units 5-6. This fuel is currently stored in pools within the containment and the storage space is nearly full. Radioactive wastes originating at Kozloduy are stored in interim storage facilities and an on-site cementation plant for liquid wastes is being built, although with considerable delay.

Conclusions

General remarks

33. There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased.

34. The lack of SAR is a serious shortcoming for judging the safety of NPP.

Kozloduy units 1-4

35. The short term upgrading measures implemented at units 1-4 have significantly improved the safety of these units. The measures taken so far have been directed mainly to the prevention of incidents and accidents.

36. Despite the safety improvements already achieved and considering the present safety status of the plant, there are still some major safety issues which are closely linked to the original basic design of the VVER-440/230 reactors and which are difficult to be removed. Among these are the limited confinement function and capability and the vulnerability against common cause failures. For Kozloduy 1-2 the implementation of relevant measures cannot be expected taking into account the announced closure dates.

37. Further safety improvements are being implemented or planned. The current safety-upgrading programme includes the extension of the design basis to a 200-mm break and the consolidation of the confinement system improvements. The utility and its technical support are motivated to the implementation of these improvements and have announced their intention to implement safety-upgrading programmes to mirror those that have been implemented at Bohunice V1. However, the work at Kozloduy is at least three years behind that at Bohunice and consequently safety improvement is not as far advanced.

38. In view of the large amount of work required to be carried out in the next modernisation stage it is difficult to have a final judgement on the adequacy and feasibility of all measures foreseen in this programme. It seems that financial provision for continued safety improvements are inadequate for Kozloduy 1-4.

Kozloduy units 5-6

39. In principle, the main safety features of these units are similar to Western PWRs. A programme for further upgrading of these units is at an early stage and has been reviewed by Western TSOs.

40. Safety assessments done by Western TSOs for similar plants in Ukraine and Russia indicate that, with the completion of the planned safety upgrades, it could be possible to achieve a level of safety for units 5-6 that is in line with international recognised safety practices. However, to confirm this, all safety measures from the programme have to be implemented and a consistent safety case has to be established. Both have to be reviewed by the CUAEPP.

References

1. Convention on Nuclear Safety (CNS), Answers to Questions on the National Report of Bulgaria, April 1999.
2. International Conference on the Strengthening of Nuclear Safety in Eastern Europe, Vienna 14 - 18 June 1999, IAEA-CN-75.

CZECH REPUBLIC

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. A new Atomic Act (law on peaceful utilisation of nuclear energy and ionising radiation) came into force in 1997. It confirms the SÚJB as the responsible body for supervising the utilisation of nuclear energy and ionising radiation. It defines the competencies of the SÚJB for the licensing of nuclear installations as well as the assessment, inspection and enforcement activities.

2. The Atomic Act states that the operator is responsible for the safety of its installations. The company that operates the nuclear power plants is a share holder company in which the state controls the major part.

3. Since the new Atomic Act came into force in 1997, the SÚJB has prepared or revised all regulations arising from the Atomic Act. The issuing of these regulations is an important accomplishment of the SÚJB.

4. The Czech Republic is a contracting party to all key international conventions dealing with nuclear safety.

5. The nuclear legislative framework in the Czech Republic is comparable with Western European practice.

Status of the regulatory body and technical support infrastructure

6. The SÚJB is a central agency of the State Administration reporting to the Government. Its President may participate in the meetings of the council of ministers. If needed, the Vice-Prime Minister, in charge of economy and finance, ensures the link between the council of ministers and the SÚJB. The SÚJB is funded from the State budget, approved by the parliament.

7. The SÚJB is responsible for nuclear safety, radiation protection, transport of nuclear and radioactive material, international notification of incidents and accidents, the provision of information to the public, nuclear material accountancy, and the import and export of dual purpose equipment. The SÚJB plays an important role in the emergency preparedness and planning in conjunction with other administrative departments.

8. The SÚJB has the power to issue and withdraw authorisations. It also has the power to impose penalties on the operators for any violation of the conditions of an authorisation. Enforcement actions by individual inspectors can be appealed to the SÚJB President, the next level of appeal being the court of justice.

9. The SÚJB considers that its current budget is sufficient. It obtained a 13% increase in 2000 to facilitate the licensing work for the Temelin nuclear power plant. The SÚJB has a special budget for research, which is divided equally between radiation protection and nuclear safety. Certain administrative constraints arise for the SÚJB when it contracts for technical

support. Except for small contracts or matters that are urgent from a safety point of view, the SÚJB is obliged to go through an open tendering process. This does not favour the long-term contractual technical support that the SÚJB needs.

10. The SÚJB was able to recruit 30 new staff over the last 3 years, which led to a total of 161 staff (as of 1st January 2000) engaged on nuclear safety and radiation protection activities. The National Radiation Protection Institute, with a staff of 110 providing technical support on radiation protection, is under direct SÚJB supervision. From 1st January 2000 another TSO with 45 staff came under the direct control of the SÚJB. Although its main field of operations is in the area of non-proliferation of nuclear, biological and chemical weapons, it also has considerable capabilities in the area of radiation protection and emergency preparedness.

11. Technical support for nuclear safety is provided by the Nuclear Research Institute (ÚJV), Institutes of the Academy of Science of the Czech Republic, universities, private companies and foreign organisations (for example from Slovakia). But there are a limited number of experts available from within the Czech Republic, which leads to the SÚJB needing to share competencies with the operators. Moreover, the contracting procedures with which the SÚJB must comply reduce the possibility of having long-term contracts for dedicated regulatory technical support. For the future, the SÚJB would prefer to replace some of the short-term contracts with individual contractors by long-term agreements with an extended scope of support. The administrative rules should be adapted in order to make this possible.

12. It is considered that, in general, the SÚJB has a status comparable to that of Western European regulatory bodies.

Status of regulatory activities

13. Since 1992, a number of national and international evaluations of the SÚJB have taken place. The recommendations of the various missions and support programmes have been used effectively in the development of Czech regulatory activities. The SÚJB takes an active part in international regulatory co-operation.

14. The Atomic Act authorises the SÚJB to draft subordinate regulations which, after approval by a legal advisory group of the Government, are signed by the SÚJB President. The laws and decrees issued in the Czech Republic contain very detailed requirements. The SÚJB needs to provide the Government with feedback on the application of the current regulatory pyramid and, if appropriate, propose the necessary changes. The SÚJB intends to continue developing technical guidance documents on the application of these regulations for the operators as soon as resources are available after the licensing of the Temelin nuclear power plant.

15. A well-defined licensing process for nuclear installations according to Western practice has been set up in the Czech Republic. It is governed by the Atomic Act and the Construction Act and includes the steps of siting, construction, operation and decommissioning. Major licences for siting, construction and permanent operation are issued by the District Authorities of the region where the installation is located. Such licences cannot be granted if the SÚJB issues a negative opinion regarding the safety of the plant. The District Authorities collect opinions from all other involved bodies of the state administration, including SÚJB. In addition to this process, there is a set of individual SÚJB approvals,

which have to be granted (in accordance with the Atomic Act), for individual steps within the siting, construction, operation and decommissioning phases of a nuclear installation. The environment impact assessment, which is part of the licensing process, includes a statement on the decommissioning options.

16. The methodology for assessment of safety related documentation is derived from US NRC practice. In addition to the assessment of the safety analysis reports, the SÚJB also assesses and approves such documents as plant technical specifications, the physical protection plan and the utility's quality assurance programme. Requirements for periodic safety reviews are included in licence conditions, usually requesting a review after 10 years of operation. However, when a plant is undergoing a modernisation programme, the periodic safety review is regarded as part of that programme.

17. The SÚJB inspection activities also derive from US NRC practices. They are based on a biannual inspection plan. The inspection plan and the inspection committees are the foundation of the SÚJB system of experience feedback. The SÚJB has established event-reporting requirements for the licensee and has developed a system for analysis and feedback of the licensee's operating experience. This is similar to Western European practice. The SÚJB also actively participates in the INES and international event reporting systems. In addition to its participation in the VVER regulators' forum, the SÚJB has an international agreement with Slovakia and Hungary to share the experiences gained at Dukovany, Bohunice, Mochovce and Paks.

18. The SÚJB has established two advisory committees, one for nuclear safety, the other for radiation protection. This provision is recognised as a good practice. In addition, special advisors have also been contracted for the licensing of the Temelin nuclear power plant.

19. In summary, the SÚJB has developed a series of regulatory practices that compare favourably with those of Western European nuclear regulators. The SÚJB is giving high priority to the licensing of the Temelin nuclear power plant and will resume the development of guidance documents after this period of intensive activities.

Emergency preparedness on governmental side

20. The new Act in the field of emergency preparedness and planning was passed by the Parliament in June 2000. In the case of an emergency situation of any kind, the co-ordination of all activities is the responsibility of the Inter-Ministerial Crisis Co-ordination Committee. This is composed of sub-committees such as the one for protection of the public, of which the SÚJB President is a member.

21. In the case of a nuclear emergency, the SÚJB has a role to advise the authority responsible for the protection of the public. To this end it has created an emergency response centre.

22. On-site emergency plans are approved by the SÚJB. It also ensures their consistency with the off-site plans that are approved by the head of the District Authority.

23. Neighbouring countries, e.g. Austria, have been invited as observers during emergency exercises. The national organisation for emergency preparedness needs to be further tested during exercises. However, the SÚJB considers that it will be difficult to test the national organisation in an exercise prior to the implementation of the new Act. The Czech Republic has participated in INEX-2 international exercises.

24. It is concluded that the SÚJB has taken the appropriate steps to fulfil its role in emergency preparedness.

Conclusions

25. The regulatory regime and regulatory body in the Czech Republic are comparable with those in Western Europe. Nuclear Safety legislation establishes the roles and responsibilities of the utility and the regulatory body. The regulatory body is well engaged in the state control of nuclear activities and the national emergency organisation is defined. A well-defined licensing process according to Western practice has been set up in the Czech Republic.

26. It is recommended that the Government of the Czech Republic consider the following:

- The implementation of the new Act on emergency preparedness and planning needs to be given a high priority,
- It seems that the documents in the regulatory pyramid in some cases may contain too detailed requirements. The SÚJB should be requested to suggest simplifications,
- The contracting rules of the SÚJB need to be adapted so that it can obtain, when appropriate, the necessary high quality technical support on a long term basis.

Chapter 2: Nuclear power plant safety status

Data

1. The Czech Republic has two nuclear power plants (NPP) at Dukovany and Temelin. Temelin NPP is the only plant within EU candidate countries, which is not yet in operation. Fuel loading of unit 1 started on 5 July 2000, fuel loading for unit 2 is planned to be approximately 15 months later.

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Dukovany: (in operation)				
Unit 1	VVER-440/213	1974	02/1985	2015
Unit 2	VVER-440/213	1978	01/1986	2016
Unit 3	VVER-440/213	1978	11/1986	2016
Unit 4	VVER-440/213	1978	06/1987	2017
Temelin: (under construction)			Fuel loading	Design lifetime
Unit 1	VVER-1000/320	1986	07/2000	30 years
Unit 2	VVER-1000/320	1987	11/2001	30 years

2. The plants are owned by CEZ a.s. (Czech Power Company), a joint stock company. CEZ is the sole license holder for the construction and operation of nuclear power installations in the Czech Republic.

(i) Dukovany units 1-4

3. The information given in this report on Dukovany NPP is based on the general knowledge on VVER-440/213 plants (summarised in Annex 2), the Czech National Report for the Convention on Nuclear Safety (April 1999), IAEA documents and information provided by the SÚJB and the NPP.

   The plant specific technical statements mainly rely on information provided by the operator on the occasion of a two days expert meeting with the SÚJB and the operator in June 1999 at Dukovany. Major safety issues were discussed and a summary list of upgrading measures (already implemented or planned in the near future) was provided by the operator. A second meeting of TSO expert organisations with the regulatory authority and the operator took place in May 2000. Since Dukovany NPP was not supported by large Western TSO projects in the past, both expert meetings were most worthwhile in providing technical information on the safety status of the NPP. Other background documentation which has been used is listed in the references.

   An in-depth safety assessment of Dukovany NPP, in particular a review of the modernisation programme (MORAVA), has not been made by Western TSOs. The operator, however, offered to give further help in confirming and expanding the technical information on the plant's safety status given so far.

4. The initial design lifetime for each unit as a whole is 30 years from first criticality. For each of the reactor pressure vessels the design lifetime is 40 years.

Basic technical characteristics

Design basis aspects

5. All units of Dukovany NPP are second generation VVER-440/213 type reactors. Generic safety characteristics of these reactors are presented in Annex 2.

6. For the primary circuit and the safety-related systems, the basic design was made by Russian organisations. The specific plant design was developed and carried out by Energoprojekt Prague, a Czech company which, under Czech law, became the only responsible organisation for the design. All major parts of the primary equipment (except the main circulation pumps) as well as the equipment of the whole secondary circuit were manufactured in the former Czechoslovakia, mainly by Skoda Plzen, Vitkovice, etc. Domestic companies were also engaged in the quality control during manufacturing and construction. Since the nineties fuel manufacturing in Russia is also under Czech quality control. No major quality concerns have been identified in tests and inspections carried out since start of operation. Since the first years of plant operation, safety improvements have been made continuously. A major back-fitting programme had already started in 1991 based on the safety assessment of Greifswald unit 5, analyses and supporting programmes of the IAEA and WANO, and other international co-operation. Major safety improvements were focussed on fire protection, electrical supply, secondary side feedwater supply, and the installation of an emergency response centre. Further improvements are either under design or planned.

Reactor pressure vessel and primary pressure boundary

7. Reactor pressure vessel integrity (especially safety margins against radiation embrittlement) appears to be adequate for all units. Due to the well-balanced composition of material impurities (low content of Phosphorus, Copper) and the protection measures to lower the embrittlement rate, it is expected that annealing will not be necessary for any of the vessels during the design lifetime. To ensure pressure vessel integrity various measures have been introduced, e.g. low leakage core configuration and pressure vessel embrittlement monitoring by a surveillance programme. In-service inspections of the reactor pressure vessels and the primary piping are conducted with state-of-the-art techniques.

8. The piping systems were designed in accordance with Russian and Czech standards. A set of primary pipe whip restraints has been partially installed. A partial leak-before-break (LBB) implementation exists, but it is not relied upon in the safety case. Several preventive measures on steam generator (SG) integrity have been implemented or are underway (e.g. N16 activity measurement on each steam line, measures for exclusion of corrosion damage at flange connections, new feedwater distributors (inside the SG) in order to exclude primary collector thermal fatigue). Accident analyses have been performed and corresponding emergency operating procedures have been revised.

9. After completion of pipe whip restraints, the integrity of the primary pressure boundary is considered to be adequately safe.

Confinement

10. The leak rates have continuously decreased since the commissioning but they are still slightly higher than those that are usually accepted in Western PWR containments. For design basis accidents, however, radiological consequences would not exceed those accepted within EU countries. The performance of the bubbler condenser system in case of Large Break LOCA has been verified in full-scope tests in the frame of the Bubbler

Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents.

<u>Safety systems and hazards</u>

11. In terms of capacity and redundancy the design of the safety systems is in general comparable to Western reactors of the same vintage (see Annex 2). Several measures against hazards (e.g. fire protection) have been taken in order to improve the separation between redundant trains. Further upgrading protection measures are completed or under way. Protection against sump screen clogging has been implemented. Secondary pipe whip restraints are scheduled to be added at the 14.7-m level in accordance with US standards, based on results of a recent analysis.

12. For improving the original generic VVER-440/213 design at Dukovany NPP, an independent Emergency Feedwater System has been installed in a separate building. Former shortcomings have been eliminated.

13. A systematic fire hazard analysis and a flooding analysis were carried out in 1997. Major weak points already have been eliminated (e.g. fire prevention measures). Further measures are underway or planned to be completed in 2000. Measures to cope with high-energy pipe breaks are under development, the completion is scheduled for 2003.

14. Seismic qualification of existing equipment is ongoing in the frame of the MORAVA Project; all new implemented equipment is qualified to withstand 0.1g which is acceptable for this site according to Western practice.

<u>I&C systems and emergency power supply</u>

15. Many improvements on I&C and electrical equipment have already been introduced or are underway. Based on insights gained from reliability analyses, proposals for modifications in the safety related I&C have been developed and will be implemented in 2001. Under current plans of the utility major upgrading of the I&C with digital systems is foreseen by 2010.

16. Various means for condition monitoring of mechanical components, e.g. vibration monitoring of reactor internals, lose part monitoring, on-line operational load measurement as well as ageing monitoring for key components, have been introduced.

<u>Beyond design basis accidents and severe accidents</u>

17. Analyses on some representative beyond design basis accidents (e.g. ATWS, total loss of heat sink, total loss of electrical power) were completed in 1998. The results of these analyses were used in the development of symptom based emergency operating procedures. Analyses on selected severe accidents with core melt scenario have been performed within the scope of a regional Phare project and in the frame of a level-2 PSA.

Safety assessments and programmes for further improvements

<u>Safety assessment and documentation</u>

18. In 1991 the former Czechoslovak Atomic Energy Commission (CSKAE) established conditions for licensing unit 1 for continued operation beyond 10 years (after 1994). In particular, this required the operator to provide a revised SAR, the so-called Operational Safety Analysis Report (OSAR). OSARs also have been prepared for units 2-4. Based on

the OSAR, the SÚJB issues time-limited licences for further operation.

The structure and content of the OSAR are in compliance with the Regulatory Guide n°5 from 1988 and, to a major extent, with the later IAEA guide for periodic safety reviews (IAEA Safety Series 50-SG-O12).

19. All the modifications and safety improvements implemented at Dukovany NPP have to be included continuously in the safety analysis reports of the corresponding unit.

Probabilistic safety assessment

20. In 1992 the first version of a level-1 PSA study for Dukovany NPP was developed by Nuclear Research Institute Rez (NRI), in co-operation with several Czech and Slovak research institutes. In 1994 the updated level-1 PSA for Dukovany NPP was completed. The study was the first level-1 PSA completed for a VVER-440/213 reactor by a Western contractor. Since 1995 NRI has regularly updated the Dukovany level-1 PSA under a living PSA project. The current version of the level-1 PSA includes internal initiating events, fires and floods. The results were used for confirmation and scheduling of upgrading measures within the scope of the MORAVA programme (see § 24) and for refining of the emergency operating procedures. Finally in 1998 the level-1 PSA study was reviewed by an IAEA IPERS mission.

21. In addition, a shutdown PSA (SPSA) has been carried out. The results of the SPSA indicate that the contribution to the total core damage frequency is comparable with that of operation at full power. The results of the SPSA are being used to improve procedures for shut down accidental conditions. First results of a level-2 PSA study are already available and they will be used as an input for severe accident guidelines.

Safety measures and further assessments

22. The Dukovany NPP is involved in international co-operation. Several IAEA missions (OSART, ASSET, IPERS, etc.) have been performed to assess plant operational safety.

    All important safety issues have been addressed in the existing safety programme and are either resolved or are underway. It is intended that the relevant measures will be resolved according to a schedule and will be complete by the year 2002 [1].

23. The Dukovany NPP is practising an extensive exchange with WANO and participates in common activities with other VVER-440/213 operators.

Programmes for safety improvements

24. An extensive modernisation programme (MORAVA) has been established based on Western nuclear safety standards and evaluation of operational experience [2]. The whole modernisation programme will be fully implemented by 2010. The major safety modifications, except I&C, will be completed by 2004. Upgrading of the safety related parts of the I&C with digital systems is planned to be implemented during refuelling outages and will be complete by 2010. The main objective of the programme is to achieve a safety level that is fully comparable with international safety standards and NPPs operating in EU countries.

25. Major upgrading measures which have already been implemented or are under way are for example:

- Automatic protection against primary circuit cold overpressure,
- Protection against sump screen clogging,
- Modification of equipment on the 14.7-m floor as pipe whip restrains, protection against missiles, replacement of valves, two additional steam relief valves, replacement and re-routing of pipes are under way,
- Modifications on the emergency feedwater system (e.g. pipe whip restraints, qualification of valves, et al.) are underway.

Furthermore, additional measures for assuring safe operation are underway, e.g.:

- Reconstruction and extension of diagnostic monitoring equipment,
- Installation of a full scope simulator.

Reconstruction of the I&C system is under preparation.

Operational safety

Organisation, procedures, operation and maintenance

26. Staff responsibilities within the NPP are clearly defined. Nuclear safety and production are separate divisions within the management organisation. The head of the nuclear safety division is a deputy director.

27. Until now the plant operational personnel have been trained at the full-scope simulator of the VUJE Education and Training Centre (Slovakia). At Dukovany, a plant specific full-scope simulator has been installed and training is planned to start there from the beginning of 2001.

28. Symptom oriented emergency operating procedures (EOPs) have been developed in co-operation with Westinghouse. The new EOPs were fully introduced in November 1999.

Safety culture and management, quality assurance

29. The safety culture of Dukovany NPP has been continuously improved. Two OSART missions in 1989 and 1991 noted a high level of nuclear safety and a professional management with competent and trained personnel. A WANO peer review was performed in 1997.

30. A comprehensive quality assurance programme (QA) was established in compliance with IAEA recommendations and regulatory requirements. A management system has been set up in order to assess the safety significance of plant modifications and to ensure their proper implementation.

Operational experience

31. The reliability of plant operation since its first start-up is an indication of the good quality of the equipment.

32. Over the last ten years the average number of unplanned shutdowns (scrams) per unit has been less than 1 per year. A system has been established to ensure efficient feedback of operational experience from Dukovany NPP and other NPPs, especially from VVER reactors.

Emergency preparedness

33. The emergency plan is regularly updated and exercises are carried out annually. The Dukovany Crisis Centre is equipped with necessary computerised support systems. The level of preparedness achieved is adequate.

(ii) Temelin units 1-2

34. Originally it was planned to build 4 VVER-1000 type reactors at Temelin. Construction of the first two units started in 1986. In the early 1990s the original plan, however, was revised. In 1993 the former government decided to complete only units 1 and 2. This decision was re-approved last year by the current government.

35. Background information on Temelin NPP is available from several IAEA documents and to some extent from bilateral co-operation with institutions from EU countries. Furthermore additional generic information on the main safety features of VVER-1000 derives from Tacis and Phare projects on other VVER-1000 plants (e.g. Rovno 3, Kozloduy 5-6).

Several IAEA documents have been used for the assessment given in this chapter, e.g. IAEA report on VVER Safety Issues Resolution at Temelin NPP (1996) [3], review mission reports on Temelin NPP - PSA and External Events (1995, 1996) [4], [5]. A general overview on the Temelin NPP safety status and Safety Analysis Report (SAR) was presented by Czech experts at the IAEA Conference on Strengthening of Nuclear Safety in Eastern Europe, Vienna 1999 [2].

Recently a conceptual safety assessment study on selected safety issues of Temelin NPP was carried out by German expert institutions (GRS et al.) in close co-operation with the SÚJB. For the purpose of the study, plant specific information was made available at several bilateral expert meetings (Dec. 1999 - May 2000) by the SÚJB and the NPP. The study performed by GRS, however, does not replace an overall safety review of the plant.

Basic technical characteristics

Design basis aspects

36. Both units of Temelin NPP are of the standard VVER-1000/320 type. Their design concept is similar to Western PWRs of the same vintage. General safety characteristics of the VVER-1000/320 are presented in Annex 2.

37. The Local Civil Constructions Authorities issued the construction permits for units 1 and 2 in 1986 based on statements of the authority (former Czechoslovak Atomic Energy Commission, CSKAE). These construction permits however were given under some specified conditions, e.g. a re-analysis of all design basis accidents using qualified calculation tools. These conditions have been fulfilled.

38. Qualified Czech (former Czechoslovak) companies manufactured major parts of the equipment (e.g. reactor pressure vessel, steam generators, pressurizer, all equipment on the secondary side). A large part of the systems and their supporting plant (e.g. electrical supply) were designed, manufactured and installed by Czech organisations. Domestic companies were also engaged in quality control of the main equipment and thereby gained knowledge and experience of quality verification.

39. The design of Temelin NPP has been the subject of continuous improvements and

modifications, which have been reviewed by many international expert groups. Numerous individual improvements were implemented already before 1990. Further safety improvements in Temelin NPP have been strongly influenced by international co-operation.

40. Major safety design changes include:

- Replacement of I&C,
- Replacement of core and nuclear fuel,
- Replacement of the original radiation monitoring system,
- Replacement and supplementing of the diagnostic system,
- Replacement of original cables with fire-retardant and fire-resistant ones,
- Significant changes in the electrical design (electrical protections, addition of 2 non-safety grade diesel generators, increased discharge time of batteries).

The most important design modifications (core design and I&C) were supplied by a Western vendor. According to the SÚJB and the NPP, the combination of Eastern and Western technologies did not cause major problems because I&C was replaced completely and there was one main contractor for accident analysis, core design and I&C. However the interfaces between the different technologies were considered continually during design and implementation.

41. Re-assessment of generic safety issues for VVER-1000 reactors was performed just before fuel loading and results were provided to IAEA as an open report to the member states [6].

42. The safety improvement programme implemented in Temelin NPP is the most comprehensive one that has been applied to a VVER-1000/320 plant so far.

Reactor pressure vessel and primary pressure boundary

43. The high quality of the reactor pressure vessel, manufactured by Skoda, Plzen, is well documented. The Nickel impurity content, however, is somewhat higher than today's more stringent specifications. To determine the effect of neutron irradiation on the material, a special irradiation programme covering end-of-life fluence condition has been performed. However, due to some uncertainties, a final assessment with regard to expected changes of material properties currently cannot be made. Therefore close attention has to be given to the monitoring of the embrittlement of the RPV during operation.

To reduce the rate of embrittlement and to ensure pressure vessel integrity, various measures have been introduced, e.g.:

- Pressure vessel embrittlement monitoring is carried out by an adequate surveillance programme which includes irradiation samples being inserted between the reactor core and the pressure vessel wall in the range of maximum neutron flux,
- Preheating of the water of emergency injection systems.

These measures minimise the risk of the RPV brittle fracture; moreover the surveillance will allow the identification of a possible acceleration of material ageing.

44. In service inspections (both from inside and outside) of the reactor pressure vessel (every four years), of the steam generators, the primary piping and other main equipment will be

conducted with state of art techniques.

45. To ensure SG integrity, design modifications and operational measures have been introduced reducing the possibility of primary to secondary leaks. Also a modified technology for the collectors manufacture has been used and, due to the replacement of the Cu alloys in the turbine condenser by pipe bundles from titanium, the water chemistry conditions (pH) of the secondary circuit have been improved.

46. Despite these improvements a leak (up to the maximum flow area of about 14 cm^2) is analysed as design basis accident. The leak size corresponds to the leakage coming from collector header lid lift up.

47. According to the results of a first PSA that was carried out in the early nineties, leakages from the primary to the secondary side have been the main contributors to the total core damage frequency. This was mainly due to conservative assumptions. Considering further updating of the PSA and thereby taking into account all preventive measures that have been introduced in the plant, it can be expected that the initiating frequency of a primary to secondary leak would be assessed more realistically, thus leading to a lower contribution to the core damage frequency.

48. LBB has been applied to the main primary piping (including the pressurizer surge line, low pressure ECCS, residual heat removal system and passive emergency cooling system) in order to reduce the probability of large primary breaks and to avoid the need for further reinforcement of existing pipe whip restraints. Therefore it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

<u>Confinement</u>

49. Constructive improvements on the pre-stressing of the containment tension cables as well as improved monitoring of the pre-stressing system and the concrete structure have been introduced. The measured containment leak rate of unit 1 is comparable to that of Western reactors. It can be concluded that the containment function can be ensured for the postulated DBA.

<u>Safety systems and protection against hazards</u>

50. In terms of its capabilities, the redundancy and separation of the safety systems (e.g. ECCS, EFWS, AC and DC emergency power supply) is comparable to that of Western PWR of the same vintage (see Annex 2).

51. A comprehensive I&C modernisation has been carried out on unit 1 and is underway on unit 2. The old I&C system (original design) was replaced by the new one which is based on modern technology (digital I&C system). Systems which are important for safety have been modernised such as Reactor Trip System, Engineered Safety Features Actuation System and Post-Accident Monitoring System. The new I&C system also covers the Reactor Power Control and Limitation System, the Unit Control System and the Unit Information System. Systematic monitoring is carried out to register failures, using such tools as automatic testers, self-diagnostics, data quality and validity tests, communication diagnostics and manual tests.

52. To ensure correct interactions between the new I&C and the original equipment, all stages of the design and implementation were carried out jointly by Energoprojekt and Westinghouse. Basic design, detailed design and the systematic analysis (functional design)

are verified with best estimate analysis. A special and comprehensive independent verification and validation programme similar to the one at Sizewell B NPP was implemented and accepted by the SÚJB.

53. Additional monitoring of the reactor coolant pressure boundary integrity has been introduced e.g. vibration monitoring of reactor internals, lose part monitoring, on-line operational load measurement as well as ageing monitoring for key components.

54. Measures to address the ECCS sump screen clogging issue in the case of a medium or large LOCA have been introduced but their effectiveness needs to be verified.

55. Far reaching measures for protection against internal hazards have been implemented:

- A systematic fire hazard analysis and a flooding analysis have been carried out,
- Comprehensive measures to increase fire protection (e.g. replacement of the original cable with fire resistant cables),
- Measures preventing consequential failures due to high energy pipe breaks. To compensate for missing spatial separation, additional pipe whip restraints have been installed at the 28.8m level as a protection against postulated steam piping and feedwater line rupture (following current US regulations).

Some specific issues, however, e.g. protection against postulated steam piping or feedwater line rupture on the 28.8-m level, need further consideration.

56. Plant-specific safety demonstration for the functioning of the main steam relief valves and the main steam safety valves under dynamic loading with a steam-water mixture still has to be fully verified. This action is underway. This function is needed to control specific primary to secondary leaks.

57. The seismic re-evaluation of the Temelin NPP locality was performed in accordance with the IAEA methodology (Safety Series 50-SG-S1) using design value of g=0.1. The new seismic analyses were performed for all safety important buildings, components, control systems, I&C and electrical systems. Based on this seismic re-evaluation, modifications and changes were carried out (e.g. installation of additional dampers at the pressurizer pipeline for cold spray).

<u>Beyond design basis accidents and severe accidents</u>

58. Compared to the original design, representative BDBAs have been analysed (e.g. station black out, total loss of heat sink, ATWS). Where it was considered necessary, corresponding measures were implemented (e.g. additional Diesel Generators, pressurizer safety valves withstanding water/steam mixtures for bleed and feed, confirmation of the gas removal system effectiveness for bleed and feed).

59. A systematic analysis of severe accident scenarios selected based on the preliminary analyses and results of PSA studies was performed which permitted the proposal and evaluation of the accident management strategies. Advanced severe accident analysis codes of Western European and US origin have been applied. At present the accident analysis is oriented to support the development and validation of methods and procedures for severe accidents management.

Safety assessments and programmes for further improvements

Safety assessment and documentation

60. Before the commissioning of Temelin NPP the Pre-Operational Safety Analysis Report (Pre-OSAR) was available. Structure and content of the Pre-OSAR is in accordance with the US NRC Regulatory Guide 1.70 taking into account characteristics of the VVER-1000 design and plant specific modifications. Additional requirements of the SÚJB were addressed.

61. The Pre-OSAR is one of the preconditions for fuel loading. For the licensing of plant operation the Pre-OSAR will be amended by results of the commissioning process (tests, etc). This new document will be the final Operational Safety Analysis Report (OSAR). According to regulatory requirements for periodic safety reviews the OSAR has to be updated after each ten years of operation.

Probabilistic safety assessment

62. In the beginning of the nineties a US consultant performed a PSA (level-1 and 2) in co-operation with the utility. The level-1 PSA also includes events during shutdown states. Interim PSA results have been used to complement design related and operational safety upgrading measures. Further updating of the PSA is underway, taking into account plant modifications, Emergency Operating Procedures, more realistic plant specific input data and results from recent accident analyses.

63. The level-2 PSA includes an analysis of the containment strength, a determination of the impact of the core melt progression on the containment structure and an evaluation of the fission product release (timing, frequency and magnitude) for various accident sequences. The Temelin level-2 PSA is one of the first analyses performed for VVER-1000 plants. The analysis involved standard scope, approach and procedures.

    All Temelin PSA models (including level-2) will be updated in order to reflect all design modifications and safety improvements, and the real operational status of the plant.

Safety missions and further safety improvements

64. During construction of Temelin NPP a series of IAEA and others missions have taken place covering various safety aspects: plant construction practice, safety systems evaluation and safety analyses, fire protection, quality assurance, resolution of safety issues, etc.

65. Investigations on selected safety issues (e.g. accident analysis) were also carried out by several Western institutions. Some preliminary Western safety assessments on Temelin NPP were carried out during bilateral co-operation.

Operational safety

Organisation, procedures, operation and maintenance

66. The training of the operating staff of Temelin NPP basically follows the same scheme as that of Dukovany NPP. Plant operators have been trained on a plant specific full scope VVER-1000 simulator at the site. The simulator will be adapted based on commissioning results and first operational experience.

67. Symptom oriented emergency operating procedures have been developed to support operator actions during accident conditions.

Safety culture and management, quality assurance

68. The plant management is committed to develop and maintain a strong safety culture. Support on this subject has been received from IAEA and Western organisations and companies.

69. The utility has developed its own competence achieving independence from the original Russian supplier. Nonetheless, the present situation at the NPP is characterised by good relations with the original designer and close co-operation with Russian experts.

70. A comprehensive quality assurance programme (QA) has been established in compliance with IAEA recommendations and has been approved by the SÚJB.

71. Final tests by NPP staff before commissioning indicated no major concerns and good quality of equipment.

Operational experience

72. Operational experience from other VVER-1000 NPPs and Western PWRs with comparable parts of equipment (e.g. digital I&C) has been examined. Further relevant information has been gained from OSART missions to operating VVER-1000 NPPs, several other IAEA missions and co-operation with WANO.

Emergency preparedness

73. The on-site emergency plan is based on the plan for Dukovany NPP but has been revised and updated. For the programme supporting emergency preparedness at Temelin, technical and normative documentation of other countries and IAEA were taken into account.

National industry infrastructure for technical support

74. The Czech Republic has a strong national infrastructure in the nuclear field due to fact that Czechoslovakia has developed its own reactor in the past and later it was the supplier of main components for VVER reactors. This infrastructure includes also research and general design at Energoprojekt Prague. Skoda is responsible for design work in the frame of the components it supplies. The Nuclear Research Institute Rez provides technical support in different areas, e.g. component integrity, especially the Reactor Pressure Vessel. As part of the infrastructure, one could also note the remaining links with the Slovakian institutions, especially the VUJE Institute.

On-site spent fuel and waste management

75. At present spent nuclear fuel from Dukovany NPP is stored for a 6-year period in the reactor storage pool and subsequently transferred into CASTOR casks. In the early years of operation, spent fuel was temporary stored in the interim storage at Bohunice NPP. All spent fuel has now been transferred back to Dukovany NPP. In 1997, at the Dukovany site a spent fuel interim storage facility with a capacity of 600 tons has been built and commissioned by CEZ.

    The radioactive liquid and solid waste is reconditioned and stored at the site.

76. It is planned that Temelin spent fuel elements will be stored in the reactor storage pool of the plant (inside the containment) for about 10 years. Subsequently the spent fuel elements will be transferred to the interim storage facility.

Conclusions

(i) Dukovany units 1-4

77. The following conclusions can be drawn:

- In the early years of Dukovany NPP operation, modifications were carried out to remove safety deficiencies in the original design,
- Dukovany´s containment structures provide adequate protection against design basis accidents and the overall radioactive releases would not be higher than would be accepted within the EU. However their leak-tightness is not as good as that of typical containments in Western Europe. This would have some influence in the progress and consequences of potential severe accident scenarios,
- Safety assessments and verification documents, e.g. periodic safety reviews, are conducted comparable to Western practice,
- Extensive PSA studies have already been performed or are currently underway,
- The safety culture has been continuously improved and appears to be adequate,
- An extensive modernisation programme (MORAVA) has been established and expected to be implemented within the next 10 years by 2010. All safety improvements except I&C replacement should be implemented by 2004.

78. By full implementation of the Modernisation Programme, it is expected that Dukovany NPP will achieve a safety level comparable to that of NPP of the same vintage operating in Western Europe.

(ii) Temelin units 1-2

79. The following conclusions can be drawn:

- The safety improvement programme for Temelin NPP is the most comprehensive one that has been applied to a VVER-1000/320 plant,
- From the beginning of plant construction improving the nuclear safety, radiation protection and accompanying safety evaluation was a continuous process,
- The combination of Eastern and Western technologies was successfully completed. I&C was replaced completely and there was one main contractor for accident analysis, core design and I&C. The interfaces between the different technologies were considered. A standard Western practice was used to combine Eastern and Western technologies including safety assessment. The commissioning process has to confirm the integration of the different technologies,
- Some safety issues still need clarification with respect to the safety of piping at the 28.8-m level and with respect to the verification of steam relief valves,
- If these issues are resolved, Temelin NPP will achieve a safety level that is comparable to that of operating Western PWR.

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HUNGARY

Chapter 1: Status of the regulatory regime and regulatory body

The information given here is based on experience gained through bilateral and multilateral assistance programmes and co-operation such as RAMG and CONCERT, IAEA missions and other open sources.

Status of the legislative framework

1. The first Hungarian regulations on nuclear safety were issued in the form of ministerial decrees in 1979, when the first two Units at Paks were under construction. These gave a framework for nuclear power plant licensing and safety inspections, and also contained technical requirements for nuclear safety. The first Atomic Energy Act was issued in 1980. The Paks nuclear power plant is owned by the state.

2. Revision of the nuclear legislation and the regulatory framework started in the early 1990s. IAEA guidance was used to help with this process and in addition the Hungarian authorities and experts became acquainted with the legislation and regulatory practices in a number of Western European countries. This international experience was reflected in the new legislation. The new Act on Atomic Energy was adopted by the Parliament of Hungary in December 1996 and the revised governmental decrees came into force in June 1997. The Atomic Energy Act establishes, among other pertinent things, the licensing process.

3. The governmental decrees issued under the Atomic Energy Act specify the duties and authority of the Hungarian Atomic Energy Commission (HAEC) and the Hungarian Atomic Energy Authority (HAEA). The Government decrees also mandate the Nuclear Safety Directorate (NSD) of the HAEA to act as the regulatory body for nuclear safety matters. Nuclear Safety Regulations, published as appendices to the governmental decree (108/1997), comprise the detailed rules and requirements for the nuclear facilities in licensing, quality assurance, design and operation as well as requirements for research reactors.

4. The Act clearly states that the operating organisation carries full responsibility for safety, and the legal status of the utility as an operating organisation is defined in the Act. The role of the regulatory body is to verify that necessary actions to ensure nuclear safety are taken by the operating organisation. The responsibilities of the regulatory body are generally well separated from the responsibilities of the operating organisation.

5. All the key international conventions related to nuclear safety have been ratified and included in national regulations. Peer evaluations of the Hungarian legislation and regulatory framework have been carried out by an IRRT mission of the IAEA.

6. It can be concluded that the legislative framework in Hungary provides a similar level of control to that generally found in Western European countries that have nuclear programmes. The legislation and other regulatory documentation is modern and comprehensive but some changes could be considered regarding their contents to strengthen the independence and co-ordination of the regulatory activities, as discussed in

the next section.

Status of the regulatory body and technical support infrastructure

7. The governmental supervision of the safety of the nuclear facilities is ensured by the HAEC, the HAEA, and relevant Ministers.

8. The role of the HAEC is to prepare proposals on nuclear issues for Government decisions, and to co-ordinate the work of state institutions in the nuclear field. The HAEC is a commission composed of senior officials of the ministries and the heads of the central public administrative organisations that perform regulatory tasks under the Act on Atomic Energy. The President of the HAEC is nominated by the Prime Minister from the members of the Government. Because of its broad mandate, the HAEC is also involved in promotion of nuclear technology activities. The President of HAEC monitors and supervises the activities of the HAEA, and reports annually on nuclear safety to the Parliament. At present the President of the HAEC is the Minister of Economic Affairs, and in this role he is in charge of energy policy.

9. The dual role of the Minister of Economic Affairs is a source of concern regarding independence of HAEC, although the statute of HAEC explicitly requires the HAEC President to act independently of his/her affiliation.

10. The HAEA is an executive body for regulation of nuclear safety and the control of nuclear materials. It has two main parts, each being headed by a deputy of the Director General. The Nuclear Safety Directorate (NSD) has responsibility for licensing, safety assessment and inspection of nuclear facilities. The General Nuclear Directorate has responsibilities regarding the safeguards of nuclear and radioactive material, transport of radioactive materials, and nuclear export and import. The Director General of the HAEA and his deputies are appointed by the Prime Minister.

11. In 1997, the work scope of HAEA was extended to include areas that were previously exclusively under the responsibility of different authorities: civil structures, radiation protection, emergency preparedness, fire safety and physical protection. These authorities still play a role in regulatory control of nuclear facilities. The HAEA also works in co-operation with authorities that regulate regional planning and environmental issues. Governmental decrees give only generic rules for co-operation with other authorities. The appropriate Ministry is responsible for nominating the co-authority. Despite recent improvements, the legal and governmental infrastructure of Hungary, with its distributed regulatory responsibilities, could be better co-ordinated in order to avoid any omission or overlap and to provide for effective co-operation between authorities.

12. Currently, HAEA has about 90 staff. Of these, about 40 are technical experts within the NSD. Taking into account the number and variety of nuclear plants, this number is comparable with Western European countries. NSD staff generally have a high level of technical competence, but in the areas of new responsibilities and inspection practices, as discussed below, there is need for training and/or an increase in the number of staff.

13. The NSD is authorised to take appropriately strong enforcement measures to ensure safety, such as ordering the shutdown of a reactor. It can also oblige the licensee to pay a fine for a violation of rules, although a need for such enforcement measure has not yet been necessary.

14. The Director General of the HAEA is the second instance for all appeals concerning the regulation of nuclear facilities and activities, and this may to some extent compromise the independence of the NSD in its regulatory decisions.

15. Radioactive waste treatment facilities as well as the interim storage of spent fuel at the plant site are regulated by NSD. However, the ultimate disposal facilities for radioactive waste and spent fuel are not classified as nuclear facilities by the present Hungarian legislation and are currently regulated by the Ministry of Health.

16. Funds for the NSD are specified in the Government budget (the major source of funds is the fees paid by the licensee), and it seems that the availability of funds has not been a limiting factor in their daily work. However, to ensure long-term stability, it would be more satisfactory if the NSD salaries were similar to those of the utility.

17. The technical know-how and staff resources within the NSD permits in-depth assessment of key safety issues. Support for this work is available from the national institutes such as KFKI AEKI (Atomic Energy Research Institute) and VEIKI (Institute for Electric Power Research), which together possess an independent, advanced safety analysis capability. KFKI and VEIKI both support the regulator and the utility; independence of work being assured by administrative rules at expert level. The technical support available to the regulatory organisation is competent and sufficient. The regulator has good access to the results of both national and international research programmes.

18. The current regulatory policy of the NSD strongly emphasises the reliance on national resources. NSD can fulfil its duties successfully without foreign assistance.

19. It can be concluded that the NSD of HAEA has duties, authorities and competence that are broadly comparable with those found in Western European nuclear regulatory organisations.

Status of regulatory activities

20. Since 1992, a number of national and international evaluations of the HAEA have taken place. The recommendations of the various missions and support programmes were effectively used in the development of Hungarian regulatory activities. The NSD has for a long time participated actively in international exchange of information.

21. The Director General has issued the Organisational and Operational Code of the HAEA, which describes in detail the regulatory regime, organisational matters, and the ways of working within HAEA. Another document issued by the Head of the NSD establishes the regulatory strategy and serves as a basis for internal Quality Assurance within the NSD. At the level of HAEA the written procedures and guidelines for internal QA, and the planning processes for management of regulatory activities, are in a development phase.

22. In addition to mandatory regulations, the regulatory documents include a set of safety guidelines that are issued by the Director General of the HAEA. There is an active programme for developing new safety guidelines and for upgrading the existing ones, as the need arises from experience. HAEA issued 33 guidelines between 1997 and 1999.

23. In connection with the Periodic Safety Reviews and the related Paks NPP operating license renewal, the NSD has developed a systematic safety assessment process. In addition, the

NSD established an extensive inspection programme. This programme is carried out by an inspection department of six experts who are permanently stationed at the Paks site. Currently, the inspection practices are being reviewed and are increasingly focusing on the work processes of the operating organisation, operating experience feedback, and integrated inspections that cover a certain area as a whole.

24. The regulatory system for analysis and feedback of operating experience from domestic events is similar to common Western European practices. HAEA/NSD is a member of the regional co-operation between the Czech Republic, Slovakia and Hungary, and a member of the VVER Regulators' Co-operation Forum.

Emergency preparedness on governmental side

25. HAEA/NSD acts in an advisory role in emergency situations. The Governmental Co-ordination Committee for emergency planning and preparedness for all kinds of emergencies is chaired by the Minister of the Interior. The Director General of HAEA is the vice-chairman in the specific case of a nuclear emergency. Under the Committee, there is an extensive national system for planning rescue measures, radiation monitoring, and providing information to the general public. HAEA/NSD has established a dedicated centre for emergency response, training and analysis (CERTA).

26. The national system for nuclear emergency preparedness has improved significantly over the past years. National exercises are carried out regularly. In addition, HAEA/NSD participates in the IAEA Central and Eastern Europe emergency-planning co-operation. NSD reviews and approves plant on-site emergency plans. The national emergency preparedness capability was tested in a large international exercise (INEX-2 HUN) in November 1998. The INEX-2 exercises and associated workshops have provided a good stimulus for recent improvements; in this respect it can be concluded that NSD has taken all appropriate steps to fulfil its role as a nuclear safety regulator.

Conclusions

27. The Hungarian approach to licensing, regulating and controlling nuclear facilities has developed strongly in the last ten years. A proper licensing process is in place. Legislation and regulations are up-to-date, and the Hungarian regulatory practices are comparable with those of Western European countries.

28. Issues that need to be considered by the Hungarian Government are the following:

- The fact that the Minister of Energy Affairs is also the HAEC President creates an apparent conflict of interest, even though the formal mandate of HAEC President precludes this,
- The number of different authorities with direct responsibilities in the regulation of nuclear facilities increases the risk that important issues may be overlooked, and reduces the efficiency of the regulatory work.

29. The NSD needs to continue its efforts to develop the inspection approach towards process oriented comprehensive team inspections.

Chapter 2: Nuclear power plant safety status

Data

1. Hungary has one nuclear power plant at Paks with four units:

Paks Unit	Reactor type	Start of construction	First grid connection	End of design life
Unit 1	VVER 440/213	1974	12/82	2012
Unit 2	VVER 440/213	1974	09/84	2014
Unit 3	VVER 440/213	1979	09/86	2016
Unit 4	VVER 440/213	1979	08/87	2017

2. The plant is owned by the Paks Nuclear Power Plant Ltd. which is 99.92% owned by the Hungarian Power Companies Ltd. The latter is owned by the state.

3. The following statements are based on information available in open literature, Convention on Nuclear Safety and other IAEA conference reports, and knowledge gained through many years of bilateral and multilateral co-operation between Hungary and WENRA member countries.

Basic technical characteristics of Paks NPP

<u>Design basis aspects</u>

4. Each unit has a design lifetime of 30 years from first criticality. Operating licenses have no final date of expiration, but are subject to renewal by the regulators every 10 years, based on a periodic safety review.

5. All units in Paks are second generation VVER-440/213 reactors. Generic safety characteristics of this type of plant are discussed in Annex 2.

6. During construction, the quality of the main equipment was controlled by the Hungarian experts. Under the then prevailing political constraints, independent quality verification during manufacturing was not possible to the extent that would have been required in the West. However, no major quality concerns have been identified in tests and inspections carried out since the plant began operating. Also the high reliability of plant operation since its first start-up is an indication of the good quality of the equipment.

7. Since the start up of the plant, many safety improvements have been made and this will continue as a matter of policy throughout the plant lifetime. One of the early improvements was a Hungarian designed core monitoring system (called VERONA). This was installed in 1988, and has since been extended to provide plant operators with information on other important safety parameters.

8. Following a thorough safety evaluation project (called AGNES), a systematic safety enhancement programme was launched in mid-1994. Among the measures already implemented are:

- Relocation of the emergency feed water system outside the turbine building. This removed the major concern about possible complete loss of decay heat removal capability as a consequence of fire or high-energy pipe break in the turbine building,

- Replacement of a number of components to improve the system performance and reliability, and to ensure adequate environmental qualification,
- New systems to improve accident management capabilities,
- Major upgrade of fire protection.

9. In 1996, a review of Paks against IAEA generic safety issues [1] was carried out by the plant staff and IAEA experts, and reached generally favourable conclusions. Although some issues will require continued attention and actions in the future, they are not considered to be significant risk factors today.

<u>Reactor pressure vessel and primary pressure boundary</u>

10. Pressure vessel embrittlement is monitored by an adequate surveillance programme. To date, all vessels have maintained their material toughness with adequate safety margins. Should annealing become necessary in the future, the required technology is available. In-service inspections of the reactor vessels and primary piping are conducted with state-of-the-art techniques. Paks is also taking measures to reduce the possibility of a large primary to secondary leak via the steam generator collector. By these means, it is considered that the integrity of the primary pressure boundary is adequately safeguarded.

<u>Confinement</u>

11. The measured leak rates reflect little variation in construction quality between units. These leak rates are generally smaller compared to other plants of the same type, although they are somewhat higher than leak rates usually associated with Western European reactor containments. The containment internal pressure driving the leak is effectively limited by the bubbler condenser function in the case of design basis accidents, and the overall radioactive releases are not higher than what is accepted within the EU.

12. The performance of the bubbler condenser system in the case of a Large Break LOCA has been verified in full-scope tests in the frame of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA).

<u>Safety systems and hazards</u>

13. In terms of their number, type, and redundancy, the Paks safety systems (diesel generators, emergency core cooling system, emergency feed water system and containment spray system) are comparable to Western reactors of the same vintage. Paks has taken proactive measures to address primary-to-secondary leaks, with modifications currently being implemented, and sump clogging during a LOCA (implemented 1996-1997, and addressing all known related concerns). Hazards including fires, floods, and high-energy pipeline breaks were analysed in the context of the Periodic Safety Review associated with recent license renewal.

<u>I&C systems and emergency power supply</u>

14. All safety-related I&C systems are being upgraded to state-of-the-art digital technology. Reactor protection signals have been modified to provide actuation through two different physical parameters for each initiating event. These signals are processed by different trains and diverse programmes of the protection systems. Emergency power supplies are already up to Western standards.

Beyond design basis accidents and severe accidents

15. The station has studied beyond design basis accidents (such as station blackout, total loss of feedwater, and Anticipated Transients Without Scram, ATWS) with help from the Hungarian research organisations and has developed guidance for operators on how to avoid severe core damage. Severe accident management procedures for Paks are being developed for implementation in the near future. These will be introduced after the completion of new symptom-oriented emergency operating procedures, in parallel with some associated improvements in plant hardware. Additional work is needed to investigate containment response to severe accident phenomena. One Phare project on these issues has been recently completed and another is underway addressing feasibility of filtered containment venting and hydrogen handling. Based on the results of these projects, a comprehensive strategy for managing severe accidents will be developed.

Safety assessments and programmes for further improvements

Safety assessment and documentation

16. The original safety-related documentation supplied with the plant followed vendor practice which then differed from the Western European approaches to safety analyses and reporting. A thorough safety evaluation of Paks was conducted in the AGNES project, which started in late 1991 and was completed by mid-1994. Both deterministic analyses as required in the licensing of Western plants and a level-1 PSA study encompassing internal events, low power and shutdown states, and flooding and fire events have been completed. A seismic PSA is underway. Much attention has been given to appropriate validation of the analysis tools. The results were well documented and were used later to prepare the technical documents of the Periodic Safety Review. They have also been used to update the Final Safety Analysis Report (FSAR). According to regulations issued in 1997 the contents of the FSAR follow the US NRC Regulatory Guide 1.70, accommodating VVER specific features. The first complete version of the FSAR has been submitted to the NSD, and approval is expected by the end of 2000.

17. A periodic safety review (PSR) of all Paks units is required by the current Act and regulations. The first PSR of Paks was initiated by a Ministerial Order issued in 1993. Specifications for that PSR took into account the relevant IAEA safety guidelines. The PSR of units 1 and 2 was completed in 1997, and that for units 3 and 4 was completed by the NPP at the end of 1999. Regulatory review of the PSR is expected to be completed in 2000. The PSR has to be repeated every 10 years. Although one goal of the initial PSR was to update the Final Safety Analysis Report, the current regulations require the licensees to keep the FSAR continuously up to date.

18. A separate project to improve seismic capability has been underway for over five years. This has examined more than 10 000 plant items to determine their vulnerability to earthquakes. A significant part of the necessary modifications has already been done, and the entire project will be completed in 2002.

Programmes for safety improvements

19. According to the best current understanding and available information, no single deficiency representing dominant risk factors remains to be addressed, but a number of measures to further reduce the remaining risk are currently being implemented. These include:

- The addition of equipment to the protection systems and engineered safety features, in order to provide diverse responses to postulated accidents such as large primary-to-secondary circuit leaks, and primary circuit overpressurization,

- Means to protect the reactor containment from phenomena that may occur after severe core damage,
- The development of a new set of emergency operating procedures,
- Condenser replacement to allow high pH secondary water chemistry in order to protect the steam generators from transportation of secondary circuit erosion products.

20. Implementation of these (and other relatively minor) measures has already been scheduled. Less urgent measures are being implemented at a rate of one unit per year, starting from 1999 for unit 1. This programme is expected to be completed by the end of 2002, when all modifications already planned will have been implemented on all units.

21. There are other refurbishment programmes which are part of plant regular maintenance and lifetime management. This mainly involves the replacement of ageing equipment with advanced modern equipment.

Operational safety

Organisation aspects

22. The financial situation of the company is stable. This is demonstrated by the fact that a significant share of annual turnover is regularly spent on investments in safety and reliable operation. Although the management of the Paks plant is today characterised by a strong commitment to safety and reliability, there is some concern that political changes in the Government tend also to induce changes in the station management. Examples of this have been seen in the past ten years.

23. Since the start of operation, the operating company has developed its competence with one aim of attaining independence from the original Russian suppliers. Today they are in a situation where the involvement of the Russian supplier organisation and its successors is no longer a necessity, but is one option in an open bidding process.

24. A positive indication of safety culture found at Paks is the extensive investment in local training facilities. A full scope plant-specific training simulator has been in use since 1988. A recent development is a maintenance-training centre where activities can be exercised before carrying out inspection and/or maintenance on real plant equipment.

25. The drive for increased safety and quality of operations is exemplified by extensive international co-operation. The plant has actively sought contacts with other utility organisations through WANO and especially with other VVER operators. A WANO team was invited to make a peer review of operating practices in 1992, with a follow-up mission in 1995. Since 1988, the plant has also received several safety missions offered by the IAEA, such as OSART, ASSET, and an IAEA safety improvement review. Both WANO and IAEA missions have made recommendations for improvements to operational safety, and these have been given due attention.

Safety culture and management, quality assurance

26. Close contacts are being maintained with European expert organisations and companies operating in the nuclear field. Support has also been received from the IAEA in the development of the plant safety culture. In addition, the plant has developed a QA system based on local regulations derived from IAEA codes and guidelines.

Operational experience

27. The reliability of plant operation and the low frequency of transient events places Paks at

the higher end of the performance ratings for the world's NPPs. This is evident from the small number of unplanned scrams and other operational events throughout the history of the plant.

<u>Emergency preparedness</u>

28. The level of emergency preparedness at Paks is comparable with that at plants in Western European countries, as demonstrated in OECD INEX-2 international exercise, which was based on an accident scenario at Paks.

National industry infrastructure for technical support

29. The VEIKI and KFKI institutes have several decades of high quality experience in fundamental safety-related research, including both reactor physics and system thermal-hydraulics. This ensures a sound domestic competence base and a strong technical support capability.

On-site spent fuel and waste management

30. Spent fuel from the early years of Paks operation has been transported to the Mayak reprocessing facility in Russia. These shipments ceased in 1995 and therefore domestic storage became necessary. Already before 1995, when the possibility to ship spent fuel to Russia became uncertain, and free storage capacity in the spent fuel pools was running low, the Paks NPP awarded a contract for the construction of a modular vault type dry storage (MVDS) system at the NPP's site. The licence for its construction was issued in February 1995 and the licence for commissioning of the first phase (3 modules, 450 assemblies each) of the project was issued in February 1997. Seven modules are now in operation, and construction of next four modules has started. These eleven modules will be able to handle up to 10 years accumulation of spent fuel from all four units, and extension of the MVDS can be made stepwise when the need for more space arises. The facility is designed for interim storage for a period of 50 years.

31. The interim storage facility for spent fuel is being administered by a special organisation designated by the Government, called Public Agency for Radioactive Waste Management (PURAM). The operating personnel are contracted from the NPP.

Conclusions

32. The following conclusions can be drawn:

- The basic technical structure of the plant is good from the safety point of view, and the key safety systems are comparable to Western plants of the same vintage. No major shortcomings in the present safety systems have been identified in any of the several, independent, in-depth assessments done so far. Also the performance of the bubbler condenser containment in case of large break LOCA has been verified in full-scope tests. There is still need for detailed analysis of the experimental results and for complementary tests of other design basis accidents (steam line break and small LOCAs),
- Paks containment structures provide adequate protection against design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU. However their leak rates are somewhat higher than those typical of Western European reactor containments,
- Operational safety aspects are generally at a level comparable to Western plants of the same

vintage. Management changes related to the political changes in the Government cause some concern,

- Periodic safety reviews are conducted in line with Western practices and have already led to an increase in safety.

References

1. IAEA-EBP-VVER-03, Safety issues and their ranking for VVER-440/213 nuclear power plants, April 1996.

LITHUANIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. Lithuania has established the basic laws and regulations related to nuclear safety. The Law on Nuclear Energy from 1996 contains general provisions about licensing, design, operation and decommissioning of nuclear facilities, export and import of nuclear materials, transportation, physical protection, accident management, civil liability, financing, labour relations and international relations concerning nuclear energy. The Law defines a licensing system under which the responsibility for safety is assigned to the licensee. The regulatory body, VATESI, is authorised to issue the licences. The development of the licensing procedure has been completed with international support and the system was successfully implemented in the licensing of Ignalina NPP (INPP) unit 1 in July 1999.

2. The Law on Nuclear Energy also requires the licensing of organisations delivering services and equipment to nuclear facilities.

3. In May 2000 a law on decommissioning of INPP unit 1 was approved by the Lithuanian parliament. The law prescribes that a programme and a plan for decommissioning should be prepared and that all the necessary preparatory measures shall be taken before January 1, 2005.

4. Besides VATESI, the Law on Nuclear Energy establishes the responsibilities of other governmental organisations with respect to the licensing of nuclear related activities. Under the Law, practical work arrangements need to be developed between the different organisations involved in licensing. A reasonable practice for co-ordination between these bodies was created in the mentioned licensing of unit 1.

5. The INPP, the only nuclear power plant in Lithuania, is owned by the state as represented by the Ministry of Economy. At present the operating organisation is not authorised to handle all management issues. For instance, the financing is controlled by the Ministry. In practice this means that the operating organisation cannot assume the full responsibility for safety. This shortfall in the legal status of INPP has been under discussion for several years without any resolution.

6. Lithuania has acceded to all key international conventions related to nuclear safety.

7. It can be concluded that the legislative system in general is in line with Western European practice. However, to be fully comparable with Western European, practice the formal licensing of vendors needs to be abandoned and the nuclear utility needs to be given a corporate legal status.

Status of the regulatory body and technical support infrastructure

8. The regulatory body, VATESI, was established in 1991 from a few specialists from the INPP organisation and the small site inspection group of USSR Gospromatomnadzor. VATESI’s responsibilities and authorities are described in its statute and in the Law on

Nuclear Energy.

9. VATESI is advised by a Board appointed by the government. The Head of VATESI reports directly to the Prime Minister on regulatory matters. Consequently VATESI is independent from that part of the state (Ministry of Economy) which is responsible for the ownership of INPP.

10. VATESI is financed through the state budget. As a consequence of the present difficult economical situation in Lithuania, VATESI, as well as other governmental organisations, suffered budget reductions in 1999 and 2000. In 2000 salaries were reduced on average by 16%. At present this restricts VATESI's possibilities to further develop the organisation, to use external expert advice and to participate in international activities.

11. VATESI has a staff of 29 in its head office in Vilnius and a further 5 persons in its resident supervision group at Ignalina NPP. 19 of these are technical experts.

12. According to its statute, VATESI has enforcement powers to withdraw the INPP operating permit for safety reasons and to impose penalties on INPP staff in cases of violation of safety rules.

13. Although in general VATESI staff are competent, more technical staff are needed to handle adequately all normal regulatory tasks, as well as to develop internal procedures and to deal with the new task of decommissioning. The government has approved a plan to increase the number of VATESI's staff. However, due to budget restrictions, new recruitment has been stopped temporarily. The salary level, although significantly lower than at INPP for corresponding work but higher than for other governmental institutions, has been good enough to recruit qualified new staff, and staff turnover is low.

14. VATESI is now in a position to develop and implement an internal Quality Management system. Such a project has been defined and is underway with international support. However there is some concern that VATESI at present is unable to cope with this development work due to its limited staff resources. Important issues to be addressed in the Quality Management system are new integrated inspection procedures, development of regulations, procedures for safety assessment, documentation management, activity planning and human resource planning.

15. The national expertise that is available to VATESI from Technical Safety Organisations (TSO) has increased in number and competence during the recent years. This expertise comes mainly from the Lithuanian Energy Institute in Kaunas but also from the technical universities in Vilnius and Kaunas and from other organisations. A special TSO Council co-ordinates activities, and considers, among other things, whether a TSO involved in any specific matter is sufficiently competent and independent from the interests of the nuclear operator. The licensing process of Ignalina unit 1 engaged these TSOs in a broad co-operation with Western TSOs. This has been of great value for the transfer of Western methods and practices to Lithuania. However the national TSO resources cannot yet be regarded as sufficient to support VATESI. For instance, no national competence is available for supporting VATESI in human factors assessments.

16. VATESI has some access to research results through the universities and the Lithuanian Energy Institute and through international bilateral contacts. International contacts are very important for Lithuania, as they are for other small nuclear countries, and need to be

strengthened.

17. It can be concluded that the resources of VATESI need to be strengthened in order to carry out its regulatory duties.

18. A reorganisation of governmental institutions reporting directly to the Prime Minister is under way in Lithuania. There is a plan to include VATESI into the regulatory sphere of the Ministry of Environment. In this reorganisation special attention needs to be given to the managerial and financial independence of VATESI.

Status of regulatory activities

19. At an early stage, VATESI introduced a system of annual permits for the operation of INPP. This practice has enabled VATESI to exercise strict regulatory control of the plant. In 1999 the INPP unit 1 was licensed in compliance with the Act on Nuclear Energy. The licence is valid until July 2004 and contains a number of conditions. The licensing was a major effort for VATESI and unique for RBMKs, with respect to the scope of safety analysis and regulatory review carried out with international co-operation. This review was done according to Western practice. The work is continuing now with the follow up of the implementation of licensing conditions and the corresponding licensing of unit 2. This licence is planned to be issued in the end of 2002. VATESI has learnt considerably from this process which has contributed to its development as a competent regulatory body.

20. To date VATESI has issued a number of licences for Lithuanian and foreign suppliers to INPP. It should be mentioned that VATESI only makes a general assessment of the Quality Management and the competence of the vendor, and it is the responsibility of the operator to make a more detailed assessment before a contract is signed.

21. During recent years VATESI has developed new regulations, allowing the resident inspector group to apply a more system oriented inspection methodology, in order not to be involved as much in plant daily activities. This development continues and is estimated to require a few more years before it is fully implemented. The end result is expected to be a clearer separation of activities between VATESI and the operator regarding the safety of INPP.

22. VATESI has an internal commission of experts that reviews on a regular basis event reports from INPP and makes recommendations on regulatory measures.

23. The decision taken to close INPP unit 1 before 2005 will require a number of actions from VATESI. The planning of these is underway. A most important task is to make sure that safety is not compromised during the last operating years. This includes technical issues as well as organisational and safety management issues.

Emergency preparedness on governmental side

24. Lithuania has adopted a national emergency preparedness plan that has been internationally reviewed. In the case of severe national emergencies, a Crisis Commission is established at governmental level for co-ordination of all rescue activities. The head of VATESI is a member of the Commission.

25. In case of an accident at INPP, the role of VATESI is to give advice to the national rescue

authorities and to supervise the accident management at INPP without taking part in the operational decisions. VATESI has developed and exercised its own emergency preparedness plan. There is a 24-h cover for on-duty and decision making. Equipping of an Emergency Operating Centre is planned with Phare support. VATESI, as well as other responsible governmental organisations, have reviewed and approved the Emergency Response Plan of INPP.

26. Lithuania participates in the IAEA Emergency Preparedness Harmonisation Project and has also participated in several INEX exercises.

Conclusions

27. The legal and regulatory system has developed substantially over the last years. A licensing system is in place and VATESI has developed its approaches to safety assessment and inspection. Further efforts are, however, needed in order to be comparable with Western European practice.

28. The following issues need to be considered by the Lithuanian government:

- The legal status of INPP needs to be changed in such a way that the operating organisation is given the full responsibility and authority to handle all financial and other management issues and thus be able to assume the full responsibility for safety,
- The full responsibility to select and assess suppliers to nuclear facilities should rest with the operating organisation; hence the legal obligation of VATESI to formally license suppliers needs to be changed, given a reasonable transition period,
- The imposed reduction of resources to VATESI in terms of staff and budget needs to be compensated as soon as possible and the resources successively strengthened in order to handle all normal regulatory tasks, the contracting of necessary technical support and the full participation in international regulatory co-operation,
- The technical support structure and access to nuclear safety research should be further strengthened in order to provide VATESI with the necessary competence to review all major safety issues,
- In the ongoing reorganisation of governmental institutions reporting directly to the Prime Minister, special attention needs to be given the independence of VATESI.

29. The following issues need to be considered by VATESI:

- The development of the internal Quality Management system needs to be given a high priority. In this development work the final step needs to be taken towards an integrated regulatory supervision of INPP, clearly separating the roles of the regulatory body and the operator in all activities. This particularly applies to the role of the resident supervision group,
- Lessons learned from the licensing of unit 1 need to be incorporated in safety assessment and licensing procedures in order to strengthen the independent integrated assessment capability of the regulatory body.

Chapter 2: Nuclear power plant safety status

Data

1. Lithuania has one nuclear power plant with two units in operation at Ignalina (INPP). These are of a later RBMK design and have the highest power rating of any RBMK reactor:

Unit	Type	Present power level		Start of construction	First grid connection	End of design life
		MWth	MWe			
INPP-1	RBMK 1500	4200	1300	1977	12/1983	2013
INPP-2	RBMK 1500	4200	1300	1978	08/1987	2017

2. The plant is owned and operated by the state. The Lithuanian parliament has recently confirmed a governmental decision to shut down unit 1 before 2005.

3. The Information in this chapter is based on first hand knowledge gained by Western TSOs during their participation in the EBRD funded safety review of the Ignalina 1, in bilateral co-operation programmes, in Phare projects and through the IAEA Extrabudgetary Programme on RBMK reactors.

Basic technical characteristics

<u>Design basis aspects</u>

4. After the Chernobyl accident in 1986, design modifications were made at the INPP as well as at other RBMK reactors. These modifications were aimed at reducing the positive void coefficient, improvement of the reactor protection system and display of the reactivity margin (see Annex 1). Compared to other RBMK reactors of the same design generation, INPP has some additional safety features, for instance the following are included in the original design basis:

- A break of the largest pipe in the primary circuit (900 mm header) and,
- Well-diversified emergency core cooling systems.

The design basis for emergency core cooling is comparable to Western plants of the same vintage with regard to loss of coolant accidents (LOCA) and operational transients. Anticipated Transients Without Scram (ATWS) and events such as fires, station blackout and seismic events were not, however, fully covered in the original design basis. With respect to station blackout -due to the large water inventory, the large heat capacity of the graphite and the relatively low power density- the grace period is about four times longer compared to typical Western light water reactors. Hence, there is more time available at INPP for accident management measures. Recent investigations by Western experts indicate that the seismic risk might be lower than earlier considered.

5. Backfitting has improved the original design on several important points:

- The reactor cavity venting system of each reactor is now capable of tolerating 9 simultaneous fuel channel breaks, without any damage to the cavity which could result in significant radiological releases,
- A new system (DAZ-system) has been installed in unit 1, protecting the reactor against the

most frequent and most severe ATWS events, e.g. loss of preferred power, loss of main heat sink, etc. The DAZ-system will also be installed in unit 2 during the maintenance period in 2000. To remove the ATWS issue as a safety concern, a new diverse shutdown system (DSS) is planned to be installed in unit 2 and brought into operation in 2003. This has not been considered for unit 1 because of the limited remaining operating time,

- Additional signals for scram have been installed in both units (low flow through group distribution headers, low operational reactivity margin (ORM), fast pressure decrease in drum separators, high temperature in reactor protection cabinets). Worn out accumulator batteries have been replaced with Western equipment,
- The control room of unit 1 has been upgraded with a new process computer, which also contains a Safety Parameter Display System with 3D neutronics calculations. For unit 2 these modifications are planned to be completed in 2002,
- The fire safety has been much improved in both units by new sprinklers, detectors, cable coating, fire doors, fire ventilation and removal of combustible material.

Status of fuel channel pressure tubes

6. The top welds of the pressure tubes have been thoroughly examined and no safety significant deficiencies have been found. All the pressure tubes have been fitted with new seals. New equipment has been delivered and its use has been validated for more accurate ultrasonic measuring of the tubes and the associated gas gap. New equipment for visual inspection of the inner surface of the pressure tubes has also been delivered and used. Together with the earlier equipment, it is now possible to get complete information about the status of the tubes and the graphite and, as a consequence, a good possibility of assessing the remaining lifetime. Irradiated parts of one tube have also been examined in Sweden and the results were used in the licensing of unit 1. The examined parts were in good material condition with regard to hydration and possible embrittlement. These results have been confirmed by an international peer-review. Further investigations will be made of irradiated parts from unit 2.

Material verification

7. The INPP units have suffered from material defects and leakages in the primary circuit piping, although this has been less than at other RBMK units. The material problems and degradation mechanisms are of the same types that have been found in Western BWRs, especially intergranular stress corrosion cracking in certain piping. Since 1992, the primary circuit has been examined with modern methods and equipment. There is now a good, and for RBMK unique, knowledge about the material status of the large diameter pipes of the primary circuit. The work continues with more refined analysis of crack characteristics and upgrade of the old, as well as installation of new, leak detection systems in unit 2. A new IAEA extrabudgetary programme will address these issues for all RBMK plants with INPP as the reference. Most of the work on non-destructive testing (NDT) is now done by INPP's own staff who are certified according to European Standards (EN 473). Investigations have started regarding implementation of risk based NDT-inspection in unit 2.

Status and capabilities of safety systems

8. INPP has a high redundancy of front line engineered safety systems. In the original design, as in most older Soviet designs, inadequate physical and functional separation made some systems vulnerable to area events and common cause failures. At INPP the fire protection has been improved to protect vital electrical systems, control and protection systems, and the emergency core cooling pumps. Based on a fire hazard analysis, further improvements will be implemented in 2000. Important dependencies in the support systems, e.g. the

service water system, have been identified by PSA and modifications have been implemented. Environmental qualification of the safety-related components needs further consideration. Development and implementation of a system for environmental qualification is planned to be completed in 2002 following recent VATESI regulations on Ageing Management.

Reactivity control

9. As mentioned in § 5, additional protection signals have been installed in order to compensate for deficiencies in the earlier Control and Protection System (CPS) design. As also mentioned in § 5, it has been decided to install a new diverse shut down system in unit 2. This is expected to be financed by the EU. The new system (DSS) is planned to be in operation in 2003. The tender request specification is currently under international review and tendering is expected soon. The system specification includes control rods plus a liquid hold down system, two different sets of instrumentation, trip logic and different clutch mechanisms. The specific design and installation will be proposed by the tendering organisations. In order to address the generic risk of power increase after potential loss of coolant from the CPS channels, part of the existing manual control rods will be replaced with rods of the cluster type. In the new rods the space occupied by coolant is small, representing less potential reactivity insertion. Since 1995, a new type of fuel with higher enrichment and burnable absorber has been loaded into both units, improving fuel economy, facilitating reactor operation and having improved safety characteristics. This fuel together with the new scram signal (mentioned in § 5) makes it much easier to keep the operational reactivity margin within the safety limits.

Status and capabilities of safety systems

10. A pressure suppression type of partial confinement called the Accident Localisation System (ALS) protects the reactor and part of the primary circuit. This system has the following features:

- About 65% of the water volume of the primary circuit, the parts situated below the top of the core, is enclosed within the confinement,
- The confinement is made up of a large number of semi-interconnected reinforced concrete compartments,
- Condensation of steam occurs in ten water pools which are separated in two groups of five,
- Spray nozzles for steam condensation are installed in several compartments,
- A controlled venting system serves to reduce both the peak and the long-term pressure.

11. The design basis for the ALS is the isolation of the steam resulting from a LOCA after the largest pipe break. Hence, the condensation capacity of the INPP ALS system is larger than in other RBMK units.

12. Recent thermal-hydraulic and structural mechanics calculations show that the maximum pressure in the ALS for the ultimate design basis accident remains more than 0.1 MPa below the design pressure and that with the upper bound pressure from calculations the structural integrity of the ALS is not endangered. However, some aspects of the performance of the ALS remain to be addressed in order to verify design basis events at a level of Western European practice. Several studies are included in the ongoing safety programme, with the objective of finalising by 2001 at the latest. The leak-tightness and structural stability of the ALS was a licensing issue for unit 1. The NDT investigations made on the as-built structure indicated no serious deviations from the design

specifications. The strength analysis is planned for completion in 2000. However, the measured leak rate is much higher than observed in Western plants of the same vintage, significantly higher in unit 1 than in unit 2. Unit 2 has steel lining inside the ALS. INPP continues to enhance the leak-tightness of the confinements.

<u>Beyond design basis accidents</u>

13. The capacity of ALS to handle a core damage or an accident state is being analysed in an ongoing level-2 PSA. Several scenarios are identified in the analysis where there is a risk of the ALS bypassing through structural leakage or failing structures. More deterministic analysis is needed in order to make reliable conclusions. The reactor cavity, which is one part of the confinement system, has a relatively low design pressure and, moreover, its hypothetical failure could lead to pressure tube ruptures, consequential failure of the fuel and a release of radioactivity through the bypass. The probability of this pressure being exceeded is however rather low taking into account the newly installed additional protection signals and the increased relief capacity of the reactor cavity.

<u>Ageing and lifetime assessments</u>

14. The closure of the gas gap between pressure tubes and graphite blocks (see Annex 1) is an ageing issue with implications for the lifetime of the fuel channels. During the recent years several hundred pressure tubes of unit 1 have been measured during outages using Western designed equipment. Some 70 tubes have been removed for precise determination of the gap width and for further analysis of the tube material and the channel graphite. Based on present knowledge of the pressure tube and graphite behaviour, the date for gap closure at unit 1 is uncertain but is estimated not to happen before 2002. VATESI monitors the situation and controls operation with annual permits. VATESI has declared that operation will be stopped as soon as gap closure is confirmed. The empirical data basis for this decision will be significantly enlarged by the new gap measurements obtained by the removal of 100 pressure tubes during the maintenance period of unit 1 in year 2000. This will provide a much more accurate lifetime projection than was previously available. In a 1994 agreement with the EBRD, the Lithuanian government declared that re-tubing of the INPP units would not be performed.

Safety assessments and programmes for further improvements

<u>Safety assessments and documentation</u>

15. The International RBMK Safety Review 1992-94 used INPP unit 2 as one of its reference plants. This review considered the safety of RBMK reactors in nine technical areas and resulted in about 300 recommendations and extensive documentation.

16. Under an agreement with EBRD, unit 1 was subjected to a comprehensive safety assessment in 1994-96. It consisted of the production of a safety analysis report (SAR) and its independent review (RSR). The safety assessment was made to Russian regulations reviewed and amended for Lithuania, IAEA codes and guides and equivalent Western standards, but considered only a limited remaining operating time before gas-gap closure was expected. It was also limited due to financial and time restraints. The SAR was reviewed by an independent international team. The Ignalina Safety Panel, an independent group of senior experts, evaluated the findings and drew conclusions based on both the SAR and the RSR.

17. A number of deterministic analyses were carried out in the licensing of unit 1 in order to supplement the SAR, e.g. accident analyses (missing in the SAR), a fire hazard analysis, a single failure analysis of the Control and Protection System (CPS), safety cases for ALS and

the structural integrity of the primary circuit. Some of these analyses will be further developed in the safety improvement programme for 2000-2005. The licensing process for unit 2 has recently started. In this work, a new specific and extended SAR will be developed and reviewed for unit 2.

18. A PSA level-1 covering full power operation has been available for unit 2 since 1994. The quantitative results are based partly on plant specific data and partly on generic data. Limitations exist in the modelling of external events and dynamic effects of LOCA, as well as in the modelling of Common Cause Failures and Human Performance. Based on this study, a level-2 study is being prepared and expected to be completed by the end of 2000.

    An updated version (phase 5) of the PSA level-1 study was recently reviewed by an IAEA IPSART mission. As a result of the mission the study will be further improved.

<u>Programmes for safety improvements</u>

19. Following the safety review recommendations the Lithuanian Government committed itself to financing a new safety improvement programme (SIP-2). The programme contains design modifications, management and organisational development and safety analyses. A first part of the SIP-2 is now finished. According to INPP, 118 of 160 planned activities were implemented by the end of 1999 including 12 specific activities for unit 1. A second part of SIP-2 is now defined for implementation during 2000-2005, which addresses the remaining issues from the safety reviews, issues specifically directed at unit 2 and high priority issues as a result of decommissioning of unit 1. The SIP-2 programme has so far suffered from financial difficulties and there is a concern that it could be delayed.

Operational safety

<u>Organisation, procedures, operation and maintenance</u>

20. INPP is a state enterprise under the Ministry of Economy. The plant Director General reports directly to the Minister.

21. A major problem for INPP is the financial difficulties caused by limited remuneration for the electricity production. The problem of limited electricity demand and insufficient payment for deliveries has been significant during recent years. The payments are just about sufficient to keep the plant operating with regard to staff salaries and operational expenses. For 1999 INPP only received a small part of the planned funding for the SIP-2 programme and there is a concern that this will also happen in 2000. In that case, INPP will face further financial problems and some planned improvement measures will need to be postponed. In any case, the complete safety improvement programme cannot be financed without foreign support.

22. As a whole, the operational and technical support staff shows a high level of technical competence. A significant transfer of Western knowledge has taken place in the last years as a result of extensive international co-operation programmes. As a result of the decision on decommissioning of unit 1, INPP has prepared a programme to reduce the number of employees mainly by outsourcing activities, like district heating, transportation, maintenance, etc. Recently the number of employees was reduced from 5 000 to 4 800. A few plant specialists have left INPP, mostly for work abroad. There is a concern that more specialists will leave as soon as other opportunities arise.

23. The operating procedures have been improved as recommended in the SAR review. Further development of the operating procedures may result from the use of the new full-

scale simulator. New symptom based Emergency Operating Procedures (EOPs), meeting Western standards, are planned to be implemented by the end of 2000. To prepare for the training of the shift teams, INPP instructors have received training in modern control room work management and methods. The new full-scope simulator will considerably contribute to a successful implementation of the EOPs.

24. The Technical Specifications document addresses the necessary operational limits and conditions although the format differs from Western practice.

25. A new computerised maintenance management system is being implemented. There is also a new procedure and computer tools for the handling of plant modification drawings according to Western standards.

26. The training system is undergoing modernisation according to the IAEA Systematic Approach-to-Training model. Training of control room operators on the new full-scale simulator began in late 1998.

Safety culture and management, quality assurance

27. INPP has been subjected to one OSART, two ASSET missions, and a number of other IAEA activities. With the support of Western experts, considerable efforts have been made since 1994 to develop management, organisation, and safety culture at INPP. Several moves towards Western practice have been made for instance the establishment of a Plant Safety Committee. However, it has been difficult for the committee to be fully accepted within the INPP organisation. A new Quality Management System based on IAEA standards has been implemented during 2000 after four years of work. To achieve deeper changes to the old safety culture is proving to be a slow process and is also dependent on changes in Lithuanian laws and regulations. The difficult economical situation and the decision on decommissioning present further challenges to the management of INPP in the development of the safety culture.

Operating experience

28. The operating history of INPP shows decreasing trends for all events categories except for leaks in the primary circuit. Up to 1990, the collective dose of INPP staff was comparable to the world average. From 1990 the dose has increased a little, mainly due to the extensive safety upgrading works. The main categories of events during the 1990s have been equipment failures, leakage in the primary circuit and Control and Protection System problems. There was also a serious bomb threat in 1994, which led to an extensive project with Western support to upgrade the physical protection of the plant. Recent statistics show a decrease in the number of more serious events during the 1990-ties and a slight increase of number of minor events. Of 81 reported events in 1999, 68% were attributed to equipment faults, 17% personal errors, and 15% to deficiencies in procedures.

Analysis and feedback of operational experience

29. Procedures exist for the analysis of events and operational experience feedback. However the communication between the different plant departments and the internal experience feedback, needs to be improved. Regulations require the reporting of ”abnormal events” to VATESI. Event reporting is well implemented. There is an exchange of operating experience between INPP and the other RBMK plants through reports, at technical meetings and telephone conferences.

Emergency preparedness

30. The earlier on-site Emergency Response Plan has been thoroughly reviewed and modified in line with Western standards. Accident classification and alarm criteria have been developed according to IAEA's RBMK guidelines, and following a final review they will be included in the plan. The new plan was exercised for the first time in 1998 and was exercised again in October 1999 using the completely reconstructed Emergency Operating Centre. The plan is now under revision again to accommodate the experience gained.

National industry infrastructure for technical support

31. Prior to taking responsibility for INPP in 1991, Lithuania had little involvement in nuclear activities. Therefore, the country lacks deeper nuclear experience and tradition. The national technical support infrastructure for INPP is improving but will not be sufficient in the near future. In particular, further Western assistance and Russian consultation will be needed for qualified engineering work.

On-site spent fuel and waste management

32. A new interim dry storage facility for 72 spent fuel casks has been built close to the site and recently started operating. A safety assessment of the present facilities for storage of solid and bitumenised waste is going on. Further measures regarding nuclear waste management are included in the present safety improvement programme and have received a high priority as a result of the decommissioning decision.

Conclusions

Design issues

33. INPP belongs to the more advanced and improved design generation of RBMK reactors. In addition, the original design has been considerably improved through the safety improvement programmes and most of the generic safety concerns with RBMK reactors have been satisfactorily addressed. More measures will be implemented as a consequence of, for instance, the installation of a new diversified and independent shut down system at unit 2. The safety situation of INPP is also better known internationally and considerably better documented than for other RBMK plants.

34. The measured leak rate of the confinement system is much higher, in particular at unit 1, than observed in Western plants. Some further aspects remain to be addressed in order to reach a complete verification of the confinement performance for the design basis events. However, compared with Western European light water reactors of the same vintage, there remain weaknesses in the design of the confinement, especially in case of a severe accident:

- The reactor cavity, which is one part of the confinement system, has a relatively low design pressure and, moreover, its hypothetical failure could lead to consequential failure of fuel integrity and a release of radioactivity through the bypass. The risk of this pressure being exceeded is however rather low taking into account the increased relief capacity and the newly installed additional protection signals,
- A LOCA in the primary circuit outside the partial confinement could lead to a release of steam that it would not be possible to isolate. Such a potential accident sequence was accepted in the original design, due to its perceived low risk regarding radioactive release to the environment, but is not acceptable according to Western European design principles.

35. It is not realistic to upgrade the confinement to include the complete reactor pressure boundary. Consequently, regarding mitigation of accidents, a safety level comparable to light water reactors of the same vintage in operation in Western Europe will not be reached. Therefore special attention must be given to prevention of accidents, including the need to ensure a high level of operational safety for the remaining operating time.

Operational safety

36. With regard to the operational safety, the following issues need to be resolved in order to be comparable with Western European plants:

- The financial situation of INPP needs to be much improved in order to cover all operational expenses as well as implementing the safety improvement measures considered necessary for the remaining operating time,
- Issues relating to safety culture need a stronger implementation in order to prioritise safety in all organisational levels, not least after the decision to shut down unit 1,
- The symptom based emergency operating procedures need to be finalised and implemented without delay,
- Due to the decision on decommissioning of unit 1, special attention needs to be given to the keeping of a sufficient number of technical specialists, as well as maintaining the motivation of the staff, for the remaining operating time of both reactors.

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ROMANIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. Legislation regulating the peaceful use of nuclear energy has existed in Romania since 1974. The current nuclear Law, in force since December 1996, defines areas of application together with the roles, duties and responsibilities of organisations involved in the licensing process. A 1998 amendment to the Law was issued to remove inconsistencies with other national Laws and to strengthen the status and the role of the Regulatory Body (National Commission for Nuclear Activities Control - CNCAN).

2. The Law clearly assigns responsibility for the safe operation of NPPs to the operator. According to the Law, CNCAN is responsible for the regulation, licensing, and control of all nuclear facilities in Romania. The legislation requires CNCAN to license not only the operator but also its subcontractors with regard to quality assurance. This approach has the potential to obscure the operator's perception of its primary responsibility for safety at the plant. There is however a plan to revise this law in the near future in order to take advantage from the lessons learned after a few years of its application.

3. The company that operates the Cernavoda NPP is the National Nuclear Company "Nuclearelectrica". It is a state company reporting to the Ministry of Industry and Trade.

4. Romania has ratified the key international conventions dealing with nuclear safety.

5. The nuclear legislative framework in Romania is generally in line with Western European Practice, even if some specific improvements are still necessary (see § 2).

Status of the regulatory body and technical support infrastructure

6. CNCAN is led by a President with the rank of State Secretary, nominated by the Prime Minister and reporting directly to the Government. CNCAN is independent from ministries and organisations that have a role in the use and promotion of nuclear energy. However, some cases of CNCAN involvement in the selection process of licensee suppliers have been noted.

7. CNCAN's organisation and staff composition has undergone several changes. The organisational structure, approved by the Government in 1998, was implemented in spring 2000. An Advisory Committee is also envisaged but this is not yet operational. A licensing board is in place to support the President in the licensing decision process. Regulatory activities relating to the licensing of the nuclear power plant and research reactors are performed by the Nuclear Safety General Division. This currently comprises 31 experts but a significant recruitment programme aimed at hiring qualified experts or new graduates was implemented last year. CNCAN has taken over the function of monitoring the national environmental radioactivity and 174 people are involved in this activity. However, due to their different professional background, these personnel cannot be used for nuclear licensing activities.

8. CNCAN is funded by the Government and, in addition, it is also allowed to retain 50% of licensing fees paid by the applicants. In the last two years priority has been given to assign most of the budget to significantly improve the headquarters infrastructure and to purchase new equipment. By comparison, the development of the technical competencies of staff and the provision of external technical support for the assessment of key regulatory issues might not have received sufficient financial resources.

9. Personnel salaries have been recently increased and this should lead to a reduction in the turnover of qualified staff.

10. In the field of NPP licensing activities, CNCAN has a limited number of senior qualified experts. There is also a limited number of experienced inspectors. Most of the recently hired CNCAN personnel need to be trained in the safety and operational features of CANDU reactors, regulatory methodology and practice, and in inspection practice. The experience gained during the licensing of Unit 1 also needs to be transferred to the new staff.

11. CNCAN has the power to issue, amend, and revoke licences but the licensee can appeal against CNCAN decisions in the Romanian courts. CNCAN has the necessary enforcement power to carry out inspections on the site and to enforce remedial actions when violations are identified.

12. Progress in the further development of regulatory management needs to continue. This should include the finalisation of a Quality Assurance manual that is currently under development using, as reference, the working procedures of the Canadian nuclear safety authority.

13. In addition to insufficient internal technical assessment capabilities, CNCAN currently does not have qualified external technical support for all safety aspects. At present some support in specific areas is provided by the Canadian nuclear safety authority, the IAEA, and private consulting organisations operating in the country. CNCAN is also investigating the possibility to develop a national TSO.

14. Progress has been made by the Romanian Regulatory Body in the recent years and in particular during the licensing process of the Unit 1 of the Cernavoda NPP. Further improvements are however necessary to reach a status comparable to Western European practice.

Status of regulatory activities

15. CNCAN is currently revising the Romanian regulations to make them consistent with the new nuclear law and to bring them into line with Western European practice.

16. According to the nuclear Law, any activity related to a nuclear installation must be licensed by CNCAN. Since April 1999 the Cernavoda plant has been operating based on a two years licence. At present the major regulatory activity is related to the definition and the implementation of a strategy for the extension of the Cernavoda licence after May 2001.

17. As required by the law, CNCAN is involved in the licensing of suppliers to the licensee. Pending amendment of the law, CNCAN has started to develop an inspection practice of the licensee's QA programme, including the control of suppliers. This is more in line with

Western European practice.

18. The Romanian licensing practice has been developed during the construction, commissioning and initial operational phases of Cernavoda Unit 1 and is defined in a regulation under revision. The basic format of the safety documentation is the requirements of US NRC Regulatory Guide 1.70, with specific provisions derived from the Canadian practice. For the licensing activities of the Unit 1 of the Cernavoda NPP CNCAN nominated a licensing manager. Technical advice was also provided by the Canadian nuclear safety authority throughout the licensing process by the presence of a Canadian expert on the site. The CNCAN assessment activities were addressed towards verifying the compliance of the plant design basis with applicable regulations and were mainly based on engineering judgement supported by a few independent analyses. IAEA expert missions also took place to provide additional support for CNCAN decisions.

19. CNCAN has established event-reporting requirements for the licensee and has developed an internal system for the assessment of the plant operating experience. CNCAN is also actively participating in international event reporting systems.

20. At present the CNCAN independent assessment capability is limited because there are insufficient experienced staff members. Some support comes from the Canadian Nuclear Safety Authority, from IAEA national and regional projects, and in the framework of the CANDU Regulators group of which CNCAN is member. A first year of regulatory assistance has been provided under the EU Phare programme.

21. CNCAN site inspection practice was initially developed based on the Canadian approach. It has been improved through the advisory service of the Canadian nuclear safety authority and also through international missions organised by the IAEA. Inspection during construction and commissioning was based on an on-site inspection unit for day to day activities, plus team inspections from the headquarters to address specific areas. Since the start up of the Cernavoda Unit 1, site inspections have been performed mainly by headquarters personnel. It is now planned to assign a resident inspector to the site mainly acting as a liaison with the CNCAN headquarters. Inspection procedures based on the Canadian practice are under development together with a training programme for the inspectors.

22. CNCAN has made good progress in performing its regulatory activities. Some improvements are however still needed to be in line with the practice of Western European Regulatory Bodies.

Emergency preparedness on governmental side

23. A national emergency plan is in place. An inter-ministerial Committee has the responsibility for the control, evaluation, and approval of the national emergency plan. As a member of this Committee, CNCAN has the role of providing technical support and advice, and notifying the public about nuclear emergencies. CNCAN also has the responsibility for approving the on-site emergency plans of nuclear facilities. Romania has participated in a number of INEX international exercises. At present, however, CNCAN does not have enough experienced personnel in emergency preparedness and a dedicated emergency centre is not available at the CNCAN headquarters. There are however plans to address these issues in the future.

Conclusions

24. Romania is taking appropriate steps to establish a regulatory regime and a regulatory body comparable with Western European practice. Roles, duties, and responsibilities of organisations involved in nuclear safety are in line with those assigned to similar organisations in Western Europe. The independence of the regulatory body from the organisations involved in the use and promotion of nuclear energy is fully established by the law and is sufficiently reflected in the practice. The regulatory regime and the regulatory body have both improved during the licensing process of Cernavoda NPP. However some improvements are necessary to reach a situation comparable with the practice in Western European countries.

25. The following recommendations need to be addressed by the Government:

- Despite the critical economic situation the availability of technical support to the regulatory body need to be improved as well as the salaries of the regulatory body personnel,
- The full responsibility to select and assess suppliers to nuclear facilities should rest with the operating organisation; hence the legal obligation of CNCAN to formally license suppliers with regard to quality assurance needs to be changed, given a reasonable transition period,
- Procedures and lines of communications between national organisations involved in the response to a nuclear emergency should be improved.

26. The following recommendations are addressed to the regulatory body:

- The independent assessment capabilities and the inspection practice need to be improved, especially in view of the future licensing activities of Cernavoda Unit 2. To this purpose adequate resources should be assigned to set up and implement a training programme for new recruits,
- In order to be able to carry out its duties and responsibilities in the area of emergency preparedness, CNCAN needs to further develop by establishing an emergency response centre and a dedicated unit in the organisation,
- The existing agreement with the Canadian regulatory body needs to be more effectively used for training purposes and for obtaining advice on regulatory issues specific to CANDU technology,
- It is recommended that a strategic plan is developed to ensure that appropriate resources are allocated to the higher priority issues,
- The ongoing revision of the regulatory pyramid needs to be continued and completed.

Chapter 2: Nuclear power plant safety status

Data

1. Romania has only one NPP into operation. It is a CANDU 6 reactor located in the Cernavoda site. Construction of five CANDU 6 reactor units started at Cernavoda in 1980 and stopped at different stages of advancement (e.g. 46% for unit 1) following the 1989 political changes. Subsequently, it was decided to concentrate on the completion of the first two units.

2. In 1991, the Romanian Electricity State Company (RENEL) signed a contract with a Western Consortium (AECL and Ansaldo), transferring to it the responsibility to complete the construction of Unit 1 and to commission and manage its initial operation. The national industry participated in the construction of conventional systems under a qualification programme supervised by the Consortium. Operating responsibility was transferred from the Western Consortium to RENEL, with commercial operation starting in July 1997 under the conditions of a provisional operating licence issued by the Regulatory Body.

Reactor	Reactor type	Electrical power (MW)	Start of construction	First grid connection	End of design life
Cernavoda Unit 1	Candu 6	705	1980	1996	2026

3. Work on Unit 2 had stopped with 80% of the civil work and 5% of the mechanical work completed. However, the Romanian Government is now committed to complete the construction of the Unit and the financial means for achieving this are being worked out.

4. The Cernavoda NPP is owned by the National Nuclear Company "Nuclearelectrica". It is a state company reporting to the Ministry of Industry and Trade.

5. Due to the close involvement of the Western Consortium in the construction project, Cernavoda has not benefited from EU industrial assistance programmes. Furthermore, CANDU reactors are not in operation in any Western European Country so their detailed safety characteristics are not known to Western European safety organisations.

6. The statements presented in this report are primarily based on information provided by the Romanian regulator and the utility. In addition the Canadian regulatory body was contacted to collect some information on the design of the CANDU 6 reactor and the evolution of regulatory requirements in Canada.

Basic technical characteristics

<u>Design basis aspects</u>

7. The Cernavoda reactor uses natural uranium as fuel and heavy water as the coolant and moderator. It is based on a standard CANDU 6 design developed in Canada in 1979 and is similar to NPPs in operation at Point Lepreau and Gentilly 2 in Canada.

8. In a CANDU 6 reactor the moderator and the coolant are separated by two concentric tubes, the pressure tube and the calandria tube. The pressure tubes (380) and the calandria tubes are housed in a cylindrical tank (calandria) that contains heavy water moderator at

low pressure, surrounded by a concrete reactor vault containing light water for biological and thermal shielding. The pressure tubes contain the fuel bundles, and the coolant is circulated through these tubes. The calandria tubes prevent the moderator from coming into contact with the high temperature coolant. There is an annulus between the pressure tube and the calandria tube, filled with gas (CO_2). The gas is monitored to provide early detection of tube failure. Refuelling is carried out with the reactor at power. This allows removal of any defective fuel from the core as soon as it is identified, so helping to keep the heat transport system essentially clean from fission product activities.

9. A disadvantage of the CANDU reactor concept is that the reactivity void coefficient is positive. The adverse effects of this during transients and accident conditions are counteracted by two independent, diverse and equally capable shutdown systems.

   Some advantages of the design are:

- Control devices cannot be ejected from the core, being located in the moderator at low pressure,
- There is the possibility of using the moderator as an emergency heat sink following severe loss of coolant scenarios with the emergency core cooling unavailable.

<u>Pressure tubes and primary pressure boundary</u>

10. The integrity of the primary pressure boundary of the Cernavoda NPP is monitored through a periodic inspection programme in accordance with Canadian Standards. The standard is based on the ASME code adapted to take into account the fact that CANDU is a pressure tube type reactor. In particular the methodology developed in the Canadian Standard is based on the inspection of samples chosen based on pre-established criteria. The pressure tubes of the Cernavoda NPP have been manufactured from a new type of material (Zirconium-Niobium 2.5%) which takes account of the lessons learned following a tube rupture at the Pickering 2 NPP in Canada. The Regulatory Body has approved the measures to ensure the integrity of the pressure tubes and of the primary system.

<u>Safety systems</u>

11. Plant systems are divided into process systems (some of which are safety related) and special safety systems. The special safety systems are the two shutdown systems (shut-off rods and liquid poison), the emergency core cooling system, the containment system and the related supporting systems. An unavailability target of 10^{-3} per year is requested for each Special Safety System as design requirement, while a strict application of the "single failure criterion" is not requested. To provide protection against common cause failures, the mitigating systems (Process and Special Safety Systems) have been divided into two independent groups such that at least one group will be capable of shutting down the reactor, cooling the fuel, containing fission products and monitoring the plant. The emergency power supply system comprises 4x50% stand-by diesel generators and 2x100% emergency power supply diesel generators. For the Cernavoda plant additional assessment is necessary to confirm the plant design margins against seismic events and the adequacy of fire protection.

<u>Confinement</u>

12. The containment provides a complete enclosure of the reactor primary circuit. It is a pre-stressed structure with plastic liner, equipped with an automatic spray system to reduce any pressure increase after an accident. The measured leak rate of the containment is comparable with that of reactors in operation in Western Europe.

Postulated design basis accidents

13 The Cernavoda NPP, like other CANDU reactors, is designed against a set of postulated events based on the concept of single / dual failure. Single failure (corresponding to a single initiating event in the terminology used in PWR plants) is a failure of any process system that is required for the normal operation of the plant. In this category there are events like large LOCA, single channel events, small LOCA, etc. Dual failure (corresponding to single initiating events with associated a degraded performance of engineered safety features in PWR reactors) is a combination of a single failure event described above and the simultaneous failure or impairment of one of the special safety systems (emergency core cooling or containment). For the single failure and the dual failure categories of events, maximum frequencies and reference dose limits for members of the public are established. Quantitative engineering limits (e.g. peak cladding temperature in case of LOCA) are not fixed at regulatory level. This is in line with the Canadian approach in which it is the licensee's responsibility to develop the general performance standards, established by the regulator, into more detailed design requirements. Conservative assumptions are adopted in the plant transients and accident analysis. Plant design bases include external events such as earthquake, flooding, missiles, and for the containment a reference aircraft impact.

Beyond design basis accidents and severe accidents

14 Beyond design basis events like Anticipated Transients Without Scram and Station Blackout are not analysed in the CANDU safety analysis. These types of scenarios are assumed to be prevented by the existing design safety features (two independent diversified and equally capable shutdown systems and a redundant number of standby and emergencies diesel generators). Concerning severe accidents the standard CANDU safety analysis already includes scenarios with the failure of emergency core cooling in which the heat removal is provided by the moderator. For scenarios with more core degradation, the capability of the calandria to provide a spreading of the corium and sufficient heat removal area for core debris as well as the additional capability of the concrete reactor vault as ultimate heat sink are still to be analysed and the corresponding management procedures defined.

In early 2000, Cernavoda NPP developed a new Safety Analysis Strategic Plan targeted towards acquiring and developing severe accident methodologies. This plan is under discussion with the Regulatory Body.

Construction and commissioning

15 The basic safety features of the CANDU 6 concept have not changed significantly over the years. When construction of Cernavoda Unit 1 restarted in 1991, design improvements were introduced similar to those already implemented in the twin plants of Wolsung (South Korea), Point Lepreau and Gentilly 2 as a result of their operating experience and PSA studies. The main improvements include for example better separation between control and shutdown system, modification of control room design and the provision for post LOCA sampling capability in the containment.

16 During commissioning, difficulties were experienced with equipment reliability. These problems mainly involved Romanian supplied equipment. The Western consortium managed the resolution of these problems often by replacing equipment. Of particular note, was the need to replace the Romanian supplied diesel generators sets with imported equipment. Deficiencies related to the construction were largely corrected through major programmes of piping / weld inspection and repair. All special safety systems and related

safety support systems were imported. From the collected information it appears that the commissioning of the plant was performed in a manner comparable to CANDU 6 plants in Canada.

Safety assessments and programmes for further improvements

17 The basic safety assessment of the plant is provided in the Final Safety Analysis Report, whose content is in line with the standard content of US and Canadian safety assessment documents. The probabilistic basis of a standard CANDU design is derived from a reliability analysis performed at system level to show compliance with established reliability targets. A level-1 PSA was developed in the past by national organisations. It is however still incomplete and not validated. The utility of Cernavoda NPP, in the framework of the new strategic plan for safety analysis, started an activity aimed at completing a PSA level-1 by years 2001-2. In the longer term it is planned to develop a level-2 PSA analysis that includes fire, flood and seismic hazards, and low power events.

18 At present there are some issues, applicable to CANDU 6 plants like Cernavoda, that have been addressed or are under discussion in Canada. These issues include fire hazard assessment, prevention of dangerous effects of secondary side pipe failure (control room habitability), clogging of containment sump filters, core cooling in absence of forced flow, hydrogen behaviour in the containment. For example, design changes for the sump filters are under evaluation at Cernavoda. The resolution of the above issues needs to be monitored both by the Operator and the Regulatory Body and an improvement programme established where necessary. At present in Cernavoda there is a continuous programme of plant modifications based on operational feedback. However, this modification programme and the possible improvement programme to address the issues discussed above, may be affected by the financial situation at the NPP (see § 21).

Operational safety

Organisation, procedures, operation and maintenance

19 Plant organisation and operational documents are based on the Canadian approach and culture. The document that replaces the traditional Technical Specifications for Operation is the Operating Policies & Principles (OP&P). This is a Canadian reference document, which defines the envelope for safe operation and also includes items related to the plant organisational structure. As a result of an agreement between CNCAN and the Utility, the OP&P now used in Cernavoda are more detailed than those used for similar Canadian plants.

20 The plant Human Resources Development Unit reports directly to the Station Manager. Training programmes have been established both for general topics and for specific job types. A training centre is available on the site, with a full-scope plant simulator, which is currently being adapted to the specific details of the Cernavoda plant. The training programme for operators now includes a supplementary programme developed together with the Polytechnic Institute. This complements the basic training for control room operators, which does not address engineering studies. A systematic training system based on the use of the simulator needs to be further developed.

21 Due to the general economic difficulties of the country, the Utility only receives a part of the payments due to it for the production of electricity. In addition, a large part of the income that is received goes to pay off the credit for plant construction. This is leading the NPP into financial difficulties.

Safety culture and quality assurance

22 The utility has developed a nuclear safety policy document, which covers both corporate and plant levels. It clearly states the overriding priority given to nuclear safety and the objective of the utility to promote a nuclear safety culture at Cernavoda NPP. It also sets out a number of policies for the achievement of good performance. On a yearly basis the Cernavoda management communicates strategic objectives of the NPP to its staff.

23 The Cernavoda NPP received an IAEA pre-OSART mission in 1994 and a WANO mission in August 1997. The WANO mission noted some positive points in the management processes, and some areas to be improved, such as maintenance, plant configuration control, training, and feedback from operating experience. Action has been taken which implements some of the recommendations, while others are still ongoing.

24 Following the transfer of operating responsibility to the Romanian utility, the on-site technical support from the Western Consortium was significantly reduced. In view of this, the current plant managers were selected from professional experts who had been working for the Cernavoda Project for several years. These received initial training in Canada and then on-the-job training while acting as deputies of the Consortium managers during the construction and commissioning phases. The plant management is aware that additional efforts are necessary to ensure that an adequate safety culture extends from the senior staff to all levels of plant personnel.

25 In Canada in 1997, an Independent Integrated Performance Assessment (IIPA) identified deficiencies in human performance and management at a number of Ontario Hydro stations. At the request of CNCAN, a systematic review of the IIPA recommendations was performed by the utility to ascertain their applicability to Cernavoda. The above mentioned WANO Peer Review also covered safety management aspects. It is planned to implement those recommendations that are applicable, but this will depend on budget availability. Both the utility and CNCAN are of the opinion that the most critical IIPA findings that led to the temporary shutdown of some Ontario Hydro units are already addressed or not applicable to Cernavoda.

26 A Plant Quality Assurance manual was issued in 1993. A revised version, based on current Canadian, Romanian, IAEA and ISO standards, is under evaluation by CNCAN.

Operational experience

27 A plant event database has been kept since start of commercial operation. These events led to a total of 7 unplanned shutdowns.

28 A new set of procedures has been in use since April 2000. These procedures will allow the systematic analysis of operational events with ASSET methodology and the assessment of external operating experience.

Emergency preparedness

29 There is an on-site emergency plan approved by CNCAN. In order to improve the evacuation routes from Cernavoda, a bridge is under construction over the Danube, although work on this has stopped because of financial problems. Periodic emergency drills and annual exercises are carried out at the plant.

30 At present there is no dedicated emergency centre available on-site or outside the NPP.

This is contrary to Western European practice.

National industry infrastructure for technical support

31 In Romania, engineering support for the nuclear programme is provided by the Centre of Technology and Engineering for Nuclear Objects (CITON), and research is carried out by the Institute for Nuclear Research (ICN). Both organisations have been supporting the national nuclear programme since the early 1970s. However, Western support and in particular Canadian consulting services are still necessary, particularly for engineering activities involving safety related equipment.

On-site spent fuel and waste management

32 Cernavoda NPP fuel is currently stored in the plant spent fuel pool whose capacity is sufficient for about 9 years operation. The project is underway for the development of an on-site facility for interim storage of spent fuel.

Low and intermediate solid wastes are currently stored in a facility located on the site. Its capacity is for 20 years of operation.

Conclusions

33 The Cernavoda NPP has a CANDU 6 reactor similar to those in operation at Gentilly 2 and Point Lepreau in Canada. The plant was constructed and commissioned under the responsibility of a Western Consortium (AECL, Ansaldo). The safety of the plant has been assessed in a complete safety analysis report approved by the Romanian regulatory body.

34 The NPP managers and the plant operators have a professional attitude and have assimilated a Western safety approach and culture.

35 The Government needs to consider the following:

- The current financial difficulties of the plant need to be solved. If not overcome they could seriously affect activities that are necessary to ensure and maintain an adequate level of safe operation,
- The national infrastructure for technical support and research needs to be improved. It is important that Western support (especially from Canadian experts) is made available when it is needed in the future.

36 The Cernavoda NPP needs to address the following:

- To confirm design safety margins against seismic events and the adequacy of fire protection,
- To monitor the resolution of specific safety issues addressed or currently under discussion in Canada for similar plants and to establish an improvement programme where necessary,
- To preserve in the longer term the current good level of qualification and safety culture of the plant managers. This safety culture should be extended to all plant personnel and to the necessary service and support interfaces existing in the country,
- To improve areas of plant operation such as accident management, emergency preparedness, training activity and feedback from operational experience,

- To undertake and complete the strategic plan concerning the development of PSA studies and severe accident strategy.

References

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2. IAEA IRRT Mission Report, February 1998.
3. Report of Romania to the Convention on Nuclear Safety.

SLOVAKIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. A new Atomic Act on the peaceful use of nuclear energy entered into force on 1 July 1998 and abrogated the previous Act of 1984. The nuclear regulatory authority (ÚJD) was established in 1993 as an independent state authority when Slovakia became an independent state. According to the Atomic Act, it is responsible for state supervision of nuclear installations, radioactive waste and spent fuel management, transport, nuclear materials, physical protection. It has a prominent role in the emergency preparedness and planning organisation in Slovakia. The ÚJD is not responsible for radiation protection or the supervision of the use of radioactive sources outside nuclear installations. The Atomic Act defines the competencies of the ÚJD for the licensing, assessment, inspection, and enforcement activities. Some overlapping of responsibility exist between the ÚJD and the occupational safety office, which may lead to conflicting requirements placed upon an operator.

2. The Slovak nuclear power plants are operated by Slovenské Elektrárne, a 100% state owned Shareholder Company. A partial privatisation could take place in the future. The legal status of the operator is well defined in the Atomic Act, which states that it is responsible for the safety of its installations. The Atomic Act also gives to the regulatory body the responsibility to deliver authorisations that have no safety significance. The regulatory body should be relieved from delivering them.

3. Slovakia is a contracting party to all the key international conventions dealing with nuclear safety.

4. The nuclear legislative framework in Slovakia is generally in line with Western European practice, even if some issues could be improved as indicated in § 2 above.

Status of the regulatory body and technical support infrastructure

5. The ÚJD Chairman reports to the government which, in practice, does not interfere with technical decisions. He has direct access to the Prime Minister and participates in the meetings of the council of ministers when the agenda includes a topic within the responsibility of the regulatory body.

6. The ÚJD is funded from the state budget. Following the recommendations from the RAMG exploratory mission in 1993, the ÚJD staff and budget were increased significantly. Taking into account Slovakia's present nuclear programme, the ÚJD current financial resources are still not sufficient and the inclusion of the Slovak contribution to the Chernobyl shelter fund should not be considered as an increase of its budget. The ÚJD would better retain its experts if their salaries were more in line with those of the operator's staff. The technical competence of the ÚJD personnel is internationally recognised. The ÚJD has a staff of 82 and ÚJD has indicated that 5 additional experts would be desirable to fit with its development plans.

7. The ÚJD has the power to issue and withdraw authorisations. It also has the power to impose sanctions on the operators for any violation of the conditions of an authorisation. Significant decisions signed by the ÚJD Chairman such as plant shutdown can be appealed to court.

8. The ÚJD has good access to technical support of several organisations based in Slovakia and the Czech Republic. However, the same technical support organisations assist the operator and this may create a conflicting situation.

9. It can be concluded that, in general, the ÚJD status is comparable to that of regulatory bodies in Western European countries. On-going developments such as those on internal quality assurance will improve its effectiveness.

Status of regulatory activities

10. Since 1992, a number of national and international evaluations of the ÚJD have taken place. The recommendations of the various missions and support programmes were effectively used in the development of Slovak regulatory activities. The ÚJD takes an active part in international regulatory co-operation.

11. Significant effort has been made to issue regulations as a consequence of the new Atomic Act. At present, 9 regulations are issued out of a total of 16 planned documents. In addition to these regulatory documents, guidelines for the practical application of the regulations by the utility are starting to be produced. The ÚJD has adopted a pragmatic approach in the review process of these guides by introducing a one-year trial of the guide by the utility to integrate, in the final document, the experience feedback.

12. A rigorous licensing review is in place, based on a Safety Analysis Report established by the operator. A licence for a nuclear installation is not issued by the ÚJD but by the authorities of the region where the installation is located. Nevertheless, a licence cannot be issued without the formal agreement of the ÚJD. The licensing steps of siting, construction, operation, and decommissioning are set up in the Atomic Act of 1998. A statement on the decommissioning option is included in the environment impact assessment established at the beginning of the licensing process.

13. The safety assessment practice is well developed and was carried out through a bilateral programme with Switzerland. This activity is now funded by the ÚJD, at least in its 2000 budget: the ÚJD should be given the financial resources to continue this activity.

14. Based on the recommendations of Western nuclear regulatory authorities, the ÚJD has developed a comprehensive inspection plan to conduct routine and daily inspections, perform special inspections and event oriented activities, with a systematic recording of the findings. Inspection procedures are clearly understood and utilised. The inspection performance corresponds to Western European practice.

15. So far, the safety re-evaluation of an installation was made on a case by case basis. The ÚJD intends to introduce a periodic safety review system according to the international practice. To this end, a draft regulation is currently being reviewed.

16. The Ministry of Health is the Authority for radiation protection supervision. Some regulatory issues in the field of nuclear safety have implications in terms of radiation

protection and vice-versa. Both Authorities have established a memorandum of understanding so as to harmonise their respective regulations and actions.

17. The ÚJD has established event-reporting requirements for the licensee and has developed a system for analysis and feedback of operating experience from domestic events similar to common Western European practice. The ÚJD is also actively participating in the INES and international event reporting system.

18. In addition to its participation to the VVER regulator's forum, the ÚJD is part of an international agreement with the Czech Republic and Hungary to share the experience feedback gained at Dukovany, Bohunice, Mochovce and Paks.

19. In summary, the ÚJD has made significant progress over the recent years and has achieved a series of regulatory practices comparable with those of Western European nuclear regulators.

Emergency preparedness on governmental side

20. The National Commission for Radiation Accidents (NECRA) involves the various state bodies playing a role in case of the activation of an off-site emergency plan and gives advice to the local authorities. The first draft of the national emergency plan is expected by the end of the year 2000. Its review, which is being co-ordinated by the ÚJD, should be given high priority.

21. The ÚJD, whose Chairman is a member of the NECRA, is in charge of advising this Commission on all nuclear safety matters in case of an emergency situation. The ÚJD also reviews the on-site and off-site emergency plans from the point of view of nuclear safety.

22. The ÚJD is operating a well-equipped emergency response centre to acquire and process the technical information needed by the NECRA to advise the local authority in charge of managing the emergency situation.

23. Although the national emergency organisation was not yet formalised in a document at that time, a national emergency exercise involving the Bohunice plant, the ÚJD, the NECRA and the local authorities, was organised in 1997. More focused exercises were also organised in 1998 and 1999. Emergency exercises are foreseen to have a periodicity of 3 years in the national Emergency plan. As soon as this plan is issued, it is recommended to organise further emergency exercises in Slovakia. Slovakia has participated in INEX-2 international exercises.

Conclusions

24. The regulatory regime and regulatory body in Slovakia are comparable with Western European practices. The independence of the regulatory body from the organisations involved in the promotion of nuclear energy is fully established in the legislation, which also clearly specifies the prime responsibility for safety of the operator. A well-defined licensing system is in place. The regulatory body is well engaged in the state supervision of nuclear activities and the national emergency organisation is under preparation.

25. It is recommended that the government of Slovakia consider the following:

- The financial and human resources of the ÚJD need to be further increased. The salaries at the ÚJD need to be made comparable with those of the operator's staff,
- The ÚJD needs to be given the resources to maintain the independent assessment capability which was initiated under Swiss assistance,
- The adoption of the national emergency plan needs to be given a high priority,
- The Atomic Act should be amended to remove some duties of the ÚJD that are not directly dealing with nuclear safety.

26. It is recommended that the ÚJD consider how to ensure a clear separation between the technical support it receives and that provided to an operator.

Chapter 2: Nuclear power plant safety status

Data

1. On the two nuclear sites at Bohunice and Mochovce, the Slovak Republic has in operation the following six nuclear power plants owned by the Slovak state company Slovenské Elektrárne (SE):

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Bohunice V1:				
Unit 1	VVER-440/230	1974	1978	2008
Unit 2	VVER-440/230	1974	1980	2010
Bohunice V2:				
Unit 3	VVER-440/213	1976	1984	2014
Unit 4	VVER-440/213	1976	1985	2015
Mochovce:				
Unit 1	VVER-440/213	1983	1998	2028
Unit 2	VVER-440/213	1983	1999	2029

2. At Bohunice, a prototype gas cooled heavy water moderated reactor called Bohunice A1 was operated for a short time in the 1970´s. The reactor was permanently shut down in 1977 after an accident that led to partial core damage and it is now being decommissioned.

3. At Mochovce, the construction of two more units, similar to units 1 and 2, has been suspended (40-50% complete), and currently there is no schedule for their completion.

4. The Slovak government has decided to close the two units of Bohunice V1 in 2006 and 2008.

(i) Bohunice V1, units 1-2

5. The statements presented in this chapter regarding the safety of the Bohunice V1 plant are based on the general knowledge of VVER-440/230 plants summarised in Annex 2, on information provided by Slovak organisations (utility and safety authority), and on records of IAEA missions. The information is confirmed and complemented by the results of the WENRA task force to Slovakia from 12 to 15 October 1999, which focused on Bohunice V1.

Basic technical characteristics

<u>Design basis aspects</u>

6. The first two units on the Bohunice site are VVER-440/230 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2. Improvements have been carried out on both units continuously since they were commissioned. So far, more than 1200 modifications of various safety importance have been implemented, and this improvement process is continuing.

7. Based on the findings of a safety assessment in 1990, the Czechoslovak Atomic Energy Commission issued a list of urgent upgrading measures which have been implemented during the period 1991-1993 and they are known as Bohunice V1 small reconstruction

programme. During 1991-1992, a safety report for a large "gradual reconstruction" was completed for Units 1-2. Results of the review and assessment of this safety report were issued in 1994 in the form of ÚJD regulatory requirements. These were the basis for the development of a major safety-upgrading programme, known as Bohunice V1 gradual reconstruction programme. The aims of this programme were:

- To establish the reactor pressure vessel status,
- To improve the function of the confinement system,
- To demonstrate the ability of the plant to cope with Loss of Coolant Accidents larger than the Design Basis using conservative analysis up to 200-mm LOCA break,
- To demonstrate the ability of the plant to cope with the complete rupture of a main primary coolant pipe (a Beyond Design Basis Accident) using best estimate analysis,
- To improve the plant behaviour in response to internal and external hazards,
- To improve system and equipment reliability,
- To improve organisational and operational safety.

8. The revised design requirements provide a coherent target for safety improvement of the plant. Completion of the long-term improvement programme is expected in year 2000. By then, the safety level of the plants will have been significantly improved when compared with the standard VVER-440/230 described in Annex 2.

Reactor pressure vessel and primary pressure boundary

9. The current condition and the surveillance programme of the reactor pressure vessels appear to be adequate. Both reactor pressure vessels were annealed in 1993. Safety assessments, supported by measurements of weld impurity concentrations, indicate that further annealing will not be necessary up to end of the expected lifetime. Measures have also been implemented on Bohunice V1 to reduce the probability of a large primary break. The majority of the calculations required to demonstrate a leak-before-break case for the 500-mm and the 200-mm primary circuit pipework have been carried out. The leak-before-break case is supported by an in-service inspection programme and by suitable instrumentation to detect incipient leaks. Revised analysis of seismic loadings to take recent modifications into account is expected in the year 2000. This could lead to further minor plant changes. There is an overlap of safety arguments for the revised design basis which covers pipe failures up to 200 mm by conservative analysis and the beyond design base studies which covers pipe failure up to 500 mm by best estimate analysis. The leak-before-break case, which covers pipework from 200-mm to 500-mm diameter and primary circuit components (valves, pumps), supports both these studies. The risk of a large primary to secondary leak, as a consequence of a steam generator collector head lift, has been reduced by the use of a new sealing technology and specific in-service inspection. Thus it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

Confinement

10. Compared to the original design, the confinement capability has been upgraded by installing jet condensers and by improving the venting flaps and the leak-tightness by two orders of magnitude. Relevant experiments have been carried out on the behaviour of essential features like the jet condensers and the venting flaps to support the confinement analyses. According to the utility’s analysis for DBA and for 500-mm break LOCA (as BDBA), there is no large challenge to the confinement function. Regarding the hydrogen management issue, the related measures have been tailored to the level of hydrogen

production that would take place for calculated cladding heat-up during the postulated accidents (DBA and 500-mm break LOCA as BDBA). It is considered that a consistent approach has been followed to demonstrate the modified confinement capabilities against the postulated accidents. However, there are smaller margins with respect to the radiological confinement function compared to those of Western reactors.

Safety systems and hazards

11. Original deficiencies in terms of the capacity and separation of safety systems have been mostly corrected, and it is planned that the remaining deficiencies will be addressed during the year 2000. Extensive measures have been taken against fire risk. Regarding site seismicity, the final value of the peak ground acceleration has not yet been defined but a conservative high value has been assumed as a basis of the upgrading programme. Also improvements to address post-LOCA coolant re-circulation (sump filter clogging) have been implemented.

12. Isolated deviations from common Western practices have been identified. The most significant one is probably the lack of specific protection against the dynamic effects that could result from potential high-energy pipeline breaks. For instance, it is assumed that there would be no break in the main steam lines downstream the isolating valves, where the current mechanical calculations show only a small safety margin to a pipeline break in certain postulated abnormal operating conditions. Consideration of this type of break might call for installation of anti-whipping devices. The utility is aware of this problem and considers its solution firmly linked to the residual lifetime of the Bohunice V1 units.

I&C systems and emergency power supply

13. The original reactor protection system has been replaced with a completely new one that compares favourably with current international practices. The system is qualified in accordance with international standard IEC 880. A new post accident monitoring system has been installed that presents the most important parameters both in the main control room and on the emergency control panel. The re-designed emergency power supply system compares with most important international safety practices, and the system is properly qualified.

Beyond design basis accidents and severe accidents

14. Some limited preventive measures have already been implemented, and further actions for prevention and mitigation are planned once the current plant modernisation process has been completed. Therefore, with regard to beyond design basis accidents, the situation has been improved. A preliminary list of BDBAs including, in addition to 500-mm break LOCA, the total loss of steam generators feed water, small and intermediate primary breaks with loss of high pressure safety injection and station black-out has been already investigated by the utility. As a result, pressurizer safety valves have been replaced to allow the use of a primary feed and bleed procedure, and a source of power from the nearby hydro-plant at Madunice in the case of station blackout has been added. Further actions are still ongoing, e.g. the completion of the list of the BDBAs in the light of the PSA results and the definition of the dedicated measures (equipment and/or emergency procedures) in order to prevent core melt. Concerning severe accidents, there are no specific requirements yet because both the regulator and the utility have put emphasis on first implementing the DBA/BDBA prevention and mitigation measures to avoid core melt. This is considered as a reasonable approach, but development of a feasible severe accident management scheme would be a logical next step, though in the course of the confinement improvements, some solutions have already been considered on how to improve core melt sequence mitigation.

Safety assessments and programmes for further improvements

Safety assessment

15. A Safety Analysis Report was prepared for the V1 plant prior to the start of the long-term improvement programme. Following completion of this programme, a Safety Analysis Report, with a content similar to those of Western plants, was presented to the ÚJD in June 2000 for review.

16. A Probabilistic Safety Analysis level-1 was carried out before and after the Bohunice V1 small reconstruction programme, and will be repeated after completion of the long term improvement programme. The scope of this PSA has covered only full power mode but considers all important initiating events including internal fires and floods.

Programme for safety improvement

17. The long-term improvement programme should be complete in the year 2000. Thereafter, modifications will be implemented based on a case-by-case analysis.

Operational safety

Organisation, procedures, operation and maintenance

18. The organisation of the Bohunice plant, including Bohunice V1 and Bohunice V2, is similar to typical Western plants' organisation. The site Manager directs the operations division (two chief engineers, one for Bohunice V1 and one for Bohunice V2), the maintenance division, the economics and commerce division, the investments division, the human resources and services division and the technical support and safety division. Compared with Western European practices, it is considered that the organisational aspects and procedures used for the V1 plant are adequate.

19. The utility has been able to implement the V1 plant modernisation program as well as other necessary modernisation and maintenance activities. To ensure that maintenance is carried out efficiently, a maintenance Division has been set up with the necessary workshops, laboratories, equipment and tools. For on-the-job training, mock-ups facilities are also available.

20. Overall, the qualification of the plant staff appears satisfactory. A comprehensive training system is in place and a multifunction simulator is used. Exchanges with Western partners are on-going either through bilateral or through multilateral co-operation programmes.

21. Technical specifications for operation have been improved, and follow a typical Western approach. Implementation of revised Emergency Operating Procedures is planned for the end of the modernisation process. Improvements are also being made to procedures for normal operation.

Safety culture and management, quality assurance

22. Two safety committees are in place, one at the plant level, the second at the company level. Contacts with Western experts have helped to promote a safety culture. A QA system is in place, covering all the main activities, including the V1 improvement activities.

Operational experience

23. Systematic investigations of plant events and operational feedback are conducted by a

dedicated plant department. At the national level, investigations are also conducted independently by the VUJE institute and in some cases by the ÚJD.

Emergency preparedness

24. The on-site emergency plan is regularly updated and exercises are carried out periodically (quarterly and yearly, depending on the type of exercise). The level achieved is adequate.

(ii) Bohunice V2, units 3-4

25. The statements presented in this chapter regarding the safety of Bohunice V2 plant are based on the general knowledge of VVER-440/213 plants summarised in Annex 2, on the joint international projects (including EU TSOs focusing at specific technical features of the VVER-440 plants), and on information provided by Slovak organisations (utility and safety authority).

Basic technical characteristics

Design basis aspects

26. Units 3-4 at Bohunice are VVER-440/213 type nuclear power plants. General safety characteristics of such plants are presented in Annex 2. Since 1990, significant improvements have been implemented at Bohunice V2. These include for example the installation of in-service diagnostic systems, the renovation of instrumentation and control systems, the improvement of electrical systems, fire and seismic upgrading, and some improvements in operational safety such as the introduction of symptom based emergency operating procedures and a new generation of normal operation procedures.

Reactor pressure vessel and primary pressure boundary

27. The current condition and the surveillance programme of the reactor pressure vessels appear to be adequate. Annealing will not be necessary up to the end of the expected lifetimes of the plants. The leak-before-break (LBB) concept for 500-mm and 200-mm primary circuit pipework has been demonstrated. The LBB is supported by an in-service inspection programme and by suitable instrumentation to detect incipient leaks. Regarding primary to secondary leakage through the steam generator collector head, the same measures as for the Bohunice V1 units are either implemented or planned. Additionally a system for monitoring primary circuit leakage in the steam generators by nitrogen 16 activity in the steam has been installed. The integrity of the primary pressure boundary is therefore, considered safeguarded to an adequate level.

Bubbler condenser containment

28. The performance of the bubbler condenser system in case of Large Break LOCA has been verified in full-scope tests in the framework of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still a need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA). The leak-tightness of Bohunice V2 containments has been improved. However, the leak rate is still somewhat higher than those achieved at Western plants. Nevertheless, the containment internal pressure driving the leak would be effectively limited by the bubbler condenser function in case of design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU.

Safety systems and hazards

29. In terms of capacity and redundancy, the safety systems are comparable to Western ones.

However, some shortcomings have been identified and are being addressed in order to achieve adequate reliability of safety systems in all operating situations. For instance, measures against sump filters clogging and measures for fire protection are being implemented in the year 2000. Necessary improvements (e.g., modification of steam generators feed water system) are planned to be implemented by about the year 2002. The seismicity of Bohunice was once again reconsidered in 1998. The related assessment and the proposed values have been assessed by the IAEA. Also, a seismic monitoring network is permanently in operation.

Beyond design basis accidents and severe accidents

30. The regional Phare assistance project PHARE 4.2.7.a/93 considered the issue of beyond design basis accidents and severe accidents in the VVER-440/213 units. Different accident sequences have been considered with different failures. This includes accidents such as ATWS, primary breaks with partial or total failure of emergency core cooling system, total loss of feed water, station blackout. Some preventive measures have been taken into consideration and mitigation measures are being implemented. A new regional Phare project started in June 2000 in order to assist the Czech, Slovak and Hungarian safety authorities to assess the proposed preventive and mitigation measures.

Safety assessments and programmes for further improvements

Safety assessment

31. In 1993, the ÚJD approved the use of US NRC RG 1.70, adopted on country specific conditions for the elaboration of innovated Safety Analysis Report for Bohunice V2 units. This innovated Safety Analysis Report was presented to the regulator in 1994, after 10 years of NPP operation. Following comments from the ÚJD, a new version was produced in 1997. Its content corresponds to what is generally expected in Periodic Safety Reviews in Western Europe, and it has been reviewed and accepted by the ÚJD as part of a rigorous licensing review. The chapter in the report related to accident analysis was reviewed by the IAEA. In addition, a Probabilistic Safety Assessment has been carried out for the plant, within the process of updating of the safety analysis report after ten years of operation. As for Bohunice V1, the scope of this PSA has covered only full power mode, considering all important initiating events, including internal fires and floods. In addition, low power and shutdown PSA study level-1 has been completed and reviewed by IAEA in 1999. Work on PSA study level-2 started in 1999.

Programme for safety improvements

32. A further extensive modernisation programme is planned for implementation between 1999 and 2006, with the major upgrades relating to safety being completed by 2002 (§ 29).

Operational safety

33. Information and conclusions presented above for the V1 units are generally applicable for the V2 units.

(iii) Mochovce units 1-2

34. The statements presented on the Mochovce plant are based on the preliminary results of an independent safety evaluation, which has been carried out by a consortium of Western European Technical Safety Organisations.

Basic technical characteristics

Design basis aspects

35. The Mochovce units 1-2 are the latest ones of the VVER-440/213 type nuclear power plants (see Annex 2). Compared to their predecessors, several modifications were included during the design phase. The most important of these are the use of higher quality equipment (e.g. a modern reactor control system, a new type of pressurizer safety valves, an upgraded feed water control system), and the improvement of systems used in accident situations (e.g. a new design for the steam dump system, an improved emergency feed-water system located outside the turbine hall, upgraded fire fighting water system, a primary circuit venting system).

36. However, some design weaknesses remained, and these were addressed in a nuclear safety improvement programme developed in 1995 for the Mochovce NPP. The programme, comprising 87 safety measures, was reviewed by Western European Technical Safety Organisations and has almost been completed. The remaining measures (e.g. completion of equipment qualification, site seismicity characterisation) are underway.

37. Compared to the original VVER-440/213 design, the safety level of the Mochovce plant has been significantly improved.

Reactor pressure vessel and primary pressure boundary

38. The condition and surveillance programmes of the reactor pressure vessels appear to be adequate. In addition to the measures implemented for the V1 and V2 steam generators, the collector heads have been replaced to reduce the size of possible leaks. The integrity of the primary pressure boundary is thus considered to be safeguarded to a level comparable to Western practices.

Bubbler condenser containment

39. The performance of the bubbler condenser system used at Mochovce containment has been studied in full-scope tests performed in the framework of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test and analysis results were reported in early 2000, and, together with the investigations financed by the plant, they demonstrate structural strength and adequate confinement of radioactive fission products in case of Large Break LOCA. There is still a need for detailed analysis of the EU project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA). The leak-tightness of Mochovce units 1 and 2 containments in case of large LOCA is comparable to those of Western European containments. Due to the bubbler condenser function, the calculated radioactive releases after design basis accidents would not be higher than at many Western plants for similar accidents.

Safety systems and hazards

40. In terms of capacity, redundancy and separation, the situation is comparable to Western European practice. The ongoing programme of equipment qualification for accident situations is in an advanced stage. Concerning site seismicity, the final value of the peak ground acceleration has not yet been defined. The utility has already undertaken actions aiming at the evaluation of the site seismic characteristics. The results are expected during 2001. Depending on results, further upgrading could be necessary.

Beyond design basis accidents and severe accidents

41. The utility has already started to analyse Beyond Design Basis Accidents on the basis of a preliminary list including ATWS, total loss of steam generators feed water, small break

LOCA in coincidence with a total loss of high pressure safety injection, steam generator tube rupture in coincidence with steam line break. This list will be reassessed in the light of the PSA results. Concerning severe accidents, the utility has the intention to use the applicable results obtained from the regional Phare project PHARE 4.2.7.a/93.

Safety assessments and programmes for further improvements

Safety assessment

42. A Safety Analysis Report (SAR) was prepared prior to the start-up of unit 1 in 1998. Its content is consistent with the content of safety reports in Western Europe, and it has been reviewed and assessed by the ÚJD as part of a rigorous licensing review. Furthermore, an independent review of the SAR by a consortium of Western European Technical Safety Organisations has been performed. A Probabilistic Safety Assessment level-1 has been developed in two phases where the pre-modifications and post-modifications states of the plant are evaluated. The pre-modification state is the plant state before implementation, and the post-modification state is the plant state after implementation of the safety measures specified in the safety improvement programme. The pre-modification PSA has been assessed by a consortium of Western European Technical Safety Organisations. The post-modifications PSA will be completed in the year 2000. The scopes of these PSAs cover full power mode and consider all important initiating events, including internal fires and floods. Moreover, first results of probabilistic seismic assessment which includes curves of seismic risk will be available as part of PSA level-1 in 2000. These evaluations will be complemented by a shutdown state PSA planned to start this year.

Programme for safety improvement

43. The safety improvement programme presented by the utility and agreed by the ÚJD is almost completed (see § 36).

Operational safety

44. The organisation of Mochovce plant is similar to that one of Bohunice plant. In addition, the preparation for plant operation has benefited from extensive national and worldwide experience, and advanced methods were brought into use prior to the first start-up of unit 1. These include, among other things, the availability of a full-scope simulator for initial training of control room operators.

National industry infrastructure for technical support

45. Slovakia has quite a strong national infrastructure in the nuclear field because some important research institutes were allocated to Slovakia at the division of Czechoslovakia. The Power Equipment Research Institute (VUEZ) supports Slovakian plants in tests of containment sealing, condensation systems, safety system design, filtration and ventilation. The Nuclear Research Institute (VUJE) provides technical support in the areas of training and safety analysis. As part of the infrastructure, one could also include the links with the Czech Republic that has a long tradition of mechanical equipment manufacturing.

On-site spent fuel and waste management

46. Up to 1986, spent fuel was sent back to Russia for reprocessing and final disposal. From 1987, Bohunice V1 and V2 spent fuel has been stored in an interim storage facility. An extension is being built which will allow the storage of V1 and V2 fuel up to the end of their expected operating lifetimes. This extension is expected to be complete during the

first half of the year 2000. The storage at the reactor pools at Mochovce NPP is designed to allow storage of spent fuel for a period of six years. The construction of interim spent fuel storage is foreseen also at the Mochovce site.

47. Currently, all NPP waste is treated on the Bohunice site. Three bitumenization facilities, as well as fragmentation and decontamination units, are under operation. The Bohunice Radwaste Treatment Facility, including cementation, incineration and compacting installations, is now under commissioning.

Conclusions

48. Operational practices at all Slovakian nuclear power plants are consistent with those in Western Europe.

(i) Bohunice V1, units 1-2

49. The following conclusions can be made:

- The revised design requirements provide a coherent target for safety improvement of the plant. The utility has made significant progress towards establishing a new design basis and implementing the relevant measures. A Safety Analysis Report, similar to those of Western plants, was presented by the utility to the ÚJD in June 2000 for review. Some work remains to be done but no technical obstacles are foreseen and completion is expected in 2000,
- In order to achieve and demonstrate adequate protection against possible loss of coolant accidents, several measures have been taken. The engineering safety features have been extended to cope with a leak that is equivalent to the leak from a double-ended guillotine break of a 200-mm pipe. A 500-mm double ended guillotine break LOCA is however evaluated with best estimate assumptions aiming to demonstrate core melt prevention and adequate confinement performance,
- Compared to the original design, the confinement capability has been upgraded. It is considered that a consistent approach has been followed to demonstrate the modified confinement capabilities against postulated events. However, there are smaller margins with respect to the radiological confinement function compared to those of Western reactors,
- Very extensive measures have been taken against fire risk. Regarding the site seismicity, the final value of the peak ground acceleration has not been defined but a conservatively high value has been assumed as a basis for the upgrading programme. Also, improvements concerning post-LOCA coolant re-circulation (sump filter clogging) have been implemented,
- There are isolated deviations from Western practices regarding steam line break postulation and dynamic effects from broken high-energy pipelines. Consideration of this type of break downstream the isolating valves may call for installation of anti-whipping devices. The utility is aware of this problem and considers its solution firmly linked to the residual lifetime of the Bohunice V1 units,
- If a solution can be found to the concerns related to the confinement ability to cope with the double-ended guillotine break LOCA, the Bohunice V1 plant should achieve a safety level comparable to that of plants of the same vintage in Western Europe.

(ii) Bohunice V2, units 3-4

50. The following conclusions can be drawn:

- Since 1990, significant improvements have been implemented at Bohunice V2. However, in order to achieve adequate reliability of safety systems in all operating situations, necessary improvements have been identified and are planned for implementation in 2000 for the measures against sump filters clogging and for fire protection, and in 2002 for the modifications to the steam generators feed water system,

- The integrity of the primary pressure boundary is considered to be safeguarded to an adequate level,
- The performance of the bubbler condenser system used at Bohunice V2 containments has been studied in full-scope tests performed in the frame of the Bubbler Condenser Experimental Qualification project sponsored by the EU. However, as for all the bubbler condensers, there is still need for detailed analysis of the experimental project results and for complementary tests for Steam Line Break and Small Break LOCA. Although, the leak rate of Bohunice V2 containments is somewhat higher than those achieved by those of Western plants, the radiological consequences in case of design basis accidents would not be higher than those accepted for Western European reactors,
- Concerning safety assessment, the content of the Safety Analysis Report is consistent with what is generally expected in Periodic Safety Reviews in Western Europe. It is complemented by a Probabilistic Safety Assessment. With regard to beyond design basis accidents and severe accidents, some preventive measures have been taken into consideration and mitigation measures are being implemented,
- An extensive modernisation programme is planned for implementation between 1999 and 2006, with the major upgrades relating to safety being completed by 2002,
- Consequently, the safety of Bohunice V2 units seems generally adequate. Once the ongoing safety upgrades have been implemented (by about year 2002), the safety level of these units is expected to be comparable to what is commonly found at units of the same vintage in Western European countries.

<u>(iii) Mochovce units 1-2</u>

51. The following conclusions can be drawn:

- Compared to their VVER-440/213 predecessors, units 1 and 2 of Mochovce included several modifications during the design phase. However, some design weaknesses remained, and a dedicated nuclear safety improvement programme, including 87 safety measures, was developed for the Mochovce NPP in 1995. This programme, which is almost complete, was reviewed by Western European Technical Safety Organisations,
- The integrity of the primary pressure boundary is considered to be safeguarded to an adequate level,
- Considering the mechanical reinforcements which improve the structural behaviour of the bubbler condenser and all the analytical and experimental work that has been performed, Mochovce bubbler condenser is presently the one that has undergone the most scrutiny. The leak rates of Mochovce units 1 and 2 containments in case of large LOCA are comparable to those achieved in Western plants. Because of the bubbler condenser function, the calculated radioactive releases after design basis accidents would not be higher than those at many Western plants under similar accident conditions,
- The content of the Safety Analysis Report, prepared prior to the start-up of unit 1, is consistent with the content of safety reports in Western Europe. It has been complemented by a Probabilistic Safety Assessment which will be extended to take account of the plant modifications and initiating events during reactor shutdown states. With regard to beyond design basis accidents and severe accidents, preventive measures have been taken into consideration and mitigation measures are being implemented,
- Although some residual work (e.g. bubbler condenser qualification, Mochovce site

seismicity characterisation) is still needed to confirm all parts of the safety analysis, the safety level of Mochovce units is comparable to the safety level of the nuclear power plants being operated in Western Europe.

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SLOVENIA

Chapter 1: Status of the regulatory regime and regulatory body

The information given here is based on experience gained through bilateral and multilateral assistance and co-operation programmes such as RAMG and CONCERT, IAEA missions and programmes and other open sources. An expert meeting took place in Ljubljana in January 2000 attended by WENRA members, SNSA and the operator of Krško NPP.

Status of the legislative framework

1. When Slovenia became an independent state in 1991, the continuity of the legal system was ensured by adopting all relevant laws from the former Federation of Yugoslavia. The main nuclear Act, the 1984 Act on radiation protection and the safe use of nuclear energy is currently under review. A new law has been under development for several years but without much progress. The 1984 Act is supported by a set of second level regulations related to specific aspects of nuclear, radiation, waste and transport safety.

2. The Slovenian Nuclear Safety Administration (SNSA) was established at the end of 1987 as an independent body dealing with all matters concerning nuclear safety. The SNSA reported directly to the Government and to Parliament until a change in legislation in 1991, since when the SNSA has been reporting to the Ministry of Environment and Spatial Planning.

3. Several deficiencies have been identified in the 1984 Act and improvements are expected in the future law. The SNSA is involved in its preparation. Firstly, the responsibility for safety needs to be clearly assigned to the licence holder. In addition, the roles and responsibilities of all governmental bodies involved in the regulatory process need to be clearly defined. Also adequate provisions need to be established to give the SNSA the authority and resources to manage the independent safety assessment required in the licensing process.

4. It is noted that the use being made of the appeal process, which allows the operator to appeal to the minister on both administrative and technical decisions, may constrain the regulator and undermine his credibility, authority and independence. Such technical appeals, which could have safety implications, need to be reconsidered in the review process of the law.

5. The licensing procedure is generally defined in the 1984 Act, and further developed, including licensing requirements and conditions, in second level regulation. The SNSA has overall control of the licensing procedure, subject to the appeal process above.

6. The Krško plant was built based on equal investment by utilities from Slovenia and Croatia. The responsibility for nuclear safety remains in Slovenia. Since the breakdown of the former Federation of Yugoslavia, a proposed agreement is under discussion among the two countries. In the mean time the Slovenian Government has issued a decree to support the necessary investment programme for the plant modernisation. According to this Decree the Nuclearna Elektrarna Krško (NEK), owner of the Krško nuclear power plant, is transformed into a public company. Future operation of Krško without a final resolution of issues related to shared ownership may affect the plant financial situation, which could

have an impact on safety.

7. Slovenia has ratified all key international conventions related to nuclear safety and they are now part of the Slovenian legislation. A peer review of the Slovenian legislation and regulatory framework has been carried out by an IRRT mission of the IAEA.

Status of the regulatory body and technical support infrastructure

8. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The Director of SNSA is appointed by the Government.

   A Nuclear Safety Expert Commission (NSEC) was established by the 1980 Act on Performing protection against ionising radiation and on measures for the safety of nuclear plants and installations. This Commission has an advisory role to the Ministries, advising on important licensing issues and reviewing the SNSA annual report. The Commission is chaired by the SNSA Director and has representatives of different ministries, experts in nuclear and radiation safety from Slovenia, one representative from the NPP Krško and one expert representing the Croatian Administration.

9. The Ministry of Defence plays, through the Administration for Civil Protection and Disaster Relief (ACPDR), the co-ordinating role in the national emergency system. The Health Inspectorate plays an active role in civil emergency plans for responding to nuclear and radiation accidents.

10. The SNSA does not have a separate and independent budget from the Ministry. No specific provisions are provided in the law to define the mechanism to fund the SNSA from the State budget. The Act on Administrative procedure enables the SNSA to charge to the licensee for the expenses related to a certain licensing process with an appropriate justification. Fees charged to the licensee do not represent a significant addition to the budget. Following the recommendations of the RAMG exploratory mission, the salaries of SNSA inspectors were raised to the similar level as the utility personnel, but this did not apply to the rest of the SNSA staff. This has resulted in some SNSA staff, in particular young engineers, leaving after their initial training. The budget and financial situation of SNSA therefore needs to be improved.

11. The staffing level of the SNSA has evolved from 5 in 1988 to the present level of 37. Therefore, 11 of 48 permanent posts are currently vacant. The SNSA has an active training programme to increase staff skills. There is still a lack of qualified and experienced personnel in most critical areas. It seems that the SNSA needs to further develop capability to perform a thorough safety review and assessment.

12. The Slovenian legislation requires the applicants to submit, in addition to the safety case, an independent assessment, with positive results, performed by an authorised organisation. The licensee contracts and manages such activity. Due to its limited resources and capabilities, the SNSA widely bases its assessment and decision making on the independent reviews by the authorised organisation. To guarantee a genuinely independent assessment, the SNSA needs to be provided with the authority and the resources to contract external organisations. Procedures need to be developed to ensure that such organisations are independent from licensee in the activities contracted.

13. The SNSA is given by law enforcement powers, including the power to stop the operation of a nuclear facility, in case of non-compliance with regulations.

14. There is no unique technical support organisation and several organisations act as TSO depending on the issues being assessed. The main national TSOs are the Jozef Stefan Institute (JSI) and the Milan Vidmar Institute (MVI), which also provide technical support to the utility. The funding for work carried out by JSI and MVI for the SNSA, is provided directly by the utility. The relationship between the TSOs, SNSA and the utility needs to be clarified to ensure there are no conflicts of interest.

15. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, the budget and financial situation of SNSA should be improved in order to ensure a complete independent safety assessment capability.

Status of regulatory activities

16. The SNSA has the powers to propose new legislation and is responsible for preparing new laws and regulations. The SNSA is also responsible for issuing and amending licences for all nuclear facilities and performs regular inspections at those facilities.

17. Though the main responsibilities and functions of the SNSA are well understood by the staff and the licensee, the policy and criteria based on which regulatory decisions are taken are not always defined. The SNSA needs to define its regulatory requirements to allow it to make the licensing decisions. The SNSA strategy needs to include, when feasible, a predefined licensing process for major safety improvements.

18. The SNSA licensing and assessment is based on US NRC practice, and is performed by competent staff in their areas of responsibility. However, due to high workload combined with limited resources, the SNSA needs to develop further its technical capability to perform a complete independent assessment. In addition, an adequate system of internal quality assurance is needed. Further efforts to establish a better co-ordination of the licensing steps and the related assessment and inspection activities are needed.

19. The SNSA inspection programme needs to be strengthened through adequate resources and further development and implementation of a systematic process, based on licensee performance and more focused on safety relevant aspects. Some of the interfaces between the different Slovenian Inspectorates are not very clearly defined, in particular with regard to the Health Inspectorate, fire protection issues, emergency preparedness and physical protection.

20. There is no formal requirement for a periodic safety review of Krško NPP and this needs to be addressed in the revised legislation, regulations and technical guidance.

21. The SNSA has established a system for the feedback of operating experience, including event reporting and assessment of lessons learned form similar operating plants. The SNSA takes part regularly in international regulatory activities aimed at fostering regulatory co-operation. In particular, Slovenia has concluded bilateral agreements with countries operating similar types of reactors.

22. In general the SNSA operates according to Western practice and methodologies. However, the management of regulatory duties could be improved and the inspection programme needs further implementation of a systematic process.

Emergency preparedness on governmental side

23. Under the present legislation, the Administration for Civil Protection and Disaster Relief (ACPDR) regulates and supervises emergency preparedness regarding the protection of the public off-site. The new updated Slovenian national plan has not been finalised. No national full scope exercise has been conducted for six years. A comprehensive exercise is needed to confirm the co-ordination among all authorities involved in responding nuclear emergencies.

24. The SNSA acts as the independent expert governmental organisation, providing advice to the National Civil Protection Headquarters, and acts as the co-ordinator with neighbouring countries and with the IAEA. The SNSA approves the on-site emergency plan. The SNSA has no emergency response centre to perform its emergency function.

25. Attention has to be given to the integration of all interfaces and relationships between the various involved organisations. This integration needs to take into account the core competence of each institution on radiation protection and nuclear safety, to be consistent in the assignment of responsibilities.

26. The Krško plant is close to the borders of Croatia and two bilateral agreements are in place to ensure adequate co-ordination and co-operation by the authorities of both countries, the most recent ratified in 1999. The interface with the Croatian authorities should be reflected in the emergency plan and tested in an integrated exercise.

27. The SNSA needs to have the capability of providing independent information to the public in the case of an emergency. The distribution of information needs to be co-ordinated with other involved parties.

Conclusions

28. The SNSA operates, in general, according to Western practice and methodologies. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, there are some issues that need to be addressed.

29. It is recommended that the Government of the Republic of Slovenia consider the following suggestions:

- The existing legislation on nuclear and radiation safety is not fully in line with current Western European practice. The review of the existing legislation needs to be completed as soon as possible,
- The existing appeal process on administrative and technical decisions could have safety implications and may constrain the regulator and undermine his independence,
- The lack of a final resolution of issues related to shared ownership of Krško NPP may affect the plant long-term financial situation and have an impact on safety,

- The SNSA needs to be provided with the authority and the resources to contract external organisations providing support for independent safety assessment,
- The budget and financial situation of SNSA needs to be improved. An increase in SNSA salaries and an improvement in the financial stability of the organisation would help to retain staff and increase independent safety assessment capability,
- The national response to nuclear and radiological emergencies needs to be improved by implementing an integrated national emergency plan. Special attention needs to be paid to the interface with the Croatian authorities.

30. It is recommended that the SNSA consider the following suggestions:

- The SNSA needs to define its regulatory requirements to allow it to make the licensing decisions, in particular by establishing a predefined licensing system for major safety activities like the periodical safety review,
- The SNSA needs to develop further its own technical capabilities in order to be able to make better independent decisions.

Chapter 2: Nuclear power plant safety status

Data

1. Slovenia has one nuclear power plant located at Krško, which was built as a joint investment by the electricity utilities of Slovenia and Croatia:

NPP	Reactor type	Electrical power (MW)		Start of construction	First grid connection	End of design life
		Gross	Net			
Krško	Westinghouse 2-loop PWR	*707	*676	1974	1981	2023

*Values after steam-generator replacement and power uprating. Original design values were 664 and 632 MWe respectively.

2. Since the breakdown of the Former Federation of Yugoslavia, a proposed agreement is under discussion between the two independent Republics of Slovenia and Croatia to define the legal provisions relating to the ownership of the plant. Pending this agreement, the Slovenian Government promulgated a Decree in July 1998 on transforming NPP Krško into a public company named "Nuklearna Elektrarna Krško" (NEK), to assure necessary financial resources for safe plant operation. The founder of the company is the Republic of Slovenia.

3. The future operation of Krško NPP is also affected by an Energy Policy resolution accepted by the Parliament. According to the Energy Policy the energy sector will be privatised and an open market established starting from 2001. NEK will, however, remain a public company. The implication for NEK to operate in the market is however not clear at the moment and will require special attention to avoid any negative impact on the current level of safe operation.

4. The statements presented in this Chapter regarding the safety of the Krško plant are based on general knowledge of the Westinghouse 2-loop PWR, on information provided by the Slovenian organisations (Regulatory Body and Utility) and on the reports of IAEA and other international missions, in particular ICISA. The information was verified and complemented by the results of a visit of WENRA experts to Slovenia from January 25 to January 28, 2000.

Basic technical characteristics

Design basis aspects

5. The design of Krško is similar to other Westinghouse PWRs of the same type in the USA, Belgium, Switzerland, Korea and Brazil. The Angra 1 plant in Brazil was the reference plant for Krško.

6. The reactor coolant system comprises two parallel loops connected to the reactor pressure vessel. Each loop contains a vertical single-stage centrifugal coolant pump and one vertical U-tube steam generator.

Reactor pressure vessel and primary pressure boundary

7. RPV surveillance is performed regularly and no operational problems have been identified.

No RPV embrittlement problems are foreseen at the moment. The reactor vessel was manufactured by Combustion Engineering using a low-copper base material. The good characteristics of the steel and the results of the regular monitoring programme (tests on samples of irradiated steel initially placed inside the RPV) have led the Krško RPV to be classified in the "no problem" area, according to US NRC R.G. 1.99 rev.2. The next sample will be tested in 2001 and is expected to confirm the positive NDT trend observed so far. Leaks have developed during past years in the original steam generator (SG) tubes due to stress corrosion cracking, and consequently more and more tubes were plugged. In 1999 the plugging level almost reached 18%. The leakage of SG tubes has caused an increase in the quantity and activity of Low and Intermediate Level waste. The SGs were replaced with ones of a new design in the regular outage of spring 2000. This made possible a 6.3% power up-rate. As part of the process of SGs replacement and power up-rate, complete mechanical and structural re-analyses and evaluation of the Reactor Cooling System (RCS) design were performed by the utility and approved by the regulator. The utility performed an analysis, according to the applicable US guides, for the application of LBB to the reactor coolant system, aimed at reducing the number of pipe restraints. The application of this is still under discussion between NEK and SNSA.

Confinement

8. The reactor and primary coolant system, including the steam generators, are located inside a double containment, which consists of a cylindrical steel shell with a hemispherical dome, an annulus and a surrounding reinforced concrete shield building. A negative pressure is established in the annulus between the primary and the secondary containment. The measured leak-tightness is similar to that of NPPs in Western European countries.

   The containment design presents margins against BDBA. conservative assumptions are used in the analysis of radiological effects and consequences of LOCA. It can be concluded that the containment function is comparable to that of Western European reactors of the same vintage.

Safety systems and hazards

9. In general, the safety systems are based on two redundant trains. Emergency core cooling is provided by one high and one low-pressure safety injection system and two pressurised accumulators. In case of a LOCA, re-circulation of the safety injection cooling water from the containment sump takes place. An evaluation of the clogging of sump strainers in the case of a LOCA has been made by the operator and is under review by the Regulator. On the secondary side, the auxiliary feed-water system consists of two separate redundant loops based on electrical and steam driven pumps.

10. Heat can be removed from the containment by means of the heat exchangers of the low-pressure injection system during sump re-circulation, and also by the containment fan coolers. The Component Cooling Water System and Essential Service Water System each consist of two redundant loops.

11. The seismic design of Krško is based on a maximum acceleration value of 0.3 g at the level of the foundations and on the design response spectrum recommended by the US NRC R.G. 1.60. Some aspects of the seismic characterisation of the site are under re-evaluation (see § 20).

I&C systems and emergency power supply

12. The reactor protection system is based on solid state logic arranged in two redundant trains

in accordance with US NRC R.G. 1.22 and IEEE 279. Qualification testing has been performed on the various items of protection system equipment. The emergency power supply has been upgraded and properly qualified to meet the requirements of US 10CFR50.63 which relates to station blackout events (see § 13).

Beyond design basis accidents and severe accidents

13. The Anticipated Transients Without Scram (ATWS) rule, US 10CFR50.62, has been implemented. Notably this includes the addition of ATWS Mitigating System Actuation Circuitry. Regarding the "Station Blackout" (SBO) event, a study has been performed by NEK using the US 10 CFR50.63 rule as reference. As result of the study an improvement programme, which included the upgrade of the DC sources and the addition of a nitrogen supply to important valves, was approved by the regulatory body and has been implemented.

14. Regarding severe accidents, a reactor vessel wet cavity strategy was implemented during the spring 2000 outage. In the case of LOCA the water in the containment would flow into and fill the reactor cavity. In addition, Severe Accident Management Guidelines have been developed using as a reference the generic Westinghouse owners' group guidelines but based on plant specific analyses and studies. They will be validated on the full-scope simulator, which is capable of simulating severe accident scenarios. The simulator entered into service before the spring 2000 outage. The issue of H_2 management is under discussion with the Regulatory Body.

Safety assessments and programmes for further improvements

Safety assessment and documentation

15. The licensing basis for Krško was a safety analysis report prepared by Westinghouse, as the main supplier of the plant. The content of SAR followed US practice. In 1991 the need was identified for a systematic review and the incorporation of design changes into the plant safety analysis report. That process led into the USAR (Updated Safety Analysis Report). The first revision was issued in 1992. Currently the USAR is a living document that is updated every year according to a written review procedure and is approved by the regulatory body.

16. A PSA has been performed at levels 1 and 2. The main contributors to core damage frequency are internal events, seismic events and internal fire. A PSA for shutdown conditions has also been performed. The findings from the integrated PSA studies were used in the development of the plant safety improvement and modernisation programme. The most important applications of PSA have been the Fire Protection Action Plan (FPAP) and the safety assessment of steam generator replacement and power up-rate. A model "NEK 2000" has been developed to represent the future risk profile of the plant after the implementation of the planned modernisation programme.

17. Since the start of operation, many safety requirements have been established by the SNSA. In addition, recommendations have been made by IAEA and WANO missions, and by the ICISA International Commission (1992-93). Most of these have been resolved and some are currently being implemented. The applicable post-TMI safety requirements have been implemented.

18. The safety related improvements already implemented can be grouped as follows: Post TMI related modifications, instrumentation improvements, improvements to enhance plant protection against special events like ATWS and Station blackout, PSA driven

improvements, development of a fire protection action plan, and major equipment replacement or upgrade.

Programmes for safety improvements

19. A large improvement programme was completed during the spring 2000 outage which included replacement of both steam generators and implementation of associated modifications, testing and validation of the full-scope simulator, implementation of remaining recommendations suggested by the fire hazard analysis, modifications related to the implementation of severe accident strategy (e.g. wet cavity). In order to license these modifications a complete set of safety analyses has been performed. They were independently reviewed by authorised organisations according to the Slovenian licensing procedure and submitted to the SNSA for their review and approval.

20. The seismic qualification of safety-related components is currently under review. Additional geophysical, geological, and seismological investigations in the area surrounding NPP Krško are being performed. Under a PHARE project, an investigation of the site seismicity is being carried out and a seismic monitoring network will be established around Krško. Based on the new structural-tectonic data, the Probabilistic Seismic Hazard Analysis (PSHA) which has already been carried out may need to be revisited.

21. It can be concluded that Krško is undergoing a continuous process of review and safety assessment. A living programme of verification of the compliance with the relevant US Regulatory requirements is in place.

Operational safety

Organisation, procedures, operation and maintenance

22. The site organisation, staff numbers, qualification and training of the personnel are similar to those of Western European NPPs. All the activities directly related to the plant operation are supported by independent review functions reporting at different levels in the organisational structure. In particular the Krško Safety Committee advises the General Director. The Safety Committee is composed of 11 members, 6 of which are external to the plant. It provides an independent review in several areas connected with plant operation and safety, including hardware and procedure changes.

23. At the moment no major financial problems are reported to exist by the NEK management. The need to pay appropriate attention to the NEK needs is recognised by the Government. The Decree establishing the Krško NPP as a public company has allowed the financing of the modernisation programme in the absence of a partnership agreement with Croatia.

24. The plant operational limits and conditions are provided in the Technical Specifications, changes to which are subject to approval by the SNSA. The content and style of the Technical Specifications follow US practice and are similar to those used in some Western European plants.

25. The operating procedures are reviewed and updated every two years in accordance with written procedures. A full set of Abnormal Operating Procedures and Emergency Operation Procedures have been developed and verified during simulator training. In 1988 symptom oriented emergency procedures were implemented.

26. The plant modification procedure is based on current Western practice that categorises

changes according to their safety relevance. Although the utility has its own engineering and technical support for safety assessment of plant modifications, it also relies on external national and international support. All design modifications are reported in advance to the regulatory body. Once implementation is completed, a documentation update is performed by NEK to reflect the changes.

27. Until now, training and retraining of licensed operators has been performed on simulators in the USA. As result of the availability of a full-scope plant specific simulator on the site the Krško NPP takes full responsibility for operator training in the year 2000.

Safety culture and quality assurance

28. A clear commitment to safety exists at management and staff level. A good management style seems to be in place with motivated and competent managers and staff. The existing stability of the personnel and the average age are good bases to preserve and to improve for the future the way the plant is operated. A proactive approach to safety improvements has been noted and the plant safety status seems to be under control.

29. During the last years the Krško NPP management has shown a policy of openness to international peer review. Relevant efforts have been devoted to accomplish the recommendations formulated by these international missions.

30. The Krško Quality Assurance Programme is implemented according to US requirements (10CFR50, Appendix B) and other international standards.

Operational experience

31. During the last three years, the average number of automatic reactor trips per year was below one.

32. The plant has an Operating Experience Assessment Programme, which analyses events and experiences and provides feedback. There is a specific group in the organisation that addresses these issues. Plant personnel are encouraged to report all in-house deviations and to maintain a correct regard for nuclear safety. The operating experience programme has been reviewed by IAEA and WANO missions. Although the feedback of operating experience programme is sufficiently complete, more contacts by NEK with European utilities would be beneficial.

Emergency preparedness

33. The Krško NPP is responsible for on-site emergency planning and maintaining on-site emergency preparedness. In developing its arrangements, Krško has made reference to IAEA guides and standards, and also to regulations and guides from the US NRC. Slovenia has participated in the INEX exercises organised by the OECD/NEA. In 1999 the on-site emergency plan was upgraded taking into account a new off-site emergency facility and severe accident emergency guidelines.

National industry infrastructure for technical support

34. The utility operating Krško NPP operates only that one unit. The plant engineering department cannot provide all the necessary technical assessment and engineering services, and support is therefore provided by outside organisations. Part of the required technical support can be provided by national organisations such as the Jozef Stefan Institute. Improvements need to continue in the development of plant engineering capabilities.

35. Because of the small size of the utility and the limited availability of national technical support, it is important that the NPP continues to maintain close contact with vendors and utilities associations to keep up with the state of the art and with general improvements in the field of nuclear and radiation safety.

On-site spent fuel and waste management

36. Spent fuel is temporarily stored on-site in a deep pool. At the current annual discharge rate the spent fuel pool capacity will be sufficient up to the year 2003 but NEK has initiated an action plan to increase the pool capacity by partial re-racking and the installation of new racks. This will extend the storage capability until the end of operational life of the plant in 2023 and beyond.

    The low level and intermediate level waste storage at the Krško NPP was designed only for the temporary storage of five years, originally having in mind the provision of designed waste treatment systems. The available storage capacity at the NPP was 90% occupied by the end of 1998. A significant volume reduction of stored solid radioactive has been achieved which will provide enough space for all the waste that is estimated to be produced during the operation life of the plant.

37. For bigger components (steam generators) a new multipurpose building was built for safe storage until the decommissioning of the plant. This building also contains a decontamination area and storage for solid radioactive waste generated during steam generators' replacement.

Conclusions

38. The safety of the Krško plant is comparable with that of NPPs of the same vintage in operation in Western European countries. The safety of the plant has been analysed and is documented in a complete safety analysis report.

39. A continuous improvement programme has been implemented in the past and a large modernisation programme has been recently completed. This includes the replacement of the steam generators, the completion of the fire protection improvements, the on-site verification of the full-scope simulator and the implementation of a severe accident strategy.

40. The utility operates only this single nuclear unit and, being relatively small, needs to continue to maintain contacts with outside organisations in order to receive adequate support.

41. The site organisation and the operational safety practice are comparable to those of Western European NPPs.

42. The following issues needs to be addressed:

- The implications on safety of the long-term ownership and the forthcoming privatisation process of the energy sector need to be carefully evaluated,
- The on-going evaluation of a few issues (e.g. the seismic characterisation of the site, clogging of containment sump, management of hydrogen) needs to be completed,

- Further attention is considered necessary regarding the spent fuel storage, given the residual capability of the pool and the need to license the related modifications,
- Improvements need to continue in the development of plant engineering capabilities to verify the deliveries from design organisations and suppliers. Contacts with vendors and utilities associations are recommended to be maintained and reinforced with particular regard to the relationships with Western European utilities,
- A programme to perform a periodic safety review needs to be finalised.

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ANNEX 1

Generic safety characteristics and safety issues for RBMK plants

Status of safety documentation

1. Western knowledge of the design characteristics of RBMK reactors has increased considerably since the Chernobyl accident in 1986, and especially after 1991 through the IAEA extrabudgetary programme on the safety of RBMK nuclear power plants as well as through other multilateral and bilateral assistance projects. Full in-depth safety analyses, based on independently validated computer codes, according to Western standards, have not yet been completed for any RBMK reactor, although the in-depth safety assessment made for Ignalina NPP, has resulted in much better safety documentation than for the other RBMK reactors. In general, however, Western expert knowledge of these reactors is not as deep as it is for Western designs.

2. The 14 RBMK reactors in operation belong to three different design generations, built to comply with different generations of Soviet safety requirements. There are considerable differences between the different generations of RBMK reactors and even significant differences among reactors within the same generation. It was a conclusion from the 1994 International RBMK Safety Review that, in order to get an accurate assessment of the safety level, it would be essential to perform plant-specific safety studies, including a Probabilistic Safety Assessment, for each reactor using state-of-the-art computer codes and methodologies. However, the basic features of the core design, the reactor cavity and the design of the main circulation circuit are common to all RMBK reactors. This implies that some specific safety issues are common to all units. These issues have been addressed to a varied extent by the RBMK operators.

Confinement issues

3. A most important safety issue regarding mitigation of accidents is the lack of a confinement or the lack of a complete confinement of the main circulation circuit, depending on design generation. A feature, common to all design generations, is that the RMBK reactor core is enclosed in a separate cavity, which is designed to handle serious damage to a very limited number of the 1661 fuel channels. Accident sequences, involving a hypothetical failure of the reactor cavity, would lead to consequential fuel failure and a release of radioactivity through the bypass with possible unacceptable environmental consequences.

4. The more modern RBMK designs have an Accident Localisation System (ALS) consisting of leak-tight compartments enclosing the parts of the main circulation circuit that are considered the most important. These plants have a pressure suppression capacity to deal with a loss of coolant accident within the ALS and well-diversified emergency core cooling systems. However, even in the most modern RBMK designs, the upper sections of the pressure tubes, the steam-water lines, the steam drum separators and the upper parts of the downcomers are not included in the ALS. A rupture in these parts of the main circulation circuit could lead to a loss of coolant that would not be possible to isolate. The first generation designs have no ALS and hence their emergency core cooling systems has fewer lines and fewer pumps, since there is no emergency core cooling re-circulation from a suppression pool. The ALS is tested for leakage every four years according to procedures. Normally the real leakage is much higher than the design leakage, even if there is a

compliance with national regulations on environmental impact. The performance of the ALS has not yet been fully validated in any plant to the extent normally required for Western reactors. It can be concluded that RBMK reactors do not have design features comparable with those required in Western European light water reactors regarding the last barrier for protection of the environment in the case of an accident.

Core characteristics

5. The Chernobyl accident highlighted the original design characteristics of the RBMK core with its positive void coefficient and demonstrated an inefficient performance of the shut down system. After the accident, a series of design changes were agreed by the Soviet authorities aimed at reducing the risk of a reactivity-induced accident. Some of these changes were felt to be so urgent that they were implemented in all plants. It was planned that other changes would wait until the mid-life refurbishment of the reactors. The main design changes involved the reduction of the positive void coefficient, improvements of the reactor protection system and display of the reactivity margin. The decrease in void coefficient was to be achieved by increasing the fuel enrichment from 2% to 2.4% and by the introduction of 80-100 additional absorber rods in the core. In the technical specifications, the number of effective equivalent control rods required in the core was increased from 26 to 48 (operational reactivity margin). These changes have reduced the void effect to less than 1 β. Furthermore the reliability and speed of the shut down system has been improved by modification of the control rod and rod drive design and installation of a new fast acting scram system with 24 rods.

6. These improvements were thoroughly evaluated and monitored in the IAEA Extrabudgetary Programme. It was established during the Programme that the two installed shutdown systems could not be regarded as fully independent and diverse. Furthermore, the fast acting scram system cannot maintain the reactor in a sub-critical state in the event of a loss of coolant accident in the control and protection system channels. Such an accident was considered by the designer as having a too low probability to be considered. However, it was agreed that in order to ensure a satisfactory reliability of the reactor shutdown function, backfitting of an additional completely independent and diverse shut down system was necessary in all RBMK reactors.

7. An important safety issue is related to the operational reactivity margin (ORM) since the ORM has to be controlled in order to maintain the void reactivity coefficient, the effectiveness of the shut down system and the power distribution within the given safety limits. In the original design it was the responsibility of the operator alone to keep the ORM within the safety limits.

8. A new type of fuel with burnable poison has recently been introduced in most RBMK reactors, giving more stable core characteristics, without additional absorbers, and has greatly reduced the need to constantly control the power distribution in the core.

9. The general complexity of the large core, with strong spatially dependent interactions between thermal-hydraulics and neutronics, puts a particular burden on the Instrumentation & Control systems. The need for several localised core control systems requires powerful computing systems to process the necessary operational data for control and protection. Although the situation differs from site to site, a generic safety concern is the status of the main computers in the RBMK plants. At Ignalina NPP, however, the main computer of unit 1 was recently replaced with a new American one and the same replacement is planned for unit 2.

10. Complex 3-D codes are necessary for calculation of core dynamics. Although some efforts have been made to develop 3D tools, further efforts are needed to develop 3D computer codes with adequate thermal-hydraulic feedback to properly take into account the spatial integration of the neutron fields with the fields of temperature and water density in the core.

11. The safety analyses done for the RMBK reactors raise some credibility concerns because the computer programmes used for reactor core transient analysis are not validated as thoroughly as the respective programmes used for light water reactor cores, including VVERs. Also, the calculations are very difficult due to the complicated core structure. Additional validation is necessary and analyses need to be performed of uncertainties in plant data and calculation methods.

Redundancy, diversification and separation of safety systems

12. In the later RBMK designs there is a high redundancy in most of the first line safety systems. This is however not the case to the same extent in the supporting systems, such as the service water and intermediate cooling systems. 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 the evaluation has to be done on a specific basis.

    A major generic safety concern, however, has been the segregation of the electronic systems and the level of diversity in the most important systems and equipment. For example, the flux control system shares many common elements with the shut down system. The emergency core cooling system is actuated by a combination of signals and is not sufficiently assured that the system responds promptly or that the actuation equipment has a low probability of common cause failure. Also the lack of separation in electrical supply and of the emergency core cooling pumps has raised concerns about the sustainability to area events like fire and flooding.

Primary circulation circuit characteristics

13. The specific RMBK core design, consisting of graphite bricks penetrated by 1661 fuel channels and a number of CPS channels, raises a number of issues. The large mass of graphite (2 000 tons) provides a good heat absorbing capacity but has a disadvantage in a severe accident regarding its flammability which was clearly demonstrated at Chernobyl.

14. The primary circuit includes a few large pressure vessels distributing the coolant flow to a number of smaller vessels and to a large number of parallel pipes connecting each core channel. The system also includes a large number of valves. The design of the primary circuit creates some problems:

    - The possibility of blockages, especially blockage of a group distribution header that distributes the flow to about 40 fuel channels. The operating history of RBMK has shown a few blockage incidents, which fortunately did not develop into serious events. Flow blockage is a large contributor to the risk of a severe accident,
    - Material degradation due to the large number of pipes and welds. The RBMK pressure circuit suffers from similar material problems and degradation mechanisms, especially intergranular stress corrosion cracking that have been seen in Western BWRs. A large number of defects have consequently been found in RBMK pipework.

Dynamic effects resulting from double-ended guillotine breaks of large diameter high-energy piping, have not been considered in the design. A successful application of the LBB concept is therefore considered to be of a high priority, where intergranular stress corrosion cracking (IGSCC) can be precluded. Actions to address the IGSCC issue for RMBK plants have been initiated for all operating reactors but are not yet complete. International co-operation to address this issue, which is now under control in Western type BWRs, is regarded as important and urgent. A "break preclusion" programme could be developed, for IGSCC sensitive RBMK piping, which has similar elements as in the LBB concept and such a programme is under way at Ignalina NPP. However a level of safety similar to that provided by LBB could not be reached.

15. The RBMK has certain design advantages over other reactors. For instance, there is about double the water inventory of that in a typical Western BWR, while the fuel ratings are about 75% of those in a BWR and about 60% of those in a Western PWR. These features play a significant role in determining the slow heat-up of fuel in many accident scenarios. On the other hand, the large water inventory means that there is also more stored energy to be handled by the confinement and pressure relief systems.

Gap closure issues

16. A specific RBMK ageing issue is the gas gap closure. The pressure tube in each fuel channel is supported inside the channel in the graphite block by a series of graphite rings. It is arranged so that, at the beginning of plant life, there is a gap of 3 mm between the graphite block and rings. In this gap, a mixture of helium and nitrogen is circulated to improve the heat transfer from the graphite to the coolant and to monitor the tube integrity. Under the influence of irradiation during normal operation, this gap slowly reduces. There is no safety justification for continued reactor operation after the gap has reduced to zero, since this is not allowed by the designer. It is not clear if continued operation will challenge the integrity of the pressure tubes but in any way, it could make re-tubing impossible. The average time to expected gap closure varies upwards from about 15 reactor years, depending on operating conditions and on specific material properties of pressure tubes and graphite blocks at each reactor unit. Mid-life re-tubing was foreseen in the RBMK design and has been carried out at Leningrad units 1 and 2 (partially in units 3 and 4) and in Kursk units 1 and 2.

Operational safety

17. It has been concluded in the international reviews that an upgrade of the operational safety is of utmost importance for the improvement of nuclear safety in the operating RBMK plants. Improvements have been recommended to be implemented in parallel with proposed design related safety improvements. There should be a balanced approach to the allocation of resources to both design and operational safety areas. The main recommendations given in the international reviews are associated with the following:

- Clarification of the management structure including responsibilities, authorities and accountabilities at all levels,
- Development of Quality Management, including independent audits and audits of suppliers. Important issues to improve have been documentation management, plant modification control, investigation of events and experience feedback and improvement of the surveillance and testing of plant functions and components,
- Enhancement of the safety culture including promotion of trust and openness,

qualification improvement, self-evaluation and self-critical attitude. Also improvement of working conditions such as procedures for normal operation and emergencies, equipment labelling, housekeeping, improvement of lighting and the access conditions for operation and maintenance,

- Improvement of the training programmes, facilities and materials, introduction of continuous training and regular training of control room operators on full-scope simulators,
- Improvement of the maintenance planning and control,
- Establishment of an ALARA-programme.

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ANNEX 2

Generic safety characteristics and safety issues for VVER plants

1. The first VVERs were built at Novovoronezh in Russia and at Rheinsberg in Germany. The two units at Novovoronezh were rated at 197 MWe and 336 MWe, and operated between 1964-1988 and 1970-1990, respectively. Rheinsberg was rated at 70 MWe and operated from 1966 to 1990.

2. The first standard series of VVERs has nominal electrical power of 440 MW, and the second standard series has a power of 1000 MW.

3. There are two generations of the VVER-440 MW reactors, which are based on different safety philosophies. Of the older VVER-440/230 generation, there are 11 units still operating, while five have been permanently closed down. Of the second VVER-440/213 generation, there are currently 16 units operating.

4. In addition, two non-standard VVER 440 units have been in operation in Finland since 1977. In the contract for these plants, the Soviet vendor was required to meet Finnish regulations, which were based on US safety rules. The original VVER design was therefore modified by incorporating safety features that provide defence-in-depth against the same type of design basis accidents that are postulated for Western designed plants. The control and protection systems were designed and supplied by Western companies. Many vital mechanical components were also purchased from Western manufacturers. Plant layout, civil structures (including fire protection and ventilation systems), and electrical systems were designed by the engineering staff of the owner utility. Western type QA was applied throughout the construction project, including quality control at the factories within the former USSR.

5. In the VVER 1000 MW series, there is a gradual design development through the five oldest plants, while the rest of the operating plants, the VVER-1000/320s are quite similar to each other. In total there are 20 operating VVER-1000s.

Extent and validation of VVER accident analysis

6. In-depth safety evaluation of VVER-440 plants has been done in a number of countries both in the West and the East. This evaluation includes analysis of postulated transients and accidents with validated computer codes. Accident analysis of the Finnish VVER-440 plants has been carried out since the early 1970s by several Finnish teams and also by a German consultant.

7. The expected behaviour of the VVER-440 reactor core is confirmed by extensive data that has been collected during plant operation. For instance, at the Finnish plants the reactor core instrumentation and monitoring system is among the most comprehensive ones in the world's power reactors and accurate records exist from 43 reactor years of operation. In recent years, advanced monitoring systems with frequent automatic calculation of all important core parameters have been installed at many VVER-440 reactors. Studies of fuel rods irradiated at the Finnish reactors have confirmed the predicted fuel properties. These studies include hot cell investigations conducted in Sweden and tests to study fuel response to fast power transients, conducted both in the OECD Halden reactor in Norway, and in

the Studsvik reactor in Sweden. The ability to calculate the behaviour of the nuclear steam supply system during normal operation and small transients (such as reactor trip, reactor coolant pump trip and loss of feedwater) has been verified in extensive commissioning tests and by analysis of operational events.

8. The validation of accident analysis codes for VVERs has been carried out by several organisations in different countries since the mid-1970s. In addition to the generic data available from international standard tests, integral experiments conducted at VVER-specific thermal-hydraulic test facilities such as REWET and PACTEL in Finland and PMK in Hungary have also provided data for this validation.

9. The most recent comprehensive analysis of the Finnish VVER-440 plants was done in connection with periodic re-licensing in 1997, using a validated state-of-the-art computer code package. Independent calculations for verification of the analysis were done by the Finnish regulator and its consultants. Similar analysis has been done for other VVER-440s by competent teams in particular in Hungary and the Slovak Republic.

10. The transient and accident behaviour of the VVER-1000 reactor has also been investigated quite extensively. For instance, a feasibility study on the licensability of an improved VVER-1000 design was done in Finland in 1992. It included a full scope analysis of postulated design basis events. The analysis was updated by a Finnish team in 1995 to support an application to build a similar plant in China.

11. Other VVER-1000 analysis has been done by Western experts for instance in Germany for a plant, which was never completed, and for the Temelin plant under construction in the Czech Republic.

12. Operational experience analyses done by national regulators and reported widely by the VVER Regulators' Co-operation Forum have further improved the understanding of these plant types.

13. In conclusion, the accident analysis of both VVER-440 and VVER-1000 designs is considered sufficient to provide an adequate understanding of the generic safety characteristics of the plants.

VVER-440/230

14. In the EU candidate countries, there are six nuclear power plant units of this type: four in Bulgaria (Kozloduy 1-4) and two in Slovakia (Bohunice V1, units 1-2).

15. The design of the VVER-440/230 was based on the exclusion of a double-ended guillotine break of the main circulation line or the pressurizer surge line in the reactor cooling system. Instead, the accident assumed as the design basis for the safety systems was a break of a pipe directly connected to the main circulation lines. Following this basic assumption, all pipe joints to the main circulation lines were equipped with throttling devices. These would limit the maximum leak rate from any broken pipe that directly joins the primary circuit to the equivalent of a 32-mm diameter break. This size of leak was the basis for designing VVER-440/230 safety systems, and consequently the capacity of the originally installed emergency core cooling systems was very small. It also meant that the design did not feature a substantial, Western-style, containment around the reactor cooling system to limit potential radioactive releases in a medium or large break loss of coolant accident. The as-built confinement system of VVER-440/230s had little overpressure capability and its leak-

tightness characteristics were very poor. On the other hand, in post-accident conditions, this confinement would operate under sub-atmospheric pressure for a significant period of time, which of course would reduce releases in design basis accidents.

16. Although a large break in a reactor coolant circuit has never occurred at any nuclear power plant, a large break LOCA is generally postulated as a design basis for safety systems in existing Western nuclear power plants.

17. In addition to the limitations in the core cooling and confinement capability, the VVER-440/230 plants had two other major safety concerns:

- Internal hazards such as fires or floods, and external hazards such as seismic events or aeroplane crash, were not adequately considered in the original design. Thus the redundant parts of the safety systems were not adequately separated from each other, and were vulnerable to common cause failures. Some important safety systems were installed close to high-energy systems or in high fire risk areas (e.g. the turbine hall). Consequently, an event in one part of the plant could have resulted in complete loss of vital safety functions,
- The auxiliary systems, such as electrical power supply or cooling systems, which support the safety functions, were designed with inadequate redundancy. Consequently, a single failure in a critical component of an auxiliary system could have resulted in the loss of that support function and thus also a loss of the main safety functions.

18. Some additional safety concerns are common to all VVER plants being operated in the EU candidate countries:

- The original quality of electrical equipment and instrumentation & control equipment was inadequate, and the equipment was not qualified to function in accident conditions,
- The reactor pressure vessel wall is exposed to higher irradiation by fast neutrons than most Western designed reactor pressure vessels, and therefore the embrittlement of the vessel material proceeds more quickly,
- The design of the main barrier between primary and secondary coolant inside the steam generators (primary collector) is less robust than the tube sheet in Western PWRs, and the possibility of large primary to secondary circuit leak therefore needs to be taken into account in the design of the safety systems. For instance, primary to secondary leaks occurred in several steam generators during an event at Rovno NPP unit 1 in 1982. Steam generator primary collector covers broke off one after another as a consequence of careless operation and negligent maintenance. Despite very large leaks the Rovno accident developed slowly enough to allow operator intervention to prevent any core damage.

19. The safety concerns with VVER-440/230 plants are discussed in detail in an IAEA report [1]. All the plants have addressed these concerns to various degrees by backfitting and design changes.

20. When assessing the overall safety of the VVER-440/230 plants, it should be noted that, like all VVER-440s, they have certain inherent safety characteristics that are superior to most modern LWR plants. The principal safety characteristic of all VVER-440 plants is the large volume of coolant both in the primary and the secondary side. These reactors have more than twice as much coolant per megawatt as any Western designed NPP. These large coolant volumes and low core power density mitigate any anticipated transients so that the plant response to transients is very smooth. For instance, in all anticipated transients, the primary pressure stays well below the opening set point of the safety valves, and large

safety margins to heat transfer crisis in the reactor core remain. Large coolant inventories also allow multiple failures to occur without damage to the reactor core, e.g. interruption of all AC power supply to plant equipment for several hours, or a complete loss of heat sink for a similar time. This robustness is demonstrated by operational experience from two major blackout incidents, the Greifswald 1 fire in 1975 and Armenia 1 fire in 1982. These safety features provide an inherent protection, far more extensive than typical for Western LWRs, against the possible escalation of transients to more severe events.

21. Other significant inherent safety features are:

- Small and robust reactor core: any oscillations in spatial power distribution quickly die away, and do not require active control as in larger reactor cores,
- Low peak fuel temperatures with good retention of fission gases within the ceramic fuel pellets,
- Low heat flux from the fuel to the coolant giving a very large margin to critical heat flux in normal operation and during abnormal transients, and slow temperature increase in loss of coolant accidents,
- Robust design of main components and piping, including the main circulation lines of the reactor cooling system, which are made of austenitic stainless steel,
- The ability to isolate any failed loop of the reactor cooling system, and after isolation to bring the plant to safe shutdown using normal operating procedures,
- Risks concerning most of the Western PWRs during outages, caused by a temporarily reduced coolant inventory, are excluded in a VVER-440 because there is no need to decrease the water level in the primary circuit during a refuelling or maintenance outage. As the residual heat removal system is connected to the secondary side only, the likelihood of leaks in general and interfacing LOCA in particular are reduced compared to Western designs.

22. The following conclusions can be made regarding the safety of VVER-440/230 plants:

- The original plant design had inadequate systems to cope with accidents that are postulated as design basis of the Westerns PWRs. Due to this reason and also due to other concerns explained above, its safety was not acceptable to Western European standards,
- However, all VVER-440/230s being currently operated in candidate states have been significantly modified to varying degrees as compared to the original design. Where the modifications have been pursued most vigorously, a new design basis accident set has been defined to include up to 200-mm breaks, the emergency core cooling systems and confinement capability have been improved to deal with it, breaks beyond the new DBA have been ruled out by implementation of leak-before-break arguments, and confinement leak tightness has been improved by up to two orders of magnitude. The generic safety issues identified by the IAEA have been addressed to varying degrees at all the plants,
- It has been shown that it is possible to remove most of the safety concerns by refurbishment and backfitting, but it requires a major investment. However it does not appear feasible to backfit the plants with a reactor containment that could provide similar protection to the containments of modern Western PWRs,
- As to the original design requirement of Western PWRs for containment to keep the radioactive releases in connection with a large break LOCA and other design accidents below a specific limit, it seems possible to also meet this target at VVER-440/230s by a combination of upgrading of the existing confinement and installation of other supporting systems,

- The difference between the VVER-440/230 confinement and the Western style containment becomes more evident when analysing the capability to prevent the releases after severe reactor core damage. Although containment buildings of Western PWRs were not designed to cope with severe accidents, they can provide good protection by limiting releases to the environment in many of the investigated accident scenarios. This was well demonstrated in the TMI accident in 1979 in the USA. Conversely the VVER confinement system relies on systems to provide sub-atmospheric conditions to prevent leakage of radioactivity following core damage,
- The proven inherent safety margins and moderate response in connection with potential design basis accidents of the VVER-440 partly compensate the remaining shortcomings of an adequately upgraded confinement structure. There are parallels with the arguments used in the Western Europe to justify the lack of LWR-like containment in gas-cooled reactors. When the transients and accidents caused by equipment failures are less severe, and their rate of progress is relatively slow compared with Western PWRs, the operators have more time to take corrective actions. Although it is difficult to quantify the safety gain associated with this, there is ample operational experience that demonstrates its value. Worth noting are the two very severe fires that resulted in several hours loss of all key safety functions, 1975 in Greifswald and 1982 in Armenia, without resulting in the release of any radioactivity to the environment.

VVER-440/213

23. In the EU candidate countries, there are 12 nuclear power plant units of this type, four in Hungary, four in the Czech Republic, and four in Slovakia.

24. The accidents used as a design basis for the VVER-440/213 safety systems are similar to those postulated in Western plants, including a double-ended guillotine break of the main circulation line in the reactor coolant system. The safety systems are quite similar to those in Western PWRs. Mostly, they consist of three redundant parts, and any one of those parts can provide the intended safety function. This goes beyond many Western designed plants, which have only two redundant parts in their safety systems.

25. VVER-440/213 reactors have bubbler condenser type pressure suppression containments that in principle closely resemble Western boiling water reactor containments. The bubbler condenser is a unique Soviet design. Although its performance during design basis accidents had been studied analytically and with model tests both in the former USSR and in the Eastern European countries, there was a common desire among the international nuclear safety expert community to confirm the results with additional large-scale and separate effects tests. Large Break LOCA tests were conducted with Western support as the Bubbler Condenser Experimental Qualification Project sponsored by the EU. These included full-scale tests at facilities built in Russia, and complementary tests in the Czech and Slovak Republics. The test and analysis results were reported in the early 2000, and provide the necessary experimental evidence that the bubbler condenser is capable of withstanding the imposed loads and maintaining its functionality following a Large Break LOCA. An in-depth assessment of the reported results by independent safety organisations is in preparation, and also performance of large scale experiments for Steam Line Break and Small Break LOCAs, with corresponding pre- and post-test calculations, has been suggested. These would be required to increase confidence in the bubbler condenser performance in all accident conditions.

26. Another concern has been the containment leak-tightness. The leak rates measured in integral tests of containments, in their initially constructed conditions, were clearly higher at

some plants than is allowed for Western containments. Improvements in leak-tightness have now been achieved at all plants, although large variation among the plants is evident. However, it should be noted that comparison with Western containments is not straightforward because, in connection with the design basis accidents, the pressure suppression system tends to cause underpressure rather than overpressure at the time period when the atmosphere of the containment has its highest contents of radioactive aerosols, and when the potential for radioactive releases would thus be the highest. The behaviour of the bubbler condenser containment under severe accident conditions has not been investigated.

27. Compared with other major safety concerns of the older VVER-440/230 plants (cf. § 17), design improvements include:

- Internal and external hazards have been addressed to various degrees on a plant specific basis, and there are major design differences between the plants. There may still be some plant-specific concerns in this area, but to a lesser extent than for the VVER-440/230s,
- Protection against single failures in the auxiliary and safety systems has generally been provided by design, although improvements in detail have been required as a backfitting measure.

28. The safety concerns with VVER-440/213 plants are discussed in detail in an IAEA report and in a German safety evaluation [2], [4]. Most of these concerns have been addressed on a plant specific basis.

29. All the inherent safety characteristics discussed in connection with VVER-440/230 plants (see § 20 and 21) are equally valid for the VVER-440/213 type. Extensive model testing and safety analysis has been done in several countries, including recent analyses with state-of-the-art computer codes. These analyses have confirmed the safe behaviour of the reactor core and its cooling system in all abnormal transients. Furthermore, it has been confirmed that these plants can be brought to safe shutdown in connection with the accidents that are generally assumed as design basis events for modern nuclear power plants.

30. The following conclusions can be made regarding the safety of VVER-440/213 plants:

- The original safety targets set for the plant design were quite similar to Western European standards at the time when most plants in operation to day within the EU were constructed. However, the implementation of the design failed to pay enough attention to details, and several safety deficiencies could be identified in safety analyses done later on. Also the quality of some equipment did not properly correspond their safety importance. At all the plants, most of the safety deficiencies have been addressed by backfitting and plant modifications,
- A general issue that needed specific studies was the performance of the reactor containment during design basis accidents. Large scale Large Break LOCA tests were conducted with Western support as a joint industrial project. The test and analysis results were reported in early 2000 and are being assessed in depth. The performance of large scale experiments for Steam Line Break and Small Break LOCAs, with corresponding pre- and post-test calculations, is still required,
- Due to the robust original design, it is quite straightforward to upgrade the safety of the original VVER-440/213 design to a level comparable with the plants currently operating in Western Europe. The safety issues that need to be addressed have been identified by the

IAEA,

- As concerns protection against severe accidents that were not part of the original design basis of any of the operating Western PWR nor a VVER, the situation is as in the case of VVER-440/230 design discussed in § 22 above: containment capability to limit releases is expected to be somewhat inferior to the Western PWR containments, but much better than in VVER-440/230s. The inherent safety features compensate this shortcoming to a considerable extent.

VVER-1000/320

31. In the EU candidate countries there are two nuclear power plant units of this type in operation, both of them in Bulgaria. In the Czech Republic, two further units are being built that were originally of a similar design but have been extensively upgraded during construction.

32. The VVER-1000 plants were designed to similar safety requirements as Western plants and have equivalent safety systems. However, compared to the VVER-440/213 plants, the overall safety level of the VVER-1000 plants seems to be lower. The reason is that the higher power VVER-1000 plants have lost nearly all the inherent safety features of the smaller VVER-440 plants.

33. The main safety concern regarding the VVER-1000 plants lies with the quality and reliability of individual equipment, especially with the instrumentation and control equipment. Also the embrittlement of the reactor pressure vessel needs continuous attention and action will need to be taken if it approaches a hazardous level.

34. The main barrier between primary and secondary coolant inside the steam generators is a greater safety concern than in the VVER-440 plants, and it has been necessary to replace a number of steam generators when failures have been observed in this barrier. It remains to be demonstrated by further successful operating experience that design and manufacturing method improvements have solved these problems.

35. The plant layout has weaknesses that make the redundant system parts vulnerable to hazardous systems interactions and common cause failures caused by fires, internal floods or external hazards.

36. The safety concerns about the VVER-1000 plants are discussed in detail in an IAEA report [3], [5].

37. The following conclusions can be made regarding the safety of VVER-1000 plants:

- The original plant design had deficiencies, which would not be acceptable by Western European standards. At all plants, many of these deficiencies have been addressed by backfitting and plant modifications,
- It is feasible to upgrade the safety of the VVER-1000 plants to a level comparable with many of the plants being operated in Western Europe. This upgrading should adequately address all the safety issues identified by the IAEA.

References

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