Safety of RBMK reactors: Setting the technical framework

The IAEA's co-operative programme is consolidating the technical basis for further upgrading the safety of Chernobyl-type reactors

by
Luis Lederman

In April 1986, unit 4 of the Chernobyl nuclear power plant in Ukraine was destroyed in the worst accident in the history of commercial nuclear power. The reactor, which started operation in 1983, was a Soviet-designed nuclear power plant known by the Russian acronym RBMK.

The RBMK evolved from Soviet uranium-graphite reactors whose purpose was the production of plutonium. The first of these plutonium production reactors began operation in 1948. Six years later, in 1954, a demonstration 5-MWe RBMK-type reactor for electricity generation began operation in Obninsk. Subsequently a series of RBMKs were developed using the combination of graphite moderation and water cooling in a channel design.

Today 15 RBMK power reactors are producing electricity in three States: 11 units in Russia, two in Ukraine, and two in Lithuania. The gross electric power rating of all but two RBMKs is 1000 MWe; the exceptions are the two units at Ignalina in Lithuania which are rated at 1300 MWe gross.

All operating RBMKs were connected to electric power grids during the period 1973 (Leningrad-1) to 1990 (Smolensk-3). They represent distinct generations of reactors having significant differences with respect to their safety design features.

Six plants are considered "first-generation" units (Leningrad-1 and -2, Kursk-1 and -2, and Chernobyl-1 and -2). They were designed and brought on line in the early-to-mid 1970s, before new standards on the design and construction of nuclear power plants (OPB-82) were introduced in the Soviet Union. Units brought on line since the late 1970s and early 1980s are generally grouped as "second-generation" RBMKs (Leningrad-3 and -4; Kursk-3, and -4; Ignalina-1; Chernobyl-3; and Smolensk-1 and -2). Ignalina-2 contains safety features beyond those of other second generation units. These RBMKs were designed and constructed in accordance with the updated standards issued in 1982.

After the Chernobyl accident, Soviet safety standards were revised again (OPB-88). One RBMK (Smolensk-3) has been built to these "third-generation" standards. Additional design changes now are being incorporated in the construction of Kursk-5.

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

This article reviews major efforts for improving the safety of RBMK reactors through a co-operative IAEA programme initiated in 1992. *(See box, page 12.)* Specifically covered are technical findings of safety reviews related to the design and operation of the plants, and the documentation of findings through an Agency database intended to facilitate the technical co-ordination of ongoing national and international efforts for improving RBMK safety.

Scope of the RBMK safety programme

The IAEA's RBMK safety programme aims to consolidate results of various national, bilateral, and multilateral activities and to establish international consensus on required safety improvements and related priorities. It assists both regulatory and operating organizations and provides a basis for technical and financial decisions. A wide range of activities are covered, and since 1992, a number of reviews and assessments have been conducted. Smolensk-3 and Ignalina-2

Mr. Lederman is Acting Head of the Safety Assessment Section in the IAEA Department of Nuclear Safety.

RBMK nuclear power plants

Operating units:

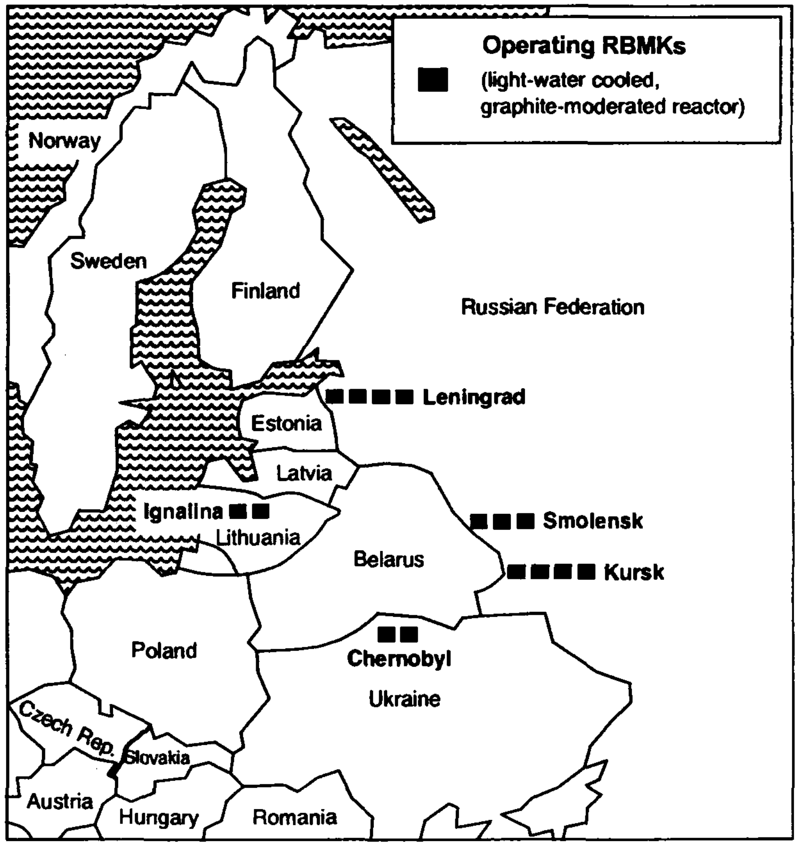
Lithuania. Two 1300-MWe units at Ignalina. *Start of commercial operation:* Ignalina-1 in 1984; Ignalina-2 in 1987

Russian Federation. Eleven 1000-MWe units: four units at Kursk; four units at Leningrad; three units at Smolensk. *Start of commercial operation:* Kursk-1 in 1977; Kursk-2 in 1979; Kursk-3 in 1984; Kursk-4 in 1986. Leningrad-1 in 1974; Leningrad-2 in 1976; Leningrad-3 in 1980; Leningrad-4 in 1981. Smolensk-1 in 1983; Smolensk-2 in 1985; Smolensk-3 in 1990.

Ukraine. Two units at Chernobyl. *Start of commercial operation:* Chernobyl-1 (780-MWe) in 1978; Chernobyl-3 (1000-MWe) in 1982. *Notes:* Chernobyl-2 shut down since 1991; Chernobyl-4 destroyed in April 1986 accident.

Units under construction:

Russian Federation. Kursk-5, a 1000-MWe unit whose construction began in December 1985.



have served as RBMK reference plants during the programme's first phase.

The IAEA conducted a first review of safety improvements proposed for RBMKs in October 1992. In June 1993, a safety assessment of design solutions and proposed safety improvements of Smolensk-3 was organized. It was conducted by an international group of experts and IAEA staff over a period of 2 weeks at the plant site. Smolensk-3 is the most advanced of the operating RBMK plants and its design incorporates safety improvements identified from analyses of the Chernobyl accident and other studies. A similar review was performed for the Ignalina units in October 1994.

Additionally, the Agency's Assessment of Safety Significant Event Teams (ASSET) have reviewed the plant-specific operational experience at all RBMK sites. An Operational Safety Review Team (OSART) mission also was conducted at Ignalina in September 1995.

Experts also have reviewed the design of plant shutdown systems at Smolensk-3. The review — the subject of an IAEA consultants' meeting in Switzerland in December 1993 — was based on IAEA Nuclear Safety Standards (NUSS) documents; national standards (Russia, Canada, and Germany); and regulatory practices of the Organization for Economic Co-operation and Development (OECD). International experts fully supported the intent of Russian designers to improve the RBMK shutdown systems.

Another matter receiving attention has been the analysis of a multiple pressure tube rupture in RBMK-type reactors. At a topical meeting in Moscow in January 1994, participants examined the relevant regulatory approaches adopted in Member States operating channel-type reactors and reviewed the methodology, criteria, and results from safety analyses. They agreed on the urgent need to validate computer codes used for studying loss-of-coolant accident (LOCA) scenarios in RBMKs. In November 1994, at an IAEA consultants' meeting in Japan, a validation matrix for code calculation was established, and in 1995 the IAEA started an international exercise based on experimental results made available by Japan.

Activities of the IAEA's RBMK safety programme are co-ordinated with those of an international consortium on the "Safety of Design Solutions and Operation of Nuclear Power Plants with RBMK Reactors" established under auspices of the European Commission. The two programmes use the same RBMK reference plants.

With the completion in 1994 of the safety reviews of Smolensk-3 and Ignalina-2, these two programmes reached important milestones. To

RBMK Safety Programme

At the start of the 1990s, the IAEA initiated a programme to assist countries of Central and Eastern Europe and the former Soviet Union in evaluating the safety of their first generation WWER-440/230 nuclear power plants. The main objectives were: to identify major design and operational safety issues; to establish international consensus on priorities for safety improvements; and to provide assistance in the review of the completeness and adequacy of safety improvement programmes. The programme's scope was extended in 1992 to include RBMK, WWER-440/213, and WWER-1000 plants in operation and under construction. It is complemented by national and regional technical co-operation projects.

Programme elements include plant-specific safety reviews to assess the adequacy of design and operational practices; reviews under the IAEA's Assessment of Safety Significant Events Team (ASSET) service; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Additionally, follow-up safety missions are conducted to nuclear plants to check the status of IAEA recommendations; assessments are made of safety improvements implemented or proposed; peer reviews of safety studies are performed; assistance is provided to strengthen regulatory authorities; and training workshops are organized. A database also is maintained on technical safety issues identified for each plant and the status of safety improvements.

As an extra-budgetary programme, activities depend on voluntary contributions from IAEA Member States. Steering Committees provide co-ordination and guidance on technical matters and serve as forums for exchange of information with the European Commission (EC) and with other international and financial organizations. The programme takes into account the results of relevant national, bilateral, and multilateral activities, and thus provides a framework in which an international consensus can be reached on the technical basis for upgrading the safety of WWER and RBMK nuclear power plants. The IAEA further provides technical advice within the co-ordination structure established by the Group of 24 OECD countries through the European Commission.

The programme's results, recommendations, and conclusions are only intended to assist national decision-makers who have sole responsibility for the regulation, upgrading, and safe operation of their nuclear power plants. They facilitate but do not replace the need for comprehensive safety assessments in the framework of the national licensing process.

make results available to the international technical community, the IAEA convened a technical meeting in May-June 1995. Results from both the IAEA and EC programmes were presented, thereby reflecting the large amount of work done by the international experts and Russian organizations to review the safety of RBMK nuclear power plants.

Both projects produced a large number of recommendations for enhancing the safety of RBMK plants. Most of them correlate with the measures already included in national programmes for RBMK units which are under way in Russia, Lithuania, and Ukraine.

Based on the initial phase of its programme, the IAEA prepared a consolidated list of design and operational safety issues for RBMKs. For this work, a database of findings and recommendations for RBMKs compiled by the IAEA was used. All findings and recommendations from the various technical meetings, safety reviews of Smolensk and Ignalina, and ASSET reports of the EC's International Consortium were collected in the database and further grouped by topical areas into safety issues. Also included is plant-specific safety information provided by the main design institute for RBMK reactors in Moscow.

The Agency's database has an interface with the databank established by the G-24 Nuclear Safety Co-ordination Group, thus making joint analyses of safety topics and assistance projects easier.

Results from the IAEA programme

The IAEA's programme identifies 58 RBMK safety issues related to seven topical areas. Issues related to six design areas are further ranked according to their perceived impact on plant safety. Safety issues connected to operational areas, particularly those related to the assurance of a high degree of safety culture, are all considered very important.

The ranking of a safety issue does not necessarily imply that all the proposed recommendations have the same urgency for implementation. Therefore, recommendations have to be further considered on a plant-specific basis.

Two broad issues addressing quality assurance (QA) and regulatory matters are not specifically attributed to any particular topical area, but they are recognized as affecting all areas. From the QA standpoint, the main concern relates to ensuring the use of the actual plant status and configuration for various analyses, safety reviews, and safety improvements. Another aspect is ensuring that the relevant design documentation

RBMK Technical Overview

The reactor core of an RBMK is constructed of closely packed graphite blocks stacked into columns and provided with axial openings. Most of the openings contain fuel channels. Some also serve other purposes (e.g. instrumentation and control) and are called "special channels". The graphite stack is contained within a cylindrical steel vessel, 14 meters in diameter, which acts as a support for the graphite stack and as a container for the helium-nitrogen gas mixture.

The total mass of the graphite within the core is 1700 tons. About 6% of the reactor's thermal energy is generated in the graphite stack. The helium-nitrogen mixture improves the heat transfer from the graphite to the channels, protects the graphite from oxidation at its operating temperature of about 650° C and, through gas sampling, forms part of the integrity monitoring system.

There are 1661 fuel channels in the vertical ducts of the graphite columns; these channels are tubes 88 millimeters in diameter made of a zirconium niobium alloy. Each fuel channel contains two fuel assemblies, one above the other, each of them containing 18 fuel rods that are 13.6 millimeters in diameter, enclosed in the zirconium niobium cladding. The total fuel length of the core is about 7 meters.

Number and Category of RBMK Safety Issues

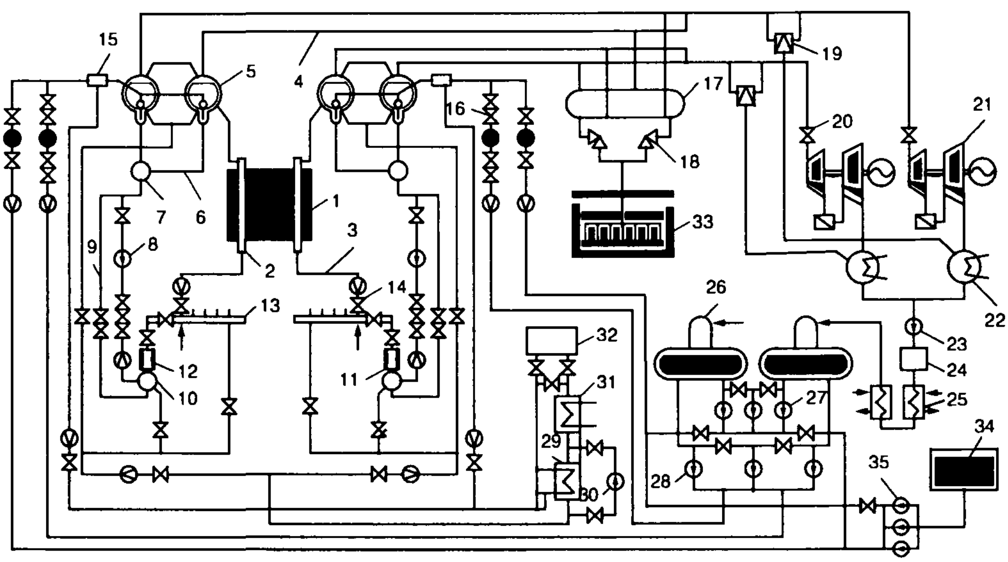
Topical area	Number of safety issues identified	Number of safety issues in category		
		High	Medium	Low
Core design and core monitoring	6	5	1	0
Instrumentation and control	7	2	5	0
Pressure boundary integrity	7	4	2	1
Accident analysis	10	3	7	0
Safety and support systems	10	4	6	0
Fire protection	5	1	3	1
Operational safety	13*	—	—	—
Total	58	19	24	2

* Not ranked, but considered very important and improvements should be implemented in parallel with design modifications.

High: Issues that reflect insufficient defense-in-depth and have a major impact on plant safety. Short-term actions have to be initiated to improve safety as applicable to each specific nuclear power plant until the issue is fully resolved.

Medium: Issues that reflect insufficient defense-in-depth and have a significant impact on plant safety. Short-term actions might be necessary to improve safety as applicable to each nuclear power plant until the issue is fully resolved.

Low: Issues that reflect insufficient defense-in-depth and have a small impact on plant safety. Actions are desirable to improve defense-in-depth, if applicable and effective from a cost-benefit point of view.



1 Reactor
2 Fuel channel
3 Water pipelines
4 Steam pipelines
5 Steam separator
6 Downcomer
7 MCP suction header
8 Main circulation pump
9 MCP header bypass
10 MCP pressure header
11 Mechanical filter
12 Flow limiter
13 Group distribution header
14 Isolation and control valve
15 Mixer
16 Feedwater valve assembly
17 Steam headers
18 Main relief valve (MRV)
19 Steam dump valve
20 Turbine trip valve
21 Turbogenerator
22 Condenser
23 Condensate pump
24 Condensate purification
25 Heater
26 Deaerator
27 Auxiliary feedwater pump
28 Feedwater pump
29 Blowdown regenerator
30 Cooldown pump
31 Blowdown afterheat
32 Bypass purification
33 ALS pool
34 Emergency water tank
35 Emergency feedwater pump

RBMK Basic Design and Safety Improvements

Major safety improvements have been implemented in RBMKs since the Chernobyl accident in April 1986. They address both the direct causes of the accident and other safety shortcomings which have been identified in various analyses.

Reactor Core. Safety modifications directly related to the Chernobyl accident focus on reducing the void reactivity coefficient and improving the control rod design. These modifications have been implemented in all RBMKs. Main measures taken to reduce the void coefficient include:

- loading additional absorbers. The number of additional absorbers varies from 85 to 103 depending on the reactor. Technical specifications require at least 81 additional absorbers.
- increasing the fuel enrichment from 2.0% to 2.4%.
- controlling the operational reactivity margin (ORM). The ORM fully inserted value is controlled between 43 and 48 equivalent control rods.

Emergency Protection System (EPS). Three safety improvements were carried out to improve the EPS efficiency and speed of response. The manual control rods were replaced by rods of an improved design. This includes elimination of the water column at the bottom end of the reactor control and protection system channels, and increasing the neutron absorbing section. The rod drives were also modified, reducing the time required to insert the rods fully into the core from 19 to 12 seconds. These two measures have improved the EPS response efficiency during the first few seconds of rod insertion. As a third measure, a fast-acting EPS was developed and installed in all operating reactors. This system can fully insert 24 control rods in less than 2.5 seconds, or in 7 seconds depending on the emergency signal activated.

Control and Monitoring System. Other measures were taken to improve the control and monitoring system. They include manual reactor trip when the power falls below 700 MW(th); and manual trip if the ORM is less than 30 equivalent control rods.

Pressure Boundary. Two independent loops provide cooling for each half of the reactor core. Each loop contains four main coolant pumps and associated piping. The pressure in the system is 7 MPa.

Emergency Core Cooling System (ECCS). For Smolensk-3, the design basis accident for the ECCS is a double-ended guillotine break of a 900 mm tube and loss of off-site electric power. This corresponds to a break in the main circulation pump pressure headers or suction header. In the event of such an accident, the ECCS makes provision for both fast-acting cooling of the core and long-term decay heat removal. The long-term cooling system comprises six emergency core cooling pumps taking suction from the accident localization system (ALS) for cooling the damaged half of reactor and three pumps taking suction from the tanks for pure condensate for cooling the non-damaged half of the reactor. Both sets of pumps are electrically driven with their power supplies backed up by diesel generators.

Design modifications not related to the causes of the Chernobyl accident are being introduced at first and second generations RBMK plants. Among other steps, these include: increasing the number of emergency feedwater pumps from three to five and the number of ECCS lines from one to two; installing additional ECCS pumps (three for cooling the damaged core side and three for cooling the undamaged side) and the associated three divisions of piping; installing check valves between the distribution group headers and the main coolant pump discharge header; and installing large capacity accumulators.

Accident Localization System. RBMKs are protected by an Accident Localization System (ALS). This pressure suppression system encloses part of the main circulation circuit and consists of leak-tight compartments. All main pipelines, headers, and components carrying cooling water are part of the ALS. The ALS differs considerably in design from one plant to another. The reactor coolant system of first-generation RBMKs is not enclosed in a leak-tight ALS, as is the case at the other RBMKs. Even in these other RBMKs, however, only part of the reactor coolant circuit is confined by an ALS of pressure compartments.

Reactor Cavity Overpressure Protection System (RCOPS). This is an important part of the RBMK's safety system. The cause of overpressurization is postulated to be a failure of pressure tubes inside the reactor cavity. Relief is provided by tubes which connect the reactor cavity to the ALS via a water lock. The design basis accident of RBMK safety analyses is the rupture of one tube. The system has the capacity for two or three channel tube ruptures (for first and second generation units, respectively) which reflects a safety margin over the design basis accident. For first-generation units, the steam discharge system vents the steam/gas mixture from the cavity to a condenser, with the gas subsequently held up and released through the stack.

To improve the capacity of the RCOPS, work is being conducted in stages for all units. At Smolensk-3, the existing system already has the capacity for the simultaneous rupture of up to nine pressure tubes under conservative assumptions of simultaneous ruptures.

is updated as the plant configuration is modified and upgraded. It is therefore of utmost importance that the organizational structure promotes awareness of safety concerns, responds quickly in evaluating these concerns, and implements timely corrective actions if they are warranted.

Exactly how and when identified safety issues are addressed is a matter to be resolved between the operating organization and the regulatory body. The IAEA safety reviews are intended to help by providing international expertise to assist in this process. The reviews draw upon the IAEA's NUSS publications, the Russian regulations, and national practices. Recommendations and conclusions are only intended to provide an additional technical basis for decisions to improve the safety of RBMKs. National authorities have sole responsibility for the regulation and safe operation of their nuclear power plants. Therefore, the results do not replace a comprehensive safety assessment which needs to be performed in the frame of the national licensing process.

Overview of Technical Findings

Core Design and Core Monitoring. The direct causes of the Chernobyl accident were related to the reactor core design. Therefore, safety improvements have been initially focused on identified shortcomings related to core physics.

To date, considerable work has been completed to decrease the core void reactivity coefficient and to increase the efficiency of the shutdown system. However, important issues remain to be resolved. They include the problem of the void reactivity associated with the loss of coolant from channels of the control and protection system (CPS); and the issue of independent and diverse reactor shutdown. International experts thus strongly support the intent of Russian designers to develop and modernize the RBMK CPS to provide a higher safety level.

Another issue of high significance to safety relates to the operational reactivity margin (ORM). The ORM has to be controlled in order to maintain the void reactivity coefficient, the effectiveness of the shutdown system, and the power distribution within defined safety limits. With the present design, it is the responsibility of the operator alone to keep the ORM within the corresponding limits. IAEA programme experts have recommended the automation of shutdown actions when the ORM value falls below the safety limits.

Other aspects of importance concern the process of analyzing RBMK design and safety. Such analyses have been done using calculational tools available at that time. These tools (e.g. computer codes) generally did not have the capability to adequately model spatial interactions between neutronics and thermohydraulics. Consequently, efforts are being directed at the development of three-dimensional methods for analyzing neutron fields, coolant density, and temperature distribution of fuel and graphite. IAEA programme experts have recommended that these methods be used to confirm results of previous safety analyses as well as for further studies.

Instrumentation and Control (I&C). The major concerns related to the segregation between the electronic systems and the level of diversity present in the most important systems and equipment. For example, the flux control system shares many common elements with the shutdown system. Although there is considerable resilience in the system due to the high level of redundancy, the two systems would be vulnerable to common mode failure, and thus control and protection could be lost simultaneously. The Emergency Core Cooling System (ECCS) is initiated by a combination of signals. However, there is no sufficient assurance that the system responds promptly nor that the actuation equipment is designed against single failures.

A safety issue ranked as "medium" is the replacement of the station's main computers. The situation differs from site to site. The equipment in Ignalina, for example, is showing distinct signs of ageing with thermally induced warping of boards and embrittlement of the plastic edge connectors.

A number of other issues related to I&C could be improved by local measures at the plant. These include test and maintenance procedures and the recording and use of failure data.

Pressure Boundary Integrity. Some primary coolant circuit components and piping are outside of the accident localization system. In first-generation RBMKs, a guillotine break in the piping can result in damage to civil structures. Application of the leak-before-break concept would reduce the risk of primary coolant circuit failures. Work is going on to demonstrate the applicability of this concept for RBMK conditions and to implement the method and techniques.

To date, there have been three single-channel ruptures due to the blockage of water flow or an imbalance in the flow-to-power ratio. Recommendations therefore call for analyzing and implementing, as feasible, the reduction of the number of in-line components, whose failure can block water flow.

The IAEA programme's safety reviews indicate that at some plants operation has continued even though the frequency and the number of examinations required by national regulations for the reactor pressure boundary are not performed, or when the results have not been satisfactory. The existing time schedules for implementing modifications, performing additional analysis, and maintaining required record-keeping are sometimes not followed. Criteria for limiting plant operation in these cases are not established.

The required volume of in-service inspection (ISI) is not fulfilled in practice. It was found that in some cases the required number of fuel channels was not inspected. The approach adopted at RBMK plants to repair identified critical defects differs from the predictive approach adopted for ISI elsewhere. Pre-service inspection records and ISI predictive records are not maintained. The existing equipment and procedures are inadequate to give reproducible measurements of small defects below the critical size.

Accident Analysis. The scope of analysis of postulated accidents available in the technical justification of safety (TOB) for RBMKs was determined by national regulations effective when the TOB was issued. Compared to current practices, it was found to be limited and the related information usually does not provide a clear description of assumptions used in the analysis. Computer codes used at the time of RBMK design were of limited modeling capability. The lack of an experimental database on pipe rupture of the primary heat transport system limited the possibility of integral code validation. More modern Russian codes and some Western codes now are being used, but they have not been sufficiently validated for modeling RBMKs.

The review further found a number of areas to be incomplete, including the analysis of Design Basis Accidents (DBA); the adequacy of the codes, database, validation, and documentation for the analysis of loss-of-coolant accidents; and the sensitivity to parameter variations and uncertainties. Additionally, anticipated transients without scram (ATWS) were considered of high safety significance and the analysis of these events needs to be performed.

The completeness of such analysis is of utmost importance to ensure the safe plant design. The analysis should identify possible shortcomings of the existing design and be performed using modern and qualified methods. As design changes are implemented in the plants, the analysis also needs to be updated. An example of this process is the safety analysis report (SAR) work now being performed for the Ignalina plant.

A useful tool in identifying weaknesses and prioritizing improvements is probabilistic safety assessment (PSA). Therefore the performance and peer review of a plant-specific PSA for all RBMKs is recommended.

Safety and Support Systems. In general, it has been found that the high redundancy which exists in several of the front-line safety systems is not present to the same extent in supporting systems, such as the service water and intermediate cooling systems. Moreover, the high level of redundancy in the safety systems cannot always be given full credit due to potential common cause failures.

The reliability of the safety systems is dependent on the system design and alignment and on operational parameters, such as maintenance and testing procedures and emergency operating procedures. Therefore, a strong tie needs to be maintained between this area and the development of emergency operating procedures and testing procedures.

In general, it has been found that the differences between the plants are so important that recommendations in this area have to be evaluated on a plant-specific basis.

Fire Protection. Passive fire protection can address fire safety problems in an effective way. Passive fire protection comprises all the measures that are put in place before the start of operation and are not expected to need any human or mechanical action in case of fire. The basis for prevention of fire damage is the minimization of the amount of burnable material and fire loads. Total elimination of burnable material is preferred, but where this is not possible, the fire load has to be separated into different fire compartments. The basis for compartmentation is the separation of safety significant equipment from each other and from hazardous substances. Compartment boundaries should consist of fixed fire barriers such as walls, floors, ceiling, and mechanical and electrical penetration seals. Also within compartments, important elements may need fire separation. This may take place through distance or local separative elements.

Fire risks were not adequately considered in the design phase of RBMK reactors when passive measures could have easily been implemented. However, much work has been done afterwards. Removal of the largest fire load, the plastic floor coating, has started gradually. Improvement of compartmentation has been carried out by upgrading fire doors and penetration sealings. Within the compartments the main improvement effort has been the covering of cables with a fire resistant protective coating.

These problems have already been tackled to some extent in some RBMK units. In other cases this has been taken care of by a newer design, by national upgrading programmes, or by bilateral/international assistance programmes.

Throughout the plant, all areas with burnable material should be provided with fire detection equipment connected to a proper alarm system. The existing systems need both extension and quality upgrading. Some plants have started these measures as a part of their own upgrading programme, and some in connection with bilaterally or internationally financed programmes.

Manual fire suppression capability is generally very strong at nuclear power plants in the former Soviet Union. This applies to the number and the training of fire brigade personnel. Deficiencies, however, exist in the personal protective equipment, communication equipment, and fire fighting equipment, such as fire extinguishers, hoses, and nozzles.

Automatic fire suppression is mainly realized through fixed water sprinkler and deluge extinguishing systems. Local carbon dioxide or foam extinguishing systems also exist. The reliability and coverage of the existing systems need evaluation. Automatic fixed water extinguishing systems should be added to some compartments which so far have not been fully protected.

The reliable supply of water assures proper availability and operation of both manual and automatic fire suppression capability. However, differences between sites and the different generations of RBMKs are extensive and measures of different magnitude are needed.

Operational Safety. Past experience in the operation of nuclear power plants confirms the important role of plant personnel in assuring nuclear safety. Considerable attention has been given to the study of human factors in plant operation. Overall, it has been found that the operational safety of RBMKs can be upgraded. Identified safety issues include those relating to the training of operators; operating and emergency management procedures; and surveillance, maintenance, and control of plant modifications. Recommendations have been made in these areas and should be implemented in parallel with proposed design and safety improvements.

A stronger technical basis

On the basis of national and multilateral safety reviews, the main safety concerns of the more modern RBMK nuclear power plants have been identified and the required safety improvements have been agreed upon.

Despite the work carried out to date, safety concerns remain, particularly those related to the first-generation units. Future IAEA activities will focus on assisting in the review of first-generation units and on streamlining efforts to resolve generic safety issues.

Up-to-date information on the plant-specific status of operating RBMKs is essential for an effective exchange of technical information and for the co-ordination of national and international efforts to improve nuclear safety. The database established by the IAEA is an effective tool to facilitate this co-ordination. The work ahead will enable the compilation of more plant-specific information in the interests of tracking the status of safety improvements and identifying areas where more efforts need to be taken. ❐

Participants in the safety review at the Smolensk plant under the IAEA's extra-budgetary RBMK safety programme.