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RBMK NUCLEAR POWER PLANTS: GENERIC SAFETY ISSUES

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

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FOREWORD

The IAEA initiated in 1990 a programme to assist the 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 of the Programme 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 scope of the Programme was extended in 1992 to include RBMK, WWER-440/213 and WWER-1000 plants in operation and under construction. The Programme is complemented by national and regional technical cooperation projects.

The Programme is pursued by means of plant specific safety review missions to assess the adequacy of design and operational practices; Assessment of Safety Significant Events Team (ASSET) reviews of operational performance; reviews of plant design, including seismic safety studies; and topical meetings on generic safety issues. Other components are: follow-up safety missions to nuclear plants to check the status of implementation of IAEA recommendations; assessments of safety improvements implemented or proposed; peer reviews of safety studies, and training workshops. The IAEA is also maintaining a database on the technical safety issues identified for each plant and the status of implementation of safety improvements. An additional important element is the provision of assistance by the IAEA to strengthen regulatory authorities.

The Programme is extrabudgetary and depends on voluntary contributions from IAEA Member States. Steering Committees provide co-ordination and guidance to the IAEA on technical matters and serve as forums for exchange of information with the European Commission and with other international and financial organizations. The general scope and results of the Programme are reviewed at Advisory Group Meetings.

The Programme, which takes into account the results of other relevant national, bilateral and multilateral activities, provides a forum to establish international consensus on the technical basis for upgrading the safety of WWER and RBMK nuclear power plants.

The IAEA further provides technical advice in the co-ordination structure established by the Group of 24 OECD countries through the European Commission to provide technical assistance on nuclear safety matters to the countries of central and eastern Europe and the former Soviet Union.

Results, recommendations and conclusions resulting from the IAEA Programme are intended only to assist national decision makers who have the sole responsibilities for the regulation and safe operation of their nuclear power plants. Moreover, they do not replace a comprehensive safety assessment which needs to be performed in the frame of the national licensing process.

EDITORIAL NOTE

CONTENTS

SUMMARY

The development of the RBMK system came out of the Soviet uranium-graphite production reactors, the first of which began operation in 1948. In 1954 a demonstration 5 MW(e) graphite-moderated, water-cooled reactor began operation in Obninsk and subsequently a series of reactors were developed using the combination of graphite moderation and water cooling in a channel design.

The RBMK is a thermal-neutron, graphite-moderated, channel-type, direct-cycle boiling water nuclear reactor with on-line refueling. Description of an RBMK NPP in this report is given based on Smolensk Unit 3 to provide the reader with some familiarity of the design of the RBMK NPP. Smolensk 3 is a third generation unit and in fact is the only 1000 MW(e) third generation unit in operation.

Currently there are 15 RBMK reactors in operation: 11 units in Russia, two in Ukraine and two in Lithuania. The connection to the grid of these units took place from 1973 (Leningrad 1) to 1990 (Smolensk 3).

Since the Chernobyl accident in 1986, a considerable amount of work has been either completed or has been under way to improve the safety of RBMK NPPs. However, safety concerns still remain for NPPs with this reactor type.

In response to a request from the former Soviet Union, made at the September 1991 IAEA Nuclear Safety Conference "Safety of Nuclear Power and the Future", an IAEA Programme on the Safety of RBMK NPPs was initiated. Other international and bilateral projects have also started.

In April 1992 an IAEA Technical Committee meeting on the safety of RBMK reactors established the general lines of the Programme with emphasis on direct improvement of RBMK safety. The IAEA started its Programme in June 1992.

The objectives of the IAEA Programme are to consolidate results from various national, bilateral and multilateral programmes and establish international consensus on the required safety improvements and related priorities. It assists both regulatory and operating organizations and provides a basis for technical and financial decisions.

In order to consolidate all results of safety reviews of RBMK NPPs, a meeting was held from 23 to 27 January 1995 at the IAEA Headquarters in Vienna. A total of 58 safety issues were consolidated in seven topical areas. Two broad issues addressing quality assurance and regulatory interface have been recognized as affecting all topical areas. The results of the meeting have been made available in June 1995 in the RBMK-SC-20 (Rev. 1) report. The report was prepared on the basis of the safety reviews of Smolensk 3 and Ignalina NPP. Therefore, it is primarily applicable to RBMK NPPs of this vintage. Whenever applicable, reference is also made to RBMKs of other generations. Specific safety issues and improvements for RBMKs of 1st and 2nd generation will be consolidated upon completion of ongoing studies for these NPPs. It is recognized that how and when the identified safety issues are addressed exactly is a matter to be resolved between the operating organization and each national regulatory body.

This report has been prepared on the basis of above mentioned report and it is intended to provide information on RBMK NPPs generic safety issues. As all other insights, recommendations and conclusions resulting from the IAEA Programme, this report is intended to assist national decision makers, who have sole responsibility for the regulation and safe operation of their nuclear power plants. It also serves to focus national and international projects on priority of the RBMK safety improvements.

1. INTRODUCTION

1.1. BACKGROUND

Various analyses of the Chernobyl NPP Unit 4 accident have revealed safety deficiencies of RBMK reactors. As a result, major design backfittings and operational changes have been implemented. In response to the request from the former Soviet Union made at the September 1991 IAEA Nuclear Safety Conference "Safety of Nuclear Power and the Future", several international projects on the safety of RBMK NPPs have been initiated.

In April 1992 an IAEA Technical Committee meeting on the safety of RBMK reactors recommended a consistent international programme with emphasis on direct improvement of RBMK safety [1]. Along these lines the IAEA launched an extrabudgetary programme on RBMK safety in June 1992.

1.2. IAEA PROGRAMME OVERVIEW ON RBMK SAFETY

1.2.1. Objectives and strategy

The objectives of the IAEA Programme are to consolidate results from various national, bilateral and multilateral programmes and establish international consensus on the required safety improvements and related priorities. It assists both regulatory and operating organizations and provides a basis for technical and financial decisions.

The Programme addresses all generations of RBMK reactors in operation, taking into account significant differences among the 15 operational RBMK units, which exist even within the same generation.

The activities focus on:

- assisting in the review of safety improvements related to plant design and operation;
- organizing specialists meeting on topical areas of RBMK safety;
- conducting plant specific safety reviews;
- reviewing results of assessment (e.g. PSA, accident analysis); and
- providing training.

1.2.2. Main activities

In October/November 1992 a Consultants meeting was organized by the IAEA in Vienna on the "Safety Assessment of Proposed Improvements of RBMK NPPs" [2]. The objective of the meeting was to review the scope and results of safety evaluations performed and the technical basis for the safety improvements implemented and planned for RBMK reactors. Areas in which further work was needed were specified and shortcomings of work performed or planned were identified.

In June 1992 at the request of the Government of Ukraine the IAEA organized an ASSET mission to review the root cause analysis of a safety significant event at the Chernobyl NPP Unit 2 [3]. The incident analyzed was a fire in the turbogenerator which occurred on 11 October 1991 and caused the collapse the building roof which disabled the emergency core cooling system. It was recognized that measures to reduce plant risk from fires required urgent action.

After completing the ASSET missions at Kursk [4], Ignalina [5], Leningrad (Sosnovy Bor) [6] and Smolensk [7] in 1992 and 1993 the IAEA convened a consultants group to review the findings of the missions and to consolidate generic insights [8].

In June 1993, a two-week safety review was conducted by the IAEA at the Smolensk Training Center at Desnogorsk, Russia [9]. Smolensk Unit 3 is the most advanced (third) generation of RBMK plants which started operation in mid-1990. A number of safety improvements identified on the basis of analyses of the Chernobyl accident and other studies had already been incorporated in the design of Unit 3.

The objective of the review was to further discuss the findings and recommendations of the IAEA-TECDOC-694, "Safety Assessment of Proposed Improvements to RBMK Nuclear Power Plants" and in particular their applicability to Smolensk Unit 3. Smolensk Unit 3 and Ignalina Unit 2 were the reference plants for the IAEA Programme.

The Smolensk Unit 3 safety analysis report (TOB) was used as the basic document for the IAEA review. The TOB information was complemented by results of specific safety analyses and other documentation related to plant design and conduct of operation. No specific additional documentation was prepared for the meeting. Members of the IAEA review team also conducted plant walkdowns specific to their respective areas of review.

Following the Smolensk 3 Safety Review, in June 1993, a consultants meeting on Prioritization of Safety Improvements of RBMK NPPs was held in Vienna, from 20 to 24 September 1993. The objective of the meeting was to consolidate the recommendations from the various RBMK safety reviews into specific safety issues and then rank these safety issues based on their importance to RBMK safety. The results were presented in the draft report RBMK-SC-011 which was revised in April 1994 [10].

The IAEA convened a consultants meeting in December 1993 in Switzerland [11] to review the implementation of the basic principles of shutdown system requirements in the design of the Smolensk 3 shutdown systems, based on the IAEA NUSS document, national standards (Russia, Canada and Germany) and OECD regulatory practices. Compatibility of the shutdown system of Smolensk 3 NPP with the requirements on independence, diversity and shutdown effectiveness were also reviewed. The intent of Russian designers to improve the RBMK shutdown system was fully supported by international experts.

A topical meeting on multiple pressure tube rupture analysis in channel type reactors was convened by the IAEA at the RDIPE (Research and Development Institute of Power Engineering) offices in Moscow in January 1994 [12]. The objective of the meeting was to exchange experience on regulatory approaches adopted in Member States operating channel type reactors and review analysis methodology, criteria and results obtained as related to multiple pressure tube rupture in RBMK type reactors. It was agreed that validation of computer codes in use to study loss of coolant accident (LOCA) scenarios in RBMK NPPs is matter of utmost importance.

In November 1994 a consultants meeting on code validation for RBMK LOCA analysis took place in Japan [14]. A validation matrix for code calculation was established and an international exercise was initiated in 1995 under IAEA coordination. The exercise is based on experimental results made available by the Government of Japan.

An IAEA review of safety improvements proposed for Ignalina NPP was conducted in October 1994 and the report has also been finalized [13].

In order to consolidate all information available, a consultants meeting was held from 23 to 27 January 1995 at the IAEA Headquarters in Vienna. The draft report RBMK-SC-011 was reviewed and updated in the light of all results available to the IAEA and the international community on the safety of RBMK NPPs [15].

The activities being conducted within the framework of the IAEA Programme are co-ordinated with those of an international consortium on the "Safety of Design Solutions and Operation of NPPs with

RBMK Reactors" established under the auspices of the European Commission. These two programmes involved the same reference plants, namely Smolensk NPP Unit 3 and Ignalina NPP Unit 2.

With the completion in 1994 by the IAEA of the safety reviews of Smolensk 3 and Ignalina 2, these two programmes have reached an important milestone.

In order to make these results available to the international technical community the IAEA Technical Committee/workshop was held from 30 May to 2 June, 1995 [16]. The presented results of the both projects reflected a large amount of work done by the international experts and Russian organizations to review the safety of the NPPs with RBMK reactors.

Both projects produced over 300 recommendations for safety enhancement of RBMK plants. Most of them correlate with the measures already included in national safety enhancement programmes for RBMK units which are under way in Russia, Lithuania and Ukraine.

The Swiss National Committee of the International Electrotechnical Commission (IEC) submitted a proposal on "Nuclear instrumentation and electrical items for the safe operation of RBMK type reactors" to the IEC Control Office in July 1992. A joint IEC-IAEA project team was set up to develop the project. The final report is being prepared [17], identifying measures that can be implemented to improve RBMK safety through enhancement of the I&C systems.

1.3. IAEA DATA BANK

1.3.1. General description

Since the beginning of the IAEA Extrabudgetary Programme on WWER reactors, a great number of safety items were collected as a result of design review and safety review missions of the WWER-440/230 type reactors. On the basis of these findings a technical database containing more than 1300 records was established to support the consolidation of the obtained information and to help in identification of safety issues.

Later the database was extended to the WWER-440/213, WWER-1000 and RBMK reactor types. Information on reviews of safety improvements, results of topical meetings, technical committee meetings, ASSET and OSART missions was included. In this work the results of other programmes (e.g. Consortium for RBMK reactors) were also taken into account.

In the structure of the databases, plant specific data is included to reflect the actions taken by individual plants in order to address safety issues.

In the development of the IAEA databases one of the most important aspects was to ensure a direct link, on the software level, with the G-24 Project Data Bank which gives an opportunity to perform analyses simultaneously using the information from both IAEA and G-24 databases.

1.3.2. RBMK database

As a result of the initial phase of the IAEA Extrabudgetary Programme on RBMK Safety, the IAEA elaborated a list of design and operational safety issues for this reactor type. For this purpose, the database on findings and recommendations for RBMK reactors was extensively used. From the working materials of previous meetings, the safety review mission reports to Smolensk and Ignalina, the ASSET reports and recommendations of the International Consortium sponsored by the European Commission, more than 1000 safety items were collected in the initial version of the database. The safety items were grouped by topical areas. Recommendations of the topical reports on multiple tube rupture and on the shutdown system of RBMK reactors have also been included into the database.

The main design institute (RDIPE, Moscow) of the RBMK reactors has made available to the IAEA data on the safety upgrading programmes of individual plants addressing the safety issues. This plant specific information is also included in the database.

The database includes all references of the data used in the original publications. An interface table with the G-24 Project Data Bank can be prepared which makes easier the joint analysis of information from both databases.

The structure of the RBMK database is given in Fig.1. This structure allows the inclusion into the database of plant specific information as it becomes available. Further, the inclusion of the full text of the technical reports in the data base is being considered.

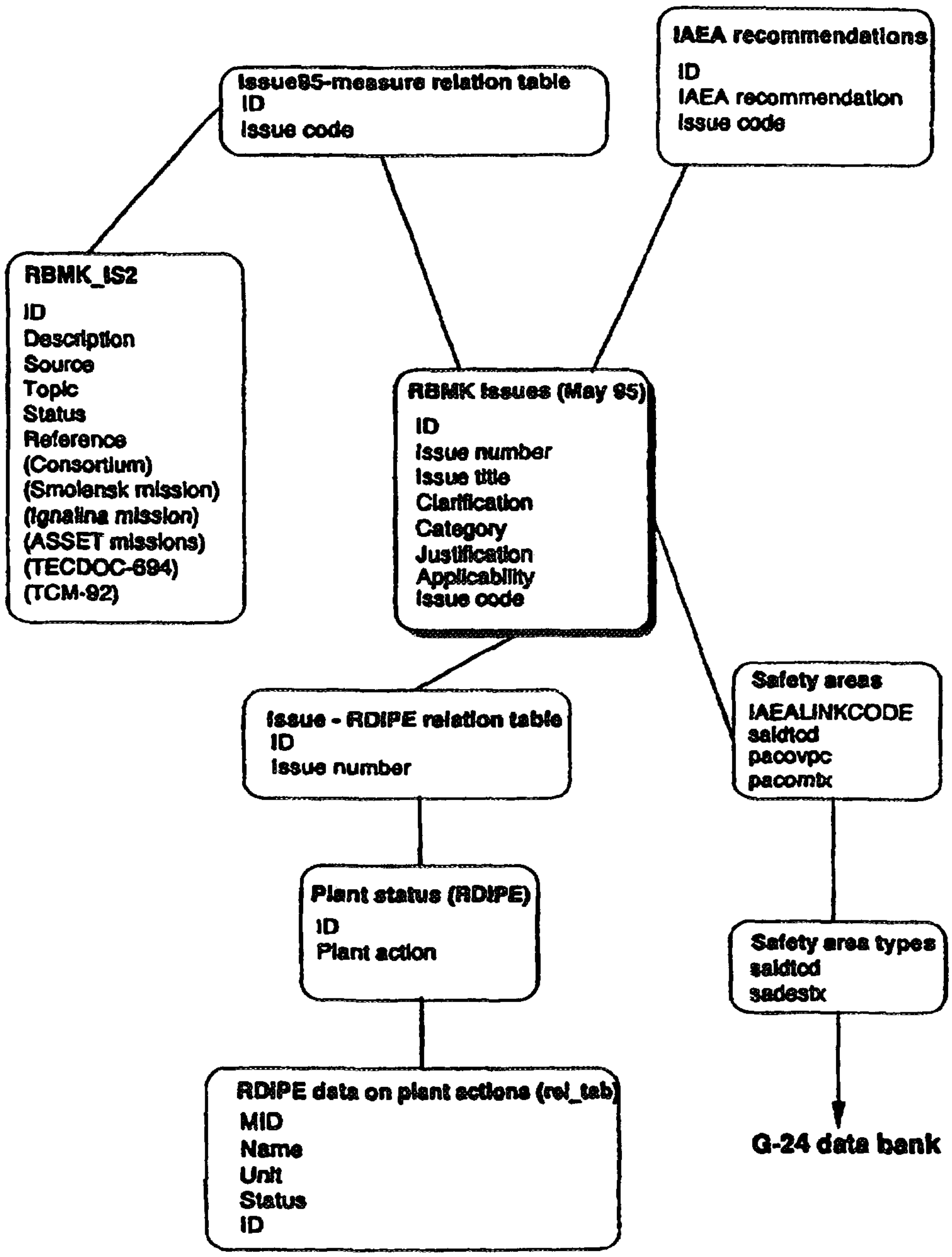


FIG. 1. The structure of the RBMK database.

2. DESCRIPTION OF THE RBMK NPP

2.1. HISTORY OF THE DEVELOPMENT

The development of the boiling water cooled graphite moderated pressure tube reactor system (RBMK) came out of the Soviet uranium-graphite Pu production reactors, the first of which began operation in 1948. In 1954 a demonstration 5 MW(e) reactor began operation in Obninsk and subsequently a series of reactors were developed using the combination of graphite moderation and water cooling in a channel design.

Currently there are 15 RBMK reactors in operation: 11 units in Russia, two in Ukraine, Chernobyl Unit 2 has been shutdown since 1991 after a major fire collapsed the turbine building roof and damaged safety equipment, and two in Lithuania. The connection to the electric power grid of these units took place from 1973 (Leningrad 1) to 1990 (Smolensk 3).

Over the course of the 20 years' development, three generations have emerged which have significant differences, particularly with respect to the safety provisions built into design. The electric power of the RBMK reactors is 1000 MW(e) except for Ignalina NPP whose one unit power is 1500 MW(e). The development of the Kursk Unit 5 has led to many design changes - hence it can be thought of as a fourth generation.

The first generation units (Leningrad 1 and 2, Kursk 1 and Chernobyl 1 and 2) were designed and built before 1982 when new standards OPB-82 for the design and construction were introduced in the Soviet Union. Since than other units have been designed and constructed in accordance to OPB-82 requirements. However, the safety standards in the USSR were revised again in 1988 (OPB-88).

2.2 GENERAL LAYOUT

Description of an RBMK NPP design in this report is given based on Smolensk Unit 3 [7,9,18], because it was the reference NPP for the first phase of the IAEA and EC projects.

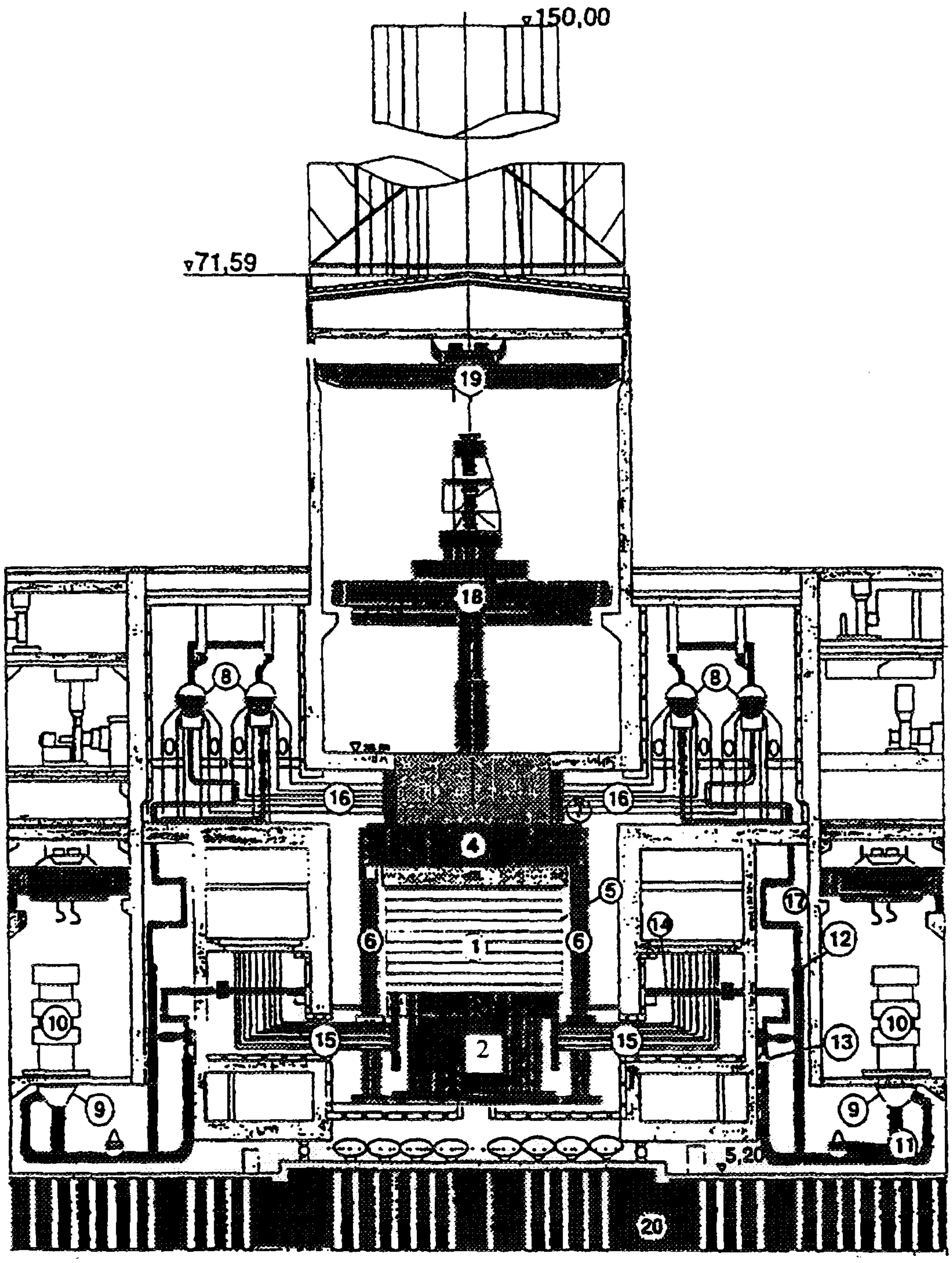
Smolensk Unit 3 is a third generation unit and in fact is the only 1000 MW(e) third generation unit in operation. Figure 2 shows a cross section of the main reactor building for Smolensk NPP Unit 3 and the main reactor parameters are given in Table I.

It is a heterogeneous pressure tube type thermal neutron reactor with a graphite moderator and a boiling light water coolant. The graphite core (Fig. 3) is enclosed in a thin-walled steel jacket within which a He/N gas mixture is slowly circulated to improve heat transfer and limit graphite corrosion. In the radial direction as well as above and below reactor it is surrounded by the primary biological shield structures.

The heat flow diagram is a typical one for a single circuit boiling water reactors (Fig. 4). The main circulation circuit consists of 2 parallel loops, each with 2 steam drum separators and 4 main circulating pumps (MCPs) and two 500 MW(e) turbines operating at 3000 r.p.m. The main high-pressure components (e.g. MCPs) are located in individual compartments connected to accident localization system (ALS). Some components of the main circulation circuit are located in those premises which are not connected to ALS.

One of the important characteristics of RBMK reactors is their on-line refueling capability. Refueling at full power is accomplished by means of the refueling machine. Under normal operation and nominal reactor power, generally two fuel assemblies are changed per day.

Below, a more detailed description of the specific parts of the NPP is presented. The presentation is structured according to the topical areas defined for the safety review and identification of safety issues.

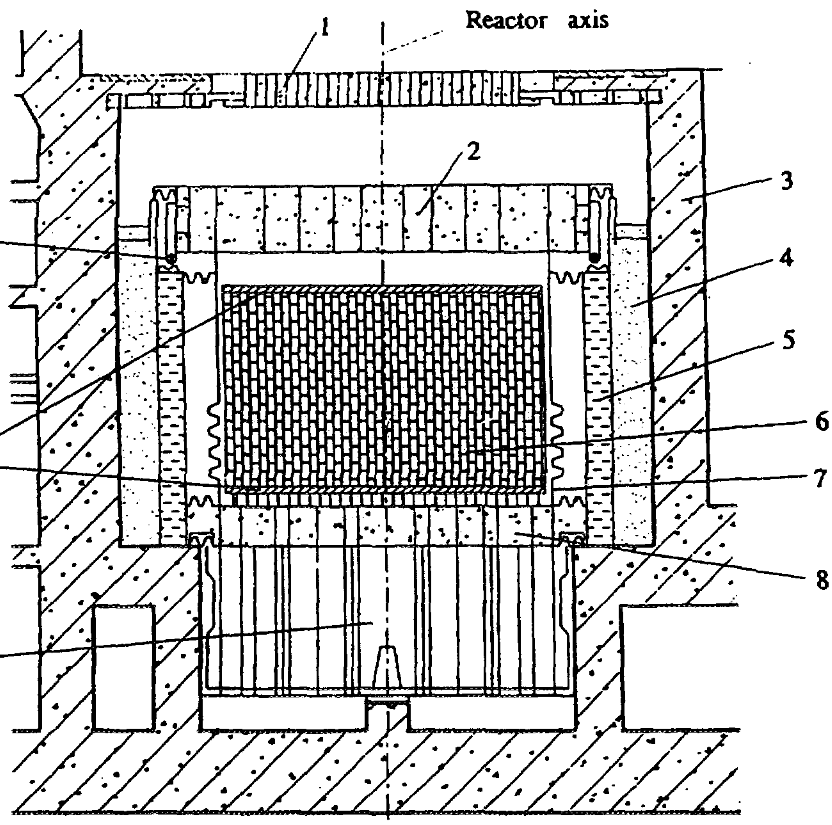


1- graphite stack, 2- metal structure 'S', 3- metal structure 'OR', 4- metal structure 'E', 5- metal structure 'KZh', 6- metal structure 'L', 7- metal structure 'D', 8- drum separator, 9- MCP bowl, 10- MCP electric motor, 11- MCP discharge valve, 12- suction header, 13- pressure header, 14- distribution group header, 15- lower water pipelines, 16- steam/water pipelines, 17- downcomers, 18- refuelling machine, 19- crane in central hall, 20- pressure suppression pool

FIG. 2. A cross section of the main reactor building of Smolensk NPP Unit 3.

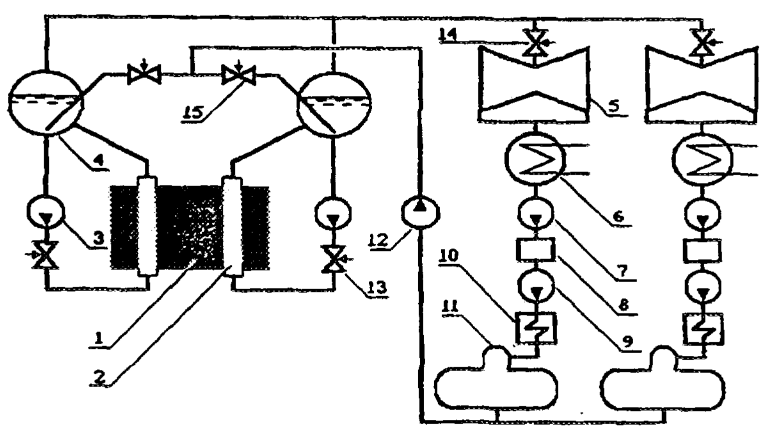
TABLE I. REACTOR PARAMETERS

DESCRIPTION	VALUE	UNITS
Electrical power	1000	MW
Thermal power	3200	MW
Circulation loops in reactor	2	
Mass flow of coolant in reactor	10416	kg/s
Pressure in steam drums	7	MPa
Mass flow of steam	1555	kg/s
Steam pressure at turbine inlet	6.5	MPa
Steam temperature before turbine	280	^{0}C
Feed water temperature	165	^{0}C
Core diameter	11.8	m
Core height	7	m
Pressure tube diameter	88	mm
Thickness of pressure tube wall	4	mm
Mean linear power	146	W/cm
Maximum linear power	350	W/cm
Fuel mass	192 000	kg
Specific power	16.7	W/gU
Number of fuel channels	1661	
Number of fuel rods in assembly	2x18	
Maximum channel power	3.0	MW
Fuel rod diameter	13.5	mm
Fuel assembly diameter	79	mm
Fuel enrichment	2-2.4	% of ^{235}U
Fuel burn-up	21.6	MW.d/kg U
Graphite mass	1700 000	kg
Graphite temperature	500/700	^{0}C
Number of control rods	211	
Core exit steam fraction	14.5	%



1- top cover, removable floor of the central hall, 2- top metal structure filled with serpentinite, 3- concrete vault, 4- sand cylinder, 5- annular water tank, 6 - graphite stack, 7 - reactor vessel, 8 - bottom metal structure, 9 - reactor support plates, 10 - steel blocks, 11 - roller supports.

FIG. 3 Cross-section of the reactor vault.



1- reactor, 2- fuel channel, 3- MCP, 4- drum separator, 5- turbines, 6- condensers, 7- CP-1, 8- condensate clean-up, 9- CP-2, 10- low-pressure heater, 11- deaerator, 12- feed pipes, 13- throttle-control valves, 14- emergency regulating valves, 15- feed units

FIG. 4 Unit thermal diagram.

2.3. CORE DESIGN AND CORE MONITORING

The RBMK reactor core is constructed of closely packed graphite blocks stacked into columns and provided with axial opening. Most of the openings contain fuel channels. A number of them 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 m in diameter, which acts as a support for the graphite stack and as a container for the helium-nitrogen graphite coolant. The total mass of the graphite within the core is 1700 tonns. Approximately 6% of the reactor's thermal energy is generated in the graphite stack. The helium-nitrogen mixture transports the heat from the graphite to the technological 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 mm in diameter made of zirconium alloy. Each fuel channel contains two fuel assemblies, one above the other, each of them containing 18 rod fuel elements which are 13.6 mm in diameter and are enclosed in zirconium alloy cladding. The total fuel length of the core is approximately 7 m. The initial fuel enrichment was 2% of ^{235}U. After the Chernobyl accident it was modified to decrease the void reactivity coefficient, and except for Ignalina, it is now 2.4%.

The reactor control and protection system (RCPS) consists of 211 movable solid absorber rods in special channels cooled by an independent water circuit. Figure 5 gives a schematic presentation of the control rod design.

In the normal operating modes, and in the event of design basis accident (DBA), the RCPS automatically maintains the present power level; initiates a swift power reduction using the automatic and manual control rods if there are signals indicating main equipment failures; terminates the chain reaction using the rods of the emergency protection system (RCPS) if there are signals indicating dangerous divergence in unit parameters or equipment failures; compensates for reactivity changes during heatup and when the reactor is being brought up to power; and controls the core power density.

The systems of RBMK reactor shutdown for the Unit 3 of the Smolensk NPP consist of:

- fast-acting emergency protection (BAZ): 24 control rods fully inserted in less than 2.5 s plus 187 control rods of other types which are fully inserted in 12-14 s;

- emergency protection (AZ-1); 24 rods of fast acting emergency protection system inserted in 7 s plus 187 control rods of other types which are fully inserted in 12-14 s.

The RCPS has a local automatic control (LAC) subsystem and a local emergency protection (LEP) subsystem. Both these systems respond to signals from in-core ionization chambers. The LAC system automatically stabilizes the main harmonics of the radial-azimuthal power density distribution, while the LEP system protects the reactor by ensuring that the prescribed power level at specific core locations is not exceeded. The axial fields are controlled by shortened absorber rods which are inserted into the core from below (32 rods).

2.4. INSTRUMENTATION AND CONTROL

The I&C process monitoring system is intended for monitoring thermal and physical parameters which characterize performance of the reactor and its support systems, and is intended for providing information for SCALA centralized computerized monitoring system and (or) for individual displaying (self-recording) instrumentation.

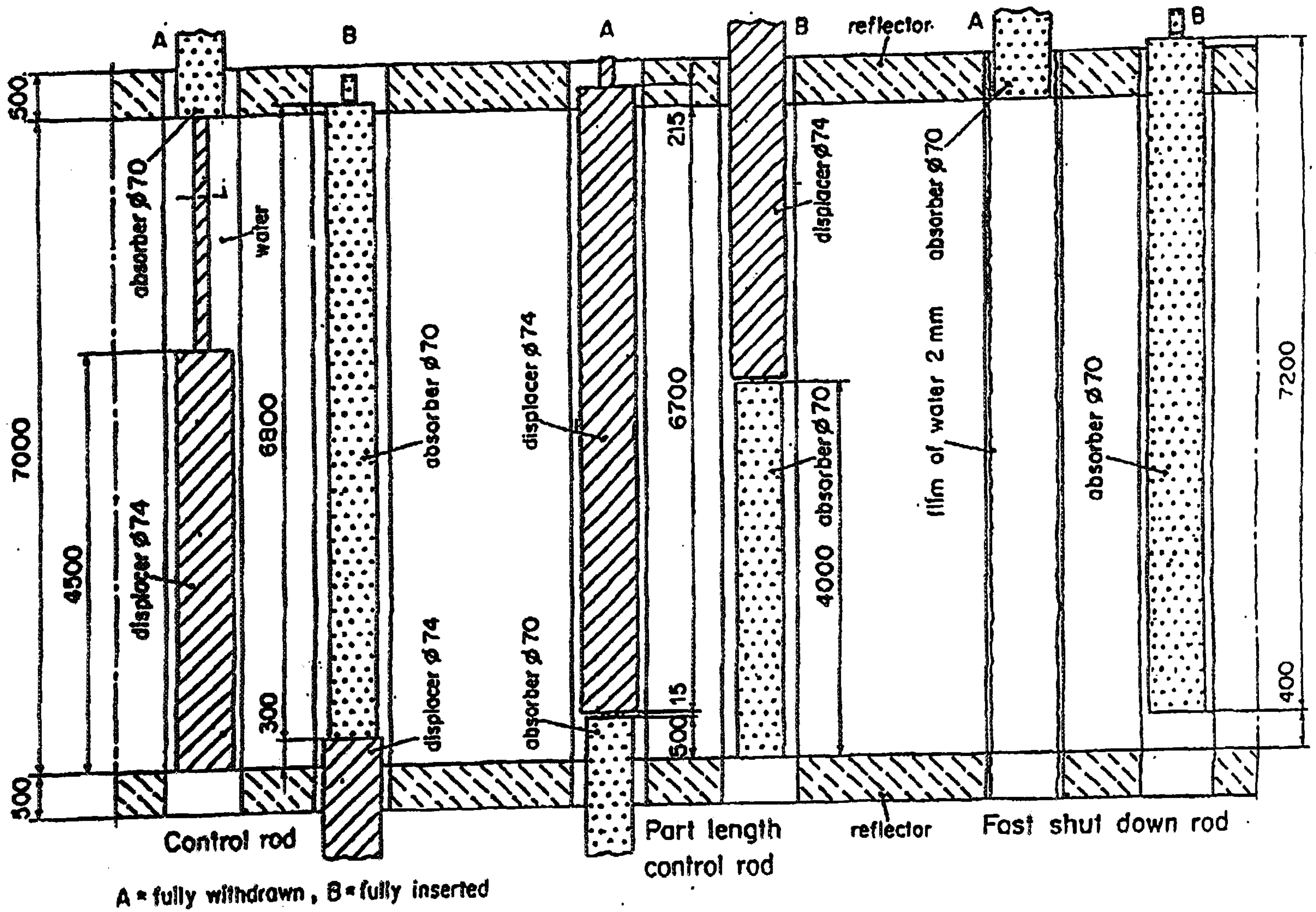


FIG. 5 Schematic presentation of the control rod design.

The process monitoring is performed by the following systems:

- monitoring of coolant flow in individual fuel and special channels;
- monitoring of graphite stack and metal structures temperature;
- monitoring of reactor channels integrity;
- physical monitoring of axial/radial power distribution in the core;
- detection of fuel clad leakage;
- monitoring of reactor main circulation circuit parameters;
- SCALA centralized monitoring.

SCALA is the data recording and handling system for the RBMK reactor. The system has many functions and performs a wide range of duties. Some of the functions are:

- to record all plant parameters;

- to evaluate the core power information and provide the operator with data on the main power and the axial and radial power profiles;

- to calculate the ORM;

- to calculate the departure from nucleate boiling (DNB) margin.

In the process monitoring system the information is obtained by direct or indirect measurements of the parameter or its calculation by computer. All parameters are entered into SCALA centralized monitoring system, and can be called at operator's request to the digital display. Independent of this system, the most important parameters are output to individual displays and (or) self-recording instrumentation in the main control room.

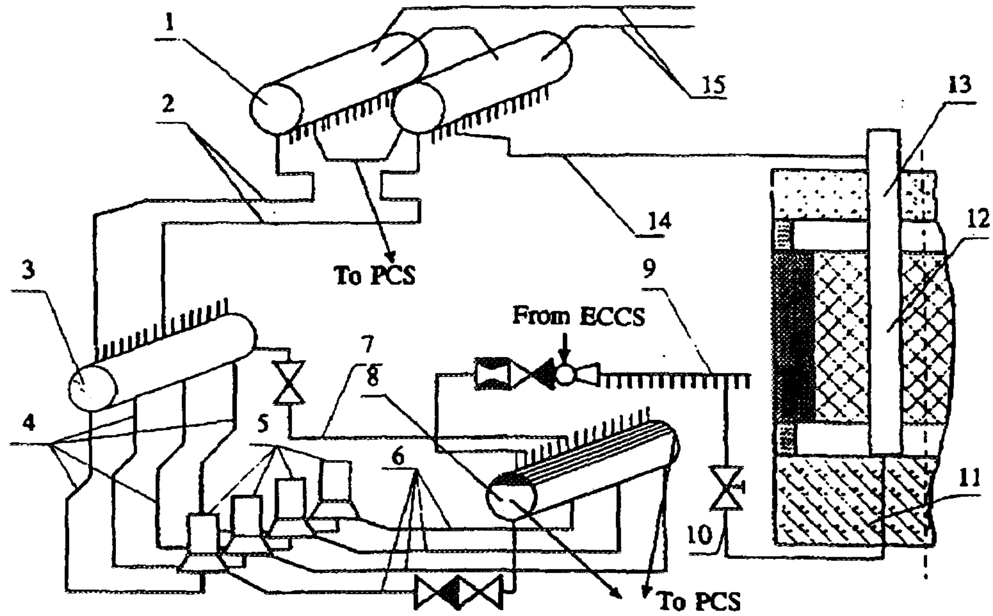
2.5. PRESSURE BOUNDARY

The pressure boundary consists of fuel channels and pipes of different diameters composing together the main circulation circuit (MC) which provides coolant circulation in the core, and the steam piping directing steam to the turbines. MC consists of two independent loops, equipment and pipelines which are symmetric about the vertical axial plane of reactor.

The steam and water mixture formed in fuel channel is supplied through separate steam-water communication lines to separator-drums, where the mixture is separated. Steam is directed to the turbines and water mixing with the feed-water comes back through 12 downcomers to the suction header of MCP (common to all the loop's pumps), then along the pipelines through the isolating valves it is supplied to the pumps' intake. From the MCP, through pipelines check valves water goes to a pressure header. Twenty two distributing group headers (325 mm diameter), each supplying 40 fuel channels, are welded to the pressure header of MCP. They supply water to every fuel channel of the reactor. A schematic presentation of one loop of the MCM is shown in Fig. 6. Figure 7 shows a distribution group header.

One of the principal distinguishing characteristics of the RBMK reactor type is that each fuel assembly is housed in an individual pressure tube called "fuel channel or technological channel". Each pressure tube has considerable autonomy. The coolant flow rate to the tube is controlled on-line by an individual isolating and control valve (ICV) making it possible to decouple the fuel channel from the main circulation circuit and to control water flow in the fuel channel according to its power density.

The pressure tube is constructed from separate end and central parts. The central part is 88 mm outside diameter (4 mm thick wall) tube made of zirconium-niobium alloy. The top and bottom parts are made from stainless steel tubes. The center and end pieces are joined by special intermediate couplings, made from steel-zirconium weld.



1 - steam separation drum, 2 - downcomers, 3 - suction header, 4 - suction piping of the MCP, 5 - MCP tanks, 6 - pressure piping of the MCP, 7 - bypass between headers, 8 - pressure header, 9 - group distribution header with flow limiter, check valve and mixer, 10 - water piping, 11 - channel to core, 12 - fuel channel, 13 - channel above core, 14 - steam-water pipes.

FIG. 6 Schematic presentation of one loop of the MC.

The fuel assembly is suspended in the center of the pressure tube by means of a special device which hermetically seals the pressure tube after the fuel assembly has been installed.

The channel tubes are set into the circular passage of the graphite stack. The upper and bottom parts of the channel tube are welded to the top and bottom metal structure plates respectively to maintain the core hermetically sealed and the bottom parts are connected with the pipe (tract) by means of a bellows compensator. The design life of the channel tube is about 15-20 years. If it is necessary, the channel tube can be replaced removing the top and bottom welds.

Each channel tube is supported inside the passage of the graphite stack by a series of graphite rings. These rings alternate in size: one fitting the pressure tube tightly, the next fitting the graphite block tightly. This arrangement gives clearance of 3 mm between the graphite block and rings at the beginning of plant life. The average energy generation required to close the gap varies from reactor to reactor because of differences in the operating conditions, but it corresponds to an average integrated generation in the range 10 000 to 15000 MW.d/channel which in turn corresponds to a predicted time before retubing is necessary of between 15 years (Leningrad Unit 1) and 21 years (Ignalina Unit 1) based on a mean channel irradiation of 700 MW.d/year. Figure 8 gives a schematic presentation of the fuel tube and graphite block arrangement of the fuel channel.

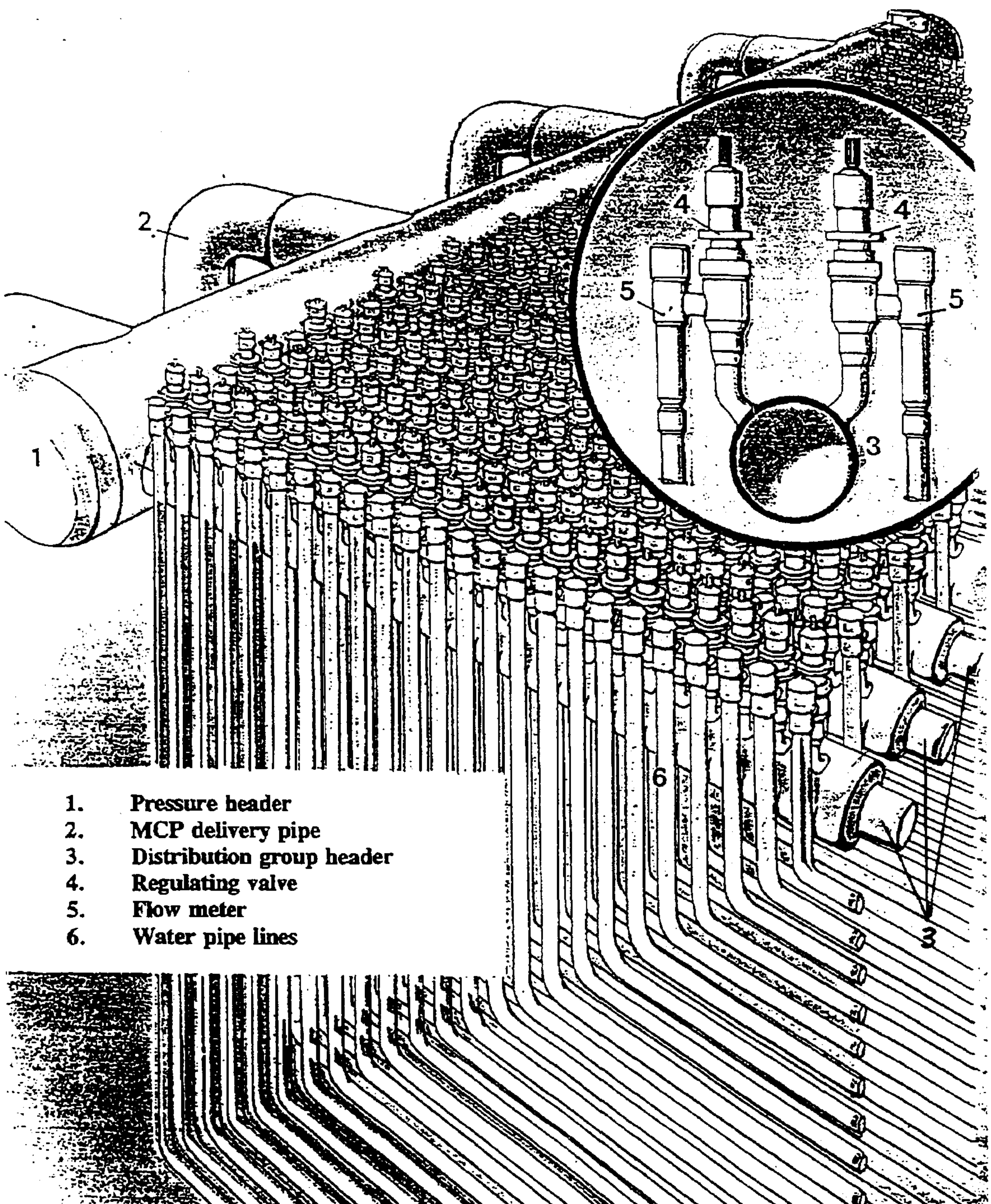


FIG. 7 Distribution Group Header.

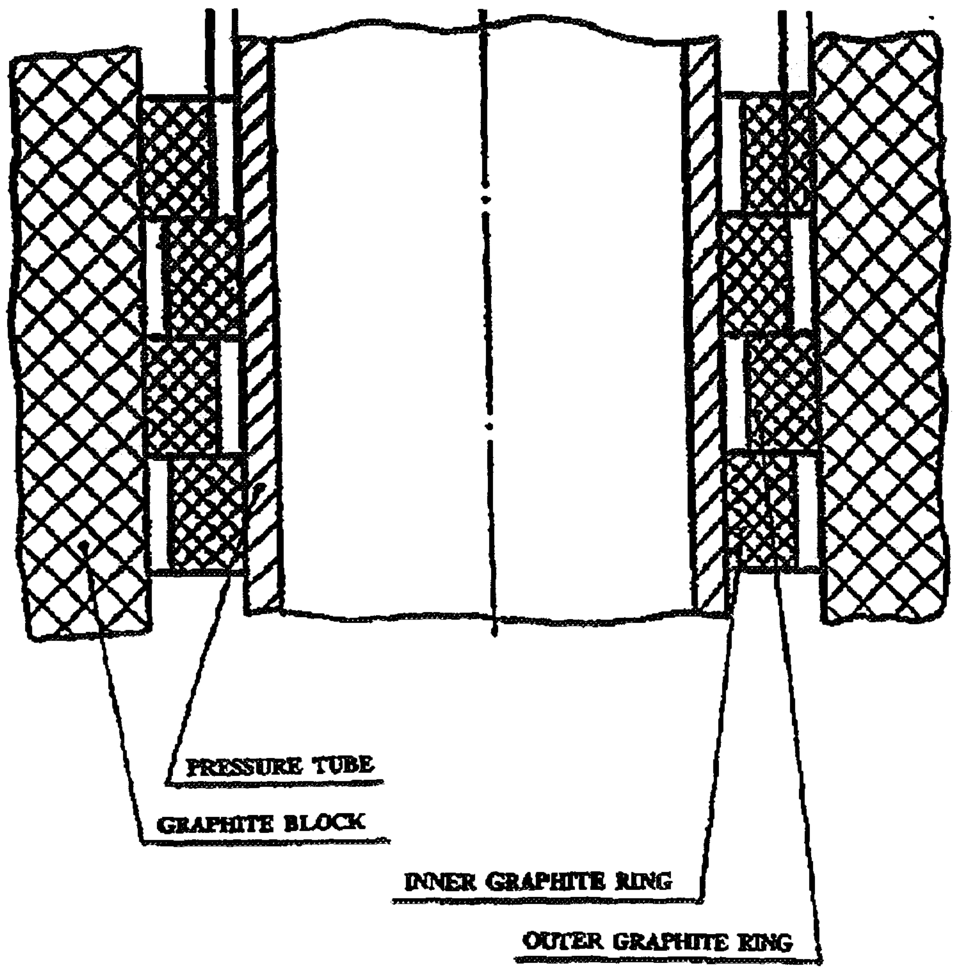


FIG. 8 Schematic of pressure tube and graphite block arrangement of a fuel channel.

To carry out fuel change at the operational reactor, a refuelling machine of RM-468 type is installed in the central hall. When this machine is connected with the pressure tube for the refuelling process it forms part of the pressure boundary.

2.6. SAFETY CRITERIA AND ACCIDENT ANALYSIS

The main principle of safety assurance used as a basis for the design of the RBMK reactor facility is to ensure that the internal and the external exposure of the operating personnel and population should not exceed the specified values; and to adhere to the standards for the content of radioactive products in the environment under normal operation and during design basis accidents. In keeping with OPB-82 requirements, the design provides for the features and organizational measures that ensure safety with any initial event, specified in this report, with coincidence of one failure of any active component (element) or of mechanical moving parts of a passive element of the safety systems occurring independently of the initial events.

The scope of accident analysis provided in the Technical Substantiation of Safety (Russian abbreviation TOB), some analogy to the Safety Analysis Report used in western countries, was determined by national regulations effective at the time the TOB was issued. Some of the computer codes used at that time for modeling and calculations were of limited capability.

During the design of the first RBMK-1000 reactor facilities the list of the accident initiating events was limited. On the basis of the experience of the reactor operation at the Leningrad, Kursk and Chernobyl nuclear power plants, and as the safety requirements became more stringent, which is a process common to nuclear power globally, the original list of initiating events was considerably expanded.

For RBMK, the list of initiating events includes more than 30 accident conditions that may be classified into 4 basic types:

- accident conditions involving reactivity variation;
- accidents in the core cooling system;
- pipe rupture accidents;
- accident conditions with equipment trip or failure.

The specific accident scenarios are developed based on the assumption that any initiating event involves an independent failure of one of the safety system components and an undetectable failure in the systems which are not normally surveyed during plant operation but contribute to the accident sequence.

As regarding evaluation of the accident conditions and development of safety measures, the RBMK design is based on the following criteria:

1. A rupture in a pipeline of maximum diameter, with coolant outflow from both ends, in operation at nominal power is considered as an ultimate design-basis accident (UDBA).

2. The safe operation limits which determine the acceptable primary coolant activity as regards the number and size of the fuel flaws, are as follows:

- 1% in case of the fuel elements fault, such as gas leaking,
- 0.1% in case of the fuel elements with the fuel directly contacting the coolant.

3. Maximum design limit of the fuel elements damage in case of pipe rupture and ECCS actuation, are as follows:

- the cladding temperature under 1200 °C,
- local depth of cladding oxidation less than 18% of the initial thickness,
- the fraction of reacted zirconium less than 1% of the total mass of claddings in channels of the core.

4. It should be made possible to unload the core and remove the fuel channel (FC) after an UDBA.

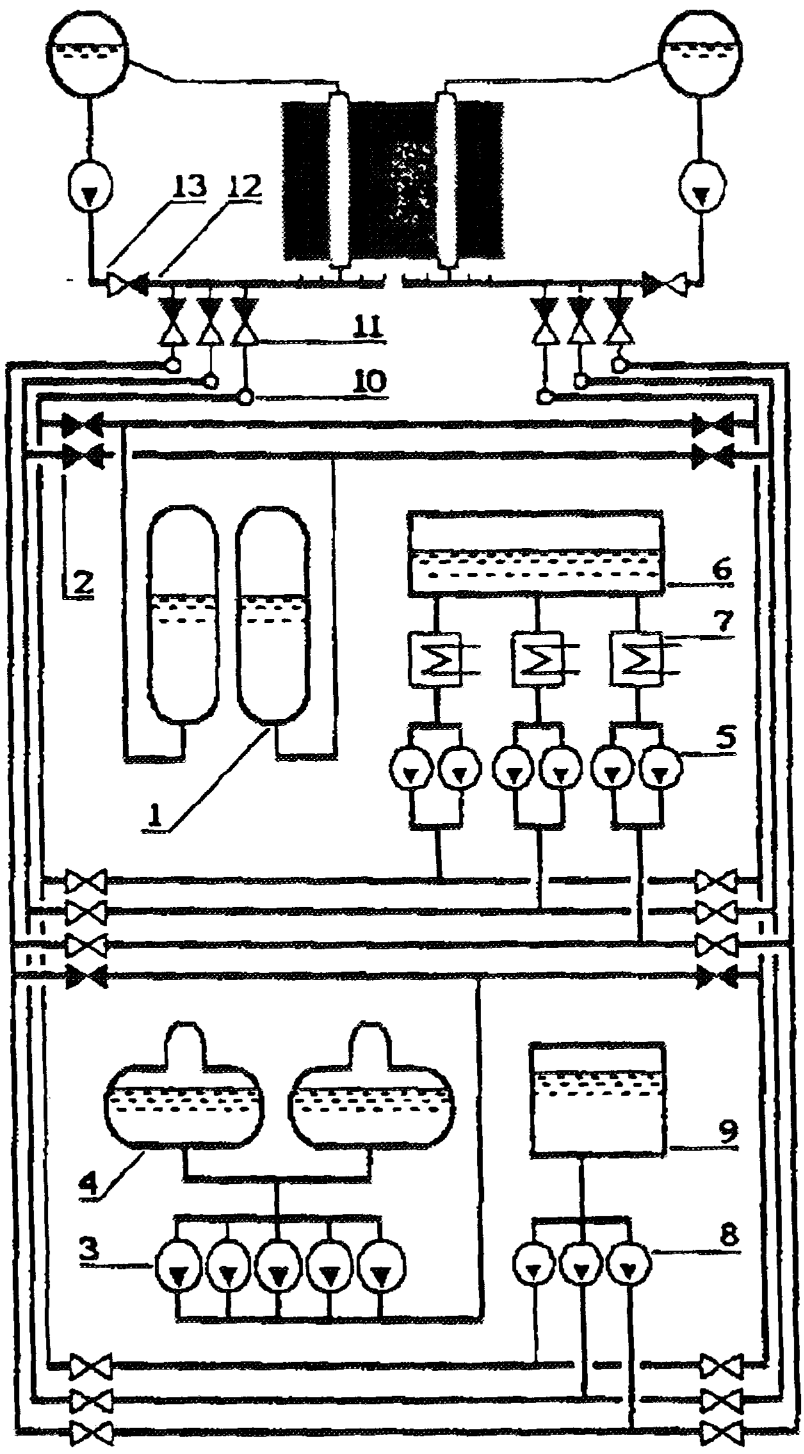
For beyond-design-basis accidents the following acceptable criteria were adopted: the probability of a core meltdown beyond-design-basis accident shall be less than 10^{-5} per reactor-year. The probability of accidents with off-site fission product release beyond acceptable limits be less than 10^{-7} per reactor-year.

The studies on beyond-design-basis accidents show that in case of second and subsequent generations of plants equipped with accident localization systems these criteria are met without implementing additional measures.

2.7. SAFETY AND SUPPORT SYSTEMS

2.7.1. Emergency core cooling system

Cooling the reactor core in case of a break in a primary circuit pipe or a failure of certain control system elements is maintained by the emergency core cooling system (ECCS) (Fig. 9).



1 - accumulator, 2 - fast-acting valves, 3 - feed pumps, 4 - deaerators, 5 - damaged side cooling pumps, 6 - suppression pool, 7 - heat exchangers, 8 - non-damaged side cooling pumps, 9 - pure condensate tanks, 10 - ECCS headers, 11 - distribution group headers (DGH), 12 - DGH check valves.

FIG. 9 Emergency core cooling system.

The design basis accident for the ECCS is a double-ended guillotine break of a 900 mm tube, corresponding to a break in the main circulation pump pressure headers or suction header (both of diameter 900 mm). Coolant is injected into the distribution group headers of the primary process loop, and provision is made for both fast-acting cooling of the core and long-term decay heat removal. The short term function is to inject water in the reactor as fast as possible and to provide cooling during two minutes. This will leave enough time for the long term subsystem to start which normally takes 40s. During the initial stage of loss of coolant accident the cooling water is provided by 5 main feedwater pumps (one pump delivers 1650t/h) which take suction from the deaerators. The capacity of the deaerator is enough to provide water to the core for several minutes.

In case of loss of off-site power simultaneously with the accident, the rundown power of the turbogenerator allows the main feedwater pumps to operate 50s. In addition, two banks of accumulators can operate for about 2 minutes.

The long-term cooling system comprises 6 emergency core cooling pumps taking suction from the accident localization system (ALS) condensing pool for cooling the damaged half of reactor and 3 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. The various valves in the system are powered from a battery backed-up power supply. Water released from the broken pipe finds its way to the pressure suppression pool and to the wastewater pool. From the latter it is transferred to the pure condensate tank through the purification system.

On first generation units there is no accident localization system and hence the emergency core cooling system has fewer channels and fewer pumps since there is only coolant injection from the clean condensate tank and none from a suppression pool. For the same reason there is no ECCS recirculation on these units.

2.7.2. Accident localization system

The RBMK NPPs is protected by a pressure suppression type containment which is referred to as Accident Localization System (ALS) (Fig. 10). This system encloses part of the main circulation circuit and consists of leaktight compartments. All main pipelines, headers and components carrying cooling water are part of the accident localization system. There are two major parts of rooms of the ALS: the compartment system of the rooms below and to the side of the reactor cavity in which the inlet pipelines to the fuel channels and distribution group headers are located (24 100m^3, 0.66 m^3/h leakage and 2.0 bar design overpressure) and the system of compartments where the main circulation pumps and the suction and pressure headers are located (11 500m^3, 0.6 m^3/h leakage and 2.65 bar design overpressure). The tightness is measured once in every four years.

The reactor cavity, the main coolant pumps and pressure header rooms, the lower water line rooms and the steam distribution corridor are connected to a pressure suppression pool system by 236 steam dumping channels (total cross-section area: 50 m^2) immersed in its water space.

The pressure suppression system consists of two floors of concrete compartments with steel liner. The water level is 1.2 m in each compartment. The total water volume, which is also the ECCS water, equals 3200 m^3 whereas the free space of the pressure suppression pool is 3700 m^3.

In addition to a system of valves between the compartments of the pressure suppression system, the localization equipment includes also a sprinkler system with pump and heat exchangers and surface condensers in the steam distribution corridor between the MCP compartments. The rooms where the steam-water lines, the drum separators and upper parts of the downcomers are located are not included in the accident localization system.

There are some differences in the design of the accident localization system of the second and third generation units.

2.7.3. Reactor cavity overpressure protection system

The reactor cavity overpressure protection system is an important part of the safety system of the RBMK. The cause of the overpressurization is postulated to be a failure of pressure tubes inside the reactor cavity. A pressure relief is provided by pressure relief tubes which connect the reactor cavity to the accident localization system via a water lock. The design basis is the rupture of one tube.

The existing overpressure protection system of Smolensk 3 has the capacity for the simultaneous rupture of up to 9 pressure tubes under conservative assumptions, thereby providing considerable margin over the DBA which involves a single pressure tube failure. (No specific requirement to accommodate 9 simultaneous failures seems to exist except to establish a larger safety margin.)

The overpressure protection system is different in the first and second generation RBMK design (i.e., without and with pressure suppression pools, respectively). In the first generation plants, the discharged steam is condensed in a surface condenser, and non-condensable gas enters a holdup tank before being released to the building stack via the gas cleanup system. In the second generation plants, the discharged steam is piped to separate zones of the bubbler-condenser pool where it is condensed. The bottom of the pipes are submerged approx. 2.8 m in the pool, so there is a 1.8 m water seal between the cavity and the pool which must be exceeded in order to initiate steam discharge.

2.7.4. Other systems

To maintain the required water quality in the MC the water is continually cleaned to remove various mechanical admixtures which are usually radioactive. Letdown is from the main circulation pump outlet headers and passes via regenerative heat exchangers and coolers to a filtration system. Cleaned water is then mixed with the main feed-water entering the drum separators. Flow is maintained by the main circulation pumps. During shutdown and post-accident the system operates in much the same way except that letdown is directly from the pipe connecting the drum separators and bypasses the regenerative heat exchangers. Flow is maintained by dedicated decay heat pumps.

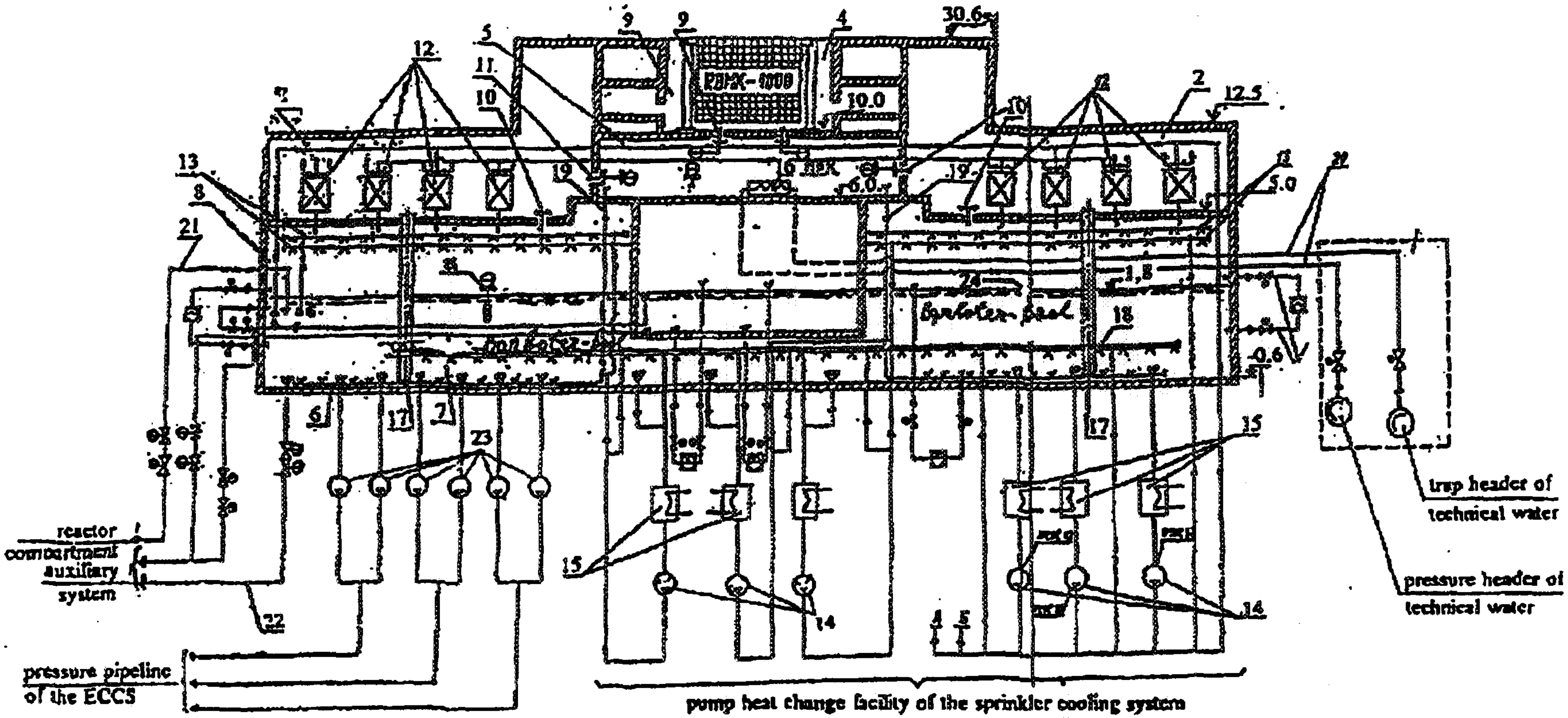
Fission products from operating reactor get to the reactor and MC compartments in various way. Special systems are used for the purpose of lowering the radioactive background in these compartments. The filtered air is released from the 150 m high venting stack. The most contaminated gases by radioactive products are gases from the so called gas circuit. These gases (helium-nitrogen mixture) are treated by special gas-discharging cleaning system.

Fire detection is provided by two different automatic systems. One consists of dual detectors in a single unit which detects both smoke (products of combustion, POC) and temperature (thermal detection) and a second system which detects high temperature. Both systems alarm in the control room.

Fire suppression consists of an automatic open head deluge system and manual fire hose stream, and portable fire extinguishing systems. The automatic deluge system actuates on operation of both the smoke and thermal detector in the first detection system or on one of the thermal detectors alone in the second system.

There are two fire suppression systems at the plant, the local automatic sprinkler suppression system for safety related cable tunnels (LASFSS) and the overall plant fire suppression system (ANFSS).

Fire suppression for a cable tunnel (for example, of a train I) is provided by trains II and III of the LASFSS. Each train is equipped with an independent tank, fire pumps, valve, and delivering and feeding pipelines. Normally LASFSS is in the state of `waiting' and its pipes have no pressure except several



1-2. leaktight compartments, 3-4. under reactor compartment, 5. steam distribution corridor, 9. membrane valves, 10. discharge valves of pressure suppression pool, 11. check vavles of leaktight compartments, 12. ejection coolers of the spray system, 13. sprinklers, 14. pumps of the spray system, 15. heat exchanger of the spray system, 16. condensers, 17. steam discharge pipes, 18. perforated header, 19. condensate discharge pipes from the steam distribution corridor, 20. service water supply to condensers, 21. system for filling in the pool, 22. system for draining the pool, 23. pumps of the ECCS.

FIG. 10 Accident Localization System.

sections that are under hydrostatic pressure. LASFSS is completely automated. Reliability of the system meets the corresponding requirements.

The overall plant fire suppression system (ANFSS) is designed for automatic fire extinguishing of non-safety related cable tunnels, transformers and oil system of the main circulation pump (MCP). The ANFSS (plant fire main loop) is supplied with water from the service water pipeline which is supplied with water from the cooling pond, not from deep wells.

Passive protection is provided by means of 1.5 h fire rated barriers (partitions) between the cables for three safe shutdown trains in the reactor building.

3. OVERVIEW OF SAFETY IMPROVEMENTS TO RBMK NPP

Since the Chernobyl accident, in 1986, a considerable amount of work has been either completed or is under way to improve RBMK NPPs safety. This overview of safety improvements to RBMK NPPs has been made, attempting to follow the same topical areas which have been used in the safety review of these reactors. The information available to the IAEA and that provided by the main design institute (RDIPE, Moscow) of the RBMK reactors about the safety upgrading programmes of individual plants have been used.

In order to cross-reference this information and the generic safety issues identified, a list of data on safety upgrading measures, available in the IAEA database, is included in Annex II.

Naturally, major safety improvements have been implemented in all RBMK units to remedy design weaknesses which had been recognized as a result of the 1986 Chernobyl accident. The majority of these modifications addressed the reactor core. All of these modifications aimed at reducing the magnitude of the void reactivity effect and of increasing the speed and effectiveness of the shutdown system. These modifications have improved the situation, as far as the reactivity induced accident is concerned, significantly.

3.1. CORE DESIGN AND CORE MONITORING

The designers' efforts were directed at decreasing the void reactivity coefficient and at improving the design of the control rods.

Void reactivity coefficient

The main measures taken to reduce the void reactivity coefficient are the following:

- loading of additional absorbers. At Smolensk unit 3, 97 additional absorbers had been loaded in June 1993. In fact, the number of additional absorbers varies from 85 to 103 depending on the reactor. The technical specifications require at least 81 additional absorbers.
- increasing the fuel enrichment from 2.0 to 2.4%. This increase is underway in all the RBMK reactors except Ignalina NPP.
- controlling the operational reactivity margin (ORM) value between 43 and 48 equivalent control rods.

Control rod design

Before the Chernobyl accident, when the Control Rods (CRs) were withdrawn out of the core, there was a column of water (neutron absorbing material) at the bottom end of RCPS channel, under the displacer. During the insertion of the CRs, this column of water was, in a first step, replaced by the graphite displacer, causing an increase in the reactivity. In addition, the CR drop-time was too long.

Three safety improvements were carried out to improve the efficiency and to increase the speed of response of the emergency protection system.

The manual control rods were replaced with ones of improved design without the water columns at the bottom end of the RCPS channels. They have a larger absorber section.

The RCPS rod drives were also retrofitted, reducing the time required to insert the rods fully into the core from 19s to 12s. The implementation of these first two measures has improved the response efficiency of the emergency protection system during the first few seconds of rod insertion.

As a third measure to improve the performance of the emergency protection system, a fast-acting emergency protection system was developed and systems of this kind were installed in all operating reactors.This new system (BAZ) consists of 24 fast acting protection rods with insertion times of 2-7 s depending upon whether BAZ or AZ-1 operational mode is called upon; these rods cause an insertion of negative reactivity of 2 β.

In addition, the number of the bottom rods has been increased (except for the first generation plants) from 24 to 32 and they are now inserted when a scram occurs.

Other measures

Following the Chernobyl accident, other measures have been taken to improve the control and monitoring system:

- operator aid display at the control desk based upon information from the SCALA system
- manual reactor trip when power falls below 700 MW(th)
- manual trip if the ORM is less than 30 equivalent control rods.

Further measures are planned:

- increase in the number of detectors for LAR-LAZ
- increase in the number of axial flux detector assemblies.

3.2. INSTRUMENTATION AND CONTROL

There have already been three significant steps in upgrading the data processing systems on the RBMK NPPs. First, the development of the TITAN system for the Ignalina NPP saw the introduction of commercial computer systems, some automatic transfer for warning set points to the indication systems, and the introduction of color displays for 2D and 3D imaging of the power and temperature profiles.

The next item was the addition of a data transfer facility at Smolensk 3 which allowed the row data to be processed by an auxiliary computer, a PC in the control room. This system still relies on the original SCALA for data collection and processing.

The upgrade exercised at Leningrad NPP has seen the introduction of SCALA-M. This system uses Intel processors. However, the system retains the original SCALA data collection equipment.

3.3. PRESSURE BOUNDARY INTEGRITY

Several modifications have been implemented to enhance the pressure boundary integrity at RBMK NPPs:

- some steam/water pipes supports have been added to deal with the situation observed related to piping vibration;

- a simplified engineering approach, the strain intensity factor concept and limit stress approach have been used to develop tables of allowable flaw sizes. These regulate which flaws must be repaired upon discovery during in-service inspection;

- probabilistic fracture mechanics calculation have been performed for the 800 mm pipe to predict leak and rupture probabilities.

A new design of flow control valve is being installed at all plants, as a measure to prevent ruptures of fuel channels that have been caused by lack of coolant flow when valves have malfunctioned.

To make expert assessment of the MC metal state in RBMK reactor there have been developed defect visualization (determination of geometrical dimensions) systems based on multifrequency acoustic holography. Two pilot systems have been tested at Leningrad NPP Unit 2 in 1992. On the basis of these systems and in accordance with the test results an ultrasonic computer flaw detector Avgur 4.1 and a portable computer system of USI Avgur 4.2 were developed to carry out data coherent processing according to revealed defects aiming at their classification and determination of geometrical dimensions.

3.4. ACCIDENT ANALYSIS

Since the Chernobyl accident there has been a significant programme underway to upgrade the computer codes for safety analysis of the RBMK NPPs. Modern Western codes like RELAP 5 and ATHLET have now been acquired and the advanced models of the RBMK reactors were developed using the RELAP5/mod3 Code for various NPPs: Ignalina Unit 1, Leningrad Units 1 and 2, Smolensk Unit 3.

The small breaks of DGH were studied using the RELAP5/mod3 and ATHLET Codes. These investigations involved the numerical analysis of the sensitivity and study on the influence of a break nodalization on the critical parameters of a channel (temperature of fuel rods and pressure tubes).

The first version of three-dimensional mathematical model, TRIADA, was developed by late 1986. The version describes three-dimensional neutron kinetics with 140 effective channels considering that up to 60 horizontal layers are utilized. The model was used to analyse the RBMK safety improvement measures and the first stage of the Chernobyl accident.

Systematic studies of RBMK PSA were started in 1988. The PSA methodology was developed and software for IBM computers was accomplished taking into consideration RBMK features in 1988-1990 (PSA-0). The second stage (1991-1992), devoted to implementing PSA for Leningrad NPP Unit 1, has incorporated categorization of states with RBMK core damage under severe accidents, analysis of severe accidents, development of NPP models before and after upgrading.

Ignalina NPP was analyzed in detail in the recently completed “Barselina” Project [19]. PSA activities were a significant part of the EC sponsored programme on “Safety of RBMK reactors”, where, in the framework of the “Task Group 9”, a Pilot Risk Study (PRS) for Leningrad NPP Unit 1 [20], Smolensk NPP Unit 3 and Ignalina NPP Unit 2 were performed [21]. A second PRS of Leningrad NPP Unit 1 was performed within the UK/RF bilateral programme to consider in more detail the value of the reconstruction measures [22]. The RDIPE pioneered the PSA for RBMKs by performing a Level 0+ study for Leningrad NPP Unit 1 [23]. Except for Ignalina PSA, no other RBMK type NPP has a full-scope Level 1 PSA completed or in progress.

3.5. SAFETY AND SUPPORT SYSTEMS

The main improvements have been carried out in the following:

Emergency core cooling system (ECCS), accident localization system (ALS) and reactor cavity overpressure protection system.

3.5.1. ECCS

Several different improvements have been implemented:

- increasing the number of emergency feedwater pumps from 3 to 5 and the number of ECCS lines from 1 to 2 for NPPs of the first generation.

- installing additional ECCS pumps (3 for damaged core side cooling and 3 for undamaged core side cooling) and the associated 3 divisions of piping.

- installing check valves between the distribution group headers (DGH) and the main coolant pump discharge header

- installing additional, larger capacity accumulators.

3.5.2. ALS

As it was previously mentioned, the reactor coolant system of 1st generation RBMKs is not enclosed in a leaktight accident localization system (ALS) as is the case in the 2nd and 3rd generation plants. Even in the 2nd and 3rd generation NPPs, only a main part of the reactor coolant circuit is confined by a system of pressure compartments of an ALS. The rooms where the steam/water lines, the steam drum separators and the upper parts of the downcomers are located are not included in the ALS.

For RBMK plants of the 1st generation, a decision was already made to construct a separate building housing a pressure suppression system. The building shall be connected to the reactor building. Therefore, this system shall allow the decrease of radioactive release to the atmosphere during DBA for these units.

Furthermore, the installation of a leaktight compartment system for the rooms of the steam/water pipelines, the steam separators and the central reactor hall is being considered.

3.5.3. Reactor cavity overpressure protection system

Protection of the reactor cavity against overpressurization is an important safety feature of the RBMK. The existing overpressure protection 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 of one channel tube rupture. The existing steam discharge system vents the steam/gas mixture from the cavity to a condenser with subsequent gas holdup and release through the gas clean-up system/stack for first generation units. For second and third generation units, the discharged steam/gas mixture is routed to the bubbler/condenser pools where the steam is condensed, and gas is retained in the leaktight spaces.

There is an intention to improve the capacity of the cavity overpressure protection system. This work is being conducted in stages which are either completed, under way, or planned for all units concerned.

1st construction stage - A modest increase in capacity amounting to one or two tubes of additional relief is achieved in the first stage by adding two vent valves to existing cavity steam discharge pipes at the top of the reactor. These valves vent the reactor space into the atmosphere upon reaching 2.8 bar absolute pressure in the cavity.

2nd construction stage - A further increase of the capacity for simultaneous rupture of nine tubes is achieved by addition of a new 600 mm steam discharge pipe into the cavity. This pipe also has two atmospheric vent valves, and additionally is routed to bubbler/condenser pools where they exist.

3rd construction stage - New buildings containing bubbler/condenser pools in sealed areas are planned for first generation units. When existing, the 600 mm steam discharge line would be routed to these pools, eliminating the need for atmospheric vent valves.

3.6. OPERATIONAL SAFETY

The matters relating organization, the adequate number of staff and training, as well as division responsibilities between NPP administration and an operating organization are presented in Quality Assurance Programs, which have been issued in 1993-1994. Recommendations on an equipment maintenance programme on updating standards, and on data of equipment availability and its performance including failures, preventive measures and assessment of their efficiency have been also incorporated into these quality assurance programmes.

Quality assurance programmes in the operation area have been developed for Leningrad NPP (all units) and Smolensk NPP (all units). Quality Assurance programmes for Leningrad NPP Unit 2 and Smolensk NPP Unit 3 were submitted to GAN for consideration and approval. The others will be submitted for approval according to schedule.

Prior to shutting a unit down for upgrading, a review to determine the safety level after a corresponding stage of up-grading should be produced according to the Gosatomnadzor (Russian Safety authority) recommendations. This was done for Leningrad NPP Unit 1 in 1989, for Leningrad NPP Unit 2 in 1992 and for Kursk NPP Unit 1 in 1994.

Organizational measures have been introduced to prevent the possibility of bringing the reactor to an unstable condition. A new operating instruction has been prepared for each RBMK NPP based on the general one developed by RDIPE in 1991. This instruction defines a number of parameters, any deviation from which can lead to an accident. The operational limits and conditions are also specified in this report.

In accordance with the instructions, the reactor should be shutdown by the operator in the case where some indicators of an accident condition have been appeared. The new requirements have been introduced for the shutdown process in the power range below 700 MW. It should be performed without any stoppage up to zero power.

Some organizational measures have been also introduced to exclude unauthorized disabling of the reactor protection systems.

4. GENERIC SAFETY ISSUES

4.1. REVIEW APPROACH

The safety review of RBMK NPP in the IAEA programme was performed by expert meetings and safety review missions to particular plants' sites. Experts from OECD countries together with RBMK specialists have reviewed the various safety issues, their ranking and recommended measures for improving the safety of RBMK reactors in the following areas:

- core design and core monitoring;
- pressure boundary integrity;
- accident analysis;
- support and safety systems;
- fire protection;
- operational safety.

The international practice reflected in the IAEA NUSS publications, the Russian regulations and national practices were used by the experts for identification and assessment of the safety issues. The rank of each design safety issue was determined using the previously agreed definitions (Table II), taking into account the safety improvements already underway or implemented.Operational issues have not been ranked because it was considered that the agreed definitions of the issue categories do not fully apply to operational related matters. However, it was agreed that RBMK operational safety can be upgraded. The operational safety improvements should be implemented in parallel with the design modifications and a balanced approach should be achieved in allocating resources to both design and operational safety areas.

TABLE II

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 NPP, 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 specific NPP, 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.

Individual generic safety issues identified are presented in this report according to the following format:

1.	AREA/NUMBER	Topic area under review/ code number (one of seven topical areas).
2.	ISSUE TITLE	Short name of the safety issue.
3.	ISSUE CLARIFICATION	Concise description of the identified safety issue which characterizes the safety concern.
4.	ISSUE CATEGORY	Category of the identified issue according to the definitions of the safety categories agreed for the design issues.
5.	JUSTIFICATION OF THE CATEGORY	Brief justification for the selection of the safety issue category in item 4 (consideration is given to safety improvements already implemented.)
6.	APPLICABILITY	Nature of the safety issue relative to the NPPs, e.g. generic, generation specific, plant specific.
7.	RECOMMENDATIONS	Consolidated measures proposed to resolve safety issues identified in item 3. For each measure an indication of its adequacy to resolve the issue (e.g. partially addressing the issue or fully resolving the issue) is given.

In order to cross-reference the safety issues in this report with the various reviews, and to enable the reader to identify the findings and recommendations related to a specific issue in the various reports and reviews, a list of agreed recommendations is included in Annex I. It is important to note that some of the recommendations listed came from different reports and are repetitive. Moreover, some of the recommendations from early reviews are superseded by later reviews which include improved knowledge and understanding of the RBMK systems. In some cases, actions to address specific recommendations are already underway or completed.

It should be noted that this report was prepared on the basis of safety reviews of Smolensk 3 and Ignalina NPP. Therefore, it is primarily applicable to RBMK NPPs of this vintage.

Whenever applicable, reference is also made to RBMKs of other generations. Specific safety issues and improvements for RBMKs of 1st and 2nd generation will be consolidated upon completion of ongoing studies for these NPPs.

4.2. OVERVIEW OF GENERIC SAFETY ISSUES

A total of 58 safety issues in the seven topical areas were identified and reviewed in accordance with the approach specified above. The ranking was applied only for the six areas related to plant design. Operational safety issues, particularly those related to ensuring that a high safety culture is an underlying basis for operation, were considered very important for all NPPs with RBMK reactors and all efforts should be made to implement the related recommendations along with design modifications.

Table III presents the number of the safety issues identified for each one of the seven topical areas referred to in Section 1 and their distribution between different categories. The detailed description of each identified issue and proposed recommendations are in Section 4.3.

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

The ranking represents the agreement reached through discussions and expert judgment based on current technical understanding of the safety issues. *(In the accident analysis area RBMK specialists did not agree with category High for LOCA analysis issue since this analysis had already been performed for all RBMK NPPs before their commissioning and now this analysis is only to be reassessed and extended. This situation differs from that of ATWS which is considered by RBMK specialists as a High category issue.)*

Two broad issues addressing quality assurance (QA) and regulatory interface have not been specifically attributed to any particular topical area, but were recognized as affecting all of them.

The main concern relative to QA relates to ensuring that the design bases for the various analyses, safety reviews and safety upgrading actions are based on the actual plant status and configuration. Another aspect of this issue is ensuring that the relevant design documentation is updated as the plant configuration is modified and upgraded. It is therefore of utmost importance that the organizational structure promotes the raising of safety concerns, responds quickly in evaluating these safety concerns and implements timely corrective actions if they are warranted. Some of these aspects are addressed under the topical area of operational safety (see safety issues Operational safety 1, 2, 3, and 11) where they were dealt with according to specific recommendations from the various reviews and reports.

It is recognized that exactly how and when the identified safety issues are to be addressed is a matter to be resolved between the operating organization and the regulatory body. The expert meetings and the safety reviews are intended to help this process by bringing international expertise to assist with the identification of possible solutions or methods to resolve specific issues. However, recommendations and conclusions resulting from the IAEA Extrabudgetary Programme are only intended to provide an additional technical basis for decisions to improve the safety of RBMKs and assist national decision makers who have sole responsibility for the regulation and safe operation of their nuclear power plants. Therefore, these results do not replace a comprehensive safety assessment which needs to be performed in the frame of the national licensing process.

TABLE III

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 to design modifications.

4.2.1. Core design and core monitoring

A total of six safety issues related to core design and core monitoring were identified and ranked. Five of them, placed in High category, are related to features of core design and the adequacy of the RBMK shutdown systems. One issue was ranked Medium in the area of spatial power control and protection.

It is recognized that work has been either completed or is under way to change neutron characteristics of the core and to modify the control and protection systems. Considerable work has been completed to decrease the void reactivity of the primary system and increase the efficiency of the shutdown system. However, the problem of the void reactivity associated with the loss of coolant from the channels of the control and protection system (CPS) and the issue of independent and diverse reactor shutdown - both issues of high safety relevance - have not yet been resolved. In particular, the shutdown issue was thoroughly discussed during a special meeting held in 1993 [9]. During that meeting the intent of the Russian designers to develop and modernize the RBMK CPS in order to provide a higher safety level was strongly supported by the international experts. This intent was confirmed during this review and it was recommended that the additional shutdown system should be developed with due regard for ongoing developments in the area of core design.

The safety significance of the safety issue related to the operational reactivity margin (ORM) is high since the ORM has to be controlled in order to maintain the void reactivity coefficient, the effectiveness of the shutdown system (insertion rate, shutdown subcriticality) and the power distribution within the given safety limits. With the present design it is the responsibility of the operator alone to keep ORM within these safety limits. It was recommended to automate the shutdown actions when the ORM value falls below the safety limits to reduce the safety significance of the manual ORM control.

The design and safety analysis of the RBMK reactors were performed with the calculational tools available at that time. These tools (usually computer codes) did not have the capability to adequately model spatial interactions between neutronics and thermohydraulics. Taking into account that this is very important for RBMK safety analysis and core design, experts recommended that development of 3-D methods for predicting space-time dependent depletion, neutron fields, coolant density and temperature distribution of fuel and graphite be continued. These methods should be used to confirm the results of the safety analysis and to undertake studies to help in defining further actions.

4.2.2. Instrumentation and control

Instrumentation is a major aspect related to nearly all plant systems. However, in reviewing this area, experts have tried to concentrate on the safety related instrumentation issues rather than on those relating to control functions (e.g. core monitoring, safety and support systems).

A total of seven safety issues related to I&C were identified and ranked. Two of them, placed in High category, are related to the diversity and segregation of the I&C systems and to the initiation of emergency core cooling system (ECCS) and other safety systems.

The major concern is the segregation of the electronic systems and the level of diversity present in most important systems and equipment. This issue is seen as being very important as the defense-in-depth of the reactor is compromised. For example, the flux control system shares many common elements with the shutdown system and, 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, thus control and protection could be lost simultaneously.

The ECCS is initiated by a combination of signals. However, there is no sufficient assurance that the system responds promptly nor that the actuation equipment could have a common cause failure.

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

A number of areas related to I&C could be improved by local measures at the plant. These include in the main test and maintenance procedures and the recording of failure data and its subsequent use to plan maintenance and support safety submissions.

4.2.3. Pressure boundary integrity

In the pressure boundary area a total of seven safety issues were identified and ranked. Four of them, placed in High category, are related to fulfillment of the inspection requirements, in-service inspection (ISI), break of critical components, and fuel channel and tract integrity. Other safety issues were ranked Medium and Low.

The reviews indicate that at some plants operation has continued even though the frequency and the number of examinations required by the national regulations for the reactor pressure boundary are not performed or when the results are not satisfactory. The existing time schedules for implementation of modifications and additional analysis as well as the requirements for record keeping are not followed. Criteria for limiting plant operation in these cases are not established.

The required high volume of in-service inspection is not fulfilled in practice. It was found in the review that 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 in-service inspection predictive records are not maintained. The equipment and procedures used are inadequate to give repeatable measurements of subcritical defect size.

Some primary coolant circuit components and piping are outside of the accident localization system. The guillotine break in 800 mm diameter piping in the first generation of RBMK NPPs could result in damage to civil structure. Application of the leak-before-break (LBB) concept would reduce risk of primary coolant circuit failures. But the applicability of this concept for RBMK conditions is not fully demonstrated and LBB method and techniques are not in use.

To date, there have been three single channel ruptures due to water flow blockage or power flow imbalance. It was recommended to analyse and to implement, as feasible, the reduction of the number of in-line components, the failure of which can result in a water flow blockage.

The last issue identified in this area, and ranked as Medium, dealt with seismic and aging assessment which may lead to additional safety improvements not yet defined.

4.2.4. Accident analysis

A total of ten safety issues related to accident analysis, including methodology, scope and a specific issue reflecting the need of completeness were identified and ranked. Three issues, placed in High category, are related to LOCA analysis, cavity overpressure protection and anticipated transient without cram (ATWS) analysis. (However, RBMK specialists did not agree with category High for LOCA analysis issue since this analysis had already been performed for all RBMK NPPs before their commissioning and now this analysis is only to be reassessed and extended. This situation differs from that the ATWS which is considered by RBMK specialists as a High category issue.) Other issues were ranked Medium.

The scope of analysis of postulated accidents available in the Techical Justification of Safety (TOB) was determined by national regulations effective at the time 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 the assumptions used in the analysis. The computer codes used at the time of RBMK design were of limited modeling capability. The lack of an experimental data base on pipe rupture of primary heat transport system limited the possibility of integral code validation. Presently, more modern Russian codes and some western codes are being used, but these codes have not been sufficiently validated for modeling RBMKs.

The development, the validation and use of codes for accident analysis do not follow rigorous principles of quality assurance.

The completeness of analysis is, of course, a requirement for ensuring 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 to perform a PSA analysis. Therefore the performance and peer review of a plant specific PSA for all RBMKs is recommended.

4.2.5. Safety and support systems

A total of ten safety issues concerning safety and support systems were identified and ranked. Four issues placed in category High are related to ECCS capability, performance and reliability, and to the adequacy of the confinement function.

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 the supporting systems such as the service water and intermediate cooling systems. Moreover, the high level of redundancy in the safety systems cannot always be given full credit due to potential common cause failures.

Several of the recommendations given for one specific plant may be applicable also to other plants. In general, it has been found that the differences between the plants are so important that each recommendation has to be evaluated on a plant-specific basis.

The reliability of the safety systems is, to a large extent, dependent not only on the system design and alignment, but also on operational parameters, such as maintenance and testing procedures and emergency operating procedures, enabling the operators to identify accident situations and to initiate correct actions. Therefore, there is a strong tie between this area and generic recommendations require the development of emergency operating procedures and testing procedures.

4.2.6. Fire protection

A total of five safety issues related to fire protection were identified and ranked. They follow the structure of the IAEA Guidelines for Inspection of Fire Protection Measures and Fire Fighting Capability at NPPs. One issue placed in High category is related to passive fire protection. Other issues were ranked as Medium or Low.

Passive fire protection can address the fire safety problems in an effective way. Passive fire protection comprises all the measures that are settled 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, 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 the 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, some in connection with bilaterally/internationally financed programmes.

Manual fire suppression capability is traditionally very strong at NPPs 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 of the troops, communication equipment and fire fighting equipment, such as 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 confirms proper availability and operation of both manual and automatic fire suppression capability. However, differences between sites and generations are extensive and measures of different magnitude are needed. Fire extinguishing water supply is in category Low, because related shortcomings are minor.

4.2.7. Operational safety

A total of 13 safety issues related to operational safety were identified and reviewed. They were not ranked, as explained in Section 1. However, upgrade of operating safety is a requirement of utmost importance to improve nuclear safety of operating NPPs.

Past experience in the operation of NPPs 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. In order to make full use of the potential of people in the NPP it is important to have good documentation and working conditions.

The analysis of operating capability conducted in RBMK NPPs shows that operational safety can be upgraded. With this objective in mind, recommendations from various reviews have been generated. The related recommendations should be implemented in parallel with the proposed design related safety improvements and a balanced approach should be achieved in allocating resources to both design and operational safety areas.

4.3. INDIVIDUAL SAFETY ISSUES

4.3.1. Core design and core monitoring

1. Area / Number: Core design and core monitoring / 1

2. Issue title: Core design and core design methods.

3. Issue clarification: In the RBMK reactors, there are strong spatially dependent interactions between thermohydraulics and neutronics. To take them properly into account, three-dimensional neutronic codes with thermohydraulic feedback are needed for both static and dynamic calculations, which requires the improvement of basic data libraries. The protection against local power excursion should be improved to better predict thermohydraulic instabilities. The present design is based on calculation tools which were available at the design time and which did not have these capabilities.

4. Issue category: High

5. Justification of the category: The defense-in-depth concept requires adequate design methods. Computer codes without the capability to model spatial interactions between neutronics and thermohydraulics are not considered adequate for RBMK safety analysis and core design.

6. Applicability: Generic

7. Recommendations:

7.1. The safety evaluation relies on the capability of codes describing the full core behaviour. The need to improve 3-D reactor core models including thermohydraulic feedback and to develop coupling between 3-D core codes and thermohydraulic modelling of the main coolant circuit are strongly emphasized. International collaboration in this field will be useful.

It is recommended that the 3-D code development and validation, which is currently under way, be continued to provide improved calculational technique in the following areas:

- more accurate interpretation of the measured void coefficient;
- fuel burn-up distribution;
- OEM calculations under all conditions;
- space and time dependent neutron, water density, fuel temperature and graphite temperature distributions;
- predicting thermohydraulic instabilities.

7.2. 3-D analysis of thermohydraulic instabilities should be performed over the whole power range and for natural circulation conditions.

7.3. Validation of each step involved in the calculational routes should be performed including nuclear constant libraries as well as 3-D complex core codes. These validations have to be based on experimental and operational RBMK data. For this purpose the knowledge of RBMK core parameters such as 3-D burnup distribution has to be improved. The validation should include thermohydraulic phenomena and the capability of 3-D codes to predict thermohydraulic instabilities.

1. Area / Number:	Core design and core monitoring / 2
2. Issue title:	Core design void reactivity coefficient of the primary and CPS circuit.
3. Issue clarification:	The positive void reactivity coefficient in the RBMK primary system has been of concern since it has been identified as a major contributor to the Chernobyl accident. A large amount of effort and many positive measures have been taken to reduce its absolute value and its significance in consideration of the safety of the RBMK. However, little has been done so far to reduce the void effect in the CPS cooling circuit and the void effect in the primary coolant circuit, at low power still has to be assessed.
4. Issue category:	High
5. Justification of the category:	The issue has major impact on plant safety because of the role of the void reactivity coefficient in the Chernobyl accident. It continues to be important because of the role of voiding the core or the CPS channels in accidents, transients and particularly in ATWS transients.
6. Applicability:	Generic
7. Recommendations:	
7.1.	The intent of the designer to modify the control rod design in order to reduce the CPS void effect is strongly supported.
7.2.	Verify and validate calculation and measurement of void reactivity coefficient of the primary coolant circuit.
7.3.	Burnable poisoning (homogeneous solution) to reduce the void effect in the primary coolant circuit instead of additional absorbers (heterogeneous) should be investigated in order to reduce local effects.

1. Area / Number:	Core design and core monitoring / 3
2. Issue title:	Spatial power control and protection
3. Issue clarification:	Presently the spatial power control and protection system in some RBMK NPPs is based on a 9-zone system. This system does not provide adequate protection against erroneous rod withdrawal accidents since it cannot detect or cope with single rod withdrawals out of 40 core positions (dangerous rods). This is the reason why a change in local power control system is needed. On the other hand, the SCALA and TITAN computer systems appear to be overloaded and currently have a slow cycle time.
4. Issue category:	Medium
5. Justification of the category:	The defense-in-depth is not sufficient since the fuel clad integrity is not assured under rod withdrawal accident (RWA) conditions. However, according to the designer's analyses, the mechanical integrity of the pressure tube is maintained during such accidents.
6. Applicability:	Generic
7. Recommendations:	
7.1.	The twelve zone local automatic control/local emergency protection system which reduces the number of potentially dangerous rods to four should be implemented in all RBMKs.
7.2.	Independent analyses are required to confirm the designer's analyses.
7.3.	Rod insertion limits on the dangerous rods are recommended in order to limit the reactivity worth of the rods in jeopardy.

1. Area / Number:	Core design and core monitoring / 4
2. Issue title:	Operational reactivity margin (ORM)
3. Issue clarification:	The ORM is the excess reactivity required for safe operation of the RBMK reactor. It is expressed in terms of the equivalent number of fully inserted rods, necessary for its compensation. The ORM must be kept between two safety limits: a high limit to maintain a sufficient subcriticality shutdown margin, and low limit to keep an acceptable void reactivity coefficient. With the present design it is the responsibility of the operator alone to keep the ORM within the safety limits. Manual shutdown is claimed as a line of defense and is indeed the only line of defense for keeping the ORM within these.
4. Issue category:	High
5. Justification of the category:	The ORM has to be controlled in order to maintain the void reactivity coefficient, the effectiveness of the shutdown system (insertion rate, shutdown subcriticality) and the power distribution within the given safety limits. The burden put on the operator by the ORM concept is high, and safe shutdown is too susceptible to human errors.
6. Applicability:	Generic
7. Recommendations:	
7.1.	The safety significance of the manual implementation of the ORM should be reduced. Automation of the shutdown actions should be implemented when the ORM value falls below the safety limits.
7.2.	The efforts to reduce the contribution of the ORM to the reduction of the void effect are strongly supported.
7.3.	The homogeneous spatial distribution of ORM in the core should be guaranteed to prevent high positive local void effects.

1. Area / Number: Core design and core monitoring / 5

2. Issue title: Additional shutdown system

3. Issue clarification: In accordance with the basic principles of the shutdown system requirements as defined in international codes and guidelines and corresponding to international practice, two fully independent and diverse systems are required for safe reactor shutdown. However, the two shutdown systems installed in the RBMK reactor cannot be considered as fully independent and diverse shutdown systems; despite the fact that the rod designs and rod cooling are different, there are no displacers on the fast rods, and the speed of insertion is different.

4. Issue category: High

5. Justification of the category: Two existing shutdown systems can cope with any transient or accident to be considered in the safety analysis of the plant. However, the fast acting scram system (FASS) considered by the designers as independent scram system cannot maintain the reactor in the subcritical state in the event of CPS LOCA accident which is considered by the designer as a beyond-design-basis accident (BDBA) with extremely low probability.

6. Applicability: Generic

7. Recommendations:

7.1. The intention of the Russian side to develop and implement a fully independent and diverse additional shutdown system for all RBMK reactors should be realized with high priority. The additional shutdown system should be developed with due regard for ongoing developments in the area of core design.

1. Area / Number: Core design and core monitoring / 6

2. Issue title: Subcriticality margins in first generation RBMKs

3. Issue clarification: It has been reported that on occasions there have been difficulties in maintaining the first generation reactors subcritical after shutdown. In any circumstances, once triggered, the shutdown systems must assume both an immediate and a long-term core subcriticality with sufficient safety margin.

4. Issue category: High

5. Justification of the category: The requirement for subcriticality is 2%. In the Chernobyl 1 reactor, assuming the most reactive rod stuck, this requirement cannot be met. As a consequence, the shutdown system cannot keep the reactor in a subcritical state.

6. Applicability: Generation specific.

7. Recommendations:

7.1. Particular attention has to be paid to the long term subcriticality of the first generation reactors without reconstruction (like Chernobyl 1 and 2). The time and method of the implementation of additional control rods as done or scheduled for Leningrad 1 and Kursk 1 and 2 should be decided for all first generation units.

7.2. The loading patterns for fixed additional absorbers should be evaluated in order to reduce the efficiency of control rods located at the eastern core boundary of RBMKs.

7.3. Independent calculations to assess the margins of RBMK reactor using analysis assumptions used in OECD countries and to carry out comparative analysis of RBMKs and relevant plants in OECD countries, based on results of independent calculations, have to be performed.

4.3.2. Instrumentation and control

1. Area / Number: I&C / 1

2. Issue title: Diversity and segregation of I&C systems

3. Issue clarification: The segregation of the instrumentation and electronic systems for control and protection of the reactor and its supporting systems is consistent with some but not all of the requirements of current standards. There also appears to be insufficient diversity in some important elements when referenced against modern standards. These deficiencies with respect to diversity and segregation were recognized and significant modifications were made to the Smolensk 3 NPP before it went into service. The key areas modified were the ECCS and related emergency systems. However, problems remain with these systems at other reactors and there remains a generic issue associated with the reactor shutdown and control systems which share many common elements.

4. Issue category: High

5. Justification of the category: The use of common elements and common locations for I&C equipment and systems could lead to the complete loss of level 2 or mainly level 3 protection with respect to the defence-in-depth concept as expressed in INSAG 3. The high level of commonality gives concern with respect to the availability of systems in the event of an incident and the defence-in-depth they provide.

6. Applicability: Generic

7. Recommendations:

7.1. The diversity and segregation of the existing systems should be reviewed against: national standards, IAEA recommendations and international best practice. The review would enable support to be generated to show where defence-in-depth can be claimed and to identify areas of weakness where action is needed to eliminate the use of common elements.

7.2. The existing electronic equipment for the 24 fast action rods should either be segregated physically and electrically from the electronic equipment for the other rods or a separate train of electronics should be introduced with an additional shutdown mechanism.

7.3. The cables between the core and the electronics for the flux instruments and control rod drives should be separated out to form at least three segregated groups.

7.4. Measures should be taken to bring the ECCS and other support systems at all plants to the standard of segregation found for the ECCS electronics in the Smolensk 3 plant. This has been partially addressed as part of the upgrade programme e.g. at Leningrad Units 1 and 2 and shortly at Kursk 1.

7.5. The diversity of the means that generate the input to the safety and control systems should be examined. This exercise needs to be led by those looking at core monitoring and at support systems who will define the requirements. Once the requirements have been produced an I&C review must be completed to ensure that the sensors and electronics are diverse and segregated to give the required level of defence-in-depth.

1. Area / Number: I&C / 2

2. Issue title: Initiation of ECCS and other safety systems

3. Issue clarification: The ECCS is a key safety system for ensuring the cooling of the fuel elements when a LOCA occurs. It is initiated by a combination of signals. However, there is no assurance that the system responds in a timely manner and that the actuation equipment is robust to failures.

4. Issue category: High

5. Justification of the category: ECCS failure to start on demand can cause fuel and channel overheating and damage.

6. Applicability: Generic

7. Recommendations:

7.1. The standard of the ECCS initiation equipment in the RBMK NPPs should be brought at least to the level of that at Smolensk-3.

7.2. There are many systems important to safety which need to operate timely and reliably. A review should be completed for the control part of the systems important to safety as was done for the Smolensk-3 ECCS. The systems should be brought to the standard of the ECCS as is thought necessary to perform their safety function.

1. Area / Number:	I&C / 3
2. Issue title:	I&C system maintenance and periodic testing
3. Issue clarification:	Most I&C systems used in a safety role and particularly standby or poised systems must be subject to periodic testing to ensure that they are still operable. However, the periodic test programme is not fully in line with good international practice in respect to both preventive and remedial maintenance actions and the regime optimized to maximize the availability and minimize the level of interference with the system.
4. Issue category:	Medium
5. Justification of the category:	Failure to monitor and maintain systems particularly standby or poised safety systems can result in them being unavailable in the case of a demand. It is very important, as the failure of a system and particularly the undetected failure of that system is a threat to plant operation and in the case of protection systems plant safety as it can lead directly to the loss of a line of defence.
6. Applicability:	Generic

7. Recommendations:

7.1.	The maintenance procedures and test records should be identified and audited to ensure that their scope is satisfactory and they are being correctly implemented.
7.2.	It is recommended that the scope of the detector periodic testing programme be extended to include discriminator curve measurements to check on the performance of the detectors.
7.3.	Quality assurance arrangements for the completion and recording of maintenance activities should be brought into line with IAEA recommendations 50-C-QA and technical report 268 Manual on Maintenance of Systems and Components Important to Safety STI/DOC/10/268 .
7.4.	The scope of equipment failure data base should be reviewed and arrangements made to allow trend monitoring of equipment performance and the early identification of equipment ageing problems. There would be benefit if this data could be exchanged more effectively between the NPP and with the equipment designers and manufacturers.

1. Area / Number: I&C / 4

2. Issue title: Reliability of I&C systems

3. Issue clarification: The reliability and failure behavior of the monitoring, control and protection systems are essential elements of the plant safety case. In the RBMK NPPs, equipment failures and spurious generation of trip signals leading to actuation of the trip system have been reported. For these plants there is not sufficient information to judge that RBMK NPPs I&C systems have the high level reliability to meet safety and quality requirements.

4. Issue category: Medium

5. Justification of the category: The failure of the I&C control and protection systems could result in the loss of their functions. Operational experience shows that many incidents have been caused by I&C system failures.

6. Applicability: Generic

7. Recommendations:

7.1. Given that the ability to shut the reactor down must be maintained at all times, it is essential to review a reliability analysis, including fault tree and failure modes and effects analysis, to ensure the systems meet the safety and operational requirements.

Sample checks should be made of the existing reliability analysis using techniques such as failure modes and effects analysis to confirm both the numerical values for reliability and check that dangerous outcomes of a failure within a system have been identified. The power supply diode buffers, electromagnetic rod brake, high voltage breakers could be used as examples.

7.2. The results of the reliability analysis should be compared with failure data to ensure the fidelity of the results and performance of the equipment in service is as good as claimed in the safety case.

1. Area / Number: I&C / 5

2. Issue title: Replacement of NPP main computer

3. Issue clarification: The computers are essential to provide the operators with frequent and reliable calculated information e.g. DNBR and linear heat flux. In the absence of the computers, the NPPs cannot be operated at full power or for an extended time. The SCALA and TITAN computers systems are nearing the end of their working lives; they appear to be overloaded and they currently have a slow cycle time.

4. Issue category: Medium

5. Justification of the category: As the computers age reliability falls and spare parts become scarce posing a significant impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. The computer systems should be replaced at the earliest opportunity and in advance of further degradation in performance or the maintenance load reaching a high level so impacting on plant operation.

7.2. It is suggested that a modern distributed system be employed with dedicated units for functions important to safety.

7.3. It is recommended that the existing arrangements for signal distribution to safety displays and to the data collection system be fully isolated.

7.4. It is suggested that an incremental approach is taken to algorithm upgrade and the introduction of new functions, to facilitate parallel running of the systems to allow correct operation to be confirmed prior to the system entering service.

7.5. It is recommended that the rate of signal processing of the new computer is enhanced and a systematic approach to updating of parameters based on safety significance is adopted.

7.6. Additional screens and displays should be employed to improve the operator interface and make better use of the data.

1. Area / Number:	I&C / 6
2. Issue title:	I&C equipment upgrades
3. Issue clarification:	Many I&C systems on the RBMK plant are ageing and are in need of upgrade: the systems currently use static logic instead of dynamic logic, the number and capacity of PCs are not sufficient, the data processing computers should be upgraded, the electrical equipment is not qualified under harsh environmental conditions, the response time of the silver detectors is too long.
4. Issue category:	Medium
5. Justification of the category:	The ageing of I&C systems and the insufficient I&C support to new systems impact the performance of the safety systems and functions.
6. Applicability:	Generic
7. Recommendations:	
7.1.	The modernized versions of the I&C systems developed for the RBMK should be installed as soon as possible and the increased levels of automation should be carefully considered particularly where they can improve safety.
7.2.	Static logic is prone to unrevealed failures: these are often in the form of stuck at faults. It is proposed that dynamic logic be used more widely to assist in the provision of fault detection and detection of this and other types of fault.
7.3.	Electronic storage and retrieval systems should be considered for operating rules and key data, these should be colour coded to the equipment labels.
7.4.	The slow self powered detectors should be replaced by prompt detectors and the detector distribution optimized in order to minimize the number required for effective core monitoring and protection.
7.5.	The specification of the environmental qualifications for all new equipment should be checked against the environment in which they are to be placed e.g. for temperature, vibration, EMC, prior to installation. In case where the environment is harsh, the possible modes of failure should be checked for impact.

1. Area / Number: I&C / 7

2. Issue title: Operator support

3. Issue clarification: In the RBMK NPPs the operator has to use a considerable amount of information when making decisions. The way this information is currently made available to operators does not help him to make appropriate decisions. In addition, many safety functions rely on operator action. Therefore, it is necessary to upgrade human system interface in the control room to make easier and safer the operator's task.

4. Issue category: Medium

5. Justification of the category: Badly presented or erroneous information could cause the operator to make an incorrect decision which can initiate or worsen an accident.

6. Applicability: Generic

7. Recommendations:

7.1. All areas of the man machine interfaces should be improved in a coherent manner using a plan and concepts developed in consultation with plant staff.

7.2. The content of the parameter display in the emergency control room should be reviewed and if found deficient a new panel should be produced for the RBMK. The review should also consider the habitability of the area under conditions that oblige the operators to leave the main control room and bring the systems into use.

7.3. The station computers should be upgraded with respect to speed of operation and the form of the data displays.

4.3.3. Pressure boundary integrity

1. Area / Number: Pressure boundary integrity / 1

2. Issue title: Fulfillment of inspection requirements

3. Issue clarification: Plant operation continues even when the frequency and the volume of examinations required by the national regulations for the reactor pressure boundary are not performed or when the results are not satisfactory. The existing time schedules of implementation of modifications and additional analysis and requirements for record keeping are not followed. Criteria for limiting plant operation in these cases are not established. The integrity of the pressure boundary being a key safety function, the inspection requirements must be properly defined and fulfilled.

4. Issue category: High

5. Justification of the category: If the inspection programme prescribed to ensure the reactor pressure boundary integrity is not carried out, defense-in-depth is insufficient. This poses a major impact on safety due to each of the proper means to ensure safety for extended plant operation.

6. Applicability: Generic

7. Recommendations:

7.1. Examination requirements for the reactor pressure boundary integrity should be followed and effective verification mechanisms should be established.

7.2. To ensure compliance, limitation of the length of plant operation should be established based on the outstanding work.

7.3. Authorization for further plant operation should be obtained by the plant based on relevant criteria to be agreed upon with the regulatory body.

1. Area / Number: Pressure boundary integrity / 2

2. Issue title: In-service inspection

3. Issue clarification: The required high volume of in-service inspection is not fulfilled in practice. It was found that the required number of fuel channels was not inspected. The approach adopted at RBMK plants to repair critical defects found is different from the predictive approach adopted for ISI elsewhere. Pre-service inspection records and in-service inspection predictive records are not maintained. The equipment and procedures used are inadequate to give repeatable measurements of subcritical defect size.

4. Issue category: High

5. Justification of the category: In-service inspection is a measure which reduces the risk of failure of the pressure boundary through prediction of potential material failure on a critical defect basis. The implementation of the present practice of in-service inspection has a major impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Predictive ISI approach of following defects development and using fracture mechanics analysis should be implemented.

7.2. Comprehensive ISI documentation should be established and maintained at the plants.

7.3. ISI should be optimized with respect to inspection locations, frequency and techniques.

7.4. Modern equipment to provide for repeatable measurements of sub-critical defect sizes should be implemented. Consideration should be given to computerized ISI data acquisition, handling and storage systems.

1. Area / Number: Pressure boundary integrity / 3

2. Issue title: Break of critical components

3. Issue clarification: Some primary coolant circuit components and piping are outside of the accident localization system. The guillotine break in 800 mm diameter piping in the first generation of RBMK NPPs is considered as a major accident and could result in damage to civil structure.

Application of the leak before break (LBB) concept would reduce risk of coolant circuit failures. But the applicability of this concept is not fully demonstrated and LBB method and techniques are not in use.

4. Issue category: High

5. Justification of the category: The LBB method and techniques to critical components need to be fully implemented, because the guillotine break of these components, especially for the first generation of RBMK NPPs, where the accident localization system have limited capabilities, poses major impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. The activities to apply the leak before break concept for primary coolant circuit components inside and outside of the accident localization system should be completed. In particular attention should be given to stainless steel clad complex structures such as the steam drums, pressure headers etc.

7.2. Leak detection system, validated with respect to the plant configuration and the LBB analyses, should be developed and implemented. If humidity measurement is used for this purpose, attention should be given to adequate reduction of leaks from components.

7.3. Continue to develop leak detection systems which can be set for leakage levels.

1. Area / Number: Pressure boundary integrity / 4

2. Issue title: Fuel channel and tract integrity

3. Issue clarification: To date there have been three single channel ruptures due to water flow blockage or power flow imbalance. The rupture results in releases of radioactivity into the reactor cavity and may result in a release into the environment if the confinement safety system does not function properly according to the design.

The design of the fuel channel and tract has a significant number of circumferential welds, some between dissimilar metals. Fuel channels are not sufficiently inspected to ensure the low risk of tube rupture.

Note: The fuel channel pressure boundary is divided by the designer into the proper fuel channel and the tract. The tract is defined as the inferior and superior parts of this pressure boundary from the bottom bellows to a specified weld and from another specified weld to the channel plug.

4. Issue category: High

5. Justification of the category: The multiple rupture of fuel channels (exceeding the venting capacity of the reactor cavity overpressure protection system) poses a major impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Both non-destructive examination and post-irradiation destructive testing of the fuel channels should be continued and strengthened. Automatic inspection equipment would enable repeatable testing to be done more quickly and reliably.

7.2. The reduction of the number of in line components, which on failure can result in a water flow blockage, the use of more reliable components, and backfitting of bypasses should be analyzed and implemented if feasible.

1. Area / Number: Pressure boundary integrity / 5

2. Issue title: Special channel integrity

3. Issue clarification: Special channels are used for housing various shutdown, control and measurement devices inside the core. They are cooled by a separate cooling circuit which operates at low temperature and pressure. Special channels are not foreseen to be replaced during the lifetime of the reactor. Material degradation of special channels (i.e. hydrogen uptake) could lead to their failure from non-related events, such as pressure from a single fuel channel failure, and thus to drainage of the special channels' cooling circuit.

4. Issue category: Low

5. Justification of the category: Drainage of the special channels' cooling circuit due to channel failures during operation would add reactivity at a slow rate.

6. Applicability: Generic

7. Recommendations:

7.1. A system which monitors leakages in the special channel coolant system should be introduced. A high leakage rate or low water volume in this coolant system should result in an automatic shutdown of the reactor.

7.2. A special channel in the high flux area should be periodically removed and the material properties including hydrogen intake should be measured.

7.3. The in-service inspection programme for special channels should be optimized but on a limited basis.

1. Area / Number:	Pressure boundary integrity / 6
2. Issue title:	Fuel handling
3. Issue clarification:	Fuelling of the reactor is done on power using a single refuelling machine which is located on top of the reactor. Since several channels are refuelled every day the fuelling machine forms part of the primary coolant circuit for a significant proportion of the time. The fuelling machine is aligned to the channel by an operator using manual controls. This manual alignment may add additional loads to the channel which may result in additional stresses and may contribute to leakages through closure plugs and channel welds.
4. Issue category:	Medium
5. Justification of the category:	Disruption of the refuelling machine connection will result in a small LOCA with direct connection to the atmosphere having a significant impact on plant safety.
6. Applicability:	Generic

7. Recommendations:

7.1.	Automation of the final refuelling machine positioning and of the remaining functions of fuel replacement up to closing of the plug should be implemented.
7.2.	Improved sealing of fuel channel plug should be used to eliminate leakages of steam to the reactor hall.
7.3.	Any analysis of the primary coolant circuit system should also consider the configuration with the fuelling machine attached to the circuit.

1. Area / Number: Pressure boundary integrity / 7

2. Issue title: Seismic and ageing assessment

3. Issue clarification: A number of areas were identified in, or as pertinent to the pressure boundary, where additional analysis or assessments are needed to either confirm the present design or to propose modifications which would improve safety. Areas which require this type of work are, in particular, the areas of seismic analysis and assessment of ageing of materials.

4. Issue category: Medium

5. Justification of the category: While analysis and assessment does not in itself lead to increased safety, it may either confirm the design or lead to modifications which may improve safety significantly.

6. Applicability: Generic

7. Recommendations:

7.1. Additional seismic analysis on the pressure boundary for piping, components, and components' supports is recommended.

7.2. Additional work on the ageing of metals and concrete to determine changes in properties and their possible effects on structural behaviour is recommended.

4.3.4. Accident analysis

1. Area / Number: Accident analysis / 1

2. Issue title: Scope and methodology for accident analysis

3. Issue clarification: The scope of accident analysis available in the TOB was determined by the national regulations effective at the time 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 the assumptions used in the analysis. The computer codes used at the time of RBMK design were of limited modelling capability. The lack of an experimental data base on pipe rupture of primary head transport system limited the possibility of integral code validation. Presently, more modern Russian codes and some western codes are being used, but these codes have not been sufficiently validated for modelling RBMKs.

4. Issue category: Medium

5. Justification of the category: The defense-in-depth concept requires adequate design and accident analysis performed with computer codes with sufficient modelling capability and validated with appropriate experimental database. The absence of such tools places a significant impact on plant safety because the appropriate function of all engineered safety systems can not be ensured without a comprehensive accident analysis based on modern qualified (validated, documented and traceable) codes.

6. Applicability: Generic

7. Recommendations:

7.1. Develop RBMK models using state of the art codes. Validate these models and document the analysis in a traceable way.

7.2. Perform additional accident analysis to enlarge the scope of information available, including additional cases, sensitivity studies and partial power initial conditions.

7.3. In addition to conservative analyses, perform Best Estimate analyses to develop Emergency Operating and Symptom Oriented Procedures and to justify proposed additional trip signals.

1. Area / Number: Accident analysis / 2

2. Issue title: LOCA analysis

3. Issue clarification: The loss of coolant accident is a major accident which needs to be carefully analyzed in order to define and implement measures to mitigate it. The completeness of DBA analysis, the adequacy of the codes/ database/ validation/ documentation for the LOCA analysis, understanding of sensitivities to parameter variations and uncertainties, and completeness of results does not follow the required principles of quality assurance.

4. Issue category: High[1]

5. Justification of the category: An additional and comprehensive spectrum of LOCAs needs to be analysed to provide the design basis for accident mitigation features such as ECCS (both short term and long term cooling), pressure mitigation (to assure survival of the confinement boundaries) and cavity overpressure protection. This activity should be considered in connection with other activities (e.g. Safety and support system sheet 1) in the sense that additional LOCA analyses are needed to assess the real capability and performance of existing or modified safety systems, particularly where possible shortcomings in or lack of existing analyses have been identified by the reviews.

6. Applicability: Generic

7. Recommendations:

7.1. Establish the rationale for the DBA definition, including conservative assumptions and RBMK specific acceptance criteria.

7.2. Perform additional LOCA analysis with state-of-the-art validated codes to complete the spectrum of cases analysed, and perform sensitivity studies to assess uncertainties.

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[1]RBMK specialists did not agree with category High for the LOCA analysis issue since the analysis had already been performed for all RBMK NPPs before their commissioning. Now this analysis is only to be reassessed and extended.

1. Area / Number: Accident analysis / 3

2. Issue title: Cavity overpressure protection

3. Issue clarification: In case of a pressure tube failure, the pressure strongly increases inside the reactor cavity. This can lead to severe accident and to radiological release outside the reactor building. The capacity of the cavity overpressure protection system (in number of simultaneous pressure tube failures), and the adequacy of additional systems to mitigate overpressure and radiological consequences have not been adequately analyzed. Plant safety improvements are being proposed to increase the capacity of the cavity overpressurization protection system. However, the accident sequences which could lead to multiple pressure tube ruptures have not been defined and the adequacy of the additional system(s) to mitigate resulting overpressure and radiological consequences have not been sufficiently analyzed.

4. Issue category: High

5. Justification of the category: The consequence of overpressurizing the reactor cavity in the RBMK system involves lifting of the reactor upper shield structure and associated multiple simultaneous pressure tube failures. This leads to subsequent significant impact on plant safety with radiological consequence.

6. Applicability: Generic

7. Recommendations:

7.1. The analysis of pressure tube failures in accident sequences, both DBA and BDBA, and the design basis of systems to mitigate pressure effects from such failures requires special attention for the RBMK system.

The increase of cavity overpressure protection system capability is supported, but adequate support analysis should be performed with validated computer codes.

7.2. Evaluation of the proposed solution should consider the tradeoffs between the protection of the reactor cavity and the radiological consequences of the releases.

1. Area / Number: Accident analysis / 4

2. Issue title: Steam line break analysis

3. Issue clarification: The steam line break is an important accident scenario which needs to be carefully analyzed. Steam line break analyses were performed by Russian designers using two separate computer codes for the thermohydraulics and the neutronics modeling, with manual transfer of data between the two codes. This approach may lead to errors in the calculation of the feedback mechanism that connects coolant temperature to neutron power.

4. Issue category: Medium

5. Justification of the category: The use of the less advanced codes in the calculations of the feedback mechanism that connects coolant temperature to neutron power may result in incorrect simulation of steam line break leading to inadequate functioning of relevant safety systems based on such simulation.

6. Applicability: Generic

7. Recommendations:

7.1. The steam line break accident should be reanalysed using state-of-the-art codes with coupled neutronic and thermohydraulic models.

7.2. Additional cases of steam line break should be analysed including partial power initial conditions and different break locations (inside and outside drum separator compartments).

7.3. Best estimate analysis of steam line break should be performed to evaluate existing and proposed ECCS initiation signals.

1. Area / Number: Accident analysis / 5

2. Issue title: Pipe whip analysis

3. Issue clarification: In case of pipe break, there are possibilities of pipe whip which can lead to damages to structures and equipment. In the Russian analysis, the possibilities of pipe whip were not systematically taken into account when estimating accident results or investigating possible accident scenarios. Loads on mechanical restrains were not verified.

4. Issue category: Medium

5. Justification of the category: Pipe whip is a possible source of common cause failure which can disable the plant safety functions.

6. Applicability: Generic

7. Recommendations:

7.1. Pipe whip analysis should be systematically conducted, with special emphasis on feedwater and steam lines: Loading on mechanical restrains following breaks should be verified. Critical situations regarding spatial separation should be identified. For cases where damage by pipe whip cannot be excluded, sequences with consequential breaks should be analysed.

1. Area / Number: Accident analysis / 6

2. Issue title: Loss of power

3. Issue clarification: Loss of power scenarios need to be analysed because they pose a major threat to plant safety. Safety analysis normally consider the loss of preferred power, while total blackout (i.e. on site and off site loss of electric power supply) are analysed as a beyond design basis accident. In RBMK safety analysis blackout scenarios were not analysed. Even for the loss of power scenarios analysed, the calculations were not conducted until the time in which the plant reaches stable conditions.

4. Issue category: Medium

5. Justification of the category: Analysis of loss of power scenarios made without taking into account the particular features of the various unit as well as calculations limited to the time in which the plant has not yet reached stable conditions does not allow development of accident management procedures which fully reflect possible situations.

6. Applicability: Generic

7. Recommendations:

7.1. It is necessary to perform analysis of loss of power scenarios taking into account the particular features of the various units.

Loss of power analysis should be repeated on a plant specific basis and the analysis should be carried out until the plant reaches stable conditions (cooldown and depressurization).

7.2. Additional analysis considering total loss of power (blackout) should be carried out to estimate grace times (time available for operator actions) and to develop accident management strategies.

1. Area / Number: Accident analysis / 7

2. Issue title: Radiological consequence analysis

3. Issue clarification: In case of pressure tube rupture, additional releases due to possible mechanical fuel damages can occur and should be analyzed. In the TOBs, radiological consequence analysis is available, but previous review and studies have shown some limitations in the scope of the methodology and assumptions as compared to good international practice although this topic has not been reviewed in great detail.

4. Issue category: Medium

5. Justification of the category: The design and evaluation of modifications to the confinement systems requires the analysis of radiological consequences to evaluate the potential risk of various accident scenarios. These calculations are also necessary for emergency planning.

6. Applicability: Generic

7. Recommendations:

7.1. An independent evaluation of the methodology used for radiological consequence analysis should be performed. Comparison with state-of-the-art methods and codes used in the Western countries should be carried out.

1. Area / Number: Accident analysis / 8

2. Issue title: Performance and utilization of PSA

3. Issue clarification: Probabilistic Safety Assessment (PSA) is an important tool to identify safety deficiencies and to prioritize plant improvements. PSAs have been initiated only in a few RBMK plants in order to evaluate proposed plant modifications. These PSAs have not yet been independently reviewed as required by international practices.

4. Issue category: Medium

5. Justification of the category: Programmes for safety improvement which do not include insights from PSAs may focus on issues which are not most relevant from the risk reduction point of view. This has a significant impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Plant specific PSA should be performed for each RBMK plant and the results should be used in identifying plant weaknesses, and assessing and prioritizing plant modifications. The result should also be used as input for accident management.

7.2. A list of RBMK specific initiating events should be developed including operational experience analysis.

7.3. RBMK specific reliability data should be systematically collected and a related data base, including information from all plants, should be developed.

7.4. Before utilizing PSA results, it is recommended that PSAs be reviewed by independent experts.

1. Area / Number: Accident analysis / 9

2. Issue title: Anticipated transient without scram (ATWS)

3. Issue clarification: ATWS scenarios are generally included in safety analysis. ATWS was not considered in the original design of RBMKs. Due to the complex nature of the scram system (and considering the proposed installation of an additional scram system) the identification of ATWS sequences and estimation of their probability for RBMKs is not straightforward.

4. Issue category: High

5. Justification of the category: The ability of the plant to withstand with anticipated transients with failure of the scram system has a major impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. In light of their possible severe consequences it is very important to carry out comprehensive ATWS studies. This will assist in the design of the additional shutdown system.

7.2. Event trees to identify cases of ATWS sequences that are relevant to RBMKs should be developed. The probability of these sequences and their consequences should be evaluated.

7.3. A comprehensive ATWS analysis should be carried out.

1. Area / Number: Accident analysis / 10

2. Issue title: External hazards

3. Issue clarification: The analysis of external hazards is very important, especially in the nuclear plants where a confinement does not exist. For the RBMK NPPs, the analyses of external events seem to be incomplete. Some seismic analysis has been performed but is still not considered sufficient.

4. Issue category: Medium

5. Justification of the category: External hazards, if they are not properly considered and taken into account, pose a significant impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Comprehensive external event hazard analysis should be carried out. Events to be considered should include airplane crash, flooding and explosion.

7.2. Comprehensive seismic analysis should be performed, including the definition of site parameters, evaluation of seismic stability of structures, components and supports.

4.3.5. Safety and support systems

1. Area / Number:	Safety and support systems / 1
2. Issue title:	ECCS - capability and performance
3. Issue clarification:	The design basis of ECCS for first generation plants was limited to a small break size and a few break locations. Therefore the ECCS is not capable to cope with a full spectrum of LOCAs. For later generation plants, the ECCS capability is adequate, but the performance of ECCS operation depends on the proper alignment of the main coolant system.
4. Issue category:	High
5. Justification of the category:	The ECCS is the front line of accident mitigation systems, and in the case of its improper performance following any type of anticipated LOCA poses a major impact on plant safety.
6. Applicability:	Generation specific
7. Recommendations:	
7.1.	Upgrade of ECCS capability for first generation plants
7.2.	Improvement of primary coolant circuit component design and reliability is required to warrant the ECCS function.

1. Area / Number:	Safety and support systems / 2
2. Issue title:	Long-term cooling and water make up
3. Issue clarification:	When the reactor is shutdown, the long-term cooling of the fuel assemblies must be ensured. Therefore, it is necessary to identify all scenarios which could impair the capability of this safety function. Shortcomings in design and reliability have been identified in some plants in the provision for continued removal of decay heat for an indefinitely long length of time.
4. Issue category:	Medium
5. Justification of the category:	Shortcomings in design related to long-term cooling and water makeup and insufficient reliability of the system affect the plant capability for decay heat removal for an indefinite length of time.
6. Applicability:	Generation specific
7. Recommendations:	
7.1.	Improve water make up system reliability to make sure long term cooling can be warranted.
7.2.	Improve the residual heat removal system reliability to enable cold shutdown conditions.

1. Area / Number: Safety and support systems / 3

2. Issue title: ECCS reliability improvement

3. Issue clarification: In case of break on the main circulation circuit, the cooling of the core must be continuously ensured to avoid damage of the fuel. Therefore, a highly reliable ECCS is required. In first and second generation units, neither the ECCS nor the electrical power supply systems have reached a sufficient level of redundancy. The defense against common cause failure is not well addressed, either physically operating redundant equipment or making the design diverse.

4. Issue category: High

5. Justification of the category: The reliability of the ECCS has a major impact on plant safety.

6. Applicability: Generation specific

7. Recommendations:

7.1. Review the ECCS layout with respect to the physical separation requirements to improve system reliability.

7.2. Install, where appropriate, an additional diverse emergency feedwater system.

7.3. Improve the ECCS equipment redundancies for the first generation plants.

1. Area / Number: Safety and support systems / 4

2. Issue title: Reactor trip and ECCS - actuation signals

3. Issue clarification: In case of LOCA, the ECCS has to be triggered as soon as possible to avoid damage of the fuel assemblies. For the RBMK NPPs, it is questionable whether the reactor trip and ECCS will actuate for certain accident conditions.

4. Issue category: High

5. Justification of the category: The absence of signals to trip the reactor and actuate ECCS in certain LOCA accident conditions have a major impact on the early reactor shutdown and ECCS actuation.

6. Applicability: Generic

7. Recommendations:

7.1. In case of total loss of feedwater (main and emergency) the ECCS automatic start-up on appropriate initiating signals should be investigated. (e.g. low level in drum separators)

7.2. Reactor trip and if necessary, ECCS actuation in case of low flow in many channels should be implemented.

7.3. The use of diverse actuation signals for ECCS is recommended, in addition to the existing ones.

1. Area / Number: Safety and support systems / 5

2. Issue title: Interfacing system LOCA

3. Issue clarification: A break located outside the confinement on a pipe connected to the primary coolant system might result in significant radioactive releases outside the confinement. The reliability of the isolation equipment on such pipes has not been investigated for RBMKs.

4. Issue category: Medium

5. Justification of the category: The potential radioactive release outside the confinement due to LOCA of the pipe outside the confinement connected to the primary coolant system has a significant impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. A study on interfacing LOCAs could be undertaken in the framework of a PSA.

1. Area / Number:	Safety and support systems / 6
2. Issue title:	Adequacy of confinement function
3. Issue clarification:	RBMK does not have full containment. In addition, questions have been raised with respect to the adequacy of existing systems to provide containment function (mitigate pressure increase under accident conditions to avoid damage to structures and equipment; mitigate radiological releases) for the three RBMK generations.
4. Issue category:	High
5. Justification of the category:	The containment function is to prevent releases of radioactive materials to the site and environment in the event of an accident; its effectiveness has a profound influence on the consequence of an accident insofar as the effect on public and land, if such an accident were to occur.
6. Applicability:	Generic
7. Recommendations:	
7.1.	In accordance with international practice, the installation of steam lines isolation valves in RBMK reactors should be considered.
7.2.	The accident localization system (ALS) leaktightness should be improved and substitution of suppression pool discharge valves should be studied.
7.3.	The old RBMK generation plants should be upgraded in terms of confinement capabilities.
7.4	A possible upgrade of the upper rooms (reactor hall and steam separator rooms) with respect to leaktightness and confinement function should be studied, including stress and seismic analyses.
7.5.	The main steam safety relief valves, which continuously leak into the ALS, should be replaced by better quality equipment.

1. Area / Number: Safety and support systems / 7

2. Issue title: Reliability of ultimate heat sink

3. Issue clarification: Under all conditions the energy release by the core must be removed. Therefore, it is necessary to ensure that, upon loss of natural feed water flow, a back up system is available. Shortcomings have been identified in the service water system that could impair its operation.

4. Issue category: Medium

5. Justification of the category: A complete loss of service water, due for instance to a common mode failure, would prevent operation of all safety and non-safety related equipment.

6. Applicability: Generic

7. Recommendations:

7.1. Investigate and, where necessary, improve the service water system to avoid single failures (active and passive) that produce the complete loss of the system.

7.2. The automatic isolation of the non-essential components served by the SW System should be considered.

7.3. Review if the SW system operation is adequately monitored to alert the operator of failures that require timely actions in order to prevent the complete loss of the system.

1. Area / Number: Safety and support systems / 8

2. Issue title: Reliability of electrical system

3. Issue clarification: A nuclear plant needs permanent electricity supply to ensure the proper performance of the safety functions. Therefore, a reliable electrical supply should be available at all times. For the RBMK NPPs, the reliability of electrical system does not always meet current standards.

4. Issue category: Medium

5. Justification of the category: Electrical system failures may be a cause of common mode failure and can have a significant impact on plant safety and support systems.

6. Applicability: Generic

7. Recommendations:

7.1. Based on existing recommendations and operating experience, update electrical power systems to current standards.

7.2. The increase of battery depletion time up to one hour should be considered.

1. Area / Number:	Safety and support systems / 9
2. Issue title:	Electrical equipment qualification
3. Issue clarification:	In the NPPs, permanent electricity supply is of utmost importance, particularly in the case of accident. Therefore, the electrical equipment should be qualified under harsh environmental conditions, e.g. high temperature, pressure, radiation, fire, etc. For the RBMK NPPs, there is no assurance that electrical equipment is qualified to the environmental conditions of accidents requiring operation of this equipment.
4. Issue category:	Medium
5. Justification of the category:	The absence of environmental qualification of electrical equipment could lead to loss of safety functions and consequently pose a significant impact on plant safety.
6. Applicability:	Generic
7. Recommendations:	
7.1.	Further investigations should be performed regarding seismic and environmental qualification of electrical equipment.

1. Area / Number:	Safety and support systems / 10
2. Issue title:	Diesel generator reliability
3. Issue clarification:	In case of an external grid loss, the diesel generator (DG) system is the ultimate power supply for the NPP. Therefore, it is essential to ensure high reliability level of the DG system which has to feed the essential safety systems.
4. Issue category:	Medium
5. Justification of the category:	The poor reliability of the emergency DG power supply sources required to feed essential systems components in case of unavailability of the external grid poses a significant impact on plant safety.
6. Applicability:	Generic
7. Recommendations:	
7.1.	The increase of the undervoltage set point for automatic DG start-up should be considered.
7.2.	Improvement of the diesel generator relay controls should considered.
7.3.	The improvement of the emergency DG cooling system should be considered.

4.3.6. Fire protection

1. Area / Number: Fire protection / 1

2. Issue title: Passive fire protection

3. Issue clarification: In a NPP, fire is the most serious internal hazard because it can lead to common cause failure of safety systems. In the RBMK NPPs, there are unnecessary fire loads in several places. There are deficiencies in fire compartmentation and separation. These deficiencies include walls with unprotected openings, substandard fire doors and deficient fire basic penetration seals.

4. Issue category: High

5. Justification of the category: The deficiencies in passive fire protection can lead to fire propagation and have a major impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Statistical data on RBMK reactor fires should be used to identify the highest risks.

7.2. All avoidable fire load should be removed, including

- floor plastic
- flammable cables
- general housekeeping.

7.3. Venting of released gaseous flammables should be confirmed.

7.4. Fire walls should be built and fire doors/locks be installed to improve the fire separation between

- redundant safety trains
- control cabinets in control room
- control room and the adjacent electric rooms
- turbine hall and intermediate building
- the main transformers and the turbine hall
- auxiliary diesel generators and oil tanks

7.5. Fire insulation should be installed on

- important power cables
- load bearing steel structures
- pipe and cable penetrations

7.6. Fire resistant ventilation should be installed to the control room and electrical rooms.

1. Area / Number:	Fire protection / 2
2. Issue title:	Automatic fire detection
3. Issue clarification:	In the NPPs, in order to prevent the propagation of fire, an early automatic detection system is needed. Recent reviews and studies performed at the RBMK NPPs have identified deficiencies in the fire detection system, both in terms of its reliability and scope of coverage.
4. Issue category:	Medium
5. Justification of the category:	Deficiencies in the automatic fire detection is an essential element of the overall fire protection programme in order to assure prompt notification of the NPP operating and fire brigade personnel should a fire occur. Therefore, it has a significant impact on plant safety. Reliability of fire detection systems on the RBMK NPP is of a special safety significance because of the incompleteness of the passive fire protection.
6. Applicability:	Generic
7. Recommendations:	
7.1.	Upgrade the existing fire detection system by advanced, reliable equipment and increase the coverage.

1. Area / Number: Fire protection / 3

2. Issue title: Manual fire suppression capability

3. Issue clarification: Manual fire suppression capability is traditionally strong in Soviet designed NPPs. This applies to number and training of fire brigade personnel. Deficiencies, however, exist in the personal protective equipment, communications equipment and fire fighting equipment.

4. Issue category: Medium

5. Justification of the category: Should fire occur in a NPP, prompt extinguishment capability is an essential element of the plant fire protection programme to assure that fire does not spread from the area of origin, and to prevent unacceptable damage to plant safe shutdown capability. Manual fire fighting capability is one of the parts of the plant fire suppression capability. Deficiencies in this pose a significant impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Measures for the fire fighters to carry out their duties with maximal effectiveness, have to be improved. Protective clothing and up-to-date fire fighting equipment such as nozzles, fire hoses and portable equipment are needed. Provide the fire fighters with new high quality personal protective equipment.

7.2. Provide fire hose and water cannon installations with new replacement hoses, adjustable fog/ straight stream nozzles, and the dry stand pipes to the roof of the turbine hall with permanent hoses.

7.3. Provide the plant with portable extinguishers corresponding to western density and quality requirements.

7.4. Add, repair and maintain normal and emergency lighting in the plant.

1. Area / Number: Fire protection / 4

2. Issue title: Automatic fire suppression capability

3. Issue clarification: In addition to the manned fire suppression system, an automatic fire suppression capability is installed to fight fire at early stages. In the RBMK NPPs, the existing automatic fire suppression system has quantitative and qualitative shortcomings with respect to the quality of the systems.

4. Issue category: Medium

5. Justification of the category: Should fire occur in a NPP, prompt extinguishment capability is an essential element of the plant fire protection to assure that fire does not spread from the area of origin, and to prevent unacceptable damage to plant safe shutdown capability. Automatic fire suppression capability is the first line of defence in this respect.

6. Applicability: Generic

7. Recommendations:

7.1. Fixed automatic water extinguishing systems should be installed to the main turbine generator bearings, the underside of the turbine hall main deck to protect the steel structures, and to the underside of the turbine hall roof to protect the steel roof structures.

7.2. The coverage and quality of the existing fixed automatic water extinguishing system in cable tunnels and rooms should be evaluated.

1. Area / Number: Fire protection / 5

2. Issue title: Fire water supply

3. Issue clarification: A reliable water supply with adequate volume and pressure is essential to guarantee proper availability and operation of both manual and automatic fire suppression capability. Although water supply for fire protection has been judged adequate, the water pumps connected to the supply are manually operated and must rely on operator's action during fire emergency.

4. Issue category: Low

5. Justification of the category: The risk of the human errors associated with fine water supply are considered to have small impact on plant safety.

6. Applicability: Generic

7. Recommendations:

7.1. Proper automatic sequential starting of fire protection water pumps based on pressure drop in the system will assure that design water volume and pressure will always be available for both automatic and manual fire suppression activities. Manual stop for fire pumps will assure that they do not shut off prematurely (as with a timer stop) even though fire flow demand may still be high.

Revise fire water pump start to take place automatically on pressure drop.

4.3.7. Operational safety

1. Area / Number: Operational safety / 1

2. Issue title: Organization and staffing

3. Issue clarification: Many aspects of NPP management do not correspond to good international practice including: clear responsibilities, lines of communication, goals, organizational structure to support nuclear safety (nuclear safety committee), control room staff importance, adequate staff, division of responsibilities, performance indicators.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Review organizational structure of NPP management taking into account:

- management organization and responsibilities providing safe operating conditions in normal and abnormal situations.
- organizational structure provides for maintenance of equipment, the analysis of root causes and development and implementation of active corrective measures.
- the ability of the existing organizational structures to fulfill its function of safely managing nuclear power plants.
- best examples of the international NPP management.

7.2. Establish a nuclear safety committee at the nuclear power plant.

7.3. Carry out periodic independent review of the management of the NPP (peer review).

7.4. Responsibility and accountability should be placed at the lowest levels.

1. Area / Number: Operational safety / 2

2. Issue title: Quality assurance

3. Issue clarification: The quality assurance should encompass systematic and preset actions needed to ensure that a product or a service will meet the quality requirements. The existing QA programmes are not adequately developed in accordance with good international practice.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. To complete development of the overall QA programme for operation of the NPP and include the following:

- operational management in normal and accident condition.
- maintenance and repair. Any work should not be done unless spare parts are available and there are adequate trained personnel and adequate procedures.
- include root cause analysis as part of the QA programme.

7.2. The QA programme should be developed taking into account experience available through international co-operation. After the implementation stage, an independent assessment of programme effectiveness should be considered.

7.3. All supplied equipment for reconstruction and modernization projects must be manufactured and repaired in strict compliance with the requirement of the QA programme. Suppliers of equipment and companies providing engineering services for NPPs should comply with the NPPS requirements for QA.

1. Area / Number: Operational safety / 3

2. Issue title: Safety culture

3. Issue clarification: Existing safety culture as compared to INSAG-4 is not fully effective for all aspects of plant operation and not effectively communicated by management to all station personnel. This includes the role of procedure usage in accomplishing safety goals.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Management of NPPs shall establish a relationship between personnel which is based on trust and openness in order to implement the principles of safety culture. Management should encourage the willingness of personnel to improve their qualification, self-evaluation and to contribute to the improvement of safety. The self-critical attitude, an attitude that refuses to accept second best, should be encouraged and developed in all levels in the organization.

7.2. The NPP management should create good conditions for personnel for qualification improvement, self-evaluation and satisfaction of their career aspiration. Administration should provide personnel with adequate tools for training, and make available to personnel the best available knowledge in the area. Management should respond to legitimate requests from plant personnel to improve conditions.

7.3. Management must take into account that key role of control room personnel with requests safe operation of the plant.

7.4. Management should publish their policy on safety culture to all staff, develop a specific safety culture document and educate staff on safety culture.

1. Area / Number: Operational safety / 4

2. Issue title: Management of documents

3. Issue clarification: A programme and facilities exist to ensure that the required, correctly structured documents are stored, amended, issued and distributed effectively. However, this programme is not in full compliance with good international practice.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. The operational documentation should be kept in good conditions and be available for use in the control room and for use by field personnel.

7.2. Documentation should be regularly checked . All necessary changes should be introduced.

7.3. State of the art computer techniques should be used for documentation storage and modification.

7.4. Copies of basic operational and design documentation must be securely stored in a separate location taking into account fire protection, flood, etc.

1. Area / Number: Operational safety / 5

2. Issue title: Material condition

3. Issue clarification: Equipment is not always labelled and maintained in a state of readiness so that it can reliably operate. Housekeeping conditions do not fully correspond to good international practice to reduce industrial safety hazards to personnel, to better maintain equipment, to diminish fire hazards and to establish improved working conditions.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Special attention should be given to housekeeping. Equipment must be labelled correctly and lighting must be adequate. Equipment must be clean and there must be good access to operate and maintain equipment. Leaks from equipment must be minimized.

1. Area / Number: Operational safety / 6

2. Issue title: Training programmes and materials

3. Issue clarification: The prescribed scope of training covers only operating staff. There is no formal training in nuclear operational concepts and particularly in safety matters, which takes into account the design basis and the analysis of events, for all power plant technical and managerial staff. The amount and frequency of simulator training do not correspond to good international practice.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Outside the current formal programme, seminars should be arranged to actively promote a strong nuclear safety culture in staff at all levels.

The concept of continuous training should be introduced. Adequate and updated training materials are necessary for effective and consistent training of plant personnel and instructors. This includes general education and self training facilities.

Review training programme, facilities and material to bring their level to international practice:

- ensure training of all personnel at NPP which are connected with safety;
- creation of full scale simulator for operation staff training is a very important task to ensure operational safety of NPP;
- analyse the system of personnel licensing and in case of need introduce necessary corrections;
- ensure safety culture is stressed.

1. Area / Number: Operational safety / 7

2. Issue title: Operating procedures for normal operation

3. Issue clarification: Instructions on the preparation, format, content, review, updating, and approval of station procedures are not clear and not provided for all groups (maintenance, operations, etc.). Procedures are not fully consistent with technical specifications (TOB) and do not fully reflect operating experience. Fully effective alarm response procedures are not available.

Operating personnel are not provided with detailed instructions for routinely checking the status of the emergency cooling system.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. The enhancement of normal operational procedures may be achieved by QA programmes in operation. Guideline development for writing operating procedures within the framework of this programme will ensure the necessary quality and completeness of procedures.

1. Area / Number: Operational safety / 8

2. Issue title: Emergency operating procedures

3. Issue clarification: Human factor considerations and operating experience are not fully included in the emergency operating procedures. It is recognized that full development of symptom-based procedures is a long-term effort.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Continue work with the help of international co-operation in developing symptom based emergency procedures and train personnel to use them. A step-by-step format should be used. Procedures should include accident management instructions.

1. Area / Number:	Operational safety / 9
2. Issue title:	Experience feedback and event investigation
3. Issue clarification:	The methods and procedures used for event investigation are not in full agreement with good international practices. Feedback of experience to plant personnel based on the relevant events both in the plant and at other plants are not routinely carried out.
4. Issue category:	
5. Justification of the category:	
6. Applicability:	Generic
7. Recommendations:	
7.1.	The ASSET methods analyzing the root cause of events should be used. Action plans must be developed to resolve the issues identified by this process. It is necessary that the NPP personnel and utility personnel and regulatory bodies participating in the analysis of the root cause events undergo special training on this methodology. It is necessary to introduce a special programme to carry out corrective measures and check the completion of the same measures at other power plants.
7.2.	This methodology must also be used to recognize events which may have occurred if the scenario had developed differently.

1. Area / Number: Operational safety / 10

2. Issue title: Maintenance programme

3. Issue clarification: The maintenance programmes to improve maintenance standards, equipment history, procurement and storage of spare parts do not correspond to good international practices. Assessment of its effectiveness of corrective, preventive, and predictive maintenance needs improvement. Computers, independent verification principles and post-maintenance testing are not fully in use.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. Maintenance programmes should be updated to include all the elements of fully effective preventive, predictive and breakdown maintenance programmes. Special attention should be paid to feedback of experience.

1. Area / Number:	Operational safety / 11
2. Issue title:	Modification control
3. Issue clarification:	Modification control does not correspond to good international practices. Responsibilities and specification of the requirements for training of personnel who have responsibilities for operating, testing and maintaining equipment affected by the modification are not clearly defined.
4. Issue category:	
5. Justification of the category:	
6. Applicability:	Generic

7. Recommendations:

7.1. A procedure for temporary and permanent modification should be established.

7.2. Effective control of the implementation of modifications should be established. This should help to ensure that the modifications are consistent with overall plant safety.

1. Area / Number: Operational safety / 12

2. Issue title: Surveillance test programme

3. Issue clarification: Surveillance programmes which provide guidance on how surveillance tests are tracked, trended and scheduled do not correspond to good international practices.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. During preparations for surveillance tests the period of testing must be established based on reliability data of the component taking into account the operating experience. Special attention is necessary concerning the documentation of the results of surveillance testing and its analysis.

7.2. Surveillance tests should be carried out so that the intended function of the system is confirmed, especially for the elements important for safety such as relief valves, ECCS valves, shutdown systems, etc.

7.3. Personnel should be provided with detailed instructions and acceptance criteria for tests that verify important safety parameters and functions of systems and trains. These procedures should encompass all data necessary to determine the performance of plant equipment.

1. Area / Number: Operational safety / 13

2. Issue title: Radiation protection programmes

3. Issue clarification: A programme to reduce radiation doses to as low as reasonably achievable (ALARA) is not in place at all plants. Remote control metal inspection equipment is not yet extensively used to bring significant reduction in the collective dose.

4. Issue category:

5. Justification of the category:

6. Applicability: Generic

7. Recommendations:

7.1. ALARA programmes should be in place at all plants.

7.2. Remote controlled in-service inspection equipment should be used in high dose areas.

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ANNEX I

CROSS-REFERENCE BETWEEN GENERIC SAFETY ISSUES AND RELATED RECOMMENDATIONS CONTAINED IN THE IAEA DATA BANK

This is a compilation of all relevant and agreed recommendations made for the improvement of the RBMK NPPs contained in the IAEA Extrabudgetary Programme Data Base, which were used by the experts for the identification of the safety issues presented in Section 3 of this report.

The items listed below are taken out of the context of different reports in which they appear. It is strongly recommended that they should be read or quoted in conjunction with the report referenced in the Source column to obtain a full appreciation of them and the context within which they have been made.

ID	Description	Source
1R2.10-3	Sensitivity analysis including review of model nodalisation should be carried out for partial breaks of the distribution group header.	Consortium
1R2.10-9	The need of bypass lines between main circulation pump suction and pressure headers should be reassessed. They should either be removed or fitted with motorised isolation valves in all RBMK.	Consortium
1R2.8-1	It is assumed that the subcriticality analysis of the spent fuel storage pond includes the drop of a single assembly, which subsequently falls from the vertical, striking and damaging a second assembly. If not, this case should be analysed or all assemblies should be stored in protective cylinders.	Consortium
1R3.1.10-1	The adequacy of the KRITIKA code in assessing the drum separator level behaviour and the affected primary loop flow in the case of feedwater line break downstream of the check valve has to be assessed. Check calculations should be performed by best-estimate codes with qualified models for the drum separator under the specific flow regimes.	Consortium
1R3.1.3-2	The reliability of timely FASS actuation should be reviewed for the case of DGH ruptures downstream of the check valve. If a significant unreliability remains with the first actuation parameter (pressure rise in the water line compartment), a sensitivity calculation with respect to a delay in reactor trip would be recommended with high priority	Consortium
1R3.1.3-3	If a significant uncertainty remains on the prediction of the swell level in the drum separators in situations with DGH ruptures, sensitivity studies with regard to core cooling conditions should be performed with special consideration of the code models describing the drum separator level.	Consortium
1R3.1.6-2	Further analysis for ruptures of a drum separator downcomer line should be performed with failure of the earliest signal contributing to reactor scram and ECCS actuation (pressure increase in the leaktight or drum separator compartments depending on rupture location) to identify the mode and time of the alternative ECCS actuation and demonstrate that a margin to fuel failure still exists.	Consortium
1R3.1.7-1	Although the break of a communication line between drum separators is believed to be bounded by the downcomer rupture the analysis should be performed and documented.	Consortium
1R3.1.7-2	The analysis of the break of a communication line between drum separators should be repeated assuming failure of the signal for reactor scram and ECCS actuation derived from pressure increase in the drum separator compartment, to demonstrate that fuel failures do not occur.	Consortium
1R3.1.8-3	An automatic scram signal should be derived for the water pipeline break taking into account the results of analyses of recommendation 1R 3.1.8-2.	Consortium

ID	Description	Source
1R3.1.9-2	An automatic scram signal should be derived for the steam/water line pipe rupture using results of analyses performed with best-estimate codes, in order to reduce the risk of consequential failures and uncontrolled releases (see also recommendation R3.1.8-3).	Consortium
1R3.12-1	A comparison of Eastern and Western statistical methodologies used to calculate the margin to critical power should be performed.	Consortium
1R3.2.2-4	Best-estimate analyses of rod withdrawal accidents should be performed using the current Smolensk 3 core as well as a fresh first core status. The real time delay of the CPS and the assumption of failure of first scram signal should also be considered.	Consortium
1R3.2.3-1	It is recommended to provide adequate documentation of the analysis of rod drop out of the bottom of the reactor.	Consortium
1R3.2.5-1	It is recommended to provide adequate documentation for the case of gas ingress from the ECCS accumulators for assessment.	Consortium
1R3.2.5-3	The actual time delay of CPS response should be considered when analysing cases with gas ingress from the ECCS accumulator tanks. Gas injection should be directed into one core half.	Consortium
1R3.2.7-1	It is recommended to provide adequate documentation for the drop of a fresh fuel element into reactor assessment.	Consortium
1R3.2.7-2	An evaluation of possible consequential failures of pressure tubes and fuel elements due to vibration should be included in the analysis of the fuel element drop.	Consortium
1R3.3.2-1	It is recommended to analyse the simultaneous trip of all operating main circulation pumps from full power and from low power with a state-of-the-art computer code.	Consortium
1R3.3.5-1	It should be checked that the operator can diagnose the MCP check valve disk break-away and take appropriate measures.	Consortium
1R3.3.6-1	Code simulations of a DGH check valve quick closure resulting from a break opening should be examined to verify that the flowrate acceleration does not exceed 10 kg/s2.	Consortium
1R3.3.6-2	The probability of a DGH check valve disk rupture should be carefully estimated in case of a break opening upstream of the check valve. Depending on this value, accidental sequences with failure of one or more than one check valve should be also considered in safety analyses.	Consortium

ID	Description	Source
1R3.3.8-1	In case of complete closure of the pressure tube isolation control valve, very high temperatures are calculated in the cladding (about 1600 ^{o}C) when *no-flow* conditions are conservatively simulated. The behaviour of the fuel under such conditions and under reflooding after the channel break should be extensively investigated, identifying all the experimental results available and applicable to the specific case. Bounding evaluations should be performed in order to investigate all the conceivable consequences of the accident, e.g. fuel behaviour under severe water/zirconium reaction rates, thermal stresses induced by the injection of water.	Consortium
1R3.5.2-1	For the case of trip of both turbine generators, analysis of additional system failures should be carried out, or any further existing sensitivity analysis should be documented.	Consortium
1R3.5.5-1	It is recommended to document the analysis of excessive feedwater flow events and to investigate situations with different initial power, different void coefficients, and different feedwater flowrates.	Consortium
1R3.5.6-1	It is recommended to document the analysis of cases with reduction of feedwater temperature and to investigate situations with different initial power, different void coefficients, and different feedwater temperatures.	Consortium
1R3.5.7-1	It is recommended to analyse and to document cases with excessive steam demand for different initiating events.	Consortium
1R3.7.2-3	The implementation of an automatic scram signal on flow reduction in several pressure tubes connected to the same distribution group header is highly recommended.	Consortium
3R4.2.2-2	Analyses of design basis accidents should be undertaken with different Western calculational tools in order to evaluate the results presented for safety analyses. The goal of these analyses is to improve RBMK calculational methods in the field of space time dependent core parameters.	Consortium
3R4.2.2-3	For all accident analyses, conservatism of modelling and initial situations assumed in the general safety analysis studies should be peer reviewed for each RBMK unit.	Consortium
3R4.2.3-12	According to Western methodology, operational transients and accidental situations should be analysed in order to verify the consistency of design criteria for CPS and operational procedures. The framework of multilateral collaborations is necessary to achieve this goal.	Consortium
IGN004	Physics work and dynamic analyses related to the safety analyses report for Ignalina should be carried out using western and advanced codes and techniques.	Ignalina Report
IGN005	Sensitivity studies should be undertaken to determine the effect of crediting the second trip parameter and of delaying the ECCS water reaching the fuel.	Ignalina Report

ID	Description	Source
IGN034	Development of the Safety Analysis Report for the Ignalina NPP is supported. Since the current TOB does not reflect the actual INPP operation status (e.g., the core physics design does not correspond to the operating reactor), the INPP Safety Analysis Report should correct existing deficiencies on documentation as well as improve the accident analyses available at the plant.	Ignalina Report
IGN035	Since the experience with the RBMK accident analysis is limited outside of Russia, it is desirable to involve the Russian expertise in the above work.	Ignalina Report
IGN036	Since the validation of computer models is rather limited for RBMK accident conditions, any activities that compare the Western models against Russia data, or the Russian models against Western data, are supported. In this context, the Russian development of validation matrix is supported.	Ignalina Report
IGN037	Development of symptom-based Emergency Operation Procedures currently in progress is supported. It this context, the INPP should strengthen its understanding of the basis for the EOP's. It should also establish and maintain ties with the other RBMK plants to ensure an efficient communication of operating experience and of any generic developments regarding plant safety.	Ignalina Report
RB205	Reviews in the form of specialist meetings should be held on computer code documentation and validation standards.	TECDOC-694
RB262	The safety report would be more independent if the pertinent sections of the documents referred to were included in it.	TECDOC-722
RB282	Analyse the cladding failure of a control rod and ensure that the energy release is insufficient to cause stagnation in the low pressure cooling circuit.	TECDOC-722
RB314	Results of the CATHARE analyses should be reviewed.	TECDOC-722
RB315	The sensitivity of results of DGH break analysis to: (i) delay time in reactor trip; and (ii) delay time in ECCS injection should be investigated.	TECDOC-722
RB316	This case of DGH break downstream of the check valves should also be calculated without assumption of loss-of-power. Then it is not clear how ECCS will be actuated or how long that will take.	TECDOC-722
RB318	The results of the CATHARE analysis should be peer reviewed.	TECDOC-722
RB319	The applicability of the RELAP code to the thermal hydraulic analysis of multiple parallel channels should be assessed further. The results of any comparisons of code results to applicable experiments on the thermal hydraulic behaviour of multiple parallel channels under "stagnated" conditions, and conditions where channel failure terminates the stagnation in the remaining channels, should be reviewed.	TECDOC-722

ID	Description	Source
RB324	For all accident sequences a systematic analysis of the sequence of all scram signals should be performed.	TECDOC-722
RB334	In addition to licensing type calculations with a great number of conservative assumptions, also best estimate type calculations should be performed, e.g., for the development of accident procedures and for operator instruction on a realistic accident scenario basis.	TECDOC-722
RB338	The applicability of the RELAP code to the thermohydraulic analysis of multiple parallel channels operating at full power and under conditions of significantly reduced flow should be investigated. Results of code comparisons to relevant experiments should be reviewed.	TECDOC-722
RB585	Events involving reactor transients should be analyzed to confirm they comply with design expectations and to determine if reactor protection settings are optimum.	ASSET RBMK
RB596	Since actual results were not presented, and since better tools are now available to undertake these analyses, it is recommended that the Kurchatov staff re-analyse the scenarios they propose to better define the channel temperatures.	RBMK-RD-005
RB597	It is necessary to continue investigations on potential scenarios which can lead to multiple pressure tube rupture (MPTR).	RBMK-RD-005
RB598	It has been shown from experimental results for the advanced thermal reactor (ATR) in Japan that the critical heat flux for downward flow (reversal flow) is lower when the flow rate is small. The film boiling heat transfer coefficient after the crisis for downward flow is also low compared with that for upward flow. These two characteristics have significant effects on cladding and pressure tube temperature transients and need to be factored into the calculations. Therefore, they should be clarified with an experimental facility simulating the RBMK flow channel and a fuel bundle, and incorporated into the RELAP5 code as user's option. After that, the code capability should be checked using Russian data or data published in literature.	RBMK-RD-005
RB599	It is also recommended to validate the RELAP5 code with results of systems tests that simulate the conditions peculiar to the partial break on the DGH with injection of subcooled water. The nodalization of the inlet tube between the DGH and the core of the RBMK should be changed.	RBMK-RD-005
RB600	It is necessary to consider the practical value of experiments on the rigs of Electrogorsk and numerical investigations of DGH breaks with six pressure tubes using RELAP5, COBRA TF and other codes.	RBMK-RD-005
RB601	It is necessary to continue verification of experimental correlations of RELAP5/mod3 for reflooding, down flow and heat transfer of top flooding for fast pressure transients to conform better with experimental results.	RBMK-RD-005

ID	Description	Source
RB602	Thermomechanical data for graphite columns and for pressure tubes should be implemented in the RELAP5/mod3 for evaluation of multiple pressure tube ruptures. It is necessary to continue the investigations for scenarios that can lead to a beyond design basis accident involving multiple pressure tube ruptures.	RBMK-RD-005
RB603	It is recommended to continue independent calculations of a partial break of a DGH using the RELAP5 and CATHARE codes.	RBMK-RD-005

ID	Description	Source
1R2.10-6	Analysis should be carried out to demonstrate plant safety in faults from full power initiated by rupture of an ECCS nozzle connection to a distribution group header, with failure of a different train of the main or long-term ECCS subsystems connected to the same distribution group header.	Consortium
1R2.11-6	The performed UDBA and DGH rupture analyses should be reassessed with regard to the single failure of valves affecting the under reactor compartments or the reactor cavity.	Consortium
1R2.12-7	Analyses should be performed to assess whether the consequences of possible failures of the ALS pools steam suppression function during loss of coolant accidents could be unduly severe. In particular, the effects of such accidents on the reactor cavity should be studied.	Consortium
1R3.1.11-1	The case of a feedwater line rupture downstream of the feed assembly room with failure of one check valve to close should be considered in more detail. In particular, the time delay to allow manual actuation of the ECCS by operator should be determined.	Consortium
1R3.1.15-3	The maximum loss of coolant accidents outside of the hermetic compartments in combination with loss of preferred power should be analysed in order to verify, whether the time available for required operator actions is acceptable, and to verify that safety-grade sources of water are available for long term primary system refilling.	Consortium
1R3.1.16-14	The management of large quantities of contaminated water to be expected in some accident scenarios requires the availability of adequate systems for their transfer and treatment. This item needs further review.	Consortium
1R3.1.2-1	A check calculation using a state-of-the-art system code is recommended for the case of DGH ruptures upstream of the check valve.	Consortium
1R3.1.3-1	A check calculation using a state-of-the-art system code is recommended (if it has not yet been performed with RELAP5) for the case of DGH ruptures downstream of the check valve.	Consortium
1R3.1.3-4	For the case of DGH ruptures downstream of the check valve, a sensitivity study with respect to vapour pull-through during liquid reverse flow from drum separators is recommended, to ensure adequate core cooling conditions.	Consortium
1R3.1.3-5	The case of DGH ruptures downstream of the check valve should also be analysed without assuming simultaneous loss of power.	Consortium
1R3.1.4-1	Sensitivity calculations are recommended for DGH breaks downstream of the check valve with check valve failure in a neighbouring DGH; particularly, break flow and condensation rate modelling should be varied.	Consortium

ID	Description	Source
1R3.1.4-2	DGH breaks downstream of the check valve with a partial rupture of the check valve disk in a neighbouring DGH should be studied, especially for investigating scenarios with possible long-term flow stagnation in the pressure tubes connected to the this DGH.	Consortium
1R3.1.5-1	Considering the consequences of a partial DGH rupture downstream of the check valve, it is recommended to continue the investigations of all possible events resulting in a break of 10 % to 15 % of the DGH cross section.	Consortium
1R3.1.5-2	Sensitivity calculations for cases with partial DGH ruptures downstream of the check valve are recommended with high priority. The conditions of the scenario (break location, break size) and code modelling options (nodalisation of the DGH, nodalisation of the vicinity of the break, number of groups of channels represented) should be varied; check calculations with other codes should be performed.	Consortium
1R3.1.5-3	The computer codes utilised for cases with partial DGH ruptures downstream of the check valve should be verified for conditions with flow instabilities in parallel channels.	Consortium
1R3.1.5-4	The case of a rupture of a ECCS header- should be analysed with the assumption that one of two other ECCS trains is not available. This case corresponds to a partial break of the pressure header or of the DGH upstream of the check valve with only one out of three ECCS trains able to provide water to the affected reactor half.	Consortium
1R3.1.6-1	The analysis of the rupture of a drum separator downcomer line should be repeated with actuation of the BRU-K valves, to maximise the initial rate of loss of mass inventory.	Consortium
1R3.1.8-2	Best-estimate analyses for different break sizes and break locations of single water pipeline ruptures should be performed using state-of-the-art thermal hydraulic codes in order to: - determine break sizes and locations which lead to fuel element overheating; - provide a basis for derivation of an automatic scram signal in case of a waterpipeline break.	Consortium
1R3.7.2-1	A check calculation using a state-of-the-art code is recommended for cases with flow stagnation in a group distribution header.	Consortium
1R3.7.2-2	Sensitivity tests should be performed, changing the flowrate through the bypass line from MCP pressure header to the DGH, to determine conditions of multiple pressure tube ruptures for cases with flow stagnation in a group distribution header.	Consortium
1R3.7.5-1	The limiting fuel conditions resulting from leakages in the case joint of the main coolant pump should be checked by independent, best-estimate calculations and, possibly, the limiting times at which radiative cooling could successfully remove the residual heat together with the long term behaviour of the overall reactor materials should be found.	Consortium
IGN005	Sensitivity studies should be undertaken to determine the effect of crediting the second trip parameter and of delaying the ECCS water reaching the fuel.	Ignalina Report

ID	Description	Source
IGN006	Further information is needed to perform analyses of the material properties of fuel cladding and pressure tubes under UDBA conditions as a function of lifetime (neutron irradiation, corrosion, H2 pickup, etc.) In addition, mechanical interaction of the pressure tube and graphite stack and with the fuel assembly (ballooning, rod bow, etc.) should be considered in this analysis.	Ignalina Report
RB200	Rationalize the definition of design basis LOCAs and identify the worst design basis LOCA sequences for the RBMK .	TECDOC-694
RB248	The RBMK specialists consider the worst design basis LOCA to be guillotine rupture of a pressure header with failure to close the check valve of one distribution group header. The RBMK specialists consider partial pipe/header ruptures to be highly improbable in that any partial break capable of heat removal deterioration is assumend to be greater than the critical crack length and therefore will result in complete guillotine rupture. However, partial breaks could result in periods of flow stagnation in one or more fuel channels which would result in a fuel cladding temperature in excess of those quoted in the reference design basis LOCA analysis. A pipe/header rupture combined with a seismic event could also pose problems if some of the mitigating systems credited are not adquately seismically qualified.	TECDOC-694
RB311	The complete list of required systems (pump seals injection, control rods cooling and so on) including supporting systems should be considered in the analysis of the accident sequences.	TECDOC-722
RB317	Further analysis of partial breaks should be performed to investigate the effect of delayed trip times on fuel and fuel channel behaviour.	TECDOC-722
RB320	Assumed steam flow rates should be high enough to drive the exothermic Zr/steam reaction but low enough that convective heat transfer is negligible.	TECDOC-722
RB321	Although it is recognized that the following sequence is beyond what the RBMK specialists consider to be a credible DBA, analysis of fuel and pressure tube heatup should be performed for the following conditions: (i) partial DGH break, between the ECCS connections and the first channel, which causes flow stagnation in the 42 channels connected to the DGH; (ii) no credit for heat removal due to low, oscillatory flows predicted by RELAP; (iii) failure of ECCS; (iv) delayed scram (i.e. failure of the overpressure signal in the accident localization system). It is recognized that this sequence (beyond DBA) could lead to multiple pressure tube failures. The objective is to determine how long the operator has, after receiving low flow indications in multiple channels on the same DGH, to initiate manual shutdown of the reactor. The timings of channel failures also provide timing information for the design and implementation of additional signals for reactor scram and ECCS initiation.	TECDOC-722
RB322	Analysis of local PT overheating due to contact between overheated fuel rods and the PT should be performed.	TECDOC-722

ID	Description	Source
RB325	Further review is needed concerning this accident with coincident check valve failure on a single DGH.	TECDOC-722
RB326	Partial breaks of the MCP header were not presented at the meeting, but the RBMK specialists provided information that such investigations have been carried out in the frame of beyond design basis accidents. Therefore, fuel and pressure tube heatup calculations should be reviewed for partial breaks in the MCP pressure header which could lead to stagnation conditions. Sensitivity analyses should be reviewed to determine the impact on fuel and pressure tube behaviour of delays in scram and/or ECCS initiation. Low, oscillatory flows in channels, as potentially predicted by RELAP, should not be credited to remove heat. The objective of the analysis is to determine the timing of pressure tube rupture and the number of channel failures. From the sequence of these channel rupture times, the time the operator has to manually scram the reactor and/or initiate ECCS prior to the failure of an unacceptable number of channels can be determined. The timing of channel failures also provides information to indicate what the timing of additional trips and/or signals for scram and/or ECCS initiation must be in order to preclude multiple channel failures.	TECDOC-722
RB588	An independent verification of the core power increase during the first few seconds should be performed	RBMK-RD-005
RB589	Partial breaks of a PH with failure of one check valve in a DGH should be studied with advanced system thermohydraulic codes (e.g. RELAP5/mod3, CATHARE, CANDU codes, etc.).	RBMK-RD-005
RB590	Partial breaks of a DGH should be studied to determine what additional capability for venting the reactor cavity is required. French specialists recommend that partial breaks of a DGH should be studied without crediting any cooling due to oscillations. After the first pressure tube rupture the simulation should be continued to look for other possible flow stagnation periods of time and the maximum number of pressure tube ruptures should be determined. If it exceeds the venting capacity of the reactor cavity, additional venting systems should be implemented. This assumption will certainly lead to very high temperatures. Canadian specialists recommend that more realistic calculations are needed before increasing the reactor core cavity venting capacity.	RBMK-RD-005
RB591	The implementation of ECCS injection in steamwater lines from the drum separator could provide a better protection for this scenario.The RBMK specialists indicated that such a system is under design at RDIPE. If this modified ECCS is shown to be effective, and does not raise new problems for other scenarios, it should be implemented.	RBMK-RD-005
RB592	In the design of Smolensk 3 none of the pipes connected to the DGH correspond to the dangerous break size. It should be clarified as to whether this is true for all the remaining RBMK units.	RBMK-RD-005

ID	Description	Source
1R2.11-5	In order to assess the accident localisation system (ALS), analyses of the ALS behaviour must be presented at least with regard to the simultaneous rupture of 1, 5 and 9 pressure tubes.	Consortium
1R2.12-2	The studies directed to systematically investigate all the possible causes leading to single or multiple tube ruptures in the reactor cavity should be continued. All the feasible preventive measures should be taken with priority in order to avoid the actuation of the *safety device* relieving the reactor cavity pressure directly to the atmosphere.	Consortium
1R2.12-3	The assumptions made in the analysis of overheating and in determining the conditions for pressure tube rupture in the reactor cavity (temperature and pressure) should be confirmed by specific tests performed on channels in the worst conditions which can be encountered during reactor operation with respect to hydration and to channel-graphite interface, in addition to the experiments already performed in the past on new tubes.	Consortium
1R2.12-4	The multiple pressure tube rupture in the reactor cavity which requires a greater relief capacity than the normal relief system should be considered an accident sequence for which envelope boundary conditions should be established. This event should be thoroughly analysed considering all the aspects related to the assumed accident sequence, and a sound basis for the *Safety Device* performance should be established.	Consortium
1R2.12-5	The results of local pressurisation in the reactor cavity following pressure tube ruptures should be checked by use of best-estimate codes capable of considering all effects, eg local pressurisation, and jet forces which can be generated under such accident conditions.	Consortium
1R2.12-6	The Safety Device, that provides additional reactor cavity protection against multiple tube ruptures, should be modified in order to avoid the unfiltered and unisolatable release of fission products from the primary system. Uncontrolled, massive air ingress into the reactor cavity should be also avoided under any accident conditions.	Consortium
1R2.12-8	A diverse parameter to assure reliable scram actuation in case of pressure tube break in the reactor cavity should be included in the reactor protection system.	Consortium
IGN048	The increased reactor cavity relief capacity which is proposed by the design organizations is supported by the reviewers. It is proposed that the Lithuanian government establish an action plan to resolve the difficulties related to the issue of nuclear liability.	Ignalina Report
RB101	Upgrading the current emergency steam-gas-dump system designed specifically for fuel channel (FC) failure, plant safety is assured under simultaneous failure of up to ten FCs.	TCM-92
RB106	Introduction of specially designed supplementary preventive device into the system of emergency steam and gas discharges from the reactor cavity to increase its capacity.	TCM-92

ID	Description	Source
RB211	The RBMK specialists should proceed with high priority to complete implementation of the improvements to the cavity overpressure protection system.	TECDOC-694
RB212	In order to establish a definitive design basis for the cavity overpressure protection system additional analyses should be performed for more severe sequences in terms of pressure tube integrity. Peer review of the results and audit calculations are recommended to support the analysis methodology and results.	TECDOC-694
RB215	The designer should present results of a series of functional tests on prototypes of the atmospheric vent valves carried out to establish performance characteristics, recognizing their existence only as an interim feature.	TECDOC-694
RB335	Since atmospheric venting is implemented to provide a large margin for protection of the reactor cavity structure from overpressurization, based on current DBA, the RBMK specialists should investigate whether that venting would be better if done into the tower shaft directly (using one-way panel blowout approach in the DS compartment). The incremental release into the atmosphere is probably small, but the benefit to reduced dose and contamination to workers could be very large by eliminating the release into the DS room which is connected to the central hall by 5 m^2 open area.	TECDOC-722
RB337	The RBMK specialists should investigate a long term solution where the discharge from the reactor cavity is directed into the pressure suppression pool of the accident localization system as proposed in TECDOC-694.	TECDOC-722
RB604	An independent evaluation of the venting system performance should be made by experts from OECD countries, for each RBMK unit.	RBMK-RD-005
RB605	Every break size, every break location and every initial operating condition should be considered from the point of thermohydraulic consequences and protection and mitigation systems design.	RBMK-RD-005
RB606	Installation of fast SCRAM system initiation by low flow detection in several channels of any DGH should be considered for each RMBK unit.	RBMK-RD-005
RB607	An ECCS injection system in the top part of each channel (e.g. Leningrad Unit 2 NPP) should be designed and peer reviewed for potential implementation in each RBMK unit. An analysis to support this suggestion was not discussed at the meeting. RDIPE Specialists informed that the system modification is already under design.	RBMK-RD-005
RB608	Modifications to increase the capacity of the reactor core cavity venting system should be considered for each RBMK unit.	RBMK-RD-005

ID	Description	Source
1R2.10-2	The analysis of rupture of the main steam lines should be repeated including water entrainment in the break flow.	Consortium
1R2.2-2	The possibility of a false "low level in drum separator" signal for ECCS actuation should be reassessed and the steam line rupture case with loss of power should be reanalysed with this assumption.	Consortium
1R2.2-5	If it cannot be excluded that water can be entrained into the main steam lines in some accident scenarios, the loading analysis and environmental qualification of the main steam lines, including all associated valves, should be reassessed.	Consortium
1R3.1.10-2	A calculation should be performed with high priority to assess the fuel conditions in case of feedwater line breaks downstream of the check valve, assuming the actual ECCS behaviour that can be expected under the specific scenario, i. e. severely reduced capabilities of the short term ECCS.	Consortium
1R3.1.12-1	Check calculations for steam line ruptures outside the drum separator rooms without loss of power using a state-of-the-art code are recommended with high priority. Attention must be paid to the coupling with the neutronic calculation, coupling with level control, to the pump operating conditions and to the time of pump trip.	Consortium
1R3.1.12-2	Another automatic signal for ECCS actuation (in addition to low drum separator pressure) is recommended to cope with steam line ruptures outside-the drum separator rooms and other cases. This could be based on the steam line flow rates or on the rate of depressurisation.	Consortium
1R3.1.13-1	A check calculation of steam line break with loss of power using a best-estimate code is recommended with a high priority. Attention must be paid to the coupling with an adequate neutronic calculation.	Consortium
1R3.1.14-1	A check calculation using a best-estimate code is recommended for the case of stuck-open main steam line safety valves. The analysis should consider two important points for accident management: - How long can the ALS operate with two open MSVs? - What are the signals to allow a diagnosis of the situation?	Consortium
1R3.1.14-2	The analysis of transients with stuck-open main steam line safety valves should be repeated not crediting the functioning of operational control systems.	Consortium
IGN047	A strategy for mitigation of the consequences of ruptures in the drum separator and MSRV compartments should be developed.	Ignalina Report
RB329	Additional analysis results should be made available such as peak cladding temperatures, core inlet temperature, subcooling margin and DNB ratio.	TECDOC-722
RB331	The cases of steamline ruptures should be calculated with one computer code which is able to simultaneously model the thermohydraulic and neutronic behaviour of the RBMK.	TECDOC-722

ID	Description	Source
1R2.6-2	The effects of dynamic forces following high energy pipe breaks of systems connected to the primary circuit should be investigated in order to assure the capability to bring the reactor to safe shutdown.	Consortium
4R6.2-11	Multiple support excitations and differential support motion: reanalysis of piping and supports taking into account differential displacement of supports, is recommended, particularly for feedwater and steam pipelines.	Consortium
4R6.2-13	A plant walk-down for inspection of piping routing should be performed in order to verify that sufficient spatial separation exists between parallel pipelines and between piping and adjacent structures (e.g. pipe penetrations through walls) so that damaging impacts in a seismic event can be safely avoided.	Consortium
RB330	The possibility of pipe wipe should be taken into account in the steamline break analysis. Mechanical restraints on the main steam lines should be verified.	TECDOC-722

ID	Description	Source
IGN046	The consequences of injecting water into the group distribution header (GDH) in a loss of off-site power and blackout situation is not clear. Thermal-hydraulic analyses should be performed to analyse the possible impact on natural circulation.	Ignalina Report
RB332	The calculation for the case of loss of preferred power should be repeated with an up-to-date computer code up to when stable conditions are reached.	TECDOC-722
RB333	It is recommended to perform accident analyses on a plant-by-plant basis, e.g., for the loss of preferred power case at Smolensk 3 credit should be given to the availability of the turbine bypass station in the first phase of the event sequence.	TECDOC-722

ID	Description	Source
1R3.1.16-10	In case of a pressure tube rupture in the reactor cavity, additional releases due to possible mechanical fuel damages should be accounted for.	Consortium
1R3.1.16-12	Independent calculations should be performed to reassess the expected amount of radioactive isotopes in the under clad region used for the analyses of radiological consequences of DBAs taking into account deviations from expected accident progressions.	Consortium
1R3.1.16-13	The retention factor in the compartments outside the ALS should be verified on the basis of conservative calculations taking into account the real characteristics of the compartments and the adopted administrative procedures.	Consortium
1R3.1.16-15	Further calculation of radiological consequences should be performed taking into account a more complete set of radioisotopes.	Consortium
1R3.1.16-16	Independent evaluations should be performed to verify the calculated doses to the most critical organ of the most exposed individual under the assumed conditions of plant releases and meteorological conditions.	Consortium
1R3.1.16-5	The transient and accident analyses should be performed against acceptance criteria in terms of radiological consequences, graduated according to the frequency intervals of their occurrence.	Consortium
1R3.1.16-6	In order to evaluate the efficiency of mitigative engineered safety features to be considered as a further barrier for the fission products (eg ALS, gas treatment systems), the fuel failures to be assumed for the evaluation should be more conservative than those derived from mechanistic approaches. For this purpose, evaluations of the consequences of possible further deterioration of the DBA scenarios due to limited deviation from the envisaged behaviour should be made.	Consortium
1R3.1.16-7	The efficiencies assumed for the Iodine filters in the analyses of radiological consequences should be related to the effective design and to the in-service tests; they should be less than 100 %.	Consortium
1R3.1.16-9	All fission product release paths have to be considered in the analyses of radiological consequences, i.e. also components of systems where water or gases from the containment are treated.	Consortium
RB204	Reviews in the form of specialist meetings should be held on radioactive release calculation.	TECDOC-694
RB323	In case of the occurrence of fuel rod failures an analysis of the radiological consequences should be performed.	TECDOC-722
RB328	The radiological aspects of operation and maintenance during and after steamline break should be analysed for all essential parts of the plant.	TECDOC-722

ID	Description	Source
6R6-1	A fuller review of selected events having the potential to be precursors of more serious events should be carried out by appropriate specialists with a view to establishing the adequacy or otherwise of the automatic protection systems. Among other aspects, this review should consider: - the potential for multiple tube failure and the adequacy of the reactor cavity overpressurisation protection system; - measures to reduce the reactivity changes associated with loss of water in the CPS cooling system and measures to prevent air entrapment, air ingress or water boiling in this system; - the adequacy of the degree of protection provided by the neutron flux and rate of change of neutron flux lines of protection,, particularly with respect to local criticality or local high power distortion fault scenarios, and the adequacy of the fast shutdown systems; - the need for reactivity measures involving draining and refilling the CPS cooling system, and if they are required, the question of whether there is a safe way to conduct them; - the high reliance of safety function invested in correct operator action and the case for increasing the scope of automated protection and control systems; - the quality and reliability of essential instrumentation and safety equipment; - the potential problems associated with circuit debris and measures to minimise/eliminate such debris. One outcome of such a review should be used to develop a more comprehensive Safety Case (TOB) and one which shows how Operating Rules and Restrictions, and protection settings are derived from appropriate fault analyses. This issue is addressed in Recommendation R-7 in Chapter 7.	Consortium
9R8-1	Plant-specific PSAs should be carried out on all plants. The PSAs should have the following objectives and priorities: Objectives: to identify plant specific deficiencies and weaknesses · to educate staff in PSA as a tool for safety management Priorities: - Scram function, including the control and protection system; - Actuation and control system for Emergency Core Cooling and Auxiliary Feedwater System; - Long-term cooling and make-up systems; - Common cause failure initiators.·	Consortium
RB203	Reviews in the form of specialist meetings should be held on reliability analysis.	TECDOC-694
RB207	An independent review of available probabilistic safety assessments (PSA) of RBMK reactors should be performed prior to using the results in support of safety decisions.	TECDOC-694

ID	Description	Source
1R3.2.8-2	Due to the relatively high probability of CPS cooling system voidage, operational transients such as gas ingress to the system should be chosen as an initial event for ATWS analysis.	Consortium
3R4.2.2-7	Considering the displacer temperature, the WIGNER energy stored should be evaluated in order to determine the risk of control rod blockage or ATWS due to an hypothetical release of that energy.	Consortium
IGN007	Taking into account up-to-date requirements for safety analysis, scenarios requiring ATWS analysis should be identified. Assessment of these ATWS sequences and their analysis should be undertaken.	Ignalina Report
RB310	The PSA studies for Smolensk 3 are to be encouraged and sequences requiring ATWS analysis should be identified. Independent assessment of these sequences and their analysis should be undertaken.	TECDOC-722

ID	Description	Source
4R6.1-13	It is necessary to discuss the sub-task Flooding in complete volume preparation of information, analysis, joint discussions, recommendations) on the next step of the project.	Consortium
4R6.2-10	The supporting steel structures of the rails of the main crane in the upper reactor hall should be reinforced for seismic intensity I = 7 for local earthquakes and I = 6 for distant earthquakes.	Consortium
4R6.2-12	It should be necessary to prepare and carry out joint plant walkdown in order to verify supports of piping and mechanical and electrical equipment.	Consortium
4R6.2-3	The use of artificial time histories derived from a large band response spectrum is recommended.	Consortium
4R6.2-4	The soil structure interaction has been considered in the seismic analysis by a simplified method and a check, taking into account parameters as radiation damping, non-linearity of soil, and embedment that certainly produces more realistic response values, is advisable.	Consortium
4R6.2-5	No studies of soil stability have been carried out for Smolensk site, but the presence of a saturated sand layer under the reactor building/13/ indicates the need for checking the liquefaction potential under cyclic load. The same recommendation holds for the two dams near the plant.	Consortium
4R6.2-6	Backfitting measures should be performed for reinforcing metal structures in the upper reactor building in order to assure seismic resistance up to intensity I = 7 for local earthquakes and up to intensity I = 6 for distant earthquakes (i.e. reinforcement of the metal roof of the upper reactor hall).	Consortium
4R6.2-7	The responses of the metal structures should be calculated using dynamic analysis methods with the objectives of achieving as realistic results as possible and of avoiding the conservative contributions resulting from former analyses using floor response spectra.	Consortium
4R6.2-8	The supports of the steam drum separators require reinforcement for seismic loads of earthquakes with intensity I = 7 for local earthquakes and I = 6 for distant earthquakes. The respective analyses should be performed by using dynamic methods of determination of appropriate reinforcing measures of the supporting devices with due consideration of realistic seismic responses of the drum separator and the connected piping.	Consortium
4R6.2-9	Confirmation of the earthquake stability of the fuel recharging machine (FRM) in operational state is needed for intensity I = 7 of local earthquakes.	Consortium
4R6.3-1	A more detailed investigation of the risk due to pressure waves to RBMK plants is recommended with particular attention to the effects resulting from the amount of explosive substances stored on the plant site itself.	Consortium

ID	Description	Source
4R6.4-1	A thorough assessment of the military flight situation in the vicinity of the Ignalina NPP is proposed with due consideration of the possible presence of military airfields next to the plant in the 3 surrounding countries (Lithuania, Latvia and Belorussia).	Consortium
4R6.5-1	Two kinds of possible external events, flooding and explosion, were almost not at all discussed in this phase of the project. Also, in all the items handled, things stayed open requiring further studies. Thus the work of the group on external events is recommended to be continued, one way or another. It is recommended to base any continuation of the work on the same organizations and the same experts.	Consortium
RB198	Development of the analysis method for pipe rupture, including zirconium, stainless steel, and carbon steel materials and taking into account earthquakes, should be continued and the probability analyses should be conducted for each specific equipment layout.	TECDOC-694
RB281	Review the seismic data available for the fast shut-down rods specifically and ensure that they will drop during a seismic event.	TECDOC-722
RB379	It is highly desirable that the database and the methodology for the evaluation of the design basis earthquake should comply with the recommendations of IAEA Safety Guide 50-SG-S1 (Rev. 1) `Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting'. In particular, the minimum recommended value of 0.1 g for SL2, associated with the appropriate response spectra and time histories should be taken into consideration as review level for the Smolensk NPP. Engineering practice shows that for structures of the type of a nuclear power plant designed using a standard response spectrum anchored to a value of 0.1 g (including effects of local earthquakes) is likely to be covered for the effects of large distant earthquakes (such as Vrancea) having a free field maximum acceleration value of 0.05 g at the site. Since a number of nuclear power plants, and in particular RBMKs, are located on the Russian platform which has been repeatedly affected by earthquakes originating in the Vrancea area, consideration may be given to carry out a special study concerning the effects of historical events from this source.	TECDOC-722
RB380	The presence of a saturated sand layer (3-4 metres thick) under the reactor building, indicates the need for checking the liquefaction potential under cyclic loading, both for near-field and far-field ground motion conditions. The data needed for this evaluation would be available only if the scope of the presently envisaged site investigations programme is expanded. The first step would involve standard penetration tests. If the safety factor for liquefaction is not found to be sufficiently large using these simple methods, more sophisticated laboratory testing (e.g. simple shear tests, dynamic triaxial testing) would be required long with appropriate computer programs for the analysis.	TECDOC-722

ID	Description	Source
RB381	International engineering practice shows that, in the soil structure interaction analysis, the 3-dimensional nature of the problem should be taken into consideration. Furthermore, such parameters as radiation damping, soil non-linearity, embedment, point of application of the design ground motion, should also be properly accounted for. International experience shows that such considerations yield more realistic response values. From these analyses, seismic loads for the evaluation of soil dynamic bearing capacity, liquefaction potential, dynamic settlement, overall building stability and floor response spectra should be derived. Dynamic soil parameters (i.e. shear wave velocity) are required down to an appropriate depth and as a minimum to -100 m. A dynamic soil testing campaign (both in situ and laboratory) should be designed to cover the data needs in sections VII.1.2 and VII.1.3.	TECDOC-722
RB382	Recommendations related to item VII.1.2 should be repeated for both dams. Different dam break scenarios should be analysed together with the safety impact of these to the plant. This issue should be included in the programme of further consideration of Smolensk NPP safety.	TECDOC-722
RB383	The result of the ongoing seismotectonic studies should be used to evaluate the benefits of such a scheme. Recommendations of the IAEA Safety Guide 50-SG-D15 Seismic Design and Qualification of Nuclear Power Plants regarding seismic instrumentation should be considered in the design of the seismic instrumentation system. Recommendations related to sites with SL-2 acceleration of less than 0.25 g may be used for the Smolensk site.	TECDOC-722
RB384	The floor response spectra should be rederived using less conservative soil structure interaction modeling as recommended in Section VII.1.3 and should be correlated to test results as described below. For existing structures, the most direct and accurate way of evaluating linear response quantities for this building (as well as other buildings, such as the turbine hall, pump house and diesel generator building) would be to conduct full scale dynamic testing preferably using a blast to induce vibrations. It will be useful to study the IAEA experience in this field. The testing results can be used to understand the true soil-structure interaction behaviour and to remove the excessive conservatism in defining seismic loads.	TECDOC-722
RB385	Engineering experience shows that for higher levels of seismic input (~0.2 g), considerations such as the provision of cross-bracing in the longitudinal direction, improved connections between precast structural elements (beams, columns and wall panels) and decrease in the dead weight of the roof significantly reduce the potential for seismically induced failures. Such considerations may be useful for RBMK building structures to be built in areas of similar seismic input levels (i.e. ~0.2 g).	TECDOC-722
RB386	The ongoing effort of the Russian specialists to experimentally determine if a multiple tube rupture can occur and the analysis to determine its probability of occurrence should continue. Results should be co-ordinated with accident analysis experts.	TECDOC-722

ID	Description	Source
RB387	Since the Russian regulatory codes require evaluations of safety related piping systems, the analytical evaluations should continue for safety related piping. However, in order to preclude the possibility of negative results due to the use of very conservative input floor spectra, the analytical effort should be based upon the use of revised, less conservative floor spectra as recommended in VII.1.3.2. The regulatory codes require that 2% dumping be used in the analysis of piping and equipment. Consideration should be given to increasing the damping values used in combination with response spectrum modal analysis methods. U.S. practice currently allows a variable damping ranging from 5% up to 10 Hz and reducing to 2% at 33 Hz response spectrum analysis.	TECDOC-722
RB388	The effort to evaluate equipment for an intensity VII earthquake should continue. As discussed in Section VII.1.3.2, the floor spectra being used are very conservative and should be revised using less conservative soil-structure interaction modeling. The current spectra may result in analytical predictions of failure of equipment anchorage or functions whereas the current installations are very likely adequate for an intensity VII earthquake. Specific recommendations regarding the findings are: (i) A thorough walkdown of the plant should be conducted to verify anchorage and to identify potential spatial systems interactions (see Section VII.3.4 for discussion of interaction issues). (ii) An evaluation of the pure condensate storage tanks should be conducted. (iii) At the next stage of the seismic capacity evaluation of the NPP, it will be necessary to thoroughly consider the available analytical and experimental studies related to active equipment and relays.	TECDOC-722
RB389	The problem of spatial systems interaction should be considered in more detail in further stages of the seismic capacity evaluation. A detailed walkdown should be conducted combined with the equipment anchorage verification walkdown (recommendation VII.3.3.2.1.) to identify potential spatial interactions. Potential interactions which are identified should be analysed and/or fixed.	TECDOC-722

ID	Description	Source
3R4.2.1-1	The safety evaluation relies on the capability of codes describing the full core behaviour. The need to improve 3D reactor core models including thermal hydraulic feedback and to develop coupling between 3D core codes and thermal-hydraulic modelling of the main coolant circuit are strongly emphasised. International collaboration in this field will be useful.	Consortium
3R4.2.1-2	Improvement of basic data libraries and their application should be obtained with respect to: - limitation of application to the validated range of data library, - extension of range of data library for accident calculations, - boundary energy limit between thermal and fast groups must be out of the large ^{240}Pu resonance (1 ev), - improvement of modelling for Xenon transient, - extension of data libraries to include released energy per fission's, and - contributions of effective delayed neutron fraction from all fission isotopes.	Consortium
3R4.2.1-3	Better information about burnup distribution is needed for the validation of codes and for redistribution and local effect evaluations. The reconstruction flux based on measurements could be used. From these considerations it will be expedient to upgrade the SKALA system.	Consortium
3R4.2.1-4	Evaluation of critical experiments without interfacing effects should be necessary to improved basic data and transport calculation modelling.	Consortium
3R4.2.1-5	A benchmark for calculating nuclear data for fuel cells and absorber cells of RBMK with detailed evaluation of reaction rates and neutron spectra should be proposed.	Consortium
3R4.2.1-6	Due to different approximations assumed for the effective delayed neutron fraction calculation, the effective delayed neutron fraction used as reactivity unit should be precise in any case.	Consortium
3R4.2.1-7	Improvement of correction methods applied for absorbers' few group physical constants should continue, a theoretical way should be searched.	Consortium
3R4.2.2-1	A parametrical analysis of void reactivity measurements based on full 3D core modelling should be undertaken. In order to determine systematic errors, this analysis should include sensitivity evaluation of the main parameters as flow rates, flow rate distributions, control rod positions (axial and radial), void fraction distribution, power level, void coefficient.	Consortium
IGN001	It is recommended that the 3-D code development and validation, which is currently under way, be continued to provide improved calculational technique in the following areas: - more accurate interpretation of the measured void coefficient; - fuel burnup distribution; - ORM calculations under all conditions; - space- and time-dependent neutron, water density, fuel temperature and graphite temperature distributions; - predicting thermohydraulic instabilities.	Ignalina Report

ID	Description	Source
IGN004	Physics work and dynamic analyses related to the safety analyses report for Ignalina should be carried out using western and advanced codes and techniques.	Ignalina Report
IGN010	Thorough investigations supported by experiments to determine flux distribution and q during refuelling are necessary. Modification and improvements of on-line computer packages should be performed.	Ignalina Report
IGN056	The need for protection against local power excursion should be confirmed. The choice of, location of, and set points of the system that must provide protection against both radial and axial power distortions should be optimized. The solution preferred by the INPP requires the replacement of the Ag detectors and gamma chambers and the adoption of a multi-section hafnium detector.	Ignalina Report
RB173	Efforts to improve 3D-methods for neutronic calculations and static and dynamic analysis of power distribution should be continued and extended to cover fuel management. A common approach agreed by RBMK and western specialists should be used.	TECDOC-694
RB180	Integrated reviews of both the neutronic and thermal hydraulic studies are needed since there is significant interaction between them.	TECDOC-694
RB283	A 3-D calculation of the void reactivity coefficient and its independent verification is highly recommended. Methods to reduce the uncertainty of the measurements have to be developed in order to confirm calculated values of the coefficients.	TECDOC-722
RB295	In view of the improvements in calculational techniques being developed (see Section I.1.3.2) more information of the core state (e.g. detailed burnup distribution) should be obtained for verifying new 3-D calculations.	TECDOC-722
RB296	The intent to develop 3-D methods for adequately predicting space- and time-dependent neutron, water density, fuel and graphite temperature fields is strongly supported.	TECDOC-722
RB297	3-D techniques with proper representation of the transient behaviour of core and channel thermohydraulics should also be applied to analyse and predict thermohydraulic stability.	TECDOC-722
RB298	The intent to develop 3-D methods for adequately predicting thermohydraulic instabilities is strongly supported. 3-D analysis of thermohydraulic instability should be performed over the whole power range and for natural circulation conditions.	TECDOC-722
RB304	Further improvements and validation of 3-D tools is encouraged.	TECDOC-722

Issue number and title: Core design and core monitoring / 2 Core design void reactivity coefficient of the primary and CPS circuit.

ID	Description	Source
1R3.2.8-1	It is necessary to reduce the reactivity effect of the CPS cooling system voidage eg: - by subdivision of this system; - by implementation of the new manual control rod including new displacer design; - by using heavy water as coolant.	Consortium
3R4.2.3-4	The design of control rods should be reviewed to limit the reactivity effect of voiding the CPS system.	Consortium
3R4.2.3-5	Homogeneous solutions (burnable poisoning) to reduce the void effect instead of heterogeneous ones (additional absorber) should be investigated in order to reduce local effects and also the ORM.	Consortium
3R4.2.3-9	Homogeneous solutions (burnable poisoning) to reduce the void effect instead of heterogeneous ones (additional absorber) should be investigated in order to reduce local effects and also the ORM.	Consortium
IGN011	The intent to modify the control rod design in order to reduce CPS void effect is strongly supported.	Ignalina Report
RB075	Dependence of reactivity on the steam content, lattice pitch and composition of the core.	TCM-92
RB081	Subcooling of inlet water having influence on the reactivity effects from the void coefficient and pump cavitation.	TCM-92
RB095	Decrease of subcriticality of the reactor due to implementation of measures to decrease void reactivity.	TCM-92
RB182	Further experiments are needed to understand the blowdown characteristics of the main circulation circuit and of the whole heat transport system and their effect on reactivity addition.	TECDOC-694
RB231	The number of channels with 2.4% fuel enrichment varies from unit to unit. There is no plan to load 2.4% enriched fuel into the Ignalina cores.	TECDOC-694
RB251	Absorber assemblies (AA) are now left permanently in the core. About 80 AAs for RBMK-1000 and about 50 for RBMK-1500 are now in place.	TECDOC-694
RB283	A 3-D calculation of the void reactivity coefficient and its independent verification is highly recommended. Methods to reduce the uncertainty of the measurements have to be developed in order to confirm calculated values of the coefficients.	TECDOC-722
RB306	Ways should be examined to subdivide the CPS cooling system in order to limit the maximum amount of reactivity which could be added on failure of the CPS cooling system piping.	TECDOC-722
RB438	Upgrade human systems interface (HSI) in main control room, other important control rooms, and at locally operated equipment in areas of labelling, color coding, mimics, lighting and ventilation.	ASSET RBMK

Issue number and title: Core design and core monitoring / 2 Core design void reactivity coefficient of the primary and CPS circuit.

ID	Description	Source
RB557	Upgrade human systems interface in the main control room and other appropriate parts of the plant.	ASSET RBMK

ID	Description	Source
1R3.2.1-1	The twelve zone LAC/LAP system which reduces the number of potentially dangerous rods down to 4 should be implemented in all RBMKs (see also recommendation 1R 3.2.2-1).	Consortium
1R3.2.1-2	An interlock system should be implemented in order to avoid inadmissible depth of insertion of potentially dangerous rods at nominal power. This recommendation applies also to the twelve zone LAC/LAP system.	Consortium
1R3.2.2-1	The twelve zone LAC/LAP system which reduces the number of potentially dangerous rods down to 4 should be implemented in all RBMKs (see also recommendation 1R 3.2.1-1).	Consortium
1R3.2.2-3	The probability of a spontaneous rod withdrawal should be assessed on the basis of the existing operating experience (see also recommendation 1R 3.2.1-3).	Consortium
2R15-4	A 12-zone control system should be installed on all RBMK plants, including Kursk, and the logic should be extended to: - automatic control of radial power peaks (via MCRs and SBRs), - local automatic control and protective setbacks at Kursk.	Consortium
3R4.2.3-6	The implementation of the 12 zone control and protection system in SMOLENSK 3 unit is strongly supported.	Consortium
RB078	Unavailability of power density distribution control system and local automatic control and local automatic protection system at low power. In addition ex-core monitors could not indicate the distribution of flux in the core.	TCM-92
RB178	The implications of the change in local power control system, e.g. with regard to change in detector type and introduction of 12 zone controllers, should be peer reviewed.	TECDOC-694
RB224	The SKALA and TITAN computer systems appear to be overloaded and currently have a slow cycle time.	TECDOC-694
RB228	Rod withdrawal accidents cause a peak in the neutron flux in the vicinity of the withdrawn rod. Depending on the specific conditions, the flux distribution can actuate either the local control system or cause the power scram system (PSS) to activate. In the case of Kursk there is only an out-of-core control system so coverage will be poorest in that reactor.	TECDOC-694
RB229	The power peaks resulting from rod withdrawal are calculated using very conservative starting assumptions. Only in the case where all conservatisms are used, will very limited fuel melting occur in one channel for those reactors with 8-zone control systems. Therefore, for those reactors, an administrative procedure has been instituted which requires that the drive mechanisms be electrically disconnected when some of the central rods are fully inserted. This procedure is followed at Kursk NPP.	TECDOC-694
RB233	The number of bottom entry rods has not been increased to 32 on first generation units, because it gave rise to problems in maintaining the reactor subcritical during/upon shutdown.	TECDOC-694

Issue number and title: Core design and core monitoring / 3 Spatial poowег control and protection

ID	Description	Source
RB234	A modernized version of the control and instrumentation system has been developed and should be installed starting in early 1993. This new system has increased diversity and segregation of functions and a more rapid cycling of the data acquisition system. Additional protective functions are contained in the new system, for example actuation of the FSS in response to signals from in-core detectors.	TECDOC-694
RB253	8 control rods were changed for 2nd generation RBMKs so that they now enter from the bottom of the core. This brings the total of bottom rods from 24 to 32.	TECDOC-694
RB288	The 12-zone regulating and protection system should be implemented as soon as possible.	TECDOC-722
RB289	Activities to automate the current manual control system using 3-D on-line power distribution analysis should be pursued.	TECDOC-722
RB290	A study should be undertaken to determine whether the prototype automatic control system developed several years ago can be upgraded and implemented on all RBMK reactors.	TECDOC-722
RB303	Ways to implement this new 12-zone monitoring concept into a reliable protection system for fuel protection should be examined.	TECDOC-722
RB305	The planned modification of the LAR system is strongly supported.	TECDOC-722
RB342	The designer's intention to check, by a test, the efficiency of the method proposed to cool the core in case of complete loss of service water should be implemented.	TECDOC-722
RB351	The recommendation on the separation of the control and protection functions is repeated. It is advised that one criteria to be used in the considerations is the establishment of if the segregation of the protection function, BAZ and AZ1, the power reduction trips AZ3 and AZ4 and the power regulation functions can be improved.	TECDOC-722
RB357	There would be some benefit by updating the equipment to allow the ORM and the departure from nucleate boiling margins to be calculated more frequently. These are essential safety parameters which the operator needs in order to operate the reactor safely and provide manually invoked plant protection. The introduction of PC systems is a sound move in this direction, as is the development of a new microprocessor-based data system.	TECDOC-722
RB358	It would be useful to establish what parameters are time-critical, during both normal and emergency situations, and to ensure that these are updated at a suitable frequency in the development of any new system.	TECDOC-722
RB359	The physical segregation of the equipment is quite variable. In some places the segregation is excellent, in others it leaves a lot to be desired. For example the BAZ and ECCS actuation logic is split into two trains of three units located in six different rooms. The essential neutronic inputs and control rod position control systems lack this in the independent protection channels which often share the same location even the same cubicle.	TECDOC-722

ID	Description	Source
2R15-12	The ORM is an important safety parameter in the operation of the RBMK. Automatic tripping on the basis of low ORM should be introduced.	Consortium
2R15-13	The ORM concept and the sophistication with which the ORM is calculated should be reviewed by a joint team of physicists and control engineers.	Consortium
3R4.2.3-1	The ORM is safety related, therefore the monitoring and limitation of the allowed ORM must be designed as a protection system. The homogeneous spatial distribution of ORM in the core should be guaranteed to prevent high positive local void effect.	Consortium
3R4.2.3-2	Automation of the shutdown action should be implemented when the ORM value falls below 30 rods.	Consortium
3R4.2.3-3	The way to reduce ORM contribution to the void effect decrease should be investigated and supported.	Consortium
IGN002	The process to reduce ORM is relatively slow when operating under steady state conditions. However, during transient conditions the value changes reasonably quickly. In order to relieve the operator of the responsibility of tripping the reactor when the value falls below 30 rods automation for the shutdown action should be implemented at the earliest possible time.	Ignalina Report
IGN003	Consideration of elimination, or at least reduction, of the safety role of the ORM was previously recommended in IAEA-TECDOC-694. The employment of burnable poison (erbium) is proposed by the designer to achieve this goal. The investigation of fuel and core performance as well as fuel testing before implementation is highly recommended.	Ignalina Report
RB079	Long period (10-15 minutes) of cycling through all measurements and for/of calculating the ORM.	TCM-92
RB091	Increase the operating margin from 26-30 to 43-48 effective manual control rods.	TCM-92
RB175	Further studies should be undertaken to eliminate the necessity of maintaining sufficient ORM in order to have an acceptable void coefficient.	TECDOC-694
RB176	The replacement of the ORM concept, which requires too much operator involvement, should be further explored.	TECDOC-694
RB226	When a violation of the ORM occurs, the operators are only required to scram by following an administrative procedure, since this has not been implemented into an automatic engineered safety feature.	TECDOC-694
RB255	The number of effective manual control rods for the ORM is now about 45 for RBMK-1000 and about 55 for RBMK-1500.	TECDOC-694
RB256	The ORM is now calculated every 5 minutes instead of every 15 minutes.	TECDOC-694
RB293	The process to reduce ORM is a relatively slow process when operating under steady state conditions. However during transient conditions the value can change reasonably quickly. In order to relieve the operator of the responsibility of tripping the reactor when the value falls below 30 rods, automation of the shutdown action should be implemented at the earliest possible time.	TECDOC-722

Issue number and title: Core design and core monitoring / 4 Operational reactivity margin (ORM)

ID	Description	Source
RB294	Consideration of the elimination or at least the reduction of the safety role of the ORM was already recommended in TECDOC-694. The investigation of the effectiveness of additional absorbers to reach this goal in the near future or short-term is highly recommended.	TECDOC-722

ID	Description	Source
1R3.2.8-4	The influence of a loss of water from CPS cooling system on the performance characteristics of the FASS rods and on the response of the in-core detectors must be analysed.	Consortium
3R4.2.3-8	Loading pattern of fixed additional absorbers should be evaluated in order to reduce the efficiency of control rods located at the Eastern core boundary of RBMKs.	Consortium
9R6.5-1	A fully diversified scram system with efficient neutron flux detectors should be introduced on all units.	Consortium
IGN008	The proposed work related to core monitoring and control to be carried out under the EBRD Programme for safety enhancement of the Ignalina NPP is: - Compact simulator - Additional emergency shutdown system - Low-flow reactor trip system - Additional neutron flux instrumentation and protection system - Upgrading strategy for the TITAN system - Low reactivity margin reactor trip system These activities are strongly supported.	Ignalina Report
IGN013	The intention to develop and implement an independent and diverse additional fast shutdown system for all RBMK reactors should be realized with high priority. However, a comprehensive R&D programme should be prepared by the designer on the long term developments considered in the area of core design, e.g. burnable poison cycle, new control rod design, additional shut down system. The programme should define goals and objectives of the developments, describe the concept, and present a working programme including all the necessary analyses and all the out-of-pile and in-pile test programmes required in support of such developments. An overall time schedule should be prepared identifying all the relevant milestones from initiation of the development work until final implementation in the NPP. All R&D efforts should be kept separately from the ongoing short-term safety improvement.	Ignalina Report
IGN053	The development of a second emergency shutdown system was one of the most important modifications considered. It is recognized that there are many difficult issues relating to the installation of an additional shutdown system, especially the physics and installation problems associated with such a system. The possible solutions should be considered quickly and viable solutions identified for early implementation.	Ignalina Report
RB093	Inclusion of shortened control rods to the shutdown system inserted up from the bottom of the core.	TCM-92
RB185	The effectiveness of the fast acting shutdown system should be increased to enhance the safety margins for reactivity induced accidents. An optimization study is necessary.	TECDOC-694
RB254	All bottom rods are now inserted when a scram occurs.	TECDOC-694
RB292	A joint review of western experts and RBMK specialists should be undertaken to evaluate the adequacy of the existing shutdown system.	TECDOC-722

ID	Description	Source
RB610	The Russian designers, regulators and OECD countries experts agreed that the systems SDS-1 and SDS-2 cannot be considered as fully independent and diverse shutdown systems. Even though the rod designs and rod cooling are different, there are no displacers on the fast rods, and the speed of insertion is different. However, Russian designers and regulators consider that the drawbacks identified by the meeting participants will not reduce the level of Smolensk 3 safety which has been reported by Russian experts in the Smolensk TOB. The intent of the Russian designers to develop and modernize the RBMK CPS in order to provide a higher safety level is strongly supported by the experts from OECD countries. They have further indicated that the design shall be in strict accordance with the recommendations of IAEA Safety Standards, specifically Safety Series No. 50-C-D (Rev. 1) "Code on the Safety of Nuclear Power Plants: Design", Sections 407 to 414.	RBMK-SC-013

ID	Description	Source
1R3.2.8-3	Any additional scram system should be able to reach a cold, unpoisoned, subcritical state in the case of the maximum reactivity insertion due to CPS cooling system voidage.	Consortium
3R4.2.2-8	Feasibility study of a second scram system based on other physical principle as homogeneous absorber shutdown system should be evaluated. This system should be able to keep the reactor in long term subcritical conditions.	Consortium
3R4.2.3-7	Particular attention has to be paid for the long term subcriticality of the first generation reactors without reconstruction (like CHERNOBYL 1 and 2). Implementation of additional control rods as done or scheduled for Leningrad 1 and 2 and Kursk-1 and 2; should be decided for all first generation units.	Consortium
RB095	Decrease of subcriticality of the reactor due to implementation of measures to decrease void reactivity.	TCM-92
RB181	Undertake independent calculations to assess the margins of RBMK reactor using typical western safety analysis assumptions and to carry out comparative analysis of RBMKs and relevant western plants, based on results of independent calculations.	TECDOC-694
RB233	The number of bottom entry rods has not been increased to 32 on first generation units, because it gave rise to problems in maintaining the reactor subcritical during/upon shutdown.	TECDOC-694
RB291	Studies should be undertaken to clarify whether an additional shutdown system capable of providing short and long term subcriticality to comply with western safety standards and licensing practice is required.	TECDOC-722

ID	Description	Source
1R2.1-3	The fire standards, fire barriers and physical separation between power cables and between monitoring cables of the main and emergency feedwater systems should be reassessed, taking into account the lessons of the Chernobyl 2 accident of 1991.	Consortium
4R6.1-1	In accordance with the IAEA Guidelines SGD2, fire safety improvement should be a continuous effort. The ideas/suggestions given in this analysis can serve as a background for the improvements or at least be a background for further discussions.	Consortium
4R6.1-11	Mitigation of fire effects: where there are fire compartments (structural units) today, improvement must be made by replacements of doors with a better fire resistance, and improvements of the ventilation system should be made, e.g., by installation of *fire dampers; in the fire cells, improvements must be made in order to create a* better separation between redundant safety components by introducing fire stops and extinguishing systems for example in pump rooms and electrical relay-rooms; the turbine hall ought to have smoke removal system; heavily densified cable areas ought to have separate fire ventilation systems.	Consortium
4R6.1-2	Fire confinement for passive fire protection needs improvement: check the classification, installation and closing devices of all fire doors as well as the fitting of the frames; repair cracking penetration sealing; separate the instrument cabinets of the three independent safety trains in the control room from each other in a safe way and provide them with metal back panels; in the opinion of the Western experts, a wall should be built to protect the turbine hall against the fire of a transformer; use fire retardant cables only; analyse sufficiency of fire dampers in venting ducts; replace plastic sheet floor coatings by a suitable epoxy material; protect the steel frame of the turbine hall against fire (insulation, sprinklers etc plus smoke ventilation); protect the feedwater pump area against the effects of fire; verify the quality of the fire resistant paste used to coat the cabling.	Consortium
4R6.1-7	Fire loads: the floor covering ought to be changed into fire retardant material; all structures indoors (concerning reactor safety) ought to be of non-combustible or difficult to combust materials; the turbine building has high fire loads from oil located close to safety systems, therefore the separation ought to be improved; in the diesel compartments the oil tanks should be better separated from the diesel engine in order to lessen the consequences of a fire; a better *housekeeping* is recommended throughout the unit.	Consortium
4R6.1-8	Fire risks: it must be verified whether the coatings - Polystop K and OPK - meets the requirements (IEC332-/6/); further research is recommended in this area; automatic cut-off-valves should be installed; in case of a hydrogen leakage, possibilities of emergency venting of hydrogen directly outside ought to be available; in case of an external oil leak in the turbine building, oil-drain-paths ought to be provided to carry away and collect the oil in a safe place.	Consortium
6R3-8	The production of more comprehensive data on fires at RBMK NPPs is recommended so that the plant areas at greatest risk and which need most attention can be clearly recognised.	Consortium
RB113	Fire separation between turbine hall and intermediate building.	TCM-92
RB115	Ventillation of unit control room and other electrical rooms.	TCM-92
RB119	Establishing additional insulation or structural fire barriers between redundant safety important items.	TCM-92

ID	Description	Source
RB223	Protection of steel columns in turbine hall by insulation is under deliberation. Replacement of glass wall by insulated cassettes seem inevitable.	TCM-92
RB361	Install non-combustible partitions between individual cabinets. Also install metal doors on the back of all cabinets.	TECDOC-722
RB362	Install a reinforced masonry fire wall between the main transformers and the turbine building to provide a minimum 15 m line-of-sight distance between all parts of the transformer and the glass window in the turbine building, or provide automatic water wall sprinkler protection on the exterior of the turbine building wall exposed to the transformer.	TECDOC-722
RB363	The EDG should serve as a model for the rest of the plant with respect to general housekeeping, and as the housekeeping relates to fire prevention and fire protection.	TECDOC-722
RB364	The criteria to evaluate fire retardant cable and fire retardant coating material should be reviewed.	TECDOC-722
RB365	The Smolensk NPP should commit to use only fire retardant cables for all new cable installed in the plant, whether control or power.	TECDOC-722
RB368	Replace all of the plastic sheet floor surfacing with a suitable epoxy material that has the necessary characteristics of impervious surface to aid decontamination, wearability, adhesion to the poor concrete surface, and adhesion to the steel floor decking.	TECDOC-722
RB376	Install standard size fire doors with closures and latches to replace existing sub-standard fire doors.	TECDOC-722
RB377	A suitable material and associated repair techniques should be developed and provided to repair all penetration seals throughout the plant as required.	TECDOC-722

ID	Description	Source
4R6.1-3	Fire detection and alarm systems need the addition of new, reliable, maintenance-free detectors.	Consortium
4R6.1-9	Fire detection and alarm system: areas which are recommended to be equipped with hydrogen detectors are the ALS (Accident Localization System) and the turbine-off-gas system; safety related rooms, which are not covered by any detection systems today, ought to have a detection system in order to get an early warning of fire.	Consortium
RB360	Upgrade the existing detection system with new improved POC detectors.	TECDOC-722

ID	Description	Source
4R6.1-10	Fire water supply and fire hydrant system: hydrant equipment ought to be changed into more modern and efficient equipment, particularly the hoses, the hydrant equipment should also, preferably, be stored in a separate cabinet in order to protect the equipment.	Consortium
4R6.1-4	Fixed-fire extinguishing systems require the installation of combined straight stream/fog nozzles to replace the fire hose and water cannon installations. Several improvements are needed to the water-based fixed-fire extinguishing systems: add a water-spray fire-suppression system to the turbine-generator bearings; (in the opinion of the Western experts, the volume of the sprinkler system in the diesel generator buildings has to be increased and the actuation made automatic, even if the current situation fulfills the Russian regulations); provide the lower elevations of the turbine building with an automatic water sprinkler system.	Consortium
4R6.1-6	Manual fire fighting capability: add, repair and maintain normal and emergency lighting in the plant; provide the fire brigade with adequate Protective equipment; analyse the need to increase the number and quality of portable extinguishers in vital places at the plant; install a hose cabin near the stairs on the roof of the turbine hall.	Consortium
RB120	Supply of portable fire fighting equipment corresponding to the western density requirements shoul be provided, as well as the relevant maintenance and filling procedures to the local user.	TCM-92
RB367	Provide pre-connected hose and nozzle for the dry standpipe in the outside stair tower from the TB floor to the roof.	TECDOC-722
RB373	Normal lighting should be kept on in all areas of Unit 3 where plant personnel are normally working or can be expected to be present on a regular basis.	TECDOC-722
RB374	Also, all emergency lighting fixtures should be equipped with bulbs. The emergency lighting should be tested monthly to assure proper operation. Records of these tests should be maintained for at least 18 months.	TECDOC-722
RB378	Provide combination straight stream/fog nozzles for all fire hose and water cannon installations as required.	TECDOC-722
RB496	An oil-free breathing air compressor should also be provided to refill exhausted air bottles of the self-contained breathing apparatus.	TECDOC-722
RB497	Provide new, personal, protective equipment for the Smolensk fire brigade. The new equipment should include: - Turnout coat; - Trouser; - Boots; - Gloves; - Helmets with face shields; - Self-contained breathing apparatus (SCBA).	TECDOC-722

ID	Description	Source
4R6.1-4	Fixed-fire extinguishing systems require the installation of combined straight stream/fog nozzles to replace the fire hose and water cannon installations. Several improvements are needed to the water-based fixed-fire extinguishing systems: add a water-spray fire-suppression system to the turbine-generator bearings; (in the opinion of the Western experts, the volume of the sprinkler system in the diesel generator buildings has to be increased and the actuation made automatic, even if the current situation fulfills the Russian regulations); provide the lower elevations of the turbine building with an automatic water sprinkler system.	Consortium
RB117	Upgrading of automatic fire extinquishing systems.	TCM-92
RB366	Install automatic sprinklers and draft curtains on the underside of the TB roof to protect the roof trusses.	TECDOC-722
RB369	Install automatic sprinklers and draft curtains beneath the roof deck of the entire to protect the roof trusses. Install suitable roof vents and/or fans to aid in removal of smoke and heat from any fire in the TB.	TECDOC-722
RB371	Fixed automatic water spray fire suppression should be provided for the main turbine generator bearings.	TECDOC-722

ID	Description	Source
4R6.1-10	Fire water supply and fire hydrant system: hydrant equipment ought to be changed into more modern and efficient equipment, particularly the hoses, the hydrant equipment should also, preferably, be stored in a separate cabinet in order to protect the equipment.	Consortium
4R6.1-5	Fire water supply installation: change fire pump starting to take place automatically due to pressure drop.	Consortium
RB372	Revise pump starting sequences to provide for automatic start on pressure drop and manual stop only of all fire pumps.	TECDOC-722

ID	Description	Source
1R2.4-1	When the updating or replacement of the operational process control system is performed, the following state-of-the-art systems should be considered: -four signals of each parameter available, -signal rejection according to specified criteria, -smoothing of the transition after rejection of any signal, -redundant self-checking processors.	Consortium
1R3.3.4-1	The redundancy of the signals generating AZ-l or AZ-4 by flow reduction should be checked.	Consortium
2R15-1	The architecture of the RBMK control and protection system should be reviewed with the specific objective of eliminating the common use of elements of the systems. In those cases where this cannot be achieved in a cost-effective manner, the number of interconnections should be minimised and the effective isolation of the systems should be ensured.	Consortium
2R15-3	The control system based on in-core sensors, the protection system responding to the fast shutdown initiators, and the system invoking slow shutdown should be physically and electrically segregated.	Consortium
2R15-5	The insertion of all rods into the core is currently achieved by actuation of all servodrives through extensive logic that is common to all systems. A study should be completed that will result in the modification of the drop mechanism associated with either the fast or slow protection systems such that they are actuated in a diverse manner. It is suggested that the means of permitting the fast rods to be inserted by cutting the power supply to the servodrive circuits be considered.	Consortium
2R15-6	Consolidate the number of regulating flux detectors (Ignalina scope) by using the same detector for different control and monitoring (but not protection) functions. This will free up enough Hf02 detectors to permit a cost-effective design for two independent safety-qualified shutdown systems.	Consortium
IGN054	The proposed changes to the control rod drive mechanisms are recommended. These changes successfully separate the control function from the safety protection function. It is agreed that the current performance of the rod drive mechanisms has been good, and that there are other items that would provide greater improvements in safety. Even though this modification is considered to be a lower priority item, it should be factored into the overall upgrade plan of the Ignalina NPP. It is also recommended that efforts continue to finalize this design and then complete a programme of testing and proving the design and construction.	Ignalina Report
IGN055	The proposed modification for the drive mechanism for the bottom entry rods appears to be a good addition, however it is considered to be a lower priority due to its relatively small safety impact.	Ignalina Report
IGN058	Introduction of the system for low-flow protection would be beneficial but an optimization exercise is needed to ensure that the protection signal will occur sufficiently early to provide effective defence against the faults. The modification takes greater significance in the context of V.7.3.	Ignalina Report
IGN059	The introduction of automatic low ORM trip is strongly supported, as it is of essential importance in ensuring plant shutdown ability. It must, however, be recognized that ORM is a derived parameter not a control variable, but, despite this, the introduction of an automatic system is beneficial.	Ignalina Report

ID	Description	Source
IGN060	This trip, based on the change rate of steam drum pressure, is planned to be installed at Ignalina. The introduction of this trip is seen as being very important as it provides early detection of a primary pressure boundary failure including those parts of the circuit outside the ALS.	Ignalina Report
IGN061	Failure detection into the cavity on vent temperature rise. This is clearly a diverse additional line of defence allowing for plant shutdown and ECCS actuation in the event of a failure into the reactor cavity and should be considered for inclusion into the protection system. Since INPP already has two sets of diverse pressure sensors for monitoring reactor cavity pressure, this proposal was considered to be low in priority. The optimization of the system will be required to sequence it with the two lines of pressure protection.	Ignalina Report
IGN062	Seismic trip. It is suggested that these measures be carefully reviewed before a significant investment is made as it would appear that there are far more pressing issues needing attention.	Ignalina Report
RB174	Minimize the high reliance of safety functions on operator action. An increase in the scope of automatic functions may be warranted. Introduce additional instrumentation and control devices, where such additions can be shown to improve safety.	TECDOC-694
RB177	Provide additional information on the control and protection strategy adopted for RBMK plants. In particular to identify the design requirements of the safety systems and clearly link them to the safety case.	TECDOC-694
RB347	Most standards strongly recommend that the protection system should be separated from the power regulation and indication systems. The current arrangements allow most systems to provide both protection and power regulation functions. While it is the operating practice to use different systems for the two functions, it is recommended that consideration be given to modifying the systems of the RBMK to conform to the advice given in these standards.	TECDOC-722
RB351	The recommendation on the separation of the control and protection functions is repeated. It is advised that one criteria to be used in the considerations is the establishment of if the segregation of the protection function, BAZ and AZ1, the power reduction trips AZ3 and AZ4 and the power regulation functions can be improved.	TECDOC-722
RB359	The physical segregation of the equipment is quite variable. In some places the segregation is excellent, in others it leaves a lot to be desired. For example the BAZ and ECCS actuation logic is split into two trains of three units located in six different rooms. The essential neutronic inputs and control rod position control systems lack this in the independent protection channels which often share the same location even the same cubicle.	TECDOC-722
RB553	To review and upgrade the electrical protection systems in terms of selectivity and co-ordination (or discrimination)	ASSET RBMK

ID	Description	Source
RB312	The reliability of the ECCS actuating signals "high pressure in the compartments" should be demonstrated to be high in order to justify the lack of diversification.	TECDOC-722
RB347	Most standards strongly recommend that the protection system should be separated from the power regulation and indication systems. The current arrangements allow most systems to provide both protection and power regulation functions. While it is the operating practice to use different systems for the two functions, it is recommended that consideration be given to modifying the systems of the RBMK to conform to the advice given in these standards.	TECDOC-722
RB359	The physical segregation of the equipment is quite variable. In some places the segregation is excellent, in others it leaves a lot to be desired. For example the BAZ and ECCS actuation logic is split into two trains of three units located in six different rooms. The essential neutronic inputs and control rod position control systems lack this in the independent protection channels which often share the same location even the same cubicle.	TECDOC-722
RB593	Check valves in the DGHs should only be installed in first generation plants in conjunction with flow bypass lines from the pump discharge header to the ECCS lines.	RBMK-RD-005
RB594	Experimental investigations including the validation of the RELAP5 predictions of bypass flows developed after a hypothetical complete blockage of a DGH flow should be performed.	RBMK-RD-005
RB595	As per the Smolensk report, an automatic reactor scram on low flow in multiple channels connected to the same DGH should be implemented.	RBMK-RD-005

ID	Description	Source
2R15-14	The plotting of detector discriminator curves should be carried out on a regular basis to ensure the correct calibration, and indeed operation of the device.	Consortium
RB348	The scope of testing of the equipment during normal operation and during the quarterly inspections should be established.	TECDOC-722
RB349	The results of the maintenance inspections and of the operational data on the occurrences of equipment failure and the spurious generation of trip signals leading to actuation of the trip system should be examined to determine and substantiate claims for equipment reliability.	TECDOC-722
RB353	The maintenance procedures as well as maintenance and test records should be audited to ensure their scope is satisfactory and they are being correctly implemented.	TECDOC-722
RB391	Consideration should be given to rationalizing station instructions in the development of the quality assurance arrangements (see VIII.1.3), in order to reduce the volume and put greater emphasis on instructions relating to nuclear safety.	TECDOC-722
RB421	Plant test performance results should be recorded and criteria specified to show that an acceptable standard has been achieved.	TECDOC-722
RB431	All procedures for the surveillance testing programme should be in the same step-by-step format. Only the controlled copy of the procedures should be used in the control room with the list of changes attached.	TECDOC-722
RB432	Performing of testing should be properly controlled and recorded. All tests should be recorded even if they have failed and are later successfully repeated. The reasons for failures should always be analysed and corrective actions taken.	TECDOC-722
RB434	A single procedure to control the hardware modifications process from initiation to the implementation should be produced. Such a procedure should include definition of responsibilities and specification of the requirements for the training of personnel who have responsibilities for operating, testing and maintaining equipment affected by the modification.	TECDOC-722

ID	Description	Source
RB203	Reviews in the form of specialist meetings should be held on reliability analysis.	TECDOC-694
RB227	The shutdown algorithms require further study by western experts and the links to the accident transients and to the plant safety case must still be made. The approach to proof and periodic testing as well as the role of such testing when making the high claims for shutdown system reliability should be discussed.	TECDOC-694
RB250	After reading the ASSET report for the Chernobyl nuclear power plant in order to review the root causes of a safety significant accident that occurred on 11 October 1991 at Unit 2, experts became concerned about the following two aspects: - the high number of cable problems reported; - Generator no.3 was reconnected to the power line 30 minutes after the turbine inlet valves were closed due to the failure of the main breaker controls. There is no automatic feature to open the disconnectors when the generator is tripped.	TECDOC-694
RB349	The results of the maintenance inspections and of the operational data on the occurrences of equipment failure and the spurious generation of trip signals leading to actuation of the trip system should be examined to determine and substantiate claims for equipment reliability.	TECDOC-722
RB350	It would be very desirable to make an examination including a failure modes and effect analysis of a sample of the circuit details and to look to see whether there are any faults inherent in the design or different practices in use or the possibility of unrevealed, dangerous failures occurring.	TECDOC-722
RB352	The reliability of the control rod electromechanical braking system should be reviewed along with an evaluation of the consequences of the switches failing to function properly.	TECDOC-722
RB427	Consideration should be given to developing methods to better utilize equipment history information by allowing easy retrieval, root cause analysis and trending of equipment failures. This process can readily provide identification of recurring failures allowing earlier correction of causes, and be particularly useful when failures occur after several years of service. Although this process can be accomplished manually, personal computers (if available) can also be effective in this application.	TECDOC-722
RB495	The reliability of the diode buffering needs to be checked as its failure could cause a significant common mode failure.	TECDOC-722
RB609	Instrumentation and control problems were indentified in all RBMK ASSET reports. The range of the problems included poor equipment, the mismatch between equipment installed and the duty environment, maintenance errors and on occasion, untraced faults (or fleeting).	ASSET RBMK

ID	Description	Source
1R2.4-1	When the updating or replacement of the operational process control system is performed, the following state-of-the-art systems should be considered: -four signals of each parameter available, -signal rejection according to specified criteria, -smoothing of the transition after rejection of any signal, -redundant self-checking processors.	Consortium
2R15-15	The role of recording should be increased. The current means of storing and processing the total system data output is too slow, but is acceptable.	Consortium
2R15-7	The lower levels of the SKALA system is used for analogue and digital signal Priority A: collection and retransmission should be rationalised and the conventional data collection technology basic units should be upgraded.	Consortium
2R15-8	The use of a distributed network should be considered for data collection and processing. The nature of the RBMK reactor results in a lot of plant data. The complex nature of the plant and the resulting complex nature of model calculations suggests a distributed system with one or two technical workstations and a larger number of graphic display systems may be the appropriate way forward.	Consortium
3R4.2.1-3	Better information about burnup distribution is needed for the validation of codes and for redistribution and local effect evaluations. The reconstruction flux based on measurements could be used. From these considerations it will be expedient to upgrade the SKALA system.	Consortium
3R4.2.3-10	SKALA output must be upgraded in order to provide readable information (firstly the printer).	Consortium
RB079	Long period (10-15 minutes) of cycling through all measurements and for/of calculating the Operational Reactor Reactivity Margin.	TCM-92
RB096	The "Skala" centralized monitoring system is replaced at its end-of-life by a basically new system which provides increased scale of monitoring and recording of operating conditions including emergency ones.	TCM-92
RB108	Increase in the reactor control and protection system (CPS) efficiency. Replacement of the existing "Skala" system with a new system "Skala-M"	TCM-92
RB224	The SKALA and TITAN computer systems appear to be overloaded and currently have a slow cycle time.	TECDOC-694
RB225	The efforts and actions to accelerate the rate of data processing and information updating are supported by the experts.	TECDOC-694
RB234	A modernized version of the control and instrumentation system has been developed and should be installed starting in early 1993. This new system has increased diversity and segregation of functions and a more rapid cycling of the data acquisition system. Additional protective functions are contained in the new system, for example actuation of the FSS in response to signals from in-core detectors.	TECDOC-694
RB284	The ability to recreate the axial burnup distribution in the additional operator aid display is strongly supported.	TECDOC-722
RB285	Further efforts to reduce the time delay of detailed 3-D information on space dependent core parameters are strongly supported.	TECDOC-722

ID	Description	Source
RB286	The planned replacement of silver detectors by hafnium detectors should be encouraged in order to shorten the delay time of the radial power distribution restoration. Such optimization will be implemented by using 36 axial detector assemblies in the designed upgraded system at Smolensk 3.	TECDOC-722
RB287	The additional operator aid display in the NPP control room should be added to all units at Smolensk as soon as possible and the concept extended to other RBMK reactors.	TECDOC-722
RB289	Activities to automate the current manual control system using 3-D on-line power distribution analysis should be pursued.	TECDOC-722
RB357	There would be some benefit by updating the equipment to allow the ORM and the departure from nucleate boiling margins to be calculated more frequently. These are essential safety parameters which the operator needs in order to operate the reactor safely and provide manually invoked plant protection. The introduction of PC systems is a sound move in this direction, as is the development of a new microprocessor-based data system.	TECDOC-722
RB359	The physical segregation of the equipment is quite variable. In some places the segregation is excellent, in others it leaves a lot to be desired. For example the BAZ and ECCS actuation logic is split into two trains of three units located in six different rooms. The essential neutronic inputs and control rod position control systems lack this in the independent protection channels which often share the same location even the same cubicle.	TECDOC-722

ID	Description	Source
2R15-16	The system currently uses static logic. This is prone to unrevealed dangerous faults, the introduction of a form of dynamic logic is strongly recommended.	Consortium
2R15-6	Consolidate the number of regulating flux detectors (Ignalina scope) by using the same detector for different control and monitoring (but not protection) functions. This will free up enough Hf02 detectors to permit a cost-effective design for two independent safety-qualified shutdown systems.	Consortium
7R2.3.3-1	Increase the number and capability of PCs in the NPP in order that the programme for the provision of useful displays can continue. Using the expertise available and state-of-the-art equipment referenced above, a display that could be used by the Unit Shift Supervisor in conjunction with Symptom Based EOPs should be developed. There may be need for collaboration between NPP and Western staff in this area of work.	Consortium
IGN056	The need for protection against local power excursion should be confirmed. The choice of, location of, and set points of the system that must provide protection against both radial and axial power distortions should be optimized. The solution preferred by the INPP requires the replacement of the Ag detectors and gamma chambers and the adoption of a multi-section hafnium detector.	Ignalina Report
IGN063	It is suggested that an additional survey of electromagnetic interference be made at INPP to identify remedial measures that might still be required, e.g. cable and connector replacement and changes to the grounding arrangements.	Ignalina Report
IGN065	The introduction of electronic data bases and diagram packages that allow ready presentation of information should be encouraged. The generation of such a system that includes existing documents and drawings is both an expensive and long-term undertaking.	Ignalina Report
IGN071	There is concern about the gravity of the consequences of this fault. However, the necessary remedial action has been completed.	Ignalina Report
RB174	Minimize the high reliance of safety functions on operator action. An increase in the scope of automatic functions may be warranted. Introduce additional instrumentation and control devices, where such additions can be shown to improve safety.	TECDOC-694
RB176	The replacement of the ORM concept, which requires too much operator involvement, should be further explored.	TECDOC-694
RB179	The data processing computers should be upgraded and better alarms and displays installed.	TECDOC-694
RB221	Further investigation concerning qualification of electrical equipment under harsh environmental conditions should be performed.	TECDOC-694
RB226	When a violation of the ORM occurs, the operators are only required to scram by following an administrative procedure, since this has not been implemented into an automatic engineered safety feature.	TECDOC-694

ID	Description	Source
RB234	A modernized version of the control and instrumentation system has been developed and should be installed starting in early 1993. This new system has increased diversity and segregation of functions and a more rapid cycling of the data acquisition system. Additional protective functions are contained in the new system, for example actuation of the FSS in response to signals from in-core detectors.	TECDOC-694
RB256	The ORM is now calculated every 5 minutes instead of every 15 minutes.	TECDOC-694
RB286	The planned replacement of silver detectors by hafnium detectors should be encouraged in order to shorten the delay time of the radial power distribution restoration. Such optimization will be implemented by using 36 axial detector assemblies in the designed upgraded system at Smolensk 3.	TECDOC-722
RB290	A study should be undertaken to determine whether the prototype automatic control system developed several years ago can be upgraded and implemented on all RBMK reactors.	TECDOC-722
RB293	The process to reduce ORM is a relatively slow process when operating under steady state conditions. However during transient conditions the value can change reasonably quickly. In order to relieve the operator of the responsibility of tripping the reactor when the value falls below 30 rods, automation of the shutdown action should be implemented at the earliest possible time.	TECDOC-722
RB294	Consideration of the elimination or at least the reduction of the safety role of the ORM was already recommended in TECDOC-694. The investigation of the effectiveness of additional absorbers to reach this goal in the near future or short-term is highly recommended.	TECDOC-722

ID	Description	Source
IGN068	It was agreed that INPP needs to develop a plan for incorporating all the various additions and modifications to the control room.	Ignalina Report
IGN069	The proposed modification to improve the interface between the control room and the refueling machine seems to have some merit and should be investigated further.	Ignalina Report
RB174	Minimize the high reliance of safety functions on operator action. An increase in the scope of automatic functions may be warranted. Introduce additional instrumentation and control devices, where such additions can be shown to improve safety.	TECDOC-694
RB179	The data processing computers should be upgraded and better alarms and displays installed.	TECDOC-694
RB224	The SKALA and TITAN computer systems appear to be overloaded and currently have a slow cycle time.	TECDOC-694
RB225	The efforts and actions to accelerate the rate of data processing and information updating are supported by the experts.	TECDOC-694
RB287	The additional operator aid display in the NPP control room should be added to all units at Smolensk as soon as possible and the concept extended to other RBMK reactors.	TECDOC-722
RB357	There would be some benefit by updating the equipment to allow the ORM and the departure from nucleate boiling margins to be calculated more frequently. These are essential safety parameters which the operator needs in order to operate the reactor safely and provide manually invoked plant protection. The introduction of PC systems is a sound move in this direction, as is the development of a new microprocessor-based data system.	TECDOC-722
RB557	Upgrade human systems interface in the main control room and other appropriate parts of the plant.	ASSET RBMK

ID	Description	Source
6R3-2	The development of performance indicators to monitor safety system reliability, availability and performance is recommended.	Consortium
6R3-9	RBMK NPPs and utilities should collate and publish fuller performance data.	Consortium
6R4-12	NPP Management and on-site GAN Inspectors should ensure that post trip evaluations always: - identify and eliminate the causes of the trip before restarting; - identify and eliminate discrepancies between expected and actual plant response during the trip and subsequent post trip operations and - that event reports indicate a statement by the operator of how he/she determined it was safe to restart the reactor.	Consortium
6R6-2	The operational arrangements for controlling the immediate response to abnormal operational conditions and events should be reviewed and the post-event procedures should be revised to give better guidance to the operator.	Consortium
6R7-4	The policy of reinforcement of the key role of the Control Room shift staff as the guardians of the safe operation of the reactor should be emphasised	Consortium
6R7-5	Establish Nuclear Safety Review Committees: - to review operational safety issues and advise plant management accordingly; and - to review modifications (changes to systems, documents, policies, etc) at all stations for sensibility and possible application across the board in the stations. This would be done on a face-to face-basis at the proposal stage, when the proposal is being submitted for formal approval to help those persons who have to approve modifications. It would also be done after implementation of modifications for an effectiveness review. This final review is also to ensure that all auxiliary activities are complete: documentation, retraining of staff, procedures are in place, and so on.	Consortium
6R7-6	The principle Business Processes should be reviewed. A program to conduct an expert assessment of the management system and business processes across the RBMK line of business in order to identify the areas which need improvement is the first necessary step. Some of the areas are evident in the comments throughout Section 7.4. Improvements can be made in the short term without waiting for such an assessment. However, a concerted programme is needed for long term correction.	Consortium
7R2.1.1-1	RBMK Management should familiarise itself with alternative management systems so that, if and when there is a necessity for change, the best of a number of alternatives can be adapted to the specific needs of the particular plant.	Consortium
7R2.1.2-1	A policy of making personnel accountable and responsible for their work, particularly in the area of operational documentation, should be introduced so that the level of responsibility is forced down to the level where the actual work is done.	Consortium
7R2.2.4-1	Whatever staff structure is adopted in the long term, identified staff groups should remain responsible for the health of not just the plant for which they are responsible, but also for its environment with respect to the deficiencies referenced in items 2.1 to 2.4. Not only should this responsibility be written into job descriptions, a proportion of bonus payment should depend on the continued improvement followed by maintenance of the plant and personnel environment.	Consortium

ID	Description	Source
8R6.3-2	A correct distribution of responsibility in relation to safety is a cornerstone of a sound regulatory regime. There must be formal responsibilities correctly defined but also an implementation praxis is needed in defining the role of each party involved and encouraging them to contribute to safety in recognition and balance with the role of other parties. This area should be closely monitored in the further development of the Russian regulatory regime.	Consortium
RB390	Consideration should be given, during the current development of quality assurance arrangements (VIII.1.3), to the advantages of simplifying the organization, in order to reduce the number of communication linkages and so avoid possible misunderstandings which could lead to errors.	TECDOC-722
RB394	Consideration should be given to the creation of a group to take a role, which is completely independent of station output responsibilities, in monitoring nuclear safety performance, recommending good safety practice and promoting a general awareness of nuclear safety throughout the site.	TECDOC-722
RB396	The station should implement a policy of drug testing for staff in safety sensitive occupations.	TECDOC-722
RB406	The philosophy that expects a senior person to manage up to 17 sections should be examined and changes made to the organizational structure below the Plant Shift Supervisor as necessary.	TECDOC-722
RB425	In developing specific contracts down to the department manager/shop supervisor level consideration should be given to ensure that each manager has a clear understanding of his/her specific responsibilities to minimize the duplication of responsibilities and to improve organizational effectiveness.	TECDOC-722
RB433	Consideration should be given to the benefits of centralizing the control of the in-service inspection programme so that it can be co-ordinated by a single group in the power plant. Such a measure would assist in ensuring that the in-service programme was completed to schedule.	TECDOC-722
RB499	Implement adequate criteria and related procedures for decision making for cases with primary circuit leakage.	ASSET RBMK
RB514	Consider a job rotation programme.	ASSET RBMK
RB517	Check key staff to prevent reliability degradation.	ASSET RBMK
RB521	Plant management should consider enhancing the routine assessment of the competence of personnel engaged on safety significant work. Reading procedures and attending training courses does not in itself establish competence and staff proficiency. Regular assessments to determine competency should be improved for both operations and maintenance staff. The full scope operation simulator will provide one means but for maintenance staff more considerations should be given to establishing "mock-ups" for personnel qualification.	ASSET RBMK
RB531	Strengthen the necessary staff resources for independent analysis of reported events, using root cause and trend analysis and analyse the effectiveness of the feedback process.	ASSET RBMK
RB536	Develop better planning procedures to ensure that people performing important activities are not overloaded.	ASSET RBMK

ID	Description	Source
RB537	Incorporate fuel movements into the senior shift engineer's responsibilities.	ASSET RBMK
RB538	Perform job task analysis for key personnel.	ASSET RBMK
RB539	Management should give consideration to the rotation within the shift staff and occasional interchange of Day staff and Shift staff.	ASSET RBMK
RB541	Improve coordination and communication throughout the plant.	ASSET RBMK
RB546	Reevaluate, and modify as needed, the shift organization and responsibilities best suited to cope with the immediate recovery actions of incidents and accidents. The intent of this recommendation is to improve the communications and recovery actions and to reduce the number of personnel involved in the decision process of the immediate recovery actions.	ASSET RBMK
RB552	When investigation teams are established for an event (a Commission) the Station Manager is recommended to ensure they have clear terms of reference which are not excessively constraining them by time or scope, to ensure a comprehensive analysis is achieved.	ASSET RBMK
RB567	Enhance the routine assessment of the competence of personnel engaged on safety significant work.	ASSET RBMK
RB584	Continue to develop the role of new "Human Factor and Personnel Training Department" by reinforcing their role to assist in the investigation of all safety related events across all Departments.	ASSET RBMK

ID	Description	Source
1R2.2-4	The quality control procedures in main steam line safety valve manufacturing should be reinforced (see also recommendation lR 2.3-2).	Consortium
1R2.3-2	The quality control procedures in manufacture of current or any replacement main steam line safety valves should be reinforced (see also recommendation lR 2.2-4).	Consortium
2R15-11	The Quality Assurance arrangements, particularly those associated with equipment repair, calibration and return to service and modification, must be improved, particularly if effective operational feedback is to be obtained.	Consortium
2R15-17	Station and design office QA should be improved to ensure that all equipment modifications are traceable and fully reviewed.	Consortium
7R2.3.1-1	Emergency procedures should always be available at the place where they may be required for use.	Consortium
7R2.4.3.4-2	Recording of surveillance results should be in special journals with individual test reading against normal and limiting parameters.	Consortium
7R2.4.3.5-1	Alarm procedures should be produced for all RBMK plants. Operators should not be required to remember all alarms.	Consortium
RB391	Consideration should be given to rationalizing station instructions in the development of the quality assurance arrangements (see VIII.1.3), in order to reduce the volume and put greater emphasis on instructions relating to nuclear safety.	TECDOC-722
RB397	The station is encouraged to complete the implementation of a QA programme at the earliest possible date, which will include the requirement for independent audit.	TECDOC-722
RB445	Ensure that during modernization and reconstruction programmes, only equipment of the highest reliability is used.	ASSET RBMK
RB446	Improve quality and usability of maintenance, test, and quality control procedures.	ASSET RBMK
RB448	Consider use of check lists for maintenance work.	ASSET RBMK
RB508	Develop and implement a more effective document review and improvement process which includes quality assurance practices. The following aspects would be involved in implementing this recommendation: - review the scope and style of documentation produced by utilities in other countries; - select criteria appropriate for LNPP documentation ; - continue the development of quality assurance aspects in revised plant documentation; - train selected and experienced staff to produce documentation of these criteria; - allocate priorities for review and improvement of documentation based on the safety significance of the procedures; - programme the updating of documentation based on these priorities; - conduct regular management reviews on the effectiveness of the documentation improvement programme; - include feedback from the plant users of the procedures in these reviews.	ASSET RBMK
RB526	For new equipment, ensure specification covers all possible operating conditions, including testing.	ASSET RBMK

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ID	Description	Source
RB527	Strengthen the Quality Assurance organization.	ASSET RBMK
RB559	To include quality assurance practices in procedures (by including check lists, acceptance/rejection criteria, drawings, hold points, allowable repair times for safety systems, etc.)	ASSET RBMK
RB560	To use quality assurance practices in the process of reviewing and updating procedures (specification of review criteria, training staff how to produce good documentation, etc.)	ASSET RBMK
RB579	Management should continue to improve their monitoring of the effectiveness of the documentation improvement programme.	ASSET RBMK

ID	Description	Source
6R5-1	The ASSET Mission process should be supported, being one internationally accepted method of encouraging safety culture enhancement.	
6R5-2	Limitations in the ASSET root cause analysis methodology should be should be recognised and other techniques should be reviewed particularly in the classification of root causes.	Consortium
6R5-5	Efforts to reduce the incidence of significant events should be directed at improving human performance and in reducing deficiencies in turbine, electrical and control and protection systems.	Consortium
6R7-1	All levels of management should encourage and develop a more open, responsible and self-critical attitude in all staff with responsibilities relevant to nuclear safety. This applies to the designers, scientific institutes, utility organisations and regulators as much as to the NPP staff.	Consortium
6R7-2	All members of the staff should be issued a copy of a document stating the Utility's safety policy within the framework of the Quality Program implementation.	Consortium
6R7-3	Develop an orientation course based on the document: INSAG-4, to be a part of every training session, and in particular have managers conduct it. The control room operators should be the first recipients of the course which should include a review of the safety case and operating policies based on it.	Consortium
6R8-4	A methodology to quantify the impact on safety levels of each individual improvement measure proposed for implementations should be developed. This methodology should include assessment of the impact on the general safety level prior to the implementation of any measure, and should include verification of the effectiveness of any measure in improving safety.	Consortium
7R2.3.2-1	Implement a programme of upgrades to communication equipment to allow for virtually instantaneous communication between MCR and field operators. There are a number of ways that such a programme could be progressed for example individual radio transmitter/receivers for all key operational personnel. The most appropriate for specific applications should be identified and implemented.	Consortium
7R2.5.7-1	Increase the efficiency of using the information about incidents and failures of NPP equipment. The most serious cases should be studied and simulated with NPP operators. For this purpose it would be useful to hold annual seminars in the TC concerning incidents with representatives of the utility, NPP and regulatory body present.	Consortium
RB392	The station is encouraged to continue its work to ensure that the new culture of emphasizing safety over production is understood and implemented at all levels in the organization. The IAEA's ASCOT seminars could be used as part of the continuing process of training.	TECDOC-722
RB522	Encourage reporting of all failures or errors, and increase emphasis on systematic root cause analysis of failures and errors.	ASSET RBMK
RB528	Management should foster a questioning attitude among personnel to suggest improvements to design, procedures & training and to understand the plant design.	ASSET RBMK
RB532	Strengthen attitudes towards safety and work by further education and training.	ASSET RBMK

ID	Description	Source
RB535	Management should take greater account of human factors and incorporate this into their safety culture.	ASSET RBMK
RB545	Management should continue to work to improve Safety Culture. They should publish their policy on safety culture to all staff, develop a specific safety culture document and educate the station staff in safety culture issues with periodic refresher programmes.The aim should be to encourage a more questioning and self-critical attitude in staff to the responsibilities.	ASSET RBMK
RB564	To encourage a questioning and self-critical attitude in staff through education.	ASSET RBMK
RB570	To promote openness among the staff to ensure that all events and occurrences are fully and openly reported.	ASSET RBMK
RB571	To promote a 'self-questioning' attitude in staff at all levels.	ASSET RBMK

ID	Description	Source
6R5-4	A team should be formed to develop and implement the database. This development should be driven by the NPPs with the main aim of improving the experience feedback process.	Consortium
6R7-10	A description of the structure of the operating documentation needs to be prepared so that it can be reviewed by all concerned: the operators, the intended safety committees, and any future reviewers. The operators should be a vital part of the preparation of any operating document.	Consortium
6R7-11	The Task Force to improve key Operating Documentation (Operating Rules and Procedures) should be led by an experienced Plant Shift Supervisor and the views of experienced plant operators should be fully taken into account in developing better operating documentation.	Consortium
6R7-12	The documents need to be in a form such that the individual operator can use what is necessary to him right at the work place, whether it is the main panel or a system in the field.	Consortium
6R7-13	Authors should be adequately trained to write easily understood Operation Documents.	Consortium
7R2.4.1-1	Provide means that will allow the production and maintenance of high quality A technical and operational documentation of the NPPs.	Consortium
7R2.4.2-1	Provide appropriate devices for compressing information, safe storage, retrieval and preservation of plant documentation.	Consortium
IGN065	The introduction of electronic data bases and diagram packages that allow ready presentation of information should be encouraged. The generation of such a system that includes existing documents and drawings is both an expensive and long- term undertaking.	Ignalina Report
RB398	The station should consider providing one central document amendment service with subsequent issue of amended document, or pages, to each original recipient. The use of numbered copies would facilitate the control of amended documents. The use of documents in loose-leaf form would facilitate the issue of (fully typed) amendment pages.	TECDOC-722
RB399	The control of documents, including drawings, should be included in the QA arrangements currently being developed, which would also provide the opportunity to rationalize and simplify the standards used in the process.	TECDOC-722
RB400	The procedure for the future handling and control of documents (items 1 and 2 above) should include similar centralized procedures for amendment of the document review date.	TECDOC-722
RB401	Duplicate master documents should be made and held in a location remote from the station registry. The use of microphotography or microfiche techniques would provide an efficient method of document duplication readily permitting storage in a fire-proof safe.	TECDOC-722
RB444	The design basis for all of the safety and safety related systems should be available to the plant staff.	ASSET RBMK

ID	Description	Source
RB556	To ensure that design basis information is available to plant staff for safety related equipment.	ASSET RBMK

ID	Description	Source
6R3-1	The removal of construction debris from around the NPPs, the provision of adequate amenities for operation and maintenance staff, and the general raising of housekeeping standards, would help to improve plant safety culture and reduce fire risk.	Consortium
RB393	The station is encouraged, by the application of the quality assurance arrangements being developed (see VIII.1.3), to strive to improve the fitness for purpose of plant and equipment, in order to reduce the number of events caused by design deficiencies and poor quality components.	TECDOC-722
RB395	The station should improve the general housekeeping to demonstrate a commitment to good safety practices.	TECDOC-722
RB408	The emergency lighting systems along routes from the main control room to the reserve control room should be repaired to ensure that they will operate satisfactorily when required in earnest. This applies to all areas where emergency lighting has been installed including routes from the main control room of Unit 3 to its reserve control room.	TECDOC-722
RB410	The cabinet in the reserve control room which houses the bypass maintenance switches for the secondary reactor safety protection system should enclose the back of the panel.	TECDOC-722
RB420	The type of labelling used in a plant should be reviewed with the aim of improving the standard in order to make it more permanent.	TECDOC-722
RB429	Smolensk NPP senior management should take prompt action to develop a plan to identify and correct equipment and labelling problems, including those listed in VIII.4.6.	TECDOC-722
RB430	The root causes of the existing material deficiencies should be determined and actions initiated to prevent similar problems from arising in the future. For example, consideration could be given to a tagging system for identified deficiencies, to strengthening the management plant walkdown programme and to a periodic independent review and assessment of the plant material condition.	TECDOC-722
RB575	The establishment of a special group at each plant responsible for developing and maintaining expertise in the ASSET methodology and for applying that methodology at the plant for the prevention of accidents should be incorporated. These groups should use common procedures and receive similar training so that the methods are applied consistently across all plants.	ASSET RBMK

ID	Description	Source
6R7-14	Prepare a refresher training program for the necessary staff which informs them of the deficiencies that exist in systems and documents, and of what to do to compensate for the shortcomings.	Consortium
7R2.1.3-1	Organisational provision should be made that would allow for the implementation of continuing training of the frequency and duration recommended at 2.5.4.2-1.	Consortium
7R2.5.2.2-1	Recognise the key role for the Smolensk Training Centre in providing a B training advisory service to locations that will enable co-ordination of improvements and all NPPs benefitting from each others' best practices. Best practices not just for improving training techniques but also for the overall management of the training process. In the case of non-Russian and Leningrad NPP the aim should be to include them even though contractual issues would need to be addressed.	Consortium
7R2.5.2.3-1	Budgetary provision should be allocated to the training centre to allow it to recover the cost of, not only the annual implementation of an enhanced portfolio of course/seminars, but also to allow for the capability for it to become an example to trainees as to what can be done with regard to material condition and work ethic.	Consortium
7R2.5.3.2-1	For all training activity it is necessary to recruit appropriate instructors in a timely manner and in such a way that they will not become permanently detached from the NPP Shop or Operations.	Consortium
7R2.5.4.1-1	Introduce a graduated programme of GET for new employees, contractors and visitors of more than a days duration.	Consortium
7R2.5.4.1-2	Recognise the key role for the Smolensk Training Centre in promulgating Safety Culture and address this as topic in training programmes for an extended list of key personnel to be trained in the TC (to include management, radiological protection, technical support and maintenance personnel).	Consortium
7R2.5.4.1-3	Education and training which takes place at other Institutions should also address Safety Culture issues. This is particularly the case for the Institute of Nuclear Engineering at Obninsk which would give the opportunity to introduce appropriate concepts to undergraduates as well as certain specialists undertaking refresher training.	Consortium
7R2.5.4.1-4	Self-training at the work place should be increased with careful management to ensure the correct use of such training devices that may be installed.	Consortium
7R2.5.4.2-1	Provide continuing training to operational staff on an annual schedule to allow all shift staff to undergo the current requirements (i.e. the reason for the government order to extend shift staffing from six to seven shifts for key posts). However, the requirements should be increased progressively to meet the IAEA recommendation of between 60 and 80 hours per annum. In the first instance, simulator refresher training should be increased to at least one week per year whilst paying especial attention to the resource requirement to fund such an increase, in terms of simulator capability, numbers of instructors and simulator specialists. In increasing continuing training frequency and duration attention must be paid to the staff who have been working in the same post for a number of years, in order that such training is beneficial to them and does in fact provide refreshment.	Consortium

ID	Description	Source
7R2.5.5-1	The current simulator in the Smolensk Training Centre should be upgraded in terms of hardware at the earliest opportunity in order to improve operator perception of the simulator, the training centre and training as a whole. (Probably when the Second Smolensk Simulator becomes available).	Consortium
7R2.5.5-2	Provision of modern, state of the art systems in the form of PCs should be made urgently for many reasons. Their use to interface to and display parameters from the simulator is one such reason.	Consortium
7R2.5.5-3	The concept of a compact simulator running full scope models should be pursued. This option should not exclude consideration of the development of a full scope replica simulator for the RBMK 1500.	Consortium
7R2.5.6-1	Training programmes that make use of full scope simulators should provide for the conduct of operations as they would be conducted in the control room. This requires that the simulators are furnished with maintained operational documents from the appropriate NPPs	Consortium
7R2.5.6-2	In order that simulator training can be conducted in the most credible manner the simulator should be maintained as near to the replica as practicable to its reference unit.	Consortium
7R2.5.7-2	Develop and set regular feedback *TC-NPP-TC* to obtain information on the professional qualities of operators during this work. This form of training programme effectiveness evaluation is difficult to achieve without such scheduled feedback sessions.	Consortium
7R2.5.8-1	Increase the number and capability of PCs in the TC to conduct training sessions and develop new technical means of education.	Consortium
7R2.5.9-1	Provide the necessary quality of production and maintenance for the training materials requirement of a modern training centre and at the same time provide devices for compressing information, safe storing, and preservation of educational/training information.	Consortium
IGN067	It was agreed that a high fidelity full scale simulator replica is the best way to train and qualify operators, and, given the unique nature of the Ignalina NPP, such simulator is highly recommended.	Ignalina Report
RB402	Formal training, including refresher training, in nuclear operational concepts and, particularly in the appreciation of safety issues, should be arranged for all power plant technical and managerial staff as soon as possible, as an extension to the existing courses.	TECDOC-722
RB403	STC should arrange seminars in nuclear safety appreciation for all technical and managerial staff at the power plants, who are not covered by existing training arrangements.	TECDOC-722
RB404	The Training Centre is encouraged to continue to improve the standards of educational materials and methods, and funds should be available for this.	TECDOC-722
RB405	The amount and frequency of simulator retraining should be increased. It is recommended that at least 60 hours per year be given. The operators should visit the simulator at least three times per year to ensure that skills do not decline with time.	TECDOC-722

ID	Description	Source
RB513	Include in training for all supervisors the necessary skills to detect human reliability problems.	ASSET RBMK
RB515	Improve qualification and training of personnel in fuel handling activities.	ASSET RBMK
RB516	Improve the understanding of the full functional design details and criteria for safety related systems.	ASSET RBMK
RB517	Check key staff to prevent reliability degradation.	ASSET RBMK
RB518	Expand the training of operational staff for response to other sequences of priority, besides fires.	ASSET RBMK
RB519	Ensure that all of the operators and supervisors understand the overall logic of operations, including cases which are outside the individual specific line of responsibility. This is commonly called "Cross-Training", and the existing succession plan for them appears to take care of this concern.	ASSET RBMK
RB521	Plant management should consider enhancing the routine assessment of the competence of personnel engaged on safety significant work. Reading procedures and attending training courses does not in itself establish competence and staff proficiency. Regular assessments to determine competency should be improved for both operations and maintenance staff. The full scope operation simulator will provide one means but for maintenance staff more considerations should be given to establishing "mock-ups" for personnel qualification	ASSET RBMK
RB529	Proceed with the training of staff in ASSET Root Cause Analysis and implement the methodology on regular basis.	ASSET RBMK
RB532	Strengthen attitudes towards safety and work by further education and training.	ASSET RBMK
RB537	Incorporate fuel movements into the senior shift engineer's responsibilities.	ASSET RBMK
RB540	Broaden the training of operational personnel on simulators.	ASSET RBMK
RB565	Extend training to ensure that appropriate staff are fully aware of the design basis safety case, the expected response of the plant in normal and fault conditions, and the logic behind the limits that are set. Job rotation and cross-posting should be considered, in this respect.	ASSET RBMK
RB567	Enhance the routine assessment of the competence of personnel engaged on safety significant work.	ASSET RBMK
RB568	Increase the use of simulator based training.	ASSET RBMK
RB569	Include the skills necessary to detect human reliability problems in the training of supervisors.	ASSET RBMK
RB581	Provide more emphasis in the training for appropriate operating staff in Safety System functions related to design basis requirements.	ASSET RBMK
RB582	Train selected and experienced staff in a range of event analysis methods, e.g. human performance enhancement systems.	ASSET RBMK

ID	Description	Source
1R2.7-1	The written procedures on the realignment of the SWS to face with any failures should be developed for accident management action.	Consortium
6R7-10	A description of the structure of the operating documentation needs to be prepared so that it can be reviewed by all concerned: the operators, the intended safety committees, and any future reviewers. The operators should be a vital part of the preparation of any operating document.	Consortium
6R7-11	The Task Force to improve key Operating Documentation (Operating Rules and Procedures) should be led by an experienced Plant Shift Supervisor and the views of experienced plant operators should be fully taken into account in developing better operating documentation.	Consortium
6R7-7	The safety cases (TOB) and analyses of initiating events and consequences are needed to provide the base for the Technological Procedures (Rules), in order to improve the operating Rules and Procedures for the operator. These programs are underway but need to be accelerated and checked for completeness.	Consortium
7R2.3.3-2	Adopt a standardised system of colour-coding of switch locations associated with particular systems and, where practicable, validate the colours and operator response using an appropriate simulator.	Consortium
7R2.4.1-2	The format of emergency and normal operating procedures should be convenient to operators and their proposals should be taken into account. When practicable changes should first be verified and trained using an appropriate simulator. This recommendation is equally applicable to all operational documents and is basically a requirement to account for human factors in the design, appearance and accessibility of such documents.	Consortium
7R2.4.3.5-1	Alarm procedures should be produced for all RBMK plants. Operators should not be required to remember all alarms.	Consortium
RB412	A procedure should be produced detailing the arrangements for reviewing documentation, which should include the requirement that all those who are required to approve documentation are involved in the process.	TECDOC-722
RB413	A format of operating instructions should be adopted which sets out the operations in a more systematic manner so that the procedures can be easily used by operators whenever the need arises, especially when carrying out a manoeuvre that is infrequently performed.	TECDOC-722
RB414	A schedule of the checks to be carried out by the control engineers on each shift to demonstrate compliance with fundamental nuclear safety requirements for the plant Technological Procedures should be developed.	TECDOC-722
RB419	An operating procedure for the reactor emergency cooling system should be prepared that defines the extent and content of the field operations checks carried out on each shift.	TECDOC-722
RB422	A procedure should be developed for the requalification of a plant following maintenance.	TECDOC-722
RB500	Improve the quality of the Technical Specifications.	ASSET RBMK

ID	Description	Source
RB502	Include "hold points" and check lists in safety related procedures.	ASSET RBMK
RB511	Plant management should consider enhancing the "useability" of procedures by ensuring the participation of both operations and maintenance staff in their preparation. The end user must be able to both follow and achieve the intended actions set down in the procedure. Additionally, the procedure must be current and have benefitted from plant history and operating experience. The involvement of experienced operations and maintenance staff with the technical staff is crucial for the success of this recommendation.	ASSET RBMK
RB534	Enhance of the operational procedure review programme.	ASSET RBMK
RB558	To make procedures user orientated, involve the users in reviews and updates of the procedures and actively seek feedback from the users	ASSET RBMK
RB561	To review and develop symptom-based emergency/transient condition procedures, start-up procedures, and test procedures.	ASSET RBMK
RB562	To upgrade the quality of the plant Technical Specifications.	ASSET RBMK
RB580	Revised documentation should give greater guidance to the person implementing the procedure by: - making greater use of step-by-step instructions; - including drawings and check-lists to aid the task performer; - including hold points where independent verification is required; - including guidance on normal and abnormal indications to monitor and record; - including acceptance/rejection criteria.	ASSET RBMK
RB583	Procedures implicated as contributory factors should continue to be reviewed as part of the plant investigation and before re-use.	ASSET RBMK

ID	Description	Source
1R2.8-2	Travel limit switches to the reactor crane should be installed to prevent routine movements over the spent fuel storage pond. Emergency procedures should be developed to defeat the reactor crane limit switches and for its subsequent use over the storage pond area in special circumstances.	Consortium
6R6-2	The operational arrangements for controlling the immediate response to abnormal operational conditions and events should be reviewed and the post-event procedures should be revised to give better guidance to the operator.	Consortium
6R7-9	The step-by-step procedures need to be completed in a priority program. Topic Group 6 supports the programme to develop symptom based procedures. These programmes are underway but also need to be accelerated	Consortium
7R2.3.1-1	Emergency procedures should always be available at the place where they may be required for use.	Consortium
7R2.4.1-2	The format of emergency and normal operating procedures should be convenient to operators and their proposals should be taken into account. When practicable changes should first be verified and trained using an appropriate simulator. This recommendation is equally applicable to all operational documents and is basically a requirement to account for human factors in the design, appearance and accessibility of such documents.	Consortium
7R2.4.3.2-1	Divide the current EOPs into independent procedures and organise them in such a way that the procedure can be easily found when necessary.	Consortium
IGN037	Development of symptom-based Emergency Operation Procedures currently in progress is supported. It this context, the INPP should strengthen its understanding of the basis for the EOP's. It should also establish and maintain ties with the other RBMK plants to ensure an efficient communication of operating experience and of any generic developments regarding plant safety.	Ignalina Report
RB413	A format of operating instructions should be adopted which sets out the operations in a more systematic manner so that the procedures can be easily used by operators whenever the need arises, especially when carrying out a manoeuvre that is infrequently performed.	TECDOC-722
RB416	The emergency operating procedures should be kept in binders of a distinctive colour to aid their identification when they are required for urgent use.	TECDOC-722
RB447	Enhance and upgrade emergency procedures to current human factors standards and strongly consider the use of symptom-based procedures.	ASSET RBMK
RB501	Initiate work on emergency and accident management procedures and assess adequate programme for implementation.	ASSET RBMK
RB504	Review for completeness and modify as necessary the procedures for coping with a variety of operational transients, including those considered within the design basis of the plant, and provide training of the operational staff for these transients on the new simulator, when available.	ASSET RBMK

ID	Description	Source
6R2-1	The business process for controlling and implementing event investigation and reporting should be reviewed and improved. Detailed recommendations on the level of resources, the role of VNIIAES and management procedures are given as R4-13, R4-14 and R4-IS in Chapter 4.	Consortium
6R2-2	The Task Force should be set up to develop an action plan to improve the quality of event investigations and reports and to ensure proper compliance with the regulations. Detailed recommendations on aspects of these problems are given as R4-4, R4-5, R4-6, R4-7 and R4-8 in Chapter 4.	Consortium
6R2-3	Improvements should be made to clarify and extend the scope of the Reporting and Investigation Regulations. Detailed recommendations are made with respect to the national Regulations (PN AE G-12-005-91) in recommendations R4-10 and R4-11 in Chapter 4.	Consortium
6R4-12	NPP Management and on-site GAN Inspectors should ensure that post trip evaluations always: - identify and eliminate the causes of the trip before restarting; - identify and eliminate discrepancies between expected and actual plant response during the trip and subsequent post trip operations and - that event reports indicate a statement by the operator of how he/she determined it was safe to restart the reactor.	Consortium
6R4-16	In the context of Reactor operation the Russian, Lithuanian and Ukrainian utilities should develop, and their Governments encourage, a rolling programme of Peer Reviews covering all RBMK stations such that each station is reviewed every 2 to 3 years. These reviews should be part of a National programme utilising indigenous staff. This programme should be supplemented by the occasional International Review to benchmark the National programme. (Via WANO or IAEA OSART Programmes.)	Consortium
6R5-3	The development of an RBMK verified event database should be given very high priority.	Consortium
RB531	Strengthen the necessary staff resources for independent analysis of reported events, using root cause and trend analysis and analyse the effectiveness of the feedback process.	ASSET RBMK
RB583	Procedures implicated as contributory factors should continue to be reviewed as part of the plant investigation and before re-use.	ASSET RBMK

ID	Description	Source
7R2.3.4-1	Consideration should be given to conducting as much repair and periodic overhaul work outside the process area.	Consortium
7R2.3.4-2	The continued development of networked computer systems should take account of the collection and processing of information concerning system and component failures for the purpose of maintenance planning and control as well as for analysis of reliability. (Such a networked system could also incorporate other operational information e.g. the requirement for fuel route record keeping on a shift by shift basis).	Consortium
RB426	Consideration should be given to review representative programmes used to promote improvements in maintenance effectiveness at NPPs in western countries such as France and the UK. Programmes that should beconsidered include independent assessment of the effectiveness NPP programmes and of maintenance goals and objectives. Subsequent to the review of other countries' programmes, a decision should be made as to whether specific aspects of these programmes would be applicable and beneficial to Smolensk NPP. The aim should be to set up a systematic assessment of the effectiveness of the maintenance programmes. The assessment should comprehensively cover all of the maintenance programmes.	TECDOC-722
RB427	Consideration should be given to developing methods to better utilize equipment history information by allowing easy retrieval, root cause analysis and trending of equipment failures. This process can readily provide identification of recurring failures allowing earlier correction of causes, and be particularly useful when failures occur after several years of service. Although this process can be accomplished manually, personal computers (if available) can also be effective in this application.	TECDOC-722
RB442	Implement appropriate indications for non-operability of safety trains during testing and maintenance.	ASSET RBMK
RB446	Improve quality and usability of maintenance, test, and quality control procedures.	ASSET RBMK
RB503	Develop criteria and procedures for allowable repair time for safety trains.	ASSET RBMK
RB506	For safety related systems, review adequacy of procedures for confirming plant fitness for service following maintenance.	ASSET RBMK
RB512	Improve attention to detail in maintenance activities (e.g. missing handwheel from valves, open electrical connection boxes and poor condition of valve steam areas).	ASSET RBMK
RB520	Plant management should continue their efforts to increase the awareness of maintenance personnel of the need for verification of their work and of its safety implications.	ASSET RBMK
RB548	Plant management should extend the requirement for systematic requalification testing to all maintenance work prior to the return of equipment back to normal operation. Improvement in the quality control of maintenance work in particular on the primary coolant circuit relief valves and other safety significant systems is rated as being very important. The station manager should review his work planning process, work issue and degree of compliance with documented procedures.	ASSET RBMK

ID	Description	Source
RB563	To improve procedures relating to post-maintenance testing.	ASSET RBMK
RB566	Raise the awareness of maintenance staff to pay attention to detail and to the importance of self-verification of their work and independent verification of safety-significant work.	ASSET RBMK
RB583	Procedures implicated as contributory factors should continue to be reviewed as part of the plant investigation and before re-use.	ASSET RBMK

ID	Description	Source
RB434	A single procedure to control the hardware modifications process from initiation to the implementation should be produced. Such a procedure should include definition of responsibilities and specification of the requirements for the training of personnel who have responsibilities for operating, testing and maintaining equipment affected by the modification.	TECDOC-722
RB523	Establish a group within the plant to process and monitor the implementation of changes recommended by the feedback of operating experience.	ASSET RBMK

ID	Description	Source
7R2.4.3.4-1	The implementation of this document should be established in all RBMK plants along with a system of collecting, processing and storing of surveillance test results.	Consortium
RB421	Plant test performance results should be recorded and criteria specified to show that an acceptable standard has been achieved.	TECDOC-722
RB431	All procedures for the surveillance testing programme should be in the same step-by-step format. Only the controlled copy of the procedures should be used in the control room with the list of changes attached.	TECDOC-722
RB432	Performing of testing should be properly controlled and recorded. All tests should be recorded even if they have failed and are later successfully repeated. The reasons for failures should always be analysed and corrective actions taken.	TECDOC-722
RB442	Implement appropriate indications for non-operability of safety trains during testing and maintenance.	ASSET RBMK
RB498	Complete the development of adequate test procedures for valves in ECCS.	ASSET RBMK
RB505	Enhance the surveillance programme to ensure that deficiencies in components, trains, or systems identified during routine surveillance are reported and analysed in a similar manner to events.Take specific account of the reliability of safety-related components when setting the required surveillance and replacement intervals.	ASSET RBMK
RB509	Test procedures for all safety system equipment testing should be reviewed to ascertain the practicality of including specific tests to search for unrevealed failures.	ASSET RBMK
RB510	As part of the existing programme for the enhancement of procedures, plant management should give particular attention to the content of pre-start and routine testing procedures to ensure that clear limits have been identified and testing acceptance criteria are specified. Detailed specifications are required to ensure that the safety-significant equipment can demonstrably meet its design intent, to the greatest extent before start-up.	ASSET RBMK
RB524	Improve surveillance test programme, e.g. by providing test for electrical protective devices.	ASSET RBMK
RB525	Develop a trend monitoring programme for safety related equipment based on reliability date.	ASSET RBMK
RB542	Implement an effective surveillance programme to monitor the efficiency of project integration into routine work and quality control.	ASSET RBMK
RB543	Develop feedback policies designed to continuously improve the control of work and the surveillance programme.	ASSET RBMK
RB551	Plant management should continue to improve the surveillance programme for detecting deficiencies (e.g. primary coolant circuit relief valves). Review of the surveillance programme should be established, drawing from the plant history and the operating experience to ensure any deficiencies in testing procedures are identified and corrected.	ASSET RBMK

ID	Description	Source
RB574	To improve the area of Surveillance Programmes. The scope and the content of the programmes should be reviewed as should their effectiveness in terms of the reduction in the frequency rate of "failure to perform as expected".	ASSET RBMK

ID	Description	Source
1R3.1.16-8	The habitability of vital areas where the plant personnel have to operate in accident situations should be assured by adequate engineered safety features; specific analyses should be performed.	Consortium
6R3-4	A review should be carried out to determine all causes of high circuit activity and high debris levels at SMP Unit 3 and remedial measures developed and applied to all other RBMKs as appropriate.	Consortium
6R3-6	A programme to reduce radiation doses to as low as reasonably practicable (ALARP) should be implemented. This should take into account the practices employed at the RBMK units with the lowest radiation doses.	Consortium
6R3-7	The provision of additional remote control metal inspection equipment would result in significant reduction in collective dose.	Consortium
7R2.2.5-1	The formal introduction of ICRP 60 and all of the latest ICRP recommendations should be considered and implemented where appropriate to do so. This will require a change in the training given to all staff, but, provided this is aimed at the needs of staff, this should not prove onerous. The Radiological staff will require little training. It will however take time to see the benefits and a progressive approach to achieving ALARA/ALARP is advised.	Consortium
RB423	A more detailed review of radiation protection should be carried out in the future.	TECDOC-722

ID	Description	Source
8R6.3-4	The set-up of a process in which temporary permits are granted to NPPs to continue operation, appears to be a sound approach for a transition to a full licensing regime. It represents, as a mater of fact, an important tool in the hands of the regulatory body to promote enhancements in safety of RBMK reactors taking into account that in the past the Regulatory Body was involved in RBMK safety assessment, but in a fragmented way, the risk that some important safety issues could have remained hidden still exists. The review of application documents for temporary permits represents, therefore, an important occasion for an overall reconsideration of the safety level of each individual NPP. In this context a necessary start is to upgrade the quality of the plant technical documentation to allow, e.g. accurate plant specific analysis. It is also necessary that some additional analyses are performed and included in the application documents. As an example, an extended scope of TOB covering the evaluation of special accident sequences (e.g. ATWS, Station Blackout) and a PSA, represent possible tools to extend safety substantiation. A feasible approach could be to require, as a condition attached to the temporary permit, the preparation of additional safety evaluations within a defined limit of time.	Consortium
RB270	A method to limit operation when inspections are not performed or when results are less than satisfactory should be implemented. One method of achieving this is through a periodic operating license renewal. This would require interaction between plant staff and the regulatory body. Licensing restrictions which would require corrective actions within specified times should be agreed upon. The expiry of an operating licence would require the station to shut down, and be placed into a safe shut-down state.	TECDOC-722

ID	Description	Source
9R2-3	Modern tools for pipe and weld inspection should be used, and adequate rules for response to detection of deficiencies and leaks should be defined.	Consortium
IGN023	Automatic remote controlled UT equipment should be used for diameter 800 mm circumferencial weld inspections in order to improve the reliability of the results and reduce radiation exposure to operators. Experience from application of such techniques should be used as a basis to optimize the inspection programme.	Ignalina Report
IGN024	X-ray radiographic techniques should be applied to 300 mm diameter pipes. Alternatively, consideration should be given to application of UT automatic remote controlled equipment	Ignalina Report
IGN025	Steam drum 300 mm diameter and 100 mm diameter dissimilar metal welds should be more frequently inspected (24 welds and 85 welds respectively every four years)	Ignalina Report
IGN026	Ultrasonic testing flaw acceptance criteria should take into account morphology (size and shape) character (linear, volumetric) and location of defects, across the wall thickness.	Ignalina Report
IGN027	NTD inspection of primary circuit should take advantage of advanced equipment and methods in order to improve quality and reliability of inspections.	Ignalina Report
IGN031	Augmented ISI programme should be applied to Ignalina NPP dissimilar metal welds. (see Section II.2.2.(6) and II.2.2 (7)).	Ignalina Report
RB097	Installation of automated metal diagnostic systems to survey the primary circuit metal components and pipelines, and introducing primary circuit integrity monitoring systems.	TCM-92
RB107	Replacement of all 1693 fuel channels together with the graphite rings for reconditioning the diameter gaps in the channel tube-graphite block system in all reactor cells.	TCM-92
RB122	Improvement of non-destructive testing using ultrasonic and radiographic inspection methods. Ultrasonic testing is not generally used for austenitic stainless steels at the plant.	TCM-92
RB123	Evaluation of Finnish and Russian procedures and equipment for penetrant and magnetic particle testing.	TCM-92
RB124	Assessment of inspector qualifications for ultrasonic testing against similar requirements to qualify inspectors for Scandinavian countries.	TCM-92
RB125	Application of mechanized ultrasonic testing and advanced (e.g. PC-SAFT) method for testing and analyzing ultrasonic signals.	TCM-92
RB126	Evaluation of the critical flaw size in the primary loop pipes based on the results of ultrasonioc testing and operating experience as well as on the relevant background data.	TCM-92

ID	Description	Source
RB187	Volumetric 100% inspections of the fuel channels should be considered for implementation to analyse the development of sub-critical cracks. An eddy current type of device carried by the fuelling machine may enable this inspection to be done with each fuel change.	TECDOC-694
RB188	Precise methods of prediction of the gap between the fuel channel and the graphite stack should be developed on the basis of graphite and tube behaviour.	TECDOC-694
RB190	Optimize the inspection programme by identifying the most risk- significant location in order to carry out fewer inspections, but of a higher level of reliability.	TECDOC-694
RB194	New methods of decontamination of the circuit should be developed to reduce radiation exposure of the in-service inspection personnel without eroding the material of the circuit.	TECDOC-694
RB195	The effectiveness of NDE methods should be more extensively established through performance demonstrations on specimens with realistic service type defects.	TECDOC-694
RB196	The inspection of the main circulation circuit should be carried out regularly (100% UT every 4 years). Steam pipes between the separators and the turbine stop valves, and the feedwater lines between the separator and the feedwater control valves should also be inspected sufficiently.	TECDOC-694
RB235	Some 46 fuel channel tubes have leaked in the dissimilar weld region and have subsequently been replaced. Leakage has been from stress corrosion cracks in the stainless steel section of this weld, associated with low titanium concentrations.	TECDOC-694
RB236	Delayed hydride cracks (DHC) have been observed on the outside diameter of fuel channel tubes. Growth of DHCs has been associated with high residual stresses, induced by the former method of straightening tubes. The method used at present has overcome this problem (reduction of residual stresses).	TECDOC-694
RB238	Fuel channels are being (or will be) replaced at all plants with new channels (of essentially the same design) after service exposure (creep) has resulted in closure of the gap between the channel tube and graphite. This extends the service life under the same operating conditions and, to a limited extent, improves safety due to enhanced pre-service inspection capabilities.	TECDOC-694
RB239	Automated equipment has been developed for inspection of the main circulation circuit welds. This equipment with imaging techniques for data display is being used on a pilot basis at the Leningrad plant before being implemented at all RBMK units.	TECDOC-694
RB263	In some cases the programme was found in practice to be not as advanced as concluded in TECDOC-694 and in the area of inspection will require coordination development and training with the station personnel to implement the programme recommended.	TECDOC-722

ID	Description	Source
RB264	In TECDOC-694 the recommendations was to continue tests on a mock-up facility located in the Ukraine. While this recommendation still holds that the objectives of these tests should be well defined and a suitable programme fully prepared prior to the start of tests.	TECDOC-722
RB265	It is also recommended to intensify the exchange of information on the non-destructive and destructive techniques of examining fuel channels among the experts of Russia, Canada and Japan.	TECDOC-722
RB266	The manufacturer should be consulted to see if a fingerprint of ultrasonic indications exists for the pressure tubes. If it does it should be brought to the station. If it does not, such fingerprints should be taken for at least the reference channels and the number should be expanded as much as possible.	TECDOC-722
RB267	Procedures should be developed to record all ultrasonic indications to the highest sensitivity of the equipment. The equipment should be calibrated to identify what indication size is being recorded.	TECDOC-722
RB268	TECDOC-694 recommends that the in-service inspection program be optimized. It is now recommended that prior to any work on the optimization a predictive type of in-service inspection program be implemented by the use of indication analysis. Ultrasonic indications should be recorded first, as soon as possible manually and in the future automatically if such a system can be successfully applied.	TECDOC-722
RB274	Examination of stainless steel cladding and cladding base metal interface using ultrasonic techniques should be developed and implemented at the plant. Its effectiveness should be demonstrated on mock-ups containing realistic defects.	TECDOC-722
RB275	To improve the reliability of ultrasonic inspection and to reduce the doses received by operators, automatic ultrasonic system which is being developed (according to ALARA concept) should be implemented.	TECDOC-722
RB277	The ongoing work on fuel channel reliability under normal operating conditions and on assessment of the effect of changes in the metal characteristics on fuel channel serviceability should be continued.	TECDOC-722
RB278	Providing the LBB concept is successfully applied, the inspection methods should be optimized accordingly. No simple reduction of examination programme should, however,be adopted.	TECDOC-722
RB279	The first circuit should be analysed with the objective of eliminating as many valves as possible to increase the reliability, without diminishing the functioning and the maintenance of the circuit.	TECDOC-722
RB533	Evaluate the effectiveness of the in-service inspection programme and develop an in-service master plan.	ASSET RBMK

ID	Description	Source
IGN033	Necessary decisions should be taken as soon as possible regarding the application of LBB concept to Ignalina NPP.	Ignalina Report
RB197	Currently available results related to application of the LBB concept do not provide a basis for decisions on RBMK safety.	TECDOC-694
RB199	Providing that the LBB concept is to be applied, plant specific work completed and planned should be analysed and peer reviewed, including stress analysis, fracture analysis, stability analysis, integrity analysis and leak detection methods.	TECDOC-694
RB276	The work on the leak before break concept should be continued for the primary circuit piping but eliminating the fuel channel.	TECDOC-722
RB278	Providing the LBB concept is successfully applied, the inspection methods should be optimized accordingly. No simple reduction of examination programme should, however,be adopted.	TECDOC-722

ID	Description	Source
1R3.3.8-2	The codes available are not adequate to describe the pellet and cladding temperature histories after fuel channel break conditions, because the relevant changes in geometry are not modelled. Bounding evaluations should be performed in order to investigate all the possible consequences, for instance in terms of the possible occurrence of fuel melt, water/graphite reactions in hot spots, distribution of the produced flammable gases in the reactor cavity.	Consortium
1R3.3.9-1	For the case of pressure tube ruptures in the reactor cavity it is recommended to perform experimental tests to improve the validation of the codes in order to verify the adequacy of the assumed heat exchange rates from the water flowing from the break to the graphite and to the metal structures.	Consortium
IGN006	Further information is needed to perform analyses of the material properties of fuel cladding and pressure tubes under UDBA conditions as a function of lifetime (neutron irradiation, corrosion, H2 pickup, etc.) In addition, mechanical interaction of the pressure tube and graphite stack and with the fuel assembly (ballooning, rod bow, etc.) should be considered in this analysis.	Ignalina Report
IGN015	An augmented fuel channel tube in-service inspection programme (NDT and destructive testing) for Ignalina Unit 1 should be established and implemented on a yearly basis starting from 1995 in order to minimize the uncertainties on remaining lifetimeestimation. Current Ignalina Unit 2 fuel channel tubes in-service inspection programme should be fulfilled and modified, if needed, according to the results from Unit 1 augmented programme.	Ignalina Report
IGN017	Ultrasonic fingerprints of Ignalina Unit 1 and 2 fuel channel tubes should be taken, at least from the reference channel taking advantage of the improved capabilities of the new automatic ultrasonic equipment already acquired by Ignalina NPP, which will be delivered to the plant in 1995.	Ignalina Report
IGN018	Detailed seismic analysis of fuel channel tubes, fast shutdown rods and refuelling machine should be performed taking into account the specific seismic loads of Ignalina Units 1 and 2	Ignalina Report
IGN045	Improved power supply configuration for ECCS / EFWS discharge valves. Since this issue is of high importance, an action plan should be developed to ensure a fast and efficient implementation. Any financing problems should be resolved.	Ignalina Report
RB104	Replacement of the distributing group header with installation of the check valves, restrictive plugs and mechanical filters at their inlets.	TCM-92
RB107	Replacement of all 1693 fuel channels together with the graphite rings for reconditioning the diameter gaps in the channel tube-graphite block system in all reactor cells.	TCM-92
RB187	Volumetric 100% inspections of the fuel channels should be considered for implementation to analyse the development of sub-critical cracks. An eddy current type of device carried by the fuelling machine may enable this inspection to be done with each fuel change.	TECDOC-694
RB188	Precise methods of prediction of the gap between the fuel channel and the graphite stack should be developed on the basis of graphite and tube behaviour.	TECDOC-694

ID	Description	Source
RB189	Fail-safe characteristics of replacement flow control valves should be fully verified. Modifications should be considered to allow non-operator-initiated response to flow restrictions.	TECDOC-694
RB235	Some 46 fuel channel tubes have leaked in the dissimilar weld region and have subsequently been replaced. Leakage has been from stress corrosion cracks in the stainless steel section of this weld, associated with low titanium concentrations.	TECDOC-694
RB236	Delayed hydride cracks (DHC) have been observed on the outside diameter of fuel channel tubes. Growth of DHCs has been associated with high residual stresses, induced by the former method of straightening tubes. The method used at present has overcome this problem (reduction of residual stresses).	TECDOC-694
RB237	A new design of flow control valve is being installed at all plants, as a measure to prevent ruptures of fuel channels that have been caused by lack of coolant flow when valves have malfunctioned.	TECDOC-694
RB238	Fuel channels are being (or will be) replaced at all plants with new channels (of essentially the same design) after service exposure (creep) has resulted in closure of the gap between the channel tube and graphite. This extends the service life under the same operating conditions and, to a limited extent, improves safety due to enhanced pre-service inspection capabilities.	TECDOC-694
RB264	In TECDOC-694 the recommendations was to continue tests on a mock-up facility located in the Ukraine. While this recommendation still holds that the objectives of these tests should be well defined and a suitable programme fully prepared prior to the start of tests.	TECDOC-722
RB277	The ongoing work on fuel channel reliability under normal operating conditions and on assessment of the effect of changes in the metal characteristics on fuel channel serviceability should be continued.	TECDOC-722
RB279	The first circuit should be analysed with the objective of eliminating as many valves as possible to increase the reliability, without diminishing the functioning and the maintenance of the circuit.	TECDOC-722
RB308	The idea of making a small hole to allow a minimum flow through the fuel channel flow control valve at all times should be analysed. This would allow discontinuation of the use of removable mechanical stops.	TECDOC-722
RB577	To continue design efforts to find a technical solution to the steam leakage that frequently exist in RBMK plants around the fuel channel gaskets, leading to small amounts of leakage through the top of the reactor structure into the reactor hall.	ASSET RBMK

ID	Description	Source
IGN016	Special attention should be paid to the supervision of special channel tubes during manufacture and operation.	Ignalina Report
RB109	Introduction of the coolant leakage monitoring system based on the parameters of the gas checking.	TCM-92
RB280	Determine the degree of hydrogen in a special channel which has more than 15 years of service and check for formation of hydrides at the points of contact of the alignment rings with the channel material.	TECDOC-722

ID	Description	Source
RB271	The fuel handling system would not result in disastrous accidents although with the uncontained reactor hall an open channel could cause measurable releases. Therefore, automating the final machine positioning and having the remaining functions of fuel changing up to the closing of the plug done automatically, possibly by digital computer control would reduce the chance of this event.	TECDOC-722
RB272	The improved seals should be installed as soon as possible to reduce leakage of steam to the deck.	TECDOC-722

ID	Description	Source
1R2.10-9	The need of bypass lines between main circulation pump suction and pressure headers should be reassessed. They should either be removed or fitted with motorised isolation valves in all RBMK.	Consortium
IGN019	The main circulation system should be fully re-analysed taking into account the specific seismic loads of Ignalina NPP Units 1 and 2. Any resulting indication of equipment or support weaknesses should be corrected.	Ignalina Report
IGN020	Permanent displacement and vibration monitoring systems should be installed on the main circulation system of Ignalina NPP Units 1and 2 to verify the results of the stress analysis and/or surveillance of potential component degradation.	Ignalina Report
IGN022	Ignalina NPP regulations related to ISI should be updated taking advantage of recent developments.	Ignalina Report
IGN028	A comprehensive reassessment of the main circulation system components should be carried-out taking into account the new standards and the specific seismic loads of Ignalina Units 1 and 2. Any indication of equipment or support weaknesses arising from such reassessment, should be corrected at the plant.	Ignalina Report
IGN030	The joint and nozzle connection of the steam-water pipes (100 mm diameter) to the steam separators should be further studied for the effects of ageing.	Ignalina Report
IGN032	Ignalina NPP should perform a residual life assessment of the main circulation system components.	Ignalina Report

ID	Description	Source
1R3.1.2-2	It is recommended to perform experiments to test the capability of the check valve of the DGH to close without damage when a pipe break occurs upstream. Then, the probability of the failure of CV disk under such quick closure circumstances should be assessed. Depending on the value of this probability, accident scenarios with a LOCA upstream of these CV should be studied further, assuming possibly more than one CV disk failure.	Consortium
1R3.2.4-1	It is recommended to provide adequate documentation for the case of spurious ECCS actuation for assessment.	Consortium
1R3.2.5-1	It is recommended to provide adequate documentation for the case of gas ingress from the ECCS accumulators for assessment.	Consortium
1R3.2.5-2	Systematic analyses should be performed for cases with gas ingress from the ECCS accumulator tanks which include any combinations of the following initial and boundary conditions: - first core inventory, equilibrium core loading (eg Smolensk 3); - nominal power, low power.	Consortium
9R2-2	Progress urgently ECCS back-fits to all the first generation plants.	Consortium
IGN044	Implementation of floating shut off valves in the accumulator tanks. The design basis of this modification must be clarified. The advantages and disadvantages should be evaluated together. Operational experience from existing installations should be examined. An independent expert opinion on this issue should be obtained by INPP.	Ignalina Report
RB100	Installation an emergency core cooling system.	TCM-92
RB105	Introduction of the emergency core cooling system (ECCS) headers with water supply from them to each DGH beyond their check valves and connection of feedwater cooling system (FWCS) lines to the ECCS headers.	TCM-92
RB243	Installation of check valves between the distribution group headers (DGH) and the main coolant pump discharge header.	TECDOC-694
RB244	Installing additional, larger capacity accumulators in the ECCS.	TECDOC-694
RB439	Analyse adequacy of specifications for ECCS valves in feedwater system.	ASSET RBMK
RB554	To review the appropriateness for purpose (duty) of valves in the ECCS systems.	ASSET RBMK

ID	Description	Source
1R2.6-1	Improvements in the residual heat removal system (blowdown and cooldown system) should be designed to ensure that it's function can be performed reliably by an appropriate system, such that single component failures or auxiliaries unavailability do not impair its capability. Alternatively, diverse ways to bring the reactor to cold, safe shutdown conditions should be demonstrated in terms of adequacy of the required components and auxiliaries against all the postulated scenarios, which include external and internal (e.g. fire) events.	Consortium
1R3.1.15-2	Reliable operation of the emergency make-up system should be ensured, avoiding the in-series operation with the emergency feedwater pumps. The design of such a modification - which was said to be in progress at RDIPE - should be continued.	Consortium
RB245	In Kursk 1 NPP the capability to make use of diverse external water sources for ECC injection has already been implemented (service water connection, artesian wells suction).	TECDOC-694
RB246	Further improvements have been planned to make the emergency water supply from external sources (i.e. lakes) available without any need for energy from in-plant energy sources.	TECDOC-694
RB436	Evaluate the need for automatic start of emergency feedwater system on loss of all normal feedwater flow, as sensed by physical parameters.	ASSET RBMK

ID	Description	Source
1R2.1-1	An additional or replacement direct and diverse emergency feedwater system should be implemented, using high head pumps taking suction from the clean condensate or alternative tanks. The system should be designed to safety system standards including separation of components from each other as well as from components of other systems, with a diverse feed water entry point to the drum separators.	Consortium
1R2.1-2	The potential for achieving better separation of pumps and pipes of the existing feedwater systems should be investigated with high priority.	Consortium
1R2.10-7	The adequacy of the design solution having all three pumps of the undamaged core side ECCS in one room with a common suction line should be reassessed.	Consortium
1R3.1.15-4	The capability of the compartments, where ECCS main components are located, to cope with severe flooding following the break of one ECCS discharge pipe, should be ensured.	Consortium
6R7-15	A review of the design philosophy with respect to the separation of control and protection channels, the extent of automatic control and protection, and the demands placed on the operator should be carried out.	Consortium
RB202	Reviews in the form of specialist meetings should be held on cross link damage.	TECDOC-694
RB216	Increase the degree of redundancy of DC buses and batteries in order to be in accordance with the existing 3 channels of the 6 kV safety supply system. These could be electrically supplied from the three existing alternating current buses. Co-ordination with the large backfit plan for new diesel generators and power circuits is necessary.	TECDOC-694
RB220	Redundant channel separation and protection against common mode failures should be further analysed on a site by site basis.	TECDOC-694
RB240	Increasing the number of emergency feedwater pumps from 3 to 5.	TECDOC-694
RB241	Increasing the number of ECCS (Emergency Core Cooling System) lines from 1 to 2 (1st stage).	TECDOC-694
RB242	Installing additional ECCS pumps (3 for damaged core side cooling and 3 for undamaged core side cooling) and the associated 3 divisions of piping (2nd stage).	TECDOC-694
RB247	This concerns the installation of an additional tank to store the water drained from the lower room located below the reactor cavity. This tank is to be added to the plants of the 1st generation. Its main purpose is to prevent flooding in this area.	TECDOC-694
RB343	An assessment of the existing means to prevent a common mode failure of the UCS pumps due to fire or flooding and other possible hazards should be performed to confirm that all necessary measures have been taken.	TECDOC-722

ID	Description	Source
1R2.1-4	The implementation of an automatic actuation of the emergency core cooling system on total loss of feedwater flow should be investigated, as outlined in section 2.1.6.	Consortium
1R2.10-5	The provision of a signal for ECCS actuation based on the rate of pressure reduction in the drum separators should be investigated anticipating results of investigations showing the need for another early actuation signal in cases of significant break sizes in the circulation circuit.	Consortium
1R3.1.12-2	Another automatic signal for ECCS actuation (in addition to low drum separator pressure) is recommended to cope with steam line ruptures outside-the drum separator rooms and other cases. This could be based on the steam line flow rates or on the rate of depressurisation.	Consortium
1R3.1.15-5	The possibility to include an ECCS start-up logic on very low level in the drum separators, without combination with other signals, should be investigated.	Consortium
1R3.2.4-3	The appropriateness of an interlock system that prevents ECCS actuation before a certain time delay in case of high level in the drum separators and high pressure in primary circuit should be studied.	Consortium
1R3.7.1-1	The possibility of ECCS actuation during major leaks in the drum separators on a false low level signal in the drum separator should be analysed.	Consortium
1R3.7.3-1	The implementation of second signal for safety system actuation derived from diverse parameters is strongly supported with regard to all relevant accident scenarios.	Consortium
IGN043	The suggestion to introduce an updated ECCS actuation algorithm covering a wider spectrum of LOCAs is supported by all partners involved in the review including the IAEA review team. It is recommended that INPP review the suggestion and implement the modification.	Ignalina Report
RB327	Consideration should be given to automate reactor trip and ECCS initiation if low flow is measured on a number of channels over a short span of time.	TECDOC-722
RB339	The reactor trip on low flow in multiple channels connected to the same DGH should be automated.	TECDOC-722
RB437	Evaluate the need for a change in ECCS logic from "and" to "or" related to low drum level and high reactor compartment pressure.	ASSET RBMK
RB555	To evaluate the need for a change in ECCS logic from "AND" to "OR" related to low drum level and high reactor compartment pressure	ASSET RBMK

ID	Description	Source
RB313	The assessment of the consequences of breaks of pipes connected to the primary coolant system and outside leaktight compartments should be reviewed.	TECDOC-722

ID	Description	Source
1R2.11-1	It is recommended to provide adequate documentation of the thermalhydraulic analyses performed for the accident localisation system.	Consortium
1R2.11-2	It is strongly recommended to investigate the possibility of an upgrading of the upper rooms (reactor hall, steam separator rooms)with respect to leak-tightness and confinement function in order to have an additional barrier in accordance with the defence-in-depth concept. It is recommended that the discharge from these rooms in the event of a major break in the primary system within these compartments should be directed into the pressure suppression system or to a filtered ventilation (see also 1R 3.1.9-3).	Consortium
1R2.11-3	A stress analysis and a seismic analysis should be performed before the reconstruction of the upper rooms (reactor hall, steam separator rooms).	Consortium
1R2.11-4	In accordance with international practice and in application of the defence-in-depth concept to the confinement function, consideration should be given to the installation of steamline isolation valves (see also recommendation 1R 2.2-1).	Consortium
1R2.11-7	The main steam safety relief valves in Smolensk 3 and in all plants with similar leak rate should be replaced by better quality equipment (see also recommendation 1R 2.2-3 and 1R 2.3-1).	Consortium
1R2.2-1	The provision of main steam isolation valves immediately downstream of the existing main safety valves should be analysed in detail for future implementation.	Consortium
1R2.2-3	The design of the continuously leaking main steam line safety valves should be reassessed (see also recommendations IR 2.3-1 and IR 2.11- 6).	Consortium
1R2.3-1	The continuously leaking main steam line safety valves should be replaced if the present deficiencies cannot be remedied (see also recommendations IR 2.2-3 and IR 2.11-7).	Consortium
1R3.1.15-1	Redundant, reliable indications of pipe breaks outside of ALS or leaktight compartments should be available to the operator in the main control room, e.g. high level in the drainage wells, high condensate level in the ventilation circuits.	Consortium
1R3.1.16-1	The leaktightness of the ALS should be improved in order to better fulfil the barrier function to fission products in case of an accident involving loss of coolant inside its compartments.	Consortium
1R3.1.16-2	The substitution of suppression pool discharge valves with check valves should be studied in order to make a better use of the confinement function of the suppression pool during loss of coolant accidents in the ALS.	Consortium
1R3.1.16-3	The non-isolatable, direct connection of the primary circuit to the atmosphere should be avoided also in case of accidents with of loss of coolant outside the ALS and in the reactor cavity (see also recommendation IR 2.12.7-6).	Consortium

ID	Description	Source
1R3.1.16-4	The gas treatment systems serving the compartments outside the ALS should be revised: Improvements should be made to avoid the spread of contamination in the plant buildings and to assure the detection of the amount of radioactive products released to the environment under accident conditions (see also 1R3.1.16-3).	Consortium
RB102	Reinforcing structures and lowering pressures in buildings due to pipe breaks to ensure integrity of structures and safety systems at all primary circuit breaks with the equivalent diameter d < 300 mm.	TCM-92
RB191	The possibility of backfitting better containment-confinement systems should be studied.	TECDOC-694
RB208	Create a pressure suppression system for the RBMKs of the 1st generation with an efficiency similar to that of the units of the 2nd and 3rd generation. The construction of a separate building seems to be an appropriate way.	TECDOC-694
RB209	Upgrade the upper rooms (reactor hall, steam separator rooms, etc.) with respect to leaktightness and confinement function. Discharge from these rooms in the event of a break in the reactor coolant system could be directed into the pressure suppression system or to a filtered ventilation.	TECDOC-694
RB210	In accordance with the international practice, consideration should also be given to the installation of steam line isolation valves.	TECDOC-694
RB214	The RBMK designers should proceed with high priority to provide bubbler/condenser pools at first generation units, together with completion of the second stage of improvement (a new 600 mm steam discharge pipe into the cavity routed to the pools), which will eliminate the need for the atmospheric vents.	TECDOC-694
RB249	For RBMK plants of the 1st generation a decision was already made to construct a separate building housing a pressure suppression system. This building shall be connected to the reactor building. It was said that the pressure suppression system capability for LOCA will be similar to that of the accident localization systems of the 2nd and 3rd generation plants.	TECDOC-694
RB336	Passageways and hatches connecting the central hall to other parts of the NPP should be kept closed and leak rate minimized to the greatest extent possible during reactor operation.	TECDOC-722
RB340	It is recommended that confinement leak test be carried out periodically at the full design pressure.	TECDOC-722
RB345	In the future, the main steam safety relief valve should be replaced by better quality equipment.	TECDOC-722

ID	Description	Source
1R2.7-2	The availability of service water to the minimum number of components of safety systems needed to mitigate accidents should be ensured also in case of single passive failures in the long term after the accident onset.	Consortium
1R2.7-3	The automatic, reliable isolation of components served by the Service Water System and not required for accident mitigation should be seriously considered.	Consortium
1R2.7-4	The instrumentation of the Service Water System should be reviewed in detail to verify if a complete set of information related to the proper operation of the parts of the systems serving each user is available to the operator for a prompt diagnosis of possible faults.	Consortium
IGN038	The problem of possible clogging of the service water system by shell fish should be adequately addressed.	Ignalina Report
RB342	The designer's intention to check, by a test, the efficiency of the method proposed to cool the core in case of complete loss of service water should be implemented.	TECDOC-722
RB436	Evaluate the need for automatic start of emergency feedwater system on loss of all normal feedwater flow, as sensed by physical parameters.	ASSET RBMK

ID	Description	Source
IGN049	The increase of the battery depletion time to enable one hour of uninterrupted supply should be considered.	Ignalina Report
IGN050	It is suggested to continue to replace the existing inverters by more reliable equipment.	Ignalina Report
RB217	When relocating the batteries into the new "emergency power supply building", no battery cell selector switches shall be used for reliability reasons. Train separation shall be maintained from the new equipment in this building down to the safety equipment inside the existing plant.	TECDOC-694
RB218	The experts therefore recommend the implementation of batteries of higher capacities of the order of 1 to 2 hours, based on an analysis of system behaviour under station blackout conditions.	TECDOC-694
RB250	After reading the ASSET report for the Chernobyl nuclear power plant in order to review the root causes of a safety significant accident that occurred on 11 October 1991 at Unit 2, experts became concerned about the following two aspects: - the high number of cable problems reported; - generator no.3 was reconnected to the power line 30 minutes after the turbine inlet valves were closed due to the failure of themain breaker controls. There is no automatic feature to open the disconnectors when the generator is tripped.	TECDOC-694
RB346	Although the Russian regulations only require that batteries have an autonomy of not less than 30 minutes, it is recommended that the discharge time of the emergency power system batteries be extended to at least one hour.	TECDOC-722
RB435	Establish an electrical system selectivity programme. Ensure proper coordination of all circuit breakers and fuses.	ASSET RBMK

ID	Description	Source
4R6.2-12	It should be necessary to prepare and carry out joint plant walkdown in order to verify supports of piping and mechanical and electrical equipment.	Consortium
RB221	Further investigation concerning qualification of electrical equipment under harsh environmental conditions should be performed.	TECDOC-694

ID	Description	Source
6R3-3	A detailed review should be carried out by the RBMK operating utilities to determine if there is a problem with diesel generator performance.	Consortium
IGN041	The issue of assuring the cooling of the emergency diesel generators at any time should be further pursued. As an alternative to the current design, a closed-loop cooling mode of the diesel generator heat exchangers should be seriously considered.	Ignalina Report
IGN052	Improvements in the diesel generators relay controls should be implemented as proposed by INPP.	Ignalina Report
RB219	The undervoltage criterion for automatic diesel generator startup should be reanalysed (80% of the nominal voltage is usually used in western plants). An optimization investigation should be carried out to avoid frequent diesel startup caused by transient voltage fluctuations of the grid.	TECDOC-694

ANNEX II

CROSS-REFERENCE BETWEEN GENERIC SAFETY ISSUES AND RELATED SAFETY IMPROVEMENTS CONTAINED IN THE IAEA DATA BANK

This is information contained in the IAEA Extrabudgetary Programme Data Base, which have been provide by RDIPE. The status of this information is as of the end of the 1994.

Generic measures	Unit Plant specific safety improvement measures	Plant
The validation of the developed models is performed in two manners: - by comparison with results of steady-state calculations performed by means of Russian thermohydraulic codes which had been validated on the base of numerous experiments (including the full-scale tests); - by comparison with Russian transient experiments in mock-ups of RBMK channels (model tests of Kurchatov Institute, full-scale mock-ups of RDIPE and Kurchatov Institute). The advanced models of the RBMK reactors were developed using the RELAP5/mod3 Code for various NPPs: Ignalina, Unit 3, Leningrad, Units 1.2, Smolensk, Unit 3. In the last years a systematic analysis of accidents was performed using the RELAP5/mod3 Code. Now the ATHLET Code (Germany) is involved in this process.		
	RELAP5/Mod3 calculation models developed.	Ignalina
	RELAP5/Mod3 calculation models developed.	Leningrad 1,2
	RELAP5/Mod3 calculation models developed.	Smolensk 3

Generic measures	Unit Plant specific safety improvement measures	Plant
The small breaks of DGH were studied using the RELAP5/mod3 and ATHLET Codes.These investigations involved the numerical analysis of the sensivity and study on the influence of a break nodalization on the critical parameters of a channel (temperatures of fuel rods and pressure tubes). The performed calculations of LOCA (ultimate DBA and DGH ruptures) by means of the RELAP5/mod3 Code gave simultaneously the estimations of the temperature of the pressure tubes in these conditions (for Smolensk NPP, Unit 3 and Leningrad NPP, Unit 2).		
	LOCA calculations (ultimate DBA and DGH ruptures) performed.	Leningrad 2
	LOCA calculations (ultimate DBA and DGH ruptures) performed.	Smolensk 3

Generic measures	Unit Plant specific safety improvement measures	Plant
Steam-gas discharge system modernization is being performed in 2 stages: 1) Ensurance of steam discharge at simultaneous rupture of up to 4 FChs (20 % of works). 2) Ensurance of steam discharge at simultaneous rupture of up to 9-10 FChs (80 % of works) (see TECDOC 722, III, 6.1.1)		
	First stage implemented.	Chernobyl
	Full-scope of works has been carried out.	Smolensk 3
	First stage implemented.	Leningrad 1,3,4
	First stage implemented.	Ignalina
	First stage implemented.	Smolensk 1,2
	Full-scope of works has been carried out.	Leningrad 2
.	First stage implemented	Kursk

Issue number and title: Accident analysis / 5 Pipe whip analysis

Generic measures	Unit Plant specific safety improvement measures	Plant
At the present time pipe whip analysis is being studied (it will be performed by MoAEP).		

Generic measures	Unit Plant specific safety improvement measures	Plant
	The analysis of incidents related to the loss of in-house power supply at NPPs according to their actual state have been accomplished within Safety Review Reports (TOB).	Kursk 3,4
	The analysis of incidents related to the loss of in-house power supply at NPPs according to their actual state have been accomplished within Safety Review Reports (TOB).	Leningrad 1,2,3
	The analysis of incidents related to the loss of in-house power supply at NPPs according to their actual state have been accomplished within Safety Review Reports (TOB).	Smolensk 1,2,3

Generic measures	Unit Plant specific safety improvement measures	Plant
The idea is strongly supported to convene a meeting of experts on radiological consequences of accidents.(9.12.94)		

Generic measures	Unit Plant specific safety improvement measures	Plant
Systematic studies of RBMK PSA were started in 1988. The PSA methodology was developed and software for IBM was accomplished taking into consideration RBMK features in 1988-1990 (PSA-0). The second stage (1991-1992), devoted toimplementing PSA for LNPP-1, has incorporated categorization of states with RBMK core damage under severe accidents, analysis of severe accidents, development of NPP models before and after upgrading. The upgrading measures reduce the frequency of core damage and severe accidents by an order of magnitude. At present the efforts have been concentrated on INPP-2 (within the framework of the BARSELINA project). The below listed measures have been recommended and are being implemented at INPP to enhance safety: 1. Providing redundancy and separation of UWDSs. 2. Changing actuation algorithm for the valves along the line between EFPs and separators (implemented). 3. Closing the valves in MCP's by-pass lines (to be implemented during the summer of 1994). 4. Changing power supply at the EFPs and ECCS pumps discharge (to be implemented during the summer of 1994). 5. Increasing carrying capacity of steam dump pipelines from RC (under development). 6. Measuring flow rate in each DGH (system design is under development).		
	PSA in the frame of BARSELINA project.	Ignalina 2
	Models before and after upgrading developed.	Leningrad 1

Generic measures	Unit Plant specific safety improvement measures	Plant
TRIPOLY code system is developed to simulate transients and emergency operating modes for RBMK-type power units. The code system comprises in-coreneutronics and thermal hydraulic models and a model of power unit heat engineering cycle. Fast response of code system versions oriented to training simulator applications or being run as a part of test rig to examine actual instrumentation of comprehensive control systems allows real-time simulation. A library of models to solve RBMK control special tasks oriented to CYBER, VAX computers and IBM PC was created on the base of TRIPOLY technology within 1990 to 1991. Engineering proposals to upgrade regulators of heat proess automatic control equipment on RBMK power units, and functions of scram system in the part of scram response to process parameters value exceeding normal operating level.		
	Proposed to use for developing a test rig to examine comprehensive control system.	Kursk 5
	Proposed to use for developing a simulator (contract with US General Physics Int.).	Leningrad
	Introduced in 1991 in computational system of Smolensk Training Centre simulator.	Smolensk

Generic measures	Unit Plant specific safety improvement measures	Plant
More complex 3D codes for reactor neutronics parameters analysis (STEPAN, SADKO)are developed, libraries of nuclear data and cross-sections are compiled. Excellent agreement with results of Western codes, and with experimental results is obtained. More detailed burn up distribution data are desired.		
STRAZH on-line 3D code (3D power density is calculated within 30 s) to calculate 3D burnup distribution is introduced in the frame of Leningrad-2 NPP reequipping with SKALA-M system. The calculation results may be applied for off-line calculations using STEPAN and SADKO codes.		
	3D burnup distribution is introduced with SKALA-M system.	Leningrad 2

Issue number and title: Core design and core monitoring / 2 Core design void reactivity coefficient of the primary and CPS circuit.

Generic measures	Unit Plant specific safety improvement measures	Plant
Additional neutron absorbers are loaded in all reactors: 80 rods did in each RBMK-1000 and 50 rods in each RBMK-1500 reactor. Verification of 3D codes and development of an advanced measuring method will be speeded up after bringing RBMK NPP Support Centre into operation (RDIPE) and equipping it with modern computers.		
	Void reactivity coefficient measured matches its calculated value.	Smolensk 3
From 80 to 95 per cent of fuel assemblies in all RBMK-1000 power units have been already substituted for fuel of 2.4 per cent enrichment.		
	Further comprehensive investigations needed on use of 2.4% enrichment.	Ignalina

Generic measures	Unit Plant specific safety improvement measures	Plant
Preliminary justification of the 12-zone control system is available. To verify the justification in accordance with modern requirements, necessary models and codes have been developed, and they are to be refined. The memory and fast-acting of available in RDIPE computers (CYBER) is 10 times less than required to make sure that obtained accuracy is sufficient. Besides, graph systems to record calculation results are necessary. Modification has been suggested and are being introduced, resulting in substantial safety enhancement.		
	12-zone control system, similar to Ignalina, introduced.	Leningrad 2
	12-zone control system introduced with non-lag sensors of different number and location as at Ignalina	Leningrad 3,4
	12-zone control system, similar to Ignalina, introduced.	Kursk 5
	12-zone control system with inertial sensors.	Ignalina
	12-zone control system introduced with non-lag sensors of different number and location as at Ignalina.	Smolensk 1,2
	The rod number of the 8-zone system has been decreased down to 10-12 (instead of 40) and is implemented	Kursk 1-4

Issue number and title: Core design and core monitoring / 4 Operational reactivity margin (ORM)

Generic measures	Unit Plant specific safety improvement measures	Plant
The share of assemblies with fuel of 2.4% enrichment makes 80-90% on the average at RBMK-1000. The operating reactivity margin (ORM) makes 43-48 manual control rods (MCR) at the stationary state according to regulations requirements for all the RBMK-1000 Units. The ORM is allowed to reduce down to 30 MCR under transients for the period of not more than twenty-four hours. In case it does not increase during the above period up to 43-48 MCR, the reactor is to be shut down. The system of automatic reactor shutdown for ORM reduction at Ignalina NPP (RBMK-1500) is under development. The elaborated approaches and equipment can also be used for RBMK-1000 reactors.		
	Automatic reactor shutdown system for ORM reduction under development.	Ignalina

Generic measures	Unit Plant specific safety improvement measures	Plant
FASS system comprising 24 rods is introduced in each NPP. FASS rod insertion time is within 1.8 to 2.5 s or 1.7 to 2.2 ß/s. This system is quite efficient (2ß), since it is selected on the basis of suppression of void reactivity, which is the maximum fast reactivity component and does not exceed 1ß. FASS system always operates with concurrent insertion into reactor core all the remainder of CPS rods.		
Verification and adaptation of Western codes to RBMK application is the first stage of joint analysis of RBMK emergency protection system adequacy. Modifications to both rod design (increased length of neutron absorber portion) and insertion rate (control rod insertion time decreased from 19 to 12 s) were incorporated at each power unit to increase efficiency of reactor shutdown system		
	Reactor CPS system efficiency is increased at Leningrad-1 NPP within 1989 to 1990 at the expense of increasing number of control rods by 12 ones, and by substituting additional 21 control rods for more efficient ones.	Leningrad 1

Generic measures	Unit Plant specific safety improvement measures	Plant
	Upgraded control and protection system (CPS-M) is incorporated in Leningrad-2 NPP at the first upgrading stage in 1991 - 1992. Incorporation of CPS-M increases power unit safety at the expense of broadened functions of ICPS, realized by: -volumetric in-core protection using nonlag in-core sensors; -protection on the basis of reactivity margin; - detailed volumetric power distribution monitoring; -increased number of neutron absorber rods; -"2 of 3" criterion application for all functions of CPS; -physical separation of CPS channels in various compartments; -dynamic testing (self-checking) of all CPS channels.	Leningrad 2
Number of shortened rods is increased (up to 32) in all RBMK NPP except ones of the first generation. The rods are inserted from beneath. Each unit of the first generation has 21 shortened rods. These rods are inserted into reactor core for 8 s on operation of fast-acting shutdown system (FASS) and emergency protection system-1 (EP-1).		
	32 shortened rods.	2nd generation
	21 shortened rods.	1st generation

Issue number and title: Core design and core monitoring/6 Subcriticality margins in first generation RBMKs

Generic measures	Unit Plant specific safety improvement measures	Plant
Joint (including Western experts) determination of new, nonstudied conditions (scenarios) should be conducted to specify reasonability of additional subcriticality investigations.		
The problem of decreasing subcriticality when taking measures on decreasing void reactivity coefficient relates to the reactors of the first generation only. As calculational and experimental investigations have shown the required reactor subcriticality is provided for of cold gas depoisoned state and accounts for 2 per cent (5 to 6 additional neutron absorbers) in reactors of the first generation, and 4 per cent in reactors of the second generation. However, additional 12 manual control rods (it has been introduced already at Leningrad NPP) are being installed in reactors of the first generation to futher increase subcriticality.		
	Additional 12 manual control rods installed.	Leningrad
	Additional 12 manual control rods are being installed.	1st generation
Existing structure of functional distribution of CPS rods ensures required subcriticality for reactors of the first generation and safety in accidents.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Plastic sheeting is used to cover the floors in the Smolensk NPP turbine building as a means of contamination control. This plastic presents unnecessary fuel load in the turbine building. Repair of the floors with an easily cleanable EPOXY material will allow for removal of this plastic material.		
	Replace plastic sheet floor surfacing along evacuation routes in the first turn (as of February 1995). The works will be implemented according to the planned stage-by-stage schedule.	Leningrad
	A design is developed to replace all of the plastic sheet floor surfacing in the turbine building with EPOXY material based on the EP-5264 fire retardant enamel. A similar design is produced to replace inflammable plastic sheet floor surfacing all along the evacuation routes.	Smolensk
Utilizing fire retardant cables for all new cable pulls will result in smaller combustible fuel loadings in the NPP.		
	All cables are coated by fire retardant paste of OPK-S, OPK-V types. New cable routes comprise no fire-spreading cables.	Leningrad
	All power cables of safety systems (SS) are not fire-spreading. The SS control cables are not the same, but design documentation to cover them with a fire-protecting compound is produced.	Smolensk
It is recommended that the emergency diesel-generator building at SNPP, LNPP be made as a housekeeping model meeting the requirements for fire prevention and fire protection.		
	The protection of the emergency diesel-generator building satisfies construction standards and regulations as well as departmental requirements for fire protection.	Smolensk

Generic measures	Unit Plant specific safety improvement measures	Plant
Existing deficient fire doors will be replaced with standard fire doors of 1 1/2 hour fire rating. Deficient fire barrier penetration seals will be repaired and/or upgraded as recommended. Deficient cable tray enclosures will be upgraded as recommended. The same concerns enclosures around trays with cables.		
	Firestopping partitions and doors were installed at emergency exit passages and staircases to meet the requirements of 4.2.6 PPB-AS-93.	Leningrad
	The existing non-standard, fire doors are being substituted with coated ones standard in dimensions.	Smolensk
	All the through penetrations at the NPP are repaired when necessary - a relevant technology was developed.	Smolensk
	Cable and construction penetrations between the adjacent rooms were sealed using superfine baselt fiber, with paste OPK used to seal all over the edges.	Leningrad
	Turbine hall is separated from an intermediate building by fire zones with of 2.5 hour fire rating each.	Smolensk
	Essential electric cables are separated to form sections and are isolated by pairs.	Smolensk
	Recommendation to install fire doors are being implemented. Drawings are handed over a Kursk factory.	Smolensk
	Additional fire barriers are installed between fire suppression pumps of safety systems being the most important in terms of equipment safety.	Smolensk

Generic measures	Unit Plant specific safety improvement measures	Plant
New fire detection systems utilizing a combination of smoke (product of Combustion or POC) detectors, spot thermal detectors, continuous line thermal detectors, and flame detectors, should be installed as needed throughout these NPPs. The systems will include control panels, alarm and annunciation capability, new signal cable if required, and power supply.		
	Because of poor quality of fire detectors manufactured in CIS, the NPP board chose several types of US detectors for testing, certifying and providing recommendations on their use. (Because of the lack of allocations the work was stopped.)	Smolensk
	Upgraded fire-warning system accommodates for installation of fire detectors DIP-3, PIN-2 and AMETIST. For Units 1,2 design documentation is produced, mounting is under way. For Units 3,4 further studies are necessary.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
New protective clothing (coat, boots, gloves, pants, helmets with fire shield,self contained breathing apparatus) should be provided for fire brigade personnel. New adjustable FOG/straight stream nozzles should be provided for hand held hoses and water canons throughout the plant. Other manual fire equipment should be installed or upgraded throughout the plant as required.		
	Water fire-extinguishing system is undergoing modernization. Water lines and HP pumping house are being mounted. Fire-extinguishers made by "Gloria" (Germany) were installed at main points in bld. 401, 601. On-site militarized fire brigade has been supplied with the special overalls provided by Finland. The NPP is equipped with manual fire fighting capability following home specification requirements. It is envisaged that compressed air cleaned from oil is supplied to equipment at the fire station.	Leningrad
	Fire suppression system includes equipment for outer fire-escapes from the turbine hall floor to the roof (fire-escapes, elevators and other) with preliminary connected fire-hoses and stream nozzles for dry stationary pipelines. The dry stationary pipelines located on the roof are supplied with water by special fire pumps.	Smolensk
All emergency and normal lighting systems should be repaired as necessary and broken and/or missing fixtures, bulbs, etc. should be replaced.		
	Normal operation and emergency lighting systems are maintained in proper working order.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
New automatic sprinkler systems should be installed throughout the plant as recommended (on the inner side of turbine hall (TH) and under roof slabs inside TH to protected roof beams).		
	In the opinion of the Western Group experts at present there is no sufficientlyeffective fire protection measures at all NPPs in CIS (independent reactor types). Since a complex fire protection of turbine halls is a special priority subject R&D program was proposed but the works were stopped at the program initial stage due to financial difficulties.	Smolensk
	Involute sprinklers O3-16, O3-25 are installed at upgraded automatic systems.	Leningrad

Issue number: Fire Protection / 5 Fire water supply

Generic measures	Unit Plant specific safety improvement measures	Plant
The western review team has recommended that all fire pumps start automatically in sequence on system pressure drop and have only manual stop capability. However, the RBMK experts have stated that the Smolensk NPP fire protection water supply is installed in accordance with Russian design standards.		
	RBMK experts found out that the fire water supply system is in compliance with Russian design standards.	Smolensk
	In compliance with 4.3, 4.5 of USN-01-87 water fire extinguishing devices are to be upgraded at the first stage of LNPP.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
RDIPE has carried out and developed basic technical requirements for the projects as follows: - a new actuator with separated instrumentation and protection circuits; - a system having two sets of sensors and electronics based on different component basis; - each of the sets incorporates protection equipment separated from control equipment both electrically and physically. The principle problem of implementing these projects is the lack of sufficient financial allocation.		

Generic measures	Unit Plant specific safety improvement measures	Plant
	The development is being carried out.	Leningrad 1
	Within the framework of EBRD specifications have been worked out and a tender has been announced to deliver a system actuating protection on decrease of water flow rate through a distributing group header.	Ignalina
	The development is being carried out	Kursk
	A system initiating emergency protection and actuating corresponding safety systems following thermal and technical parameters has been manufactured, tested and prepared for implementation.	Leningrad 2

Issue number and title: I&C / 3 I&C system maintenance and periodic testing

Generic measures	Unit Plant specific safety improvement measures	Plant
A continuous review of all procedures and regulations on test and maintenance of systems is being performed at all RBMK NPPs but RDIPE has no information on a special program in this direction at a national, bilateral or international level.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Proposed measures will be taken into account in future works on RBMK CPS reliability analysis.		
	Collection and analysis of CPS equipment reliability information under way.	Ignalina
	Collection and analysis of CPS equipment reliability information under way.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
Operator support systems connected through on-line communication with operational computers are developed for RBMK-1000 power units.		
	Engineering proposal, as applied to Leningrad-3 which is supposed to be shut down within 1995 to 1996 for long upgrading, has been developed for upgrading automatic systems in several stages: 1. Power unit equipping with a Safety Parameter Display System (SPDS); 2. Upgrading of a system, that is gas circuit control system of the power unit; 3. Step-by-step upgrading of power unit instrumentation and control system (including information computer-aided monitoring system, man-machine interface, turbine-generator and other equipment control); 4. Upgrading of control safety systems, including reactor emergency protection system.	Leningrad 3
A system to display safety parameters taking into account 3D power distribution is developed for operators as a part of Skala-M system based on STRAZH code.		
	Skala-M information and measuring system is introduced in Leningrad-1 within 1989 to 1990 and in Leningrad-2 within 1991 to 1992 during the first upgrading stage to ensure: - wider on-line monitoring of reactor system parameters; - increased reliability and fast-response in fulfilling the main computational operations; - improved information support for operating personnel; - increased fast-response and reliability in emergency diagnostic recording.	Leningrad 1,2

Generic measures	Unit Plant specific safety improvement measures	Plant
Organization of operator's work, including physical protection, reduction operator's workload etc. are the responsibility of operating organizations - LNPP(LNPP), the "Rosenergoatom" concern (KNPP, SNPP). At present the documents concerning the above-mentioned questions are being developed (9.12.94).		
Operator support systems connected through on-line communication with operational computers are developed for RBMK-1000 power units.		
	Engineering proposal, as applied to Leningrad-3 which is supposed to be shut down within 1995 to 1996 for long upgrading, has been developed for upgrading automatic systems in several stages: 1. Power unit equipping with a Safety Parameter Display System (SPDS); 2. Upgrading of a system, that is gas circuit control system of the power unit; 3. Step-by-step upgrading of power unit instrumentation and control system (including information computer-aided monitoring system, man-machine interface, turbine-generator and other equipment control); 4. Upgrading of control safety systems, including reactor emergency protection system.	Leningrad 3
A system to display safety parameters taking into account 3D power distribution is developed for operators as a part of Skala-M system based on STRAZH code.		

Generic measures	Unit Plant specific safety improvement measures	Plant
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Generic measures	Unit Plant specific safety improvement measures	Plant
The matters relating organization, the adequate number of staff and training as well as division responsibilities between NPP administration and an operating organization are presented in Quality Assurance Programs.		
	QA programmes issued in 1994.	Leningrad
	QA programmes issued in 1994.	Kursk
	QA programmes issued in 1993.	Smolensk

Issue number and title:	Operational safety / 2	Quality assurance
Generic measures	Unit Plant specific safety improvement measures	Plant
Quality Assurance Programmes in operation have been developed for LNPP (all units) and SNPP (all units). Quality Assurance programmes for LNPP-2 and SNPP-3 were submitted to GAN for consideration and approval. At the end of 1994 the programmes for SNPP-1,2 and KNPP-3,4 will be represented to GAN. The others will be submitted for approval according to schedule, available in GAN.		
	QA programme submitted to regulatory body.	Leningrad 2
	QA programme sumitted to regulatory body.	Smolensk 3
	QA programme will be submitted to regulatory body at the end of 1994.	Smolensk 1,2
	QA programme will be submitted to regulatory body at the end of 1994.	Kursk 3,4

Generic measures	Unit Plant specific safety improvement measures	Plant
Within the framework of safety culture enhancement quality assurance programmes (at all RBMK-1000 units) and personnel duties have been developed.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Document computer management programme (storage, corection, distribution) at RBMK-1000 NPPs are not available so far.		

Issue number and title:	Operational safety / 5	Material condition

Generic measures	Unit Plant specific safety improvement measures	Plant
The tasks, listed in TECDOC -722, VIII.4.6, have not been resolved yet.		

Generic measures	Unit Plant specific safety improvement measures	Plant
The General Regulations "Testing, maintenance and repair of systems, important for safety, and separate normal operation systems of RBMK-1000 NPPs. General technical requirements (General Regulations)", RD-EO-0011-93, have been issued by "ROSENERGOATOM", NPP operator, on the 14th June 1994. The General Regulations determine periodicity, scope and sequence of equipment maintenance. The General Regulations are to be effective from the 1 of February, 1995.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Both the "Beyond the Design-Basis Accidents Procedures" and "Symptom-Oriented Emergency Procedures" (jointly with American side) are being developed. In RDIPE there are development and introduction plans for these procedures.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Recommendations on equipment maintenance programme on up-dating standards, data on equipment availability and its performance including failures, preventive measures and assessment of their efficiency have been incorporated into Quality Assurance Programs issued by operating organizations.		
	Completed in 1993.	Smolensk
	Completed in 1994.	Kursk
	Completed in 1994.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
Prior to shutting a unit down for up-grading, a review to determine safety level after a corresponding stage of up-grading should be produced according to the GAN recommendations.		
	Review performed in 1989.	Leningrad 1
	Review performed in 1994.	Kursk 1
	Review performed in 1992.	Leningrad 2

Generic measures	Unit Plant specific safety improvement measures	Plant
The operating organization "Rosenergoatom" concern elaborates programmes on surveillance tests taking into account corresponding guidances and schedules. At present such programmes are being developed.		

Generic measures	Unit Plant specific safety improvement measures	Plant
If the relevant inquiry is made, more detail analysis of NPP radiation protection state can be carried out.		

Generic measures	Unit Plant specific safety improvement measures	Plant
"Regulations on the Order of Issueing RF GAN Temporary Permissions to Operate NPP Units in RF" (PD-04-01-93) has been approved (January 20, 1993) and put in force (March 1, 1993).		

Generic measures	Unit Plant specific safety improvement measures	Plant
Pilot specimens of automated systems have been fabricated and put under test to perform ultrasonic inspection of longitudinal and circumferential weldments D=Ø 800, headers D=Ø900 and steam separators, header bottoms and drum-separators.		
	A system to monitor leaktightness in equipment and pipelines of the forced recirculation circuit (FRC) according to three parameters (aerosol activity and air humidity in FRC rooms; acoustic signals from FRC components by means of elastic oscillation sensors) has been developed and is being introduced. The system allows to reveal the circuit loss of leaktightness and to early identify leak location.	Leningrad 2
A measurement and calculation system based on domestic components and software are supposed to be develop and followed by its introduction at NPPs.		
	Since 1993 all the main circulation pumps have been subjected to vibration diagnostics.	Kursk 4
To make expert assessment of FRC circuit metal state in RBMK reactor there have been developed defect visualization (determination of geometrical dimensions) systems based on multifrequency acoustic holography. Two pilot systems have been tested at LNPP-2 in 1992. On the basis of these systems and in accordance with the test results an ultrasonic computer flaw detector ("Avgur 4.1") and a portable computer system of USI ("Avgur 4.2") were developed to carry out data coherent processing according to revealed defects aiming at their classifying and determining geometrical dimensions.		
	Two pilot systems tested in 1993.	Leningrad 2
Certification tests of systems are under preparation to implement at test rigs of AEA Technology certifying centre. Certification for inspectors (in agreement with AEA Technology) will be carried out after the equipment for inspection is certified.		
	Additional system implemented to monitor leaks in channels.	Leningrad 1,2

Generic measures	Unit Plant specific safety improvement measures	Plant
Production prototypes of ultrasonic systems for examination of circular 800 mm I.D. weld seams and flaw visualization system equipped with data analysis and filing computer system compatible with IBM PC were tested at Leningrad 2 in 1992.		
	Prototypes of ultrasonic systems were tested in 1992.	Leningrad 2
Separate components of TV system for monitoring equipment condition in strong sealed compartments of RBMK reactor at Kursk NPP have been mounted and tested.		
	TV system for monitoring equipment tested.	Kursk
	Leak monitoring system to examine equipment and pipelines of forced recirculation circuit (FRC) is implemented since November 1991during the first upgrading stage of Leningrad-2 NPP, and is based on the detection of the three parameters as follows: airborne activity, air humidity in FRC compartments, acoustic signals picked off from FRC components by sensors of elastic oscillation. This system allows operating personnel to detect FRC leakage at its initial stage, locate break, and take adequate provisions.	Leningrad 2

Generic measures	Unit Plant specific safety improvement measures	Plant
"The Technique to Calculate NPP pipings within the "Leak-before-Break" Concept", M-TPR 01-93, 06.12.93, has been developed and approved by RF GAN. It allows to omit considering a possibility of a double ended guillotine rupture in pipelines and not to foresee measures for limiting consequences of such a break. Calculations and experimental studies on realizing recommendations of the Summary Report on the International project "Safety of Design Decisions and Operation of NPPs with RBMK Reactors" have been ceased idue to lack of financing.		

Issue number and title: Pressure boundary integrity / 4 Fuel channel and tract integrity

Generic measures	Unit Plant specific safety improvement measures	Plant
To upgrade the efficiency of systems for inspecting FCh after manufacturing the equipment listed in IAEA-TECDOC-694 (Section II, p.73) is needed.		

Generic measures	Unit Plant specific safety improvement measures	Plant
The last version of Procedure for in-service inspection of FChs and CPS channels metal, as well as graphite stack (N E.040-2703), which determines inspection periodicity, scope and applied methods, is issued in 1994.		
	Practically all special-purpose channels at LNPP, Units 1, 2 are substituted by new ones, which were manufactured in accordance with changed production technology, related to residual stresses relieve on outer surface.	Leningrad 1,2

Issue number: Pressure boundary integrity / 6 Fuel-handling

Generic measures	Unit Plant specific safety improvement measures	Plant
To improve the leak tightness of "FA-duct" joint new sealing elements of 2 types were developed: 1) "expanded" graphite (as a filler) 2) metal spacers		
	Cylinder multilayer spacers used.	Smolensk
	Cylinder multilayer spacers used.	Kursk
	Within the framework of research subsidized by the EBRD, INPP has declared tender on the joint ("expanded" graphite) development.	Ignalina
	Graphite spacer is installed and being tested.	Leningrad

Generic measures	Unit Plant specific safety improvement measures	Plant
At present no one unit is equipped with ECCS designed for a steam pipeline rupture, except for Smolensk NPP, Unit 3. In the logic, actuating SNPP-3 ECCS, there is a signal for the primary circuit pressure reduction rate but there is no instrument to record this value. Therefore, nowadays the ECCS gets into operation if two additional signals are available: all main circulation pumps are off and drum-separator pressure drops down to 42 atm.		
	Equipped with ECCS designed for a steam pipeline rupture.	Smolensk 3

Generic measures	Unit Plant specific safety improvement measures	Plant
Long-term cooling system, identical to installed at Smolensk NPP, Unit 3, are available at all RBMK-1000 second generation units and Ignalina NPP as well. At Leningrad NPP, Kursk Units 1,2 and Chernobyl NPP long-term cooling system has less inventory. At units' upgrading it will be substituted.		
	Upgrading is planned.	Kursk 1,2
	Long-term cooling system available.	Smolensk 3
	Upgrading is planned.	Chernobyl
	Upgrading is planned.	Leningrad
	Long-term cooling system available.	2nd generation

Issue number and title: Safety and support systems / 3 ECCS reliability improvement

Generic measures	Unit Plant specific safety improvement measures	Plant
ECCS does not have the sufficient redundancy and separation level only at first generation units. At second generation units 3 ECCS channels of reactor emergency half have physical separation for engineering features and electric power supply system; ECCS of reactor intact half has 3 channels located in one room, which is divided into 3 compartments by partitions.		
	3 ECCS trains in one room divided into 3 compartments.	2nd generation
	Lack of redundancy and separation.	1st generation

Generic measures	Unit Plant specific safety improvement measures	Plant
	Design of reactor shutdown system and ECCS actuation for flow rate reduction through the distribution group header has been developed.	Kursk 1,2

Generic measures	Unit Plant specific safety improvement measures	Plant
There are safety valves at all NPPs for pressure increase in strong leaktight compartments and accident localizing towers. Accident localizing system with a bubbler pool has been installed at Smolensk NPP, Units 1-3, Kursk NPP, Units 3, 4 and Chernobyl NPP, Unit 3. It compensates all FRC ruptures inside strong leaktight compartments as well as ruptures in BCS pipelines located in leaktight premises. The accident localizing systems of Leningrad NPP, Units 3, 4 and Ignalina NPP Units are equipped with an accident localization tower. An activity release limiting system SOVA design for LNPP, Units 2, 3 is under consideration.		
	Accident localizing system with a bubbler pool has been installed.	Chernobyl 3
	Accident localizing system with a bubbler pool has been installed.	Smolensk
	Accident localizing system with accident localization tower has been installed.	Ignalina
	Accident localizing system with a bubbler pool has been installed.	Kursk 3,4
	Accident localizing system with accident localization tower has been installed.	Leningrad 3,4
	Activity release limiting system under consideration.	Leningrad 2,3

Generic measures	Unit Plant specific safety improvement measures	Plant
Efficiency of steam-water pipelines cooling with atmospheric air has been calculated.		

Generic measures	Unit Plant specific safety improvement measures	Plant
Based on operating experience, the less reliable safety related equipment should be replaced by improved quality pieces of equipment (high voltage switches, cell switch for accumulator batteries, etc.)		
	According to engineering design rules an accumulator battery (AB) discharge time is assumed to be 0.5 hours. Calculation of AB loads and discharge time for the existing emergency power supply systems was performed according to design produced by the All-Russia Design Development Association for Research and Technology (VNIPIET).	Leningrad
Emergency power supply systems were upgraded following VNIPIET design.		
	Emergency power supply systems were upgrade following VNIPIET design.	Leningrad
The battery discharge time should be increased according to results of RBMK severe accident analysis. It is recommended by experts that high capacity accumulators with 1-2 h discharge time be installed. Regulatory body should conduct a battery discharge time design criteria analysis.		
	DGs designed for three-channel system for independent power supply will have start up time of 15 s according to PNAEG-9-027-91 and VNIPIET working documentation.	Leningrad
	The existing diesel-generators (DG) can take out the load not more than 90 min.	Leningrad

Issue number and title: Safety and support systems / 9 Electrical equipment qualification

Generic measures	Unit Plant specific safety improvement measures	Plant
Environmental qualification of electrical equipment should be checked. According to the results, the necessary measures should be taken.		
	There is a list to qualify electrical equipment according to OP8-88, PNAEG-9-027-9, PNAEG-9-026-90 (Reg.No. Eh-492 and 29-229).	Leningrad

ABBREVIATIONS

3-D	three dimensional
ALARA	as low as reasonably achievable
ALS	accident localization system
ANFSS	overall plant fire suppression system
ECCP	emergency core cooling pump
ECCS	emergency core cooling system
ASSET	Assement of Safety Significant Events Team (IAEA)
ATWS	anticipated transient without scram
BDBA	beyond design basis accident
CPS	control and protection system
CR	control rod
DBA	design basis accident
DG	diesel generator
DGH	distributing group headers
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EMC	electromagnetic compatibility
FASS (BAZ)	fast acting scram system
FC	fuel channel
GAN	Gosatomnadzor Federal Nuclear and Radiation Safety Authority of Russia
ICV	isolating control valve
IEC	Swiss National Committee of the International Electrotechnical Commission
INSAG	International Nuclear Safety Advisory Group (IAEA)
ISI	in-service inspection
LEP	local emergency protection
LASFSS	local automatic sprinkler suppression system
LBB	leak before break
LOCA	loss of coolant accident
LWL	lower water flow lines
LWR	light water reactor
MCP	main coolant programme
MCR	main coolant room
NPP	nuclear power plant
NUSS	Nuclear Safety Standards of the IAEA
OECD	Organisation for Economic Co-operation and Development
ORM	operational reactivity margin
OSART	Operational Safety Analysis Review Team (IAEA)
PRS	pilot risk study
PSA	probabilistic safety assessment
QA	quality assurance
RBMK	light water cooled, graphite moderated, channel type reactor (Soviet design)
RCPS	reactor control and protection system
RDIPE	Research and Development Institute of Power Engineering
RLC	reinforced leaktight compartment
RWA	rod withdrawal accident
SAR	safety analysis report
SCALA	data recording and handling systems for RBMK reactors
TOB	safety analysis report, Russian Federation
USI	ultrasonic inspection
UDBA	ultimate design-basis accident
WWER	light water cooled, water moderated, pressurized reactor (Soviet design)