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SAFETY ISSUES AND THEIR RANKING FOR 'SMALL SERIES' WWER-1000 NUCLEAR POWER PLANTS

A PUBLICATION OF THE
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SAFETY OF WWER AND RBMK NUCLEAR POWER PLANTS

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FOREWORD

The International Atomic Energy Agency 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 co-operation 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 maintains 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 implementation depends on voluntary extrabudgetary contributions from IAEA Member States and on financial support from the IAEA Regular Budget and the Technical Co-operation Fund.

For the extrabudgetary part, a Steering Committee provided co-ordination and guidance to the IAEA on technical matters and served as a forum for the exchange of information with the European Commission and with other international and financial organizations. The general scope and results of the Programme were reviewed at relevant Technical Co-operation and Advisory Group meetings.

The Programme, which took into account the results of other relevant national, bilateral and multilateral activities, provided a forum to establish international consensus on the technical basis for upgrading the safety of WWER and RBMK NPPs.

The extrabudgetary part of this Programme was phased out and brought to a successful completion in 1998. Assistance continues to be provided to Member States operating WWER and RBMK NPPs within the framework of the regular IAEA Programme and TC projects.

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 NPPs. Moreover, they do not replace a comprehensive safety assessment which needs to be performed in the framework of the national licensing process.

The IAEA officer responsible for this publication was J. Hoehn of the Division of Nuclear Installation Safety.

CONTENTS

SUMMARY

This report presents the safety issues in ‘small series’ WWER-1000 nuclear power plants (NPPs). Safety issues are deviations from current recognized safety practices in design and operation judged to be safety significant by their impact on the plants' defence in depth.

This report is intended to serve as reference for the development of plant specific safety improvement programmes and for the evaluation of measures proposed and/or implemented.

The identification of safety issues is based on safety studies conducted by the operators of ‘small series’ WWER-1000 units and by organizations dealing with these reactors, on findings of IAEA safety missions to ‘small series’ WWER-1000 plants in South Ukraine, at Novovoronezh and Kalinin and on information obtained from specialists from various countries during an IAEA consultants meeting, 8–12 September 1997 in Vienna, within the framework of the Extrabudgetary Programme on the Safety of WWER and RBMK NPPs.

Safety issues are first presented according to their impact on the main safety functions and are then described individually.

The safety issues are characterized by issue title and specified by issue clarification. Safety issues connected with plant design are followed by the ranking of the issue and ranking justification. Altogether 85 safety issues have been identified, 12 of which are in Category III (defence in depth is insufficient, immediate corrective action is necessary), 38 in Category II (defence in depth is degraded, action is needed to resolve the issue) and 22 in Category I (departure from international practices, to be addressed as part of actions to resolve higher priority issues). In the case of operational safety issues (13 safety issues) no ranking is provided as the available material was considered insufficient.

For each safety issue, comments and recommendations are made by the IAEA; the status of corresponding measures to improve safety implemented or planned at each site are presented in the specific plant status.

The review of the safety features of ‘small series’ WWER-1000 plants shows that the main safety concept of these reactors is similar to that of model 320 with respect to the nuclear island arrangement, the amount of safety systems and the main process parameters of the primary and secondary circuits. However, the ‘small series’ WWER-1000 plants have major deficiencies such as a lack of separation of redundant safety systems and a single set of the reactor protection system for technological parameters which do not meet the current national standards and international practice. Differences in engineering design solutions, quality of manufacture and reliability of equipment have been revealed as deficiencies. About one third of the design safety issues have been identified by operational experience. The majority of safety issues have been identified as deviations from current standards and practices which have evolved since the WWER-1000 NPPs were designed.

Much of the backfitting and upgrading work recognized as being required has been or is being performed. This activity was initiated by the WWER Owners Group and since the early 1990s international assistance has played an important role in the process of safety improvement of these NPPs.

The current status of plant specific backfittings varies from site to site, depending on national regulatory requirements and the available financial means. The review of the plant specific status also indicates that certain safety issues have already been solved in some units.

This report presents the information currently available to the IAEA on safety issues and safety improvement measures in ‘small series’ WWER-1000 plants. The IAEA intends to update this information regularly and make it available to the interested parties as part of the technical database developed within the framework of the Extrabudgetary Programme on the Safety of WWER and RBMK NPPs.

1. INTRODUCTION

The 1000 MW WWER nuclear power plants are the larger and more modern third generation pressurized water reactors of Soviet design. As of August 1997, 20 units of WWER-1000 plants were in operation, two in Bulgaria, seven in the Russian Federation and 11 in Ukraine. The design of WWER-1000 NPPs exists in four different models.

The design of the standard model 320 of WWER-1000 NPPs (WWER-1000/320) is more similar than other reactors of Soviet design to PWRs of Western design when considering design philosophy, design features and construction. There are 15 units of the WWER-1000/320 NPP under operation, and an additional seven units of the standard model are under construction in the Russian Federation (1), in the Ukraine (4), and in the Czech Republic (2). Shortcomings of the WWER-1000/320 plants have been revealed by operational experience and deviations from current safety standards. Generic safety issues were identified by the IAEA and published in the report IAEA-EBP-WWER-05 in March 1996, together with their safety significance and measures required to improve safety.

The designs of WWER-1000 NPP earlier models 187, 302 and 338 were developed in the 1970s based on the applicable standards at this time such as OPB-73 [1]. These early models have historically been called the ‘small series’ because only five units have been constructed: Novovoronezh NPP Unit 5 (model 187), South Ukraine NPP Unit 1 (model 302) and Unit 2 (model 338) and Kalinin NPP Units 1 and 2 (model 338). The most significant design weaknesses by current standards are the lack of physical separation and functional isolation and a single set of the reactor protection system.

In some cases, the ‘small series’ WWER-1000 units have positive plant specific design features such as the larger volume of emergency feedwater storage tank and the BRU-A isolation valves, so that the corresponding safety issues for the model 320 can be considered less sensitive. The issue on control rod insertion reliability/fuel assembly deformation is not applicable to Unit 5 of Novovoronezh NPP due to fuel assemblies with an outside shroud.

The main circulation isolation valves could be used to mitigate larger steam generator collector failures.

It was agreed at the Advisory Group Meeting in December 1995 to consider the safety aspects of the ‘small series’ WWER-1000 units separate from the units of the standard model WWER-1000/320.

The purpose of the IAEA mission of experts to South Ukraine NPP (SNPP) [2], on 8–19 July 1996, was to review the safety improvement programme for Units 1 and 2 and identify major design and operational deficiencies of models 302 and 338, and to advise on the completeness and adequacy of safety improvements. On 17–21 March 1997, a Technical Visit was conducted at VNIIAES and Novovoronezh NPP (NNPP) to identify safety issues with respect to design and operational features of the WWER-1000/187 unit and obtain information on the scope and status of implementation of compensatory and corrective measures [3]. The results of these IAEA activities were used primarily to develop a first draft of the ISSUE BOOK as a basis for the Consultants Meeting on Safety Issues and their Ranking for Small Series WWER-1000 NPPs which took place in Vienna on 8–11 September 1997. On 16–20 November 1998, a Technical Visit was conducted at VNIIAES and Kalinin NPP (KNPP) to specify the clarification of some issues and to complete the information on the scope and status of implementation of safety improvement measures at this site.

OBJECTIVE AND SCOPE

The objective of this publication is to present a consolidated list of safety deficiencies, called safety issues, ranked according to their safety significance and corrective measures to improve overall plant safety. It is intended for use as a reference to facilitate the development of plant specific safety improvement programmes and to serve as a basis for reviewing their implementation. To the extent that information was made available to the IAEA, the plant specific status with respect to each safety issue is described.

Section 2 provides an overview of the impact of the relevant issues on the main safety functions in different operational conditions, and of other aspects important to overall plant safety. A summary of the safety issues and their respective ranking is given in Tables I and II at the end of Section 2. Section 3 deals with individual safety issues identified in the design, which are presented according to the structure below. Section 4 presents the safety issues related to operational safety according to a similar structure but without the ranking.

STRUCTURE OF SAFETY ISSUES PRESENTATION

The evaluation process is organized on the basis of safety issues. Safety issues are deviations from currently recognized safety practices in design and operation judged to be safety significant with the potential to affect plants' defence in depth. Internationally recognized safety practices are reflected in the nuclear safety standards (NUSS) reports of the IAEA Safety Series. A safety issue and its ranking which is considered as a reference for safety improvements is no longer applicable to a specific plant if the safety deficiency is resolved.

Specific deficiencies identified in several individual units of the model 320 often indicate a safety concern generic to all units of the 'small series' WWER-1000 reactors. The present publication deals with these safety issues which are generic to design and operational practice of WWER-1000 reactors models 187, 302 and 338.

In this publication, safety issues are characterized by an ISSUE TITLE and specified by an ISSUE CLARIFICATION which includes:

- a technical description of the safety concern;
- the source of the issue, identified either by operational experience or as deviations from national standards such as OPB-88 [4], PBJa-89 [5] or international safety standards such as the NUSS of the IAEA, or a deviation from current practice;
- the way the issue was identified in detail, namely:
    - operational experience in WWER-1000 NPPs, including low reliability of equipment and material degradation
    - generic and plant specific operational experience feedback
    - results of safety reviews, and
    - results from probabilistic safety assessments (PSAs), either plant specific to 'small series' WWER-1000 or generic lessons learned from PSA studies of other plants.

The RANKING OF ISSUES follows the approach applied to WWER-440/230 NPPs in IAEA-TECDOC-640 [6].

Accordingly, four categories are considered:

Category I: Issues in Category I reflect a departure from recognized international practices. It may be appropriate to address them as part of the actions to resolve higher priority issues.

Category II: Issues in Category II are of safety concern. Defence in depth is degraded. Action is required to resolve the issue.

Category III: Issues in Category III are of high safety concern. Defence in depth is insufficient. Immediate corrective action is necessary. Interim measures might also be necessary.

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

The judgement of the safety significance of an issue is based on an evaluation of the potential degradation of defence in depth. For that purpose the evaluation follows the concept of defence in depth, as given in INSAG-3 [7] and INSAG-10 [8], which is centred on several levels of defence, including successive physical barriers preventing a radioactive release to the environment.

The objectives of the five levels of defence established in INSAG-10 [8] are:

Level 1: Prevention of abnormal operation and failures.

Level 2: Control of abnormal operation and detection of failures.

Level 3: Control of accidents within the design basis.

Level 4: Control of severe plant conditions including prevention of accident progression and mitigation of the consequence of severe accidents.

Level 5: Mitigation of radiological consequences of significant releases of radioactive materials.

The levels of defence are implemented firstly, to prevent damage to the plant and the barriers (i.e. the fuel and its cladding, the boundary of the primary circuit and the containment), and secondly, to mitigate the consequences of damage. Therefore, the impairment of defence in depth for a given issue involves a judgement on the degradation of barriers and an evaluation of the effectiveness of redundant means to perform the MAIN SAFETY FUNCTIONS AFFECTED*:* controlling the power, cooling the fuel and confining the radioactive material. This evaluation also considers the principle according to which plant conditions with relatively high probability of occurrence shall have only small consequences and plant conditions resulting in plant damage with high radioactive releases shall be of low probability of occurrence [9].

Depending on the impairment of the levels of defence for a given issue, the performance of the main safety functions to protect the integrity of barriers regarding capability and reliability can be affected to different degrees:

Deviations from current international practice, which affect plants' defence in depth and impair the performance of the safety functions for scenarios beyond the design basis (DB) envelope, should be ranked as Category I.

If one or more levels of defence are affected by the safety issue and the safety function is impaired for scenarios within the DB envelope or is questionable beyond the DB, defence in depth is considered degraded. Such issues are of Category II.

If one or more levels of defence are seriously affected by the safety issue and the safety function is questionable for scenarios within the DB envelope or is disabled beyond the DB, defence in depth is considered insufficient. Such issues are of Category III.

If one or more levels of defence are lost due to the safety issue and the safety function is disabled for scenarios within the DB envelope, defence in depth is considered unacceptable. Such issues are of Category IV.

The JUSTIFICATION OF RANKING first describes how the issue was identified, e.g. by operational experience or deviation from current standards and practices. Then it considers the impact of the given issue on the levels of defence and judges its safety significance on the performance of main safety functions in order to maintain the integrity of barriers for scenarios within the DB envelope and beyond. Where applicable, the effectiveness of interim measures demonstrated by operational experience has been taken into account.

In general, issues detected on the basis of design analysis or operational experience such as shortcomings in the actual implementation of engineering design, material degradation due to improper specification or due to loads not specified in the design, have been given higher safety importance than those potential issues identified as a deviation from current safety standards.

Two aspects of operational issues are of generic significance in WWERs. The first concerns the lack of automatic controls on WWERs in comparison to PWR reactors. This adds additional significance to the role of the operator. Secondly, the design deficiencies must be compensated for temporarily by increased operator activity.

Design issues are specific to a reactor type. However, operational issues tend to be more a function of the culture of a specific country, utility or NPP. The first review of the operational safety issues of Soviet design NPPs was reported in the IAEA-TECDOC-640 [6], with respect to the WWER-440/230, in 1992. The second review was reported in the report IAEA-EBP-WWER-05 [10] with respect to the WWER-1000/320 model. The issues reviewed in this publication were assumed, as a starting point, to be similar to those found in the standard model WWERs as identified in [10]. This was necessary because of the lack of in-depth studies of operating issues in ‘small series’ 1000 MW WWERs. In this publication some of the operational issues are modified in comparison to the analogous ones for the standard model WWER-1000/320. These modifications are mainly the result of additional information obtained from the safety missions to the NPPs of ‘small series’. Reviews such as OSARTs would be necessary to establish the full extent of operational issues.

To a large extent, the safety issues identified are being addressed by the Member States concerned within their national safety improvement programmes for ‘small series’ WWER-1000 NPPs.

Priority corrective measures important to resolve the issues are discussed under the heading COMMENTS AND RECOMMENDATIONS considering:

- The safety significance of the relevant issues;
- The overall importance of the resolution of this issue in achieving a balanced design and preserving the defence in depth of the plant, taking into account the positive features of the existing design and operational records; and
- The risk reduction effectiveness, as applicable.

The IAEA's view on priority corrective measures includes recommendations on the urgency of actions, comments on the adequacy of corrective measures proposed (including the scope and interrelations to be considered) as well as comments on the completeness of measures proposed or issues not addressed.

Interim measures have been recommended, even if costly, until more cost effective solutions can be adopted based on the full understanding of the underlying safety concern. In this respect, interim actions in operations and software-oriented activities have been recommended for implementation in the short term to improve the situation.

Rank IV issues, judged to be not applicable to WWER-1000 NPPs, either require immediate actions as compensatory measures to justify continued operation in the short term or a plant shutdown has to be considered if immediate actions to compensate for the safety concern are neither effective nor feasible.

The REFERENCES quoted in each issue and listed at the end of this report include not only publications as usually referred to in scientific journals, but also reports to the Steering Committee or material made available to the IAEA within the framework of the IAEA technical co-operation programmes, prepared under IAEA contracts or supplied by the experts participating in the IAEA consultants meetings. In some cases special work has been done by the regulatory authorities of a country or the NPP staff to verify and complete the information contained in the draft versions of this publication and has been sent to the IAEA in written form. Such documents are also listed. To a large degree, the information presented in this publication has been obtained directly from the experts participating in the meetings, either during the meeting or later in the form of written comments and contributions. Whenever information is given without reference to a specific published document, it should be understood to refer to the direct input from IAEA consultants provided during or after the meetings.

The approach described aims at providing guidance for the experts to rank safety significance of issues in a systematic and consistent way. It is recognized, however, that a degree of subjectivity will always be present in the experts' judgement. Realizing this, the IAEA has assured the participation of experts from different countries in these activities, so that the final judgement reflects current international practice.

The safety issues of this publication are generally applicable to each individual ‘small series’ WWER-1000 plant. However, due to site dependent specific design features and various improvements made in different units in the past, the PLANT SPECIFIC STATUS may differ remarkably.

The plant specific status so far made available to the IAEA for each plant is attached to each safety issue since there are only five units at three sites. However, this information should be further completed, updated and included in the IAEA database for use by Member States.

2. OVERVIEW OF SAFETY ISSUES

The approach described in the previous section has been used for the identification of generic safety issues and the ranking according to their safety significance. The impact of each issue on plant safety has been considered individually and compiled in Sections 3 and 4.

It is important to assess the combined impact of all relevant safety issues on plant safety. This overview of safety issues is necessary to develop an integrated action plan for implementation of corrective measures. The integrated action plan is to ensure the completeness of corrective measures, to ensure that they do not adversely interact with each other, and that their implementation schedule guarantees safety during the safety improvement process.

The evaluation of the impact of all issues on plant safety considers their impact on the capability of the main safety functions in case of demand, and on the plant's defence in depth. Consequently, the overview of generic safety issues is presented according to their impact on the main safety functions in different operational conditions as follows:

- Controlling the power by shutting down the reactor, and maintaining safe shutdown conditions during normal operation and in transient and accident conditions;

- Cooling the fuel in all conditions during normal operation, during transients, after loss of coolant accidents (LOCAs) and during shutdown or refuelling; and

- Confining the radioactive material during normal operation and accident conditions.

In addition, the relevant design issues in some areas will be considered separately because they affect nearly every system or equipment in a plant and thus influence the capability of all main safety functions. These issues are related to the following areas:

- Classification and qualification of components,
- Supporting systems (I&C, electrical power, water cooling and ventilation systems),
- Internal hazards,
- External hazards, and
- Accident analysis.

The safety review mission (SRM) to South Ukraine NPP and the technical visits (TV) to Novovoronezh NPP and Kalinin NPP have identified those differences in design and operating practices that directly influence the safe operation of ‘small series’ WWER-1000 NPPs.

This section concludes on the main topics in which safety improvements are needed in most ‘small series’ WWER-1000 plants.

Finally, Table I presents the safety issues by areas, Table II lists the individual issues and their ranking and Table III compares the design differences between WWER-1000 models.

2.1. CONTROLLING THE POWER

To ensure this main safety function, means have to be provided to prevent unacceptable reactivity transients and to shut down the reactor as required. Anticipated operational occurrences should be prevented from leading to accident conditions. For accident conditions including LOCA, reactor shutdown is necessary to permit acceptable cooling of the reactor core.

Therefore, the influence of the safety issues identified on this main safety function will be considered in normal operation, during transient and accident conditions and in shutdown conditions. The features and systems affected by all relevant issues will be discussed to evaluate the safety function capability in case of demand.

The reactivity control is performed by changing the boron concentration in the coolant and by actuating the reactor control and protection system which is used for both normal operation and for emergency scram. The WWER-1000 models 338 and 320 have sixty-one cluster-type control rods divided into 10 groups, each group, except group number 5, consisting of six clusters. At the prototype WWER-1000 model 187, there are 109 cluster-type control rods divided into 14 groups. The model 302 has 49 cluster-type control rods.

2.1.1. Controlling the power in normal operating conditions

This subsection deals with the plant capability to shut down the reactor during normal operating conditions.

The control rods are usually in an automatic mode of operation, but manual operation of control rods is possible.

TABLE I. SAFETY ISSUES AND CATEGORIES BY AREAS FOR 'SMALL SERIES' WWER-1000 NPPs

	Categories				
Area	I	II	III	IV	Not Ranked
Design					
General	-	2	1	-	
Reactor core	1	2	-	-	
Components integrity	-	3	4	-	
Systems	5	8	4	-	
I&C	4	6	1	-	
Electrical power supply	2	2	1	-	
Containment	1	-	-	-	
Internal hazards	1	6	1	-	
External hazards	1	2	-	-	
Accident analysis	7	7	-	-	
Sub-total	22	38	12	-	
Operation					
Operating procedures	-	-	-	-	3
Management	-	-	-	-	4
Plant operation	-	-	-	-	3
Radiation protection	-	-	-	-	1
Training	-	-	-	-	1
Emergency planning	-	-	-	-	1
Sub-total	-	-	-	-	13
Total	22	38	12	-	13

The criticality and power control is realized by lifting and inserting control rod clusters. At the nominal power, only the one group is used to control power which is performed in two modes: regulation of neutron power ("N" regime) and main steam header pressure stabilization ("T" regime) which can pass into each other automatically. The makeup system also has the function of compensating for reactivity by adjusting the boron concentration of the primary coolant.

The control rod insertion problem identified with respect to drop times of control rods exceeding the design limits and the appearance of excessive water gaps between fuel assemblies associated with fuel bow basically reflect deficient core design, in particular the fuel assembly design, which was not validated for 3 years of operation. The interim measures established so far ensure the design limits to shut down the reactor. But these interim measures have placed constraints on the power level for operation and on the full implementation of a low leakage core loading pattern.

There are no plans to operate the 'small series' units in the load follow mode of operation. The measures proposed to automatically protect the core against adverse power distribution in the base load mode can be easily realized within the I&C upgrading programme in Member States.

Safety issues related to this subsection are:

RC 1	Prevention of inadvertent boron dilution (II)
RC 2	Control rod insertion reliability/Excessive water gaps between fuel assemblies (II)
I&C 1	I&C reliability (II)
I&C 3	Automatic reactor protection for power distribution and DNB (I)
I&C 4	Human engineering of control rooms (II)
I&C 5	Reactor protection system redundancy (III)
El 5	Ground faults in DC circuits (II)
Oper. Pro. 01	Procedures for normal operation
Oper. Pro. 03	Limits and conditions

2.1.2. Controlling the power in transient and accident conditions

This subsection deals with the plant capability to shut down the reactor during transient and accident conditions.

The most important safety system for controlling power during transients and accidents is the control and protection system. Furthermore, the makeup system, the high pressure boron injection system, the high pressure injection systems, the accumulators with borated water and the low pressure injection system are activated respectively, depending on the transient and accident scenarios to achieve shutdown conditions.

The capability of achieving shutdown conditions by means of the control and protection system during transients and accidents was affected by the control rod insertion reliability at SNPP and Kalinin NPP (KNPP). This issue is not applicable to NNPP which has a different fuel assembly design with an outside shroud providing higher stiffness against fuel assembly bow. The capability of the scram system to cope with fast transients with rapid reactivity changes was impaired due to insertion times exceeding the design limit or even due to rod sticking. Transients of this type can be caused by initiating events such as control rod withdrawal of banks or single control rod ejection, inadvertent boron dilution, main steam line break with fast temperature decrease of the primary coolant or single control rod drop or insertion. All relevant transient scenarios have been carefully analysed both to demonstrate staying within the design limits and to justify interim corrective measures until the issue is fully resolved.

Controlling the power in accident conditions will also be affected in case a leak occurs in the heat exchangers of the low pressure injection system and service water enters the safety injection system. If the primary circuit is depressurized under LOCA conditions, a certain amount of unborated water can slowly or suddenly be introduced into the primary circuit, depending on the operating conditions of the low pressure injection system.

Results of accident analysis of the main steam line break showed that the reactor returns to criticality after scram, which is a deviation from current Russian standards.

In small break LOCA or anticipated transients without scram (ATWS) events, where the water level in the reactor pressure vessel may decrease below the hot leg elevations, there is a potential of so-called “inherent” boron dilution accidents. Steam condensation in the SG may lead to a cold leg loop seal of very low boron concentration which could enter the core and cause a steep power increase. These and other transient and accident scenarios with reactivity increase have not been analysed systematically so far.

Safety issues related to this subsection are:

RC 1	Prevention of inadvertent boron dilution (II)
RC 2	Control rod insertion reliability/ Excessive water gaps between fuel assemblies (II)
S 7	Emergency core cooling system (ECCS) heat exchanger integrity (II)
I&C 1	I&C reliability (II)
I&C 3	Automatic reactor protection for power distribution and DNB (I)
I&C 4	Human engineering of control rooms (II)
I&C 8	Accident monitoring instrumentation (II)
I&C 9	Technical support centre (II)
El 3	On-site power supply for incident and accident management (II)
El 4	Emergency battery discharge time (III)
AA 5	Main steamline break analysis (I)
AA 9	Severe accidents (I)
AA 10	Probabilistic safety assessment (PSA) (I)
AA 11	Boron dilution accidents (I)
AA 12	Anticipated transient without scram (ATWS) (II)
Oper. Pro. 02	Emergency operating procedures
Oper. Pro. 03	Limits and conditions

2.1.3. Maintaining the reactor in safe shutdown conditions after all shutdown actions

Controlling the power in low power and shutdown conditions (LPS) was found to be insufficiently analysed as compared with international practice. According to generic observations from PSA studies made for different plant types worldwide, accidents during LPS conditions contribute remarkably to the core damage risk. Subcriticality monitoring during reactor shutdown conditions was found not to be in accordance with international practice.

Under shutdown conditions (cold shutdown), the normal circulation of borated water in the RCS is significantly decreased, which reduces the chances of detection of inadvertent boron dilution. At the same time, the maintenance operations conducted in auxiliary circuits increase the chances of undetected penetration of unborated water into the makeup system through, e.g. pump sealing systems or leakages in heat exchangers.

Safety issues related to this subsection are:

RC 1	Prevention of inadvertent boron dilution (II)
RC 3	Subcriticality monitoring during reactor shutdown conditions (I)
S 7	ECCS heat exchanger integrity (II)
AA 8	Accidents under low power and shutdown (LPS) conditions (II)
AA 9	Severe accidents (I)
AA 10	Probabilistic safety assessment (PSA) (I)
AA 11	Boron dilution accidents (I)
Oper. Pro. 01	Procedures for normal operation

2.2. COOLING THE FUEL

To ensure this main safety function, means have to be provided to remove the residual heat from the core during normal operation and accident conditions as well as after a reactor shutdown.

The impact of the safety issues on the main safety function will be discussed in separate subsections, for operating and transient conditions in the subsection 2.2.1, for LOCA conditions in subsection 2.2.2, and for the long term residual heat removal in reactor shutdown conditions in subsection 2.2.3.

2.2.1. Cooling the fuel in operating and transient conditions

The process of decay heat removal and cooling of the primary circuit is carried out during the first cooling phase by the secondary circuit. Steam is released via a bypass-to-condenser system (BRU-K), via the relief valves (BRU-A) in case that off-site power is lost; or via the steam generator safety valves in case of failure in the other systems. If the feedwater system is not available, then feedwater supply is provided by the emergency feedwater system, in particular in case of loss of off-site power. The auxiliary feedwater system is used in the startup and shutdown stages. At NNPP there is no emergency feedwater system; therefore the auxiliary feedwater system is also used for emergency case. In the hot shutdown state, the decay heat is removed via two valves (BRU-TK) to the technical condensers or via the BRU-A if the technical condensers are not available.

Maintaining and monitoring the integrity of the reactor coolant pressure boundary

The main safety function cooling the fuel in operating and transient conditions can be fulfilled only if the integrity of the reactor coolant boundary is ensured. This includes monitoring of component integrity.

In this respect, the degraded steam generator collector was found to be the weakest element of the reactor coolant boundary. Manufacturing technology problems and "environmentally assisted cracking" were identified to be the possible causes for the weakness of steam generator collectors having a potential for primary to secondary leaks. Inadequate secondary water chemistry could lead to stress corrosion cracking of the collector. Additionally, the supply of cold water from the emergency feedwater system can produce unacceptable stresses in the degraded sections of the steam generator.

The second component of high safety concern is the reactor pressure vessel. At most of the plants, the irradiation embrittlement could progress faster than anticipated due to a higher concentration of up to 1.9 wt.% of Ni in welds' alloy. The containers with specimens have been placed such that the data obtained are not fully applicable to monitor embrittlement in the vessel wall. Therefore, the vessel status prediction is limited. Further, the humidity monitoring system in the upper reactor block is not sensitive enough to detect the leaks in the bolted joints of RPV head penetrations.

The applicability of the leak before break (LBB) concept to the primary piping which is not equipped with well designed constraints should be demonstrated.

Reliable in-service inspection (ISI) is a key element to ensure primary circuit integrity. The ISI is carried out according to the individual Member States' requirements, which are in principle based on former Soviet Codes and Standards, using various techniques and tools.

Several deficiencies have been identified related to RPV inspection from outside, testing of the underclad region, and steam generator collectors and tubing. The accessibility of some locations of the primary circuit to perform volumetric examinations is restricted, such as a RPV weld, vessel head and its penetrations, piping welds, steam generator shell welds, and specific piping nozzles. The examination techniques have to be either modified or compensatory measures implemented.

Recent results of a programme similar to PISC indicate insufficient reliability of NDT methods, tools and personnel.

Safety issues related to this topic are:

CI 1	RPV embrittlement and its monitoring (III)
CI 2	Non-destructive testing (III)
CI 3	Primary pipe whip restraints (II)
CI 4	Steam generator collector integrity (III)
CI 5	Steam generator tube integrity (II)
CI 7	Structural integrity related monitoring (II)
S 1	Primary circuit cold overpressure protection (II)
S 2	Mitigation of a steam generator primary collector break (II)
I&C 6	Condition monitoring for the mechanical equipment (I)
I&C 7	Primary circuit diagnostic systems (II)
I&C 10	Water chemistry control and monitoring equipment (primary and secondary) (I)
AA 6	Overcooling transients related to pressurized thermal shock (II)
AA 7	Steam generator collector rupture analyses (II)
AA 9	Severe accidents (I)
AA 10	Probabilistic safety assessment (PSA) (I)
Oper. Pro. 02	Emergency operating procedures
Plant Oper. 02	Surveillance programme

Primary pressure control

During normal operation, pressure control is carried out by pressurizer heaters and by spraying in the steam section of the pressurizer. If the main coolant pump of the loop 1 fails, spraying is performed by the auxiliary spray line of the makeup system, as long as there is no loss of off-site power. During transients, protection of the primary circuit against overpressure is also provided by the three safety valves of the pressurizer. The safety concern concerning overpressure control is related to the qualification of these safety valves for water flow.

In spite of the fact that some measures have been taken so far to prevent the risk of cold overpressurization, the protection needs to be improved to ensure that primary pressure is always below the permissible pressure for each value of the primary temperature during cold shutdown.

Safety issues related to this topic are:

S 1	Primary circuit cold overpressure protection (II)
S 4	Pressurizer safety and relief valves' qualification for water flow (II)
El 3	On-site power supply for incident and accident management (II)
AA 5	Main steamline break analysis (I)
AA 6	Overcooling transients related to pressurized thermal shock (II)
Oper. Pro. 02	Emergency operating procedures

Decay heat removal via the secondary system

The steam generators play a central role in cooling the core in operating and transient conditions. Therefore, the feedwater supply must be ensured in all conditions, i.e. the steam generator inventory must be preserved and common cause failures should not endanger feeding of SGs.

Improper protection of emergency feed water lines against dynamic effects may lead to reduced feed water supply. In spite of the fact that sufficient water resources are available, clear procedures to safely manage the situation are not available.

In case there is a total loss of decay heat removal via the secondary system in some other scenarios beyond DBA, primary "feed and bleed" should be made available as an ultimate option. There are no emergency operating procedures using ECCS and the pressurizer relief and safety valves which are not qualified for steam-water mixture or water flow.

Safety issues related to this topic are:

S 4	Pressurizer safety and relief valves’ qualification for water flow (II)
S 9	Steam generator safety and relief valves' qualification for water flow (II)
S 10	Steam generator relief valves' performance at low pressure (II)
S 11	Steam generator level control valves (I)
S 12	Ventilation system of control rooms (II)
S 15	Feedwater supply vulnerability (III)
I&C 8	Accident monitoring instrumentation (II)
AA 13	Total loss of the electrical power (II)
IH 7	Protection against dynamic effects of main steam and feedwater line breaks (II)
Oper. Pro. 01	Procedures for normal operation
Oper. Pro. 02	Emergency operating procedures

2.2.2. Cooling the fuel in LOCA conditions

During normal operation, leakages of the primary coolant are compensated by the makeup system. In case of loss of coolant accidents which cannot be compensated by the makeup system, the emergency core cooling system is activated. The emergency core cooling system ensures reflooding of the reactor core with cold borated water to remove residual heat in LOCA conditions, in addition to the functions already discussed in subsections 2.1.2 and 2.2.1. The emergency core cooling system consists of a high pressure boron injection system, high pressure injection system, a set of four medium pressure accumulators and the low pressure safety system, each, except the accumulators, having three redundant trains.

However, all three redundant trains are located in one room under the containment floor without physical separation.

The suction line of the high pressure injection pump is also connected to a heat exchanger on the downstream side (except the model 187). When the ECCS tanks reserve is depleted, water from the containment sump is used to supply the high pressure injection pumps. The borated water in the ECCS tanks is preheated to a temperature of 55°C in normal operation. To cope with LOCAs, the sprinkler system is also demanded to reduce the containment pressure.

Residual heat removal in case of a primary to secondary leak

A major primary to secondary leak due to a steam generator collector break would quickly overfill the steam generator and the main steam line which has not been thoroughly demonstrated to be qualified for hot water load. There is a potential for two scenarios which could lead to a bypass of the containment and the loss of the long term core cooling due to loss of primary water to the environment. Either there is a steam line break before the main steam isolation valves outside the containment or the BRU-A valve not qualified for water flow may fail to reclose and cannot be isolated by valve in the model 338 units. Insufficient EOPs to cope with these scenarios would endanger the risk of core damage and radioactive release to the environment. These beyond DBA scenarios and their consequences have not been analysed in sufficient detail so far.

Safety issues related to this topic are:

CI 2	Non-destructive testing (III)
CI 4	Steam generator collector integrity (III)
CI 6	Steam and feedwater piping integrity (III)
S 2	Mitigation of a steam generator primary collector break (II)
S 9	Steam generator safety and relief valves' qualification for water flow (II)
I&C 10	Water chemistry control and monitoring equipment (primary and secondary) (I)
IH 7	Protection against the dynamic effects of main steam and feedwater line breaks (II)
AA 7	Steam generator collector rupture analysis (II)

AA 9	Severe accidents (I)
AA 10	Probabilistic safety assessment (PSA) (I)
Oper. Pro. 02	Emergency operating procedures

Residual heat removal in case of ECCS failure

In the initial phase of a LOCA, the primary pressure is higher than that of the essential service water system. If the heat exchangers of the low pressure injection systems are damaged, there could be an ingress of primary water into the essential service water system.

During the LOCA, the high energy steam or water jets could tear off the thermal insulation from the surrounding equipment. This insulation material could clog the filters of the sump screens and/or the heat exchangers, preventing core cooling in the recirculation phase. Even if this problem is not solved internationally, every effort should be made to improve the situation.

If there is a passive failure either in the tank or in any of the three suction lines of low pressure injection pumps, though of low probability, the coolant inventory will be reduced. It threatens the core cooling, bypass the containment and there is a potential of ECCS failure due to common cause.

Insufficient cooling over the long term of the main coolant pump seal, due to loss of makeup flow and a failure of the emergency pump in the autonomous cooling circuit to provide primary water cooled by essential service water, could lead to a primary leak of the seal. If the seal survival after loss of seal injection flow cannot be demonstrated, the makeup pumps have to be backed up by diesel generators and the loss of makeup flow at containment isolation has to be prevented.

Safety issues related to this topic are:

S 3	Reactor coolant pump seal cooling system (I)
S 5	ECCS sump screen blocking (III)
S 6	ECCS suction line integrity (I)
S 7	ECCS heat exchanger integrity (II)
S 16	Physical separation and functional isolation of the ECCS (III)
S 17	Limited boric acid storage for HP injection (II)
IH 1	Systematic fire hazards analysis (II)
AA 13	Total loss of the electrical power (II)
AA 14	Total loss of heat sink (II)
Oper. Pro. 02	Emergency operating procedures

Residual heat removal capability in beyond design basis accident conditions

There are shortcomings which reflect deviations from current safety standards such as OPB-88 and PBJa-89, which evolved over the last two decades since the design of WWER-1000 plants. Provisions to cope with severe accident situations are insufficient, e.g. lack of RPV level indication, lack of accident monitoring instrumentation, deficit emergency power supply to manage emergency situations, and qualification of electrical and I&C equipment for LOCA conditions. The quality of cable connections has been realized as a generic problem, which is of high safety concern under emergency conditions.

Other issues of safety concern are related to the design of control rooms, i.e. the main control room (MCR) and the emergency control room (ECR). First, their ventilation systems are not independent, so that in the case of an emergency, both rooms may be lost (subsection 2.4.2). Secondly, the design of MCR and ECR does not comply with international practice of human engineering for presenting information to cope effectively with emergency situations. Further, a technical support centre is not available at the sites.

Safety issues related to this topic are:

G 2	Qualification of equipment (III)
S 12	Ventilation system of control rooms (II)
I&C 4	Human engineering of control rooms (II)
I&C 8	Accident monitoring instrumentation (II)
I&C 9	Technical support centre (II)
El 3	On-site power supply for incident and accident management (II)
El 4	Emergency battery discharge time (III)
AA 9	Severe accidents (I)
AA 10	Probabilistic safety assessment (PSA) (I)
Oper. Pro. 02	Emergency operating procedures

2.2.3. Cooling the fuel during cold shutdown or refuelling

This subsection deals mainly with the capability of the main safety function to ensure long term cooling of the fuel.

Long term cooling in the cold shutdown state is carried out using the low pressure safety injection system. The heat exchangers of the low pressure safety injection system are cooled by the essential service water system, which is also used for cooling the safety relevant components. Since the low pressure safety injection system is connected to the primary circuit, appropriate pressure and temperature conditions have to be ensured to avoid additional loads, such as thermal shock, to the primary circuit.

The residual heat removal from the core in cold shutdown and the primary circuit via the low pressure safety system to the ultimate heat sink has some shortcomings. There is no intermediate cooling system for models 187 and 302 if compared with Western PWRs. The heat exchangers of the low pressure safety injection system and their integrity are therefore vulnerable elements, which could also affect core cooling under certain accident conditions (subsection 2.2.2).

Safety issues related to this subsection are:

S 7	ECCS heat exchanger integrity (II)
S 16	Physical separation and functional isolation of the ECCS (III)
AA 8	Accidents under low power and shutdown (LPS) conditions (II)
AA 14	Total loss of heat sink (II)
Oper. Pro. 01	Procedures for normal operation
Oper. Pro. 02	Emergency operating procedures

2.3. CONFINING THE RADIOACTIVE MATERIAL

The full pressure containment of the WWER-1000 NPP accommodates the components of the primary circuit and the spent fuel pool. The containment is the ultimate third barrier to enclose the radioactive releases of the primary circuit in case of an accident. The containment has to withstand pressures (5 bar) and temperatures (150°C for 24 hours) under accident conditions. Leak tightness has to be ensured, i.e. the maximum leak rate shall not exceed 0.1% per day of the volume of the air within the containment at maximum accident pressure. The containment of the WWER-1000 plant is designed with a sprinkler system to limit pressure and temperature under LOCA conditions. The containment isolation system prevents any radioactive release into the environment by isolating all systems which penetrate the containment and which are not necessary to control the accident.

2.3.1. Confining the radioactive material during operating conditions

No issues have been identified which would lead to releases which are not within prescribed limits during normal operation.

2.3.2. Confining the radioactive material during accident conditions

This main safety function was found to be affected by the potential to bypass the containment and insufficient protection of the ultimate barrier.

Containment bypass

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

A similar situation may arise as discussed in subsection 2.2.2 in the initial phase of a LOCA with damaged heat exchangers of the low pressure injection system. Then, radioactivity from the primary coolant will be discharged to the essential service water system, thus bypassing the containment.

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

Containment integrity

In the 'small series' WWER-1000 plant design, the hydrogen removal system was not considered for use during DBA-LOCA and BDBA. The means to protect the ultimate barrier in emergency situations are not sufficient.

Safety issues related to this subsection are:

CI 4	Steam generator collector integrity (III)
CI 6	Steam and feedwater piping integrity (III)
S 4	Pressurizer safety and relief valves' qualification for water flow (II)
S 6	ECCS suction line integrity (I)
S 7	ECCS heat exchanger integrity (II)
S 13	Hydrogen removal system (II)
Cont. 1	Containment bypass (I)

2.4. OVERVIEW OF SAFETY ISSUES AFFECTING PERFORMANCE OF ALL SAFETY FUNCTIONS AND OVERALL PLANT SAFETY

2.4.1. Classification and qualification of components

The components for 'small series' WWER-1000 NPPs were designed based on OPB-73, associated standards/rules available at the design stage. The components important to safety were classified according to their functions as for normal operation (safety-related), for protective actuation, for accident localization, for safety systems and for safety support systems. OPB-88, which came into force in July 1990, defines a new classification system with 4 safety classes for components used in NPPs. The "Rules of Construction and Safe Operation of Equipment and Pipelines of NPPs" classifies pressure retaining components into three groups A, B and C.

Since the safety class is the essential factor in determining other classifications related to seismic, quality, etc. a retrospective review is recommended to be carried out and to identify deviations with respect to current requirements. Backfitting or compensatory measures should be developed and implemented if necessary.

The re-qualification of safety related components should be performed to demonstrate their ability to fulfil their functions. This practice of qualification of safety-related components is not evident at the WWER-1000 plants.

Reliability analyses of safety class 1 and 2 systems are necessary to confirm that the systems are as reliable as expected by the designer.

Safety issues related to this subsection are:

G 1	Classification of components (II)
G 2	Qualification of equipment (III)
G 3	Reliability analysis of safety class 1 and 2 systems (II)

2.4.2. Supporting systems (I&C, electrical power supply, water cooling and ventilation system)

Instrumentation and control reliability

Both I&C for safety systems and I&C for safety related systems are covered by the I&C systems important to safety. In 'small series'WWER-1000 NPPs, the I&C safety classification should be completed and approved for both cases, and should be carried into the procedures followed for maintenance of the I&C important to safety. The qualification list of I&C equipment important to safety should be reviewed for all design conditions. This practice of qualification is not evident at the 'small series' WWER-1000 plants.

The I&C equipment of 'small series' WWER-1000 units has a design lifetime of ten years. The first units which went into operation have reached the end of their design life. The failure modes found include relay contact oxidation, low insulation resistance of wiring and terminals, etc. Without major efforts in maintenance, the I&C reliability may have a serious impact on safety. Another problem is the poor quality of cable connections inside the containment. The cable connections are not able to withstand extreme environmental conditions, thus having a high failure potential under LOCA conditions.

I&C for protection system and safety actuation systems

Safety improvements can be made in the reactor protection system by installing additional scram signals of protection against high linear power density, DNB and high pressurizer level. The ECCS actuation circuits in the safety actuation systems are based on an energize-to-actuate principle.

Control room I&C design

The control room I&C design has deficiencies in the human factors in comparison with the most modern international practice. The design of the information display in the control rooms does not give the operator a rapid overview of information regarding the current state of plant and reactor safety as a whole. The accident monitoring instrumentation in the control rooms is not properly designed. The information needed during and after an accident is distributed throughout the control room and is not organized in a way that would support rapid and accurate diagnosis of an accident condition. Reactor pressure vessel level indication is currently not provided, and the level could be estimated only by indirect means. Recent international practice is to design an NPP with a room (technical support centre) where current plant data and status is compiled for display to technical experts who will support the operators during the management of an accident. The 'small series' WWER-1000 operating units have to be upgraded to add technical support centres.

I&C for monitoring and diagnostics

The I&C for monitoring and diagnosing the state of systems important to safety needs to be improved. The original design of 'small series' WWER-1000 units does not provide for adequate diagnostic systems to monitor the reactor coolant pressure boundary integrity and the mechanical equipment. The chemical monitoring system currently used is of the 1970's era. Assurance of reliable and accurate results requires a lot of maintenance effort. An accurate and preferably on-line chemical monitoring system is important to give the operator a possibility for a timely response to deviations in primary and secondary coolant water chemical condition indices. The current in-core monitoring and control system can detect and provide information for the operator to suppress xenon oscillations in baseload operation and for infrequent power changes. However, if the plant is to be used in load follow mode, the system needs to be improved.

Safety issues related to the I&C are:

I&C 1	I&C reliability (II)
I&C 2	Safety system actuation design (I)
I&C 3	Automatic reactor protection for power distribution and DNB (I)
I&C 4	Human engineering of control rooms (II)
I&C 5	Control and monitoring of power distributions in load follow mode (II)
I&C 6	Condition monitoring for the mechanical equipment (I)
I&C 7	Primary circuit diagnostic systems (II)
I&C 8	Accident monitoring instrumentation (II)
I&C 9	Technical support centre (II)
I&C 10	Water chemistry control and monitoring equipment (primary and secondary) (I)
I&C 11	Separation of the primary circuit instrumentation taps to I&C detectors (II)

Electrical power supply

The issues identified in the area of electrical power are related to diesel generators, diesel backed power supply, ground faults in DC circuits, emergency batteries and the qualification of I&C and electrical equipment for LOCA conditions. Most of these issues were identified also from the standard model 320.

The failure frequency of diesel generators is higher than expected in plant design. It is recommended that this reliability be increased by investigating the failure causes and by taking appropriate corrective actions.

The protection signals required to trip the diesel, avoiding heavy damage to the diesel, are considered inadequate. The measure proposed for Ukrainian NPPs for a three chain structure with a two out of three configuration for diesel protection signals is considered reasonable to increase diesel reliability.

There are several safety relevant systems without diesel backed power supply. These are systems which would be needed for proper management of incidents that entail complete loss of off-site power supply and necessitate plant cooldown to cold shutdown state.

An issue with a high safety concern is related to emergency batteries and their discharge time. The 'small series' WWER-1000 plants have three redundant batteries to provide energy, as the ultimate energy source, to vital loads. However, the designed discharge time is only in the order of 15 to 20 minutes, and this is not in compliance with modern requirements. It is recommended that the battery discharge time be increased to the order of 2 to 3 hours. A further concern is the lack of battery circuit monitors to automatically recognize galvanic interruptions within the battery.

The qualification of electrical and I&C equipment for LOCA conditions cannot be checked, since neither specifications regarding test procedures and sequences nor test reports are available at the sites.

Safety issues related to electrical power supply are:

G 2	Qualification of equipment (III)
El 1	Diesel generator reliability (I)
El 2	Protection signals for emergency diesel generators (I)
El 3	On-site power supply for incident and accident management (II)
El 4	Emergency battery discharge time (III)
El 5	Ground faults in DC circuits (II)

Water cooling systems

All systems important to safety are cooled by the essential service water system while the equipment which are not important to safety are cooled by the non-safety related service water system. An autonomous cooling system which operates as a closed loop to cool the main coolant pumps is cooled also by the essential service water system. The essential service water system consists of three independent trains and operates as a semi-open circuit for model 302. A two channel makeup system, not backed up by diesels and common to the three trains, is used to compensate for loss of water due to evaporation.

Proper operation of the heat sink depends on the capacity of the spray ponds, which constitute the heat sink water reserve of the site.

In case of loss of the makeup system, the spray ponds ensure heat removal for a period of 30 hours with extreme external summer temperatures and without exceeding a temperature of 33°C.

Loss of the essential service water system leads to a loss of the decay heat removal function via the secondary and the primary side as well. Both the emergency feedwater pumps and heat exchangers of the low pressure injection system are cooled by essential service water. Consequently, loss of essential service water leads to an unacceptable situation. The means and procedures to cope with total loss of heat sink are not available at 'small series' WWER-1000 units.

Safety issues related to water cooling are:

S 3	Reactor coolant pump seal cooling system (I)
S 7	ECCS heat exchanger integrity (II)
AA 14	Total loss of heat sink (II)
El 3	On-site power supply for incident and accident management (II)
IH 1	Systematic fire hazards analysis (II)
IH 5	Systematic flooding analysis (I)

Ventilation systems

Ventilation systems are necessary to cool compartments housing safety and safety related systems so as to keep air parameters within admissible limits. In particular, the ventilation of the main control room (MCR) and the emergency control room (ECR) should be designed to provide habitability. In the original design of all WWER-1000 plants, the MCR and ECR do not have their own separate independent and safety graded ventilation systems to ensure habitability in an emergency case.

Safety issues related to ventilation are:

S 12	Ventilation system of control rooms (II)
AA 9	Severe accidents (I)
IH 1	Systematic fire hazards analysis (II)
EH 1	Seismic design (II)

2.4.3. Internal hazards

The main issues identified in this area are related to fire protection, internal flooding, dynamic effects resulting from pipe breaks and the risk of dropping heavy loads on to the reactor or the spent fuel pool.

Fire protection is considered to be an especially important topic, since operating experience with nuclear power plants worldwide has shown that the possibility of fires cannot be fully excluded and the risk of a fire leading to a major event is not sufficiently low. According to the NUSS requirements, an adequate degree of fire protection should be achieved by a defence in depth concept in the design. A key element of this concept is the performance of a systematic fire hazards analysis prior to initial loading of reactor fuel and updating this analysis during operation. This would enable the determination of the required fire resistance of the fire compartment boundaries and requirements of the fire extinguishing systems and other features necessary to fulfill the fire protection requirements.

A systematic fire hazards analysis, as discussed here, has not been performed so far for any of the small series WWER-1000 nuclear power plants. As a consequence, the defence in depth concept of fire protection is lacking. This includes identified weaknesses in passive fire protection in general and in the cable spreading room in particular. A further safety concern in conjunction with fire protection is the possibility that all the 6 kV main distribution boards can simultaneously fail in the event of a fire, since they are not separated by fire barriers.

The occurrence of internal hazards resulting from high energy pipe breaks is an issue of high safety concern. The dynamic effects of high energy pipe breaks, such as pipe whips and jet forces due to the sudden release of liquids and steam, could lead to multiple failures of safety related equipment. Dynamic effects associated with a main coolant pipe double ended break could lead to a damage of two steam generators, other safety related equipment and structures.

The issue related to the risk of dropping heavy loads is associated with the lack of adequate interlocks in the polar crane in the reactor building. This requires plant specific studies to be carried out to either prevent or to minimize the adverse effects of dropped loads.

Safety issues related to this subsection are:

IH 1	Systematic fire hazards analysis (II)
IH 2	Fire prevention (III)
IH 3	Fire detection and extinguishing (II)
IH 4	Mitigation of fire effects (II)
IH 5	Systematic flooding analysis (I)

IH 6 Protection against flood for emergency power distribution boards (II)
IH 7 Protection against dynamic effects of main steam and feedwater line breaks (II)
IH 8 Polar crane interlocking (II)

2.4.4. External hazards

In the area of external hazards, there are safety concerns with respect to seismic design.

It is international practice, e.g. according to the NUSS requirements, to ensure that structures, systems and components of nuclear power plants are designed such that no safety functions are lost in the case of an earthquake which can be expected at the site. Some pumps from safety systems such as essential service water pumps, fire suppression system pumps and the ventilation equipment as well are not seismically qualified. It is essential that for each plant the earthquake level be determined for all systems, structures and components and that the safe shutdown capability be verified. The necessary upgrading work with respect to the site specific characteristics and parameters should be carried out as soon as possible.

Safety issues related to this subsection are:

EH 1 Seismic design (II)
EH 2 Analyses of plant specific natural external conditions (I)
EH 3 Man-induced external events (II)

2.4.5. Accident analysis

The accident analyses carried out to support design with conservative assumptions are used for the licensing of plants to demonstrate fulfillment of the safety requirements. From the operational point of view, realistic best estimate analyses are carried out to support plant operation, including training, preparation of operating instructions, setting limiting parameters and preparation of emergency procedures. Realistic (best estimate) analyses which are not available for the plant should be carried out to form a basis for development of emergency operating procedures and operator training.

The accident analysis part in the existing updated "Technical Justification of Safety" reports (TOBs) need to be improved on the coverage of the accident spectrum, the boundary conditions, the assumption used, the clarification of acceptance criteria, the quality of analysis and computer code validation.

A comprehensive list of accidents to be analysed needs to be established. Accidents not considered in the TOBs so far include overcooling transients related to pressurized thermal shock, anticipated transients without scram, boron dilution accidents, accidents during low power and shutdown conditions and severe accidents.

Prior to their use, the computer codes and plant model used in the analyses should be validated by experiments or checking with another modern computer code already validated.

Safety issues related to this subsection are:

AA 1 Scope and methodology of accident analysis (II)
AA 2 QA of plant data used in accident analysis (I)
AA 3 Computer code and plant model validation (I)
AA 4 Availability of accident analysis results for supporting plant operation (I)
AA 5 Main steamline break analysis (I)
AA 6 Overcooling transients related to pressurized thermal shock (II)
AA 7 Steam generator collector rupture analysis (II)
AA 8 Accidents under low power and shutdown (LPS) conditions (II)
AA 9 Severe accidents (I)
AA 10 Probabilistic safety assessment (PSA) (I)

AA 11	Boron dilution accidents (I)
AA 12	Anticipated transient without scram (ATWS) (II)
AA 13	Total loss of electrical power (II)
AA 14	Total loss of heat sink (II)
Oper. Pro. 02	Emergency operating procedures

2.5. OPERATIONAL SAFETY

A limited review of operational safety of 'small series' WWER-1000 within the Extrabudgetary Programme on the Safety of WWER and RBMK NPPs was performed for SNPP [2] and NNPP [3]. However, the experts involved in this work could rely on the results of other IAEA activities for standard model 320 plants in The Russian Federation, Ukraine and countries in Eastern Europe.

Thirteen safety issues have been identified in the area of operation. There was some difficulty in applying the criteria of ranking because the proposed approach was more suitable for the design issues, and the available data were insufficient for the ranking. Therefore, the safety issues have not been ranked.

The lack of ranking does not however signify that the operational safety issues are less important than the design safety improvements. The degree of automatization of WWER-1000/320 plants is generally lower than of PWRs and the role of the operator is very important for the safety of these plants. Current international practice in operational safety can always be introduced irrespective of plant type and age. Therefore, the achievement of the highest possible level of operational safety can be argued to be of greater importance for plants where meeting the current design safety standards is not always feasible.

The list of operational safety issues can be considered as preliminary and future reviews of plant operation are planned by the IAEA to finalize it. It would be very important that during these reviews the ranking of the operational safety issues be established in the light of the operating experience of the plants concerned.

There are a number of areas where operational safety can be improved. The recommended measures intend to stimulate the operating organizations and plant management to correct the identified deficiencies and achieve better alignment with international practices. Among the proposals the most important ones are:

- Operating procedures are key elements of plant safety both for the normal and emergency modes of operation. Improving the format and content of normal operating procedures and elaboration of symptom oriented emergency operating procedures are considered to be very important. In particular, senior plant management should evaluate the current use of procedures against international practices and modify the plant's philosophy to be consistent with.

- The justification of limits and conditions of safe operation needs to be developed systematically on the basis of reliability and accident analyses and operational experience, and to be included in the technical specifications.

- Many of the elements of safety culture are present at the WWER-1000 plants. The principles of safety culture should be incorporated into the incident prevention through training and qualification programmes.

- In spite of the wide variety of approaches used to feedback the safety related operational experience several improvements are recommended in this area including the reporting criteria, application of the root cause analysis methodology, setting up of multi-discipline engineering support groups and improving the co-operation between the WWER-1000 operators.

- The importance of quality assurance is generally recognized by the plant management. Review and improvement of the QA programmes are recommended in order to determine department responsibilities and maintain an independent system for verification and approval of all procedures prior to implementation, and to monitor that procedures are followed.

- The management of the plants should devote great attention to the improvement of maintenance procedures and programmes. Errors in maintenance and testing can result in erroneous functioning of safety systems or violation of defence in depth. The records and data related to different plant activities (e.g. maintenance, surveillance tests, backfitting) should be stored so that they are easily accessible and retrievable, preferably with the use of electronic data.

- The surveillance programmes of the plants need to be reviewed and improved to detect degradations or hidden failures, taking into account both equipment history and operating experience feedback and to identify procedural deficiencies. Test intervals should be considered carefully in order to ensure the functionality of equipment and to avoid unnecessary tests which could result in decreasing the equipment availability.

- Operational and maintenance staff needs to be trained to improve the abilities of the personnel to diagnose and manage plant events.

- Without adequately equipped and organized emergency centres, it is not possible to co-ordinate and carry out accident management measures. This could lead to events developing to the stage where they can affect the personnel, the public and the environment. Therefore, it is recommended to construct and equip the emergency centre including procedures and documentation and carry out the necessary drills and exercises.

- The radiation protection practices were reviewed in detail by the IAEA at the Zaporozhe NPP [10]. Generally the radiation protection practices at the WWER-1000 plants are good, and the collective dose of the personnel is kept low. However, the radiation monitoring instrumentation originally designed and supplied needs upgrading to cover the whole range of parameters including accidental conditions.

The upgrade of operating safety is a requirement of utmost importance in improving the nuclear safety. It is therefore imperative that the recommendations related to operational safety be implemented in parallel with the design related safety improvements. This would ensure that a balanced approach is achieved in improving the overall safety.

Safety issues related to operational safety are:

Oper. Pro. 01	Procedures for normal operation
Oper. Pro. 02	Emergency operating procedures
Oper. Pro. 03	Limits and conditions
Man. 01	Need for safety culture improvements
Man. 02	Experience feedback
Man. 03	Quality assurance programme
Man. 04	Data and document management
Plant Oper. 01	Philosophy on use of procedures
Plant Oper. 02	Surveillance programme
Plant Oper. 03	Communication system

Rad. Prot. 01	Radiation protection and monitoring
Training 01	Training programmes
Emerg. Plan. 01	Emergency centre

2.6. MAIN INSIGHTS

The review of the safety features of 'small series' WWER-1000 plants shows that the main safety concept of these reactor is similar to model 320 with respect to the nuclear island arrangement, amount of safety systems and main process parameters of the primary and secondary circuit. However, the 'small series' WWER-1000 plants have deficiencies such as the lack of separation of redundant safety systems and a single set of reactor protection system which does not meet the current national standards and international practice.

Differences in engineering design solutions, quality of manufacture and reliability of equipment revealed as deficiencies by operational experience, as well as differences in the standards and regulations in force at that time in the former Soviet Union and in Western countries subsequently required costly safety backfittings.

The backfitting process is not much different from that which is ongoing on operating plants worldwide, particularly those built to earlier standards. In 'small series' WWER-1000 plants, much of the backfitting and upgrading work has been recognized as necessary and has been or is being performed. This activity was initiated by the WWER Owners Club since the early nineties and international assistance has played a major role in the safety improvement process of 'small series' WWER-1000 NPPs. The status of plant specific backfittings varies in different countries, depending on the requirements of the national regulatory bodies and the available financial means.

The main topics in which safety improvements are needed in 'small series' WWER-1000 plants are the following:

- Physical separation and functional isolation between redundant systems important to safety have not been fully applied due to the lack of proper rules and standards during the design phase of 'small series' WWER-1000 NPPs (e.g. the emergency core cooling systems and emergency feedwater pumps). Electrical and I&C parts of the safety systems could be lost due to common causes, such as flooding, fire or high energy pipe ruptures.

- The one-set system of the reactor protection (except SNPP Unit 1) does not comply with national rules and international practice. This type of design makes the test of protection set during operation impossible.

- Although severe cracking of steam generator collectors has been observed, a collector failure was not originally considered in the design basis. A collector failure would quickly overfill the SG and the main steam line, which has not been demonstrated thoroughly to be qualified for the load of hot water flow. If there is a break of the main steam line before the fast isolation valve outside the containment or if the BRU-A valve unqualified for water flow fails to reclose, the containment will be bypassed and the long term core cooling may be lost due to loss of primary water to the environment. PSA studies indicate further that the loss of the SG integrity would contribute significantly to the core damage frequency (CDF) of WWER-1000 plants. The national approaches on primary to secondary leak treatment need to be developed.

- A potential safety concern in maintaining and monitoring the integrity of the reactor coolant pressure boundary is the reactor pressure vessel. Even if irradiation embrittlement is not a serious problem at the present time, it could progress faster than anticipated due to a higher concentration of Ni in the welds in the beltline. The containers with specimens have been placed such that an appropriate monitoring of embrittlement cannot be ensured. In addition, the problems of pressurized thermal shock and cold overpressure protection of the RPV remain.

- Improvements need to be made in in-service inspection and diagnostic systems which were not sufficiently developed at the time of 'small series' WWER-1000 construction.

- In-depth studies performed to date indicate that the reduction of safety margin, which results from a delay control rod insertion or even sticking, is not a major safety concern as long as it does not occur above the 'dashpot' area. Also associated with fuel bow are water gaps between fuel assemblies which are expected to be safety significant to thermohydraulic fuel safety criteria.

- Some problems were identified much later than the time of 'small series' WWER-1000 plant design and have to be solved not only in WWERs, but also in other plants in the world. An example of this is the danger of containment sump clogging during LBLOCA, with subsequent loss of water to ECCS pumps in the recirculation phase of the accident and possible core melt. Another unsolved problem is the hydrogen removal from the containment atmosphere and its recombination, both under DBA and BDBA conditions.

- Qualification of safety and safety related equipment, not only in I&C and electrical systems, but also in mechanical systems, should be reviewed. In view of the fact that the I&C systems and equipment were installed with the technical level corresponding to the former Soviet Union technology of the 1970s, the standards and regulations to be used for I&C systems design have to be redefined and the equipment exchanged with one that corresponds to contemporary technology.

- Fire protection and fighting capability improvement can be made, both through introduction of improved materials (fire resistant cables, fireproof doors, louvers, non-burning lubricants) and through design improvements (division of turbine hall into smaller fire compartments, removal of safety systems from potentially hazardous fire areas, installation of modern fire fighting systems).

- The necessary basis for the safety improvements programme is a comprehensive safety analysis of each plant, which should lead to the preparation of a complete safety analysis report based on the actual safety requirements and plant configurations.

- The importance of human factors in 'small series' WWER-1000 operation reflects all operational aspects of WWER-1000 that are very significant to safety. The present report does not rank the operational issues.

TABLE II. INDIVIDUAL SAFETY ISSUES AND THEIR RANKING FOR ‘SMALL SERIES’ WWER-1000 NPPs

Issue No.	Issue Title	Issue Rank	Page No. Issue/plant specific status
	AREA: GENERAL		
G 1	Classification of components	II	40/41
G 2	Qualification of equipment	III	43/44
G 3	Reliability analysis of safety class 1 and 2 systems	II	45/45
	AREA: REACTOR CORE		
RC 1	Prevention of inadvertent boron dilution	II	47/48
RC 2	Control rod insertion reliability/ Excessive water gaps between fuel assemblies	II	50/51
RC 3	Subcriticality monitoring during reactor shutdown conditions	I	53/53
	AREA: COMPONENT INTEGRITY		
CI 1	RPV embrittlement and its monitoring	III	55/56
CI 2	Non-destructive testing	III	58/59
CI 3	Primary pipe whip restraints	II	60/60
CI 4	Steam generator collector integrity	III	61/62
CI 5	Steam generator tube integrity	II	63/63
CI 6	Steam and feedwater piping integrity	III	65/66
CI 7	Structural integrity related monitoring	II	67/68
	AREA: SYSTEMS		
S 1	Primary circuit cold overpressure protection	II	70/71
S 2	Mitigation of a steam generator primary collector break	II	72/73
S 3	Reactor coolant pump seal cooling system	I	74/75

TABLE II. (cont.)

Issue No.	Issue Title	Issue Rank	Page No. Issue/plant specific status
S 4	Pressurizer safety and relief valves' qualification for water flow	II	77/77
S 5	ECCS sump screen blocking	III	79/80
S 6	ECCS suction line integrity	I	82/82
S 7	ECCS heat exchanger integrity	II	84/85
S 8	Power operated valves on the ECCS injection lines	I	86/86
S 9	Steam generator safety and relief valves' qualification for water flow	II	88/88
S 10	Steam generator safety valves' performance at low pressure	I	90/91
S 11	Steam generator level control valves	I	92/92
S 12	Ventilation system of control rooms	II	93/93
S 13	Hydrogen removal system	II	95/95
S 14	Boron injection system capability	III	97/97
S 15	Feedwater supply vulnerability	III	99/100
S 16	Physical separation and functional isolation of the ECCS	III	101/102
S 17	Limited boric acid storage for HP injection	II	103/103
	AREA: INSTRUMENTATION AND CONTROL		
I&C 1	I&C reliability	II	105/106
I&C 2	Safety system actuation design	I	108/108
I&C 3	Automatic reactor protection for power distribution and DNB	I	110/110
I&C 4	Human engineering of control rooms	II	112/113

TABLE II. (cont.)

Issue No.	Issue Title	Issue Rank	Page No. Issue/plant specific status
I&C 5	Reactor protection system redundancy	III	114/114
I&C 6	Condition monitoring for the mechanical equipment	I	116/116
I&C 7	Primary circuit diagnostic systems	II	118/119
I&C 8	Accident monitoring instrumentation	II	121/122
I&C 9	Technical support centre	II	123/123
I&C 10	Water chemistry control and monitoring equipment (primary and secondary)	I	125/125
I&C 11	Separation of the primary circuit instrumentation taps to I&C detectors	II	127/127
	AREA: ELECTRICAL POWER		
El 1	Diesel generator reliability	I	129/129
El 2	Protection signals for emergency diesel generators	I	131/131
El 3	On-site power supply for incident and accident management	II	133/134
El 4	Emergency battery discharge time	III	135/136
El 5	Ground faults in DC circuits	II	137/138
	AREA: CONTAINMENT		
Cont. 1	Containment bypass	I	139/140
	AREA: INTERNAL HAZARDS		
IH 1	Systematic fire hazards analysis	II	141/142
IH 2	Fire prevention	III	143/144
IH 3	Fire detection and extinguishing	II	147/148
IH 4	Mitigation of fire effects	II	150/151

TABLE II. (cont.)

Issue No.	Issue Title	Issue Rank	Page No. Issue/plant specific status
IH 5	Systematic flooding analysis	I	152/153
IH 6	Protection against flood for emergency electric power distribution boards	II	154/154
IH 7	Protection against the dynamic effects of main steam and feedwater line breaks	II	156/156
IH 8	Polar crane interlocking	II	158/158
	AREA: EXTERNAL HAZARDS		
EH 1	Seismic design	II	160/160
EH 2	Analyses of plant specific natural external conditions	I	162/162
EH 3	Man-induced external events	II	164/164
	AREA: ACCIDENT ANALYSIS		
AA 1	Scope and methodology of accident analysis	II	167/168
AA 2	QA of plant data used in accident analysis	I	170/170
AA 3	Computer code and plant model validation	I	172/173
AA 4	Availability of accident analysis results for supporting plant operation	I	174/174
AA 5	Main steam line break analysis	I	176/176
AA 6	Overcooling transients related to pressurized thermal shock	II	178/179
AA 7	Steam generator collector rupture analysis	II	180/181
AA 8	Accidents under low power and shutdown (LPS) conditions	II	182/182
AA 9	Severe accidents	I	184/184
AA 10	Probabilistic safety assessment (PSA)	I	186/186
AA 11	Boron dilution accidents	I	188/189

TABLE II. (cont.)

Issue No.	Issue Title	Issue Rank	Page No. Issue/plant specific status
AA 12	Anticipated transients without scram (ATWS)	II	190/190
AA 13	Total loss of electrical power	II	192/193
AA 14	Total loss of heat sink	II	194/194
	AREA: OPERATION		
Oper. Pro. 1	Procedures for normal operation		196/196
Oper. Pro. 2	Emergency operating procedures		197/197
Oper. Pro. 3	Limits and conditions		199/199
Man. 1	Need for safety culture improvements		200/200
Man. 2	Experience feedback		202/202
Man. 3	Quality assurance programme		204/204
Man. 4	Data and document management		206/206
Plant Oper. 1	Philosophy on use of procedures		207/207
Plant Oper. 2	Surveillance programme		209/209
Plant Oper. 3	Communication system		210/210
Rad. Prot. 1	Radiation protection and monitoring		211/211
Training 1	Training programmes		213/213
Emerg. Plan. 1	Emergency centre		215/215

TABLE III. DESIGN DIFFERENCES OF WWER-1000 MODELS

Items	standard model 320	model 187 (NNPP5)	model 302 (SNPP1)	model 338 (SNPP2, KNPP1, 2)
1. Reactor core:				
- Fuel enrichment	4.4 %	4.4%	3.6/4.4%	4.4%
- Refuelling scheme	3 year cycle	3 year cycle	2/3 year cycle	3 year cycle
- Structure of FA	shroudless	with shroud 1.1mm thick	shroudless	shroudless
- Number of FAs	163	317	163	163
- Number of FRs per FA	311	311	311	311
- Number of CA	61	109	49	61
- No. of control elements per CA	18	12	18	18
2. Sets of the reactor protection system	2	1	1	1
3. Main coolant loop	No MCI valves	2 MCI valves per loop	2 MCI valves per loop	2 MCI valves per loop

FA: Fuel assembly, FR: Fuel rod, CA: Control assembly, MCI: Main coolant isolation

TABLE III. (cont.)

Items	standard model 320	model 187 (NNPP5)	model 302 (SNPP1)	model 338 (SNPP2, KNPP1, 2)
4. ECCS systems				
- High pressure injection	3 trains 3×15 m^3 HP tanks (inside Cont.) 3x15m^3 HP boron tanks (outside Cont.)	3 trains 1×150 m^3 HP tank (outside Cont.)	3 trains 1×150 m^3 HP tank (outside Cont.)	3 trains 3×100 m^3 HP tanks (outside Cont.) (KNPP 1: 1x150 m^3 only)
- Low pressure injection	3 trains $1 \times 630m^3$ LP tank (sump, inside Cont.)	3 trains $3 \times 500m^3$ LP tanks (outside Cont.)	3 trains $3 \times 500m^3$ LP tanks (outside Cont.)	3 trains $3 \times 500m^3$ LP tanks (outside Cont.)
- Containment sumps	1	3 (semi-sumps)	3	3
- Physical separation and functional isolation	Yes	No	No	No
- HP injection pumps connected to sumps	Yes	No	Yes	Yes
5. Main steam lines				
- BRU-A has isolation valve	No	Yes	Yes	Yes
- Check valve location	Down-stream MSIV	Down-stream MSIV	Down-stream MSIV	Upstream MSIV
- BRU-A location	Upstream MSIV	Down-stream MSIV	Upstream MSIV	Upstream MSIV
6. High pressure boron injection (PT6 pumps)	Yes	No	Yes	Yes

ECCS: Emergency core cooling system, Cont.: Containment, HP: High pressure, LP: Low pressure
BRU-A: Atmospheric steam dump valve, MSIV: Main steam isolation valve

TABLE III. (Cont.)

Items	standard model 320	model 187 (NNPP5)	model 302 (SNPP1)	model 338 (SNPP2, KNPP1, 2)
7. Feed water systems	AFW & EFW systems, $3 \times 500\ m^3$ EFW tanks	AFW system, No EFW system, $2 \times 1000\ m^3$ AFW tanks	AFW & EFW systems, $3 \times 1000\ m^3$ EFW tanks	No independent AFW & EFW system, $3 \times 1000\ m^3$ EFW tanks
8. Intermediate service water system	Open circuit	Closed circuit	Semi-closed circuit	Closed circuit (for KNPP) Semi-closed (for SNPP 2)
9. Emergency diesel generators	3 separated by buildings	3 separated by walls in one building	3 separated by walls in one building	3 separated by walls in one building

AFW: Auxiliary feed water, EFW: Emergency feed water

3. DESIGN SAFETY ISSUES

3.1. GENERAL

REVIEW AREA/ISSUE NUMBER: General 1 (G 1)

ISSUE TITLE: Classification of components

ISSUE CLARIFICATION: The components for WWER-1000 'small series' NPPs were designed in parallel with the development of OPB-73 [1] and associated standards/rules available at the design stage. According to these standards the components important to safety were to be classified as normal operation system (safety related), protective actuation safety system, localization actuation safety system, protection safety system and support safety system, i.e. they were classified according to their function.

OPB-88 [4], which came into force in July 1990, defines a new classification system for components according to their safety importance.

Pressure retaining components are classified, in accordance with the "Design and Safe Operation Rules for Nuclear Power Plant Components and Piping" [14], into 3 groups A, B, and C with respect to impact on safety of the system of which they are a part.

According to OPB-88 [4], the safety class is considered to be the essential factor in determining other classifications (seismic, quality, etc.) of nuclear power plant elements, as specified in the nuclear power plant safety rules. However, other factors determining classifications are established in rules that prescribe the quality requirements. The highest quality according to the current state of manufacturing technology is ensured by group A. Consequently, safety class 1 always corresponds to group A. Safety classes 2 and 3 may belong to either group B or to group C, depending on other quality classification factors as mentioned above.

The safety concerns with respect to the components of 'small series' WWER-1000 NPPs are firstly, the non-uniform approach to the treatment of components which are expected to perform a given safety function, and secondly, certain deviations of the current requirements at the stages of design, manufacture and pre-and in-service inspections.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified by code comparison. The non-uniform treatment of elements provided to fulfil a given main safety function and deviations from current requirements for manufacturing and testing affect the safety provisions at Level 1 of plants' defence in depth. Consequently, all main safety functions may be impaired for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. Since components important to safety were designed according to old rules, a retrospective review is recommended to be carried out, and deviations with respect to current requirements are recommended to be identified. Backfitting or compensatory measures should be developed and implemented, if necessary.

2. The classification for I&C and electrical equipment should be completed and approved. The equipment that is identified by the OPB-88 classification effort as being part of the safety (protection or actuation) systems should be distinguished from the normal operational systems.

3. A higher level of checking following repairs would be appropriate for equipment performing safety related functions. This identification would reduce the likelihood of maintenance errors contributing to unsafe operation. It will also help to prevent the degradation of the independence of safety related channels resulting from errors in future modifications.

4. The maintenance, surveillance and in-service inspection procedures should be reviewed and modified, if necessary, in order to ensure compliance with the classification requirements for the related components and systems.

REFERENCES: [1, 2, 3, 4, 11, 14]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A re-evaluation of the classification according to OPB-88 (quality, safety, seismicity) has been made by Atomenergoproject. The results of the re-classification of systems and components showed that there exist deviation from the current safety standards of some reclassified systems and components. The primary cooling system and its associated subsystems, safety systems, I&C and supporting systems are not designed for seismic conditions. At the design stage in 1970s, the site was not recognized as a seismic area. Later a micro-zoning study showed that the plant site has an earthquake level of MSK 6. Russian experts also indicated that some safety systems do not meet the safety requirement for separation, and safety valves do not meet the requirement for water or steam/water mixture flow.

Kalinin Units 1 and 2:

A re-evaluation of the classification according to OPB-88 (quality, safety, seismicity) has been made by the Nishni Novgorod Atomenergoproekt. The deviations from the original classification have been analysed. The procedure and the results of the re-evaluation have already been discussed during a WANO peer review. Additionally, RISKAUDIT has reviewed the plans for improvements or replacement of the equipment, which are under consideration for implementation.

South Ukraine Units 1 and 2:

A re-evaluation of the classification according to OPB-88 (quality, safety, seismicity) has been made by the plant for almost all systems and submitted to the regulatory body for approval. The process for approval is underway: I&C and electrical systems are under evaluation. Others (primary and secondary circuits, water supply, fresh fuel, fire protection, buildings, ventilation and heating systems) have been analysed and currently the comments are under consideration for integration in the final document.

The deviations from the original classification have been analysed. Based on analyses of manufacturing, design and operating deficiencies a final decision will be taken for future improvements or replacement of the equipment. The proposed methodology includes statistical analyses of available data and reliability analyses. A first trial is proposed to be performed on a typical system, which would show the highest number of deficiencies. The final choice has not yet been made.

A programme has been developed at the plant and measures are being implemented for components to fulfil the requirements of OPB-88 and currently used safety regulations. The SNPP Units 1, 2 project provided construction of the design that was seismically proof (see report 1.1–038 D of 01.07.1983) according to seismic conditions of the site.

REVIEW AREA/ISSUE NUMBER: General 2 (G 2)

ISSUE TITLE: Qualification of equipment

ISSUE CLARIFICATION: In accordance with NUSS 50-C-D [9], Section 12, the qualification of equipment important to safety is required to demonstrate their ability to fulfil their intended functions. This qualification requirement applies to normal operating conditions, to accident conditions and to internal and external events. In addition, according to international practice, it should be possible for the plant operators and the regulatory body to examine the associated qualification reports. A major concern with respect to WWER-1000 nuclear power plants, as shown by safety reviews, is that this practice of qualification of equipment is either lacking or not evident.

An example of this is the qualification of electrical and I&C equipment, including cable connections, for LOCA conditions. Neither the specifications concerning the test procedures nor the test reports are available at the nuclear power plants. In addition, safety reviews have shown that the cable connections, especially inside the containment of WWER-1000 nuclear power plants, are not able to withstand extreme environmental conditions and consequently, they have a high failure potential under LOCA conditions.

A further example is the seismic qualification of systems important to safety, especially ventilation systems which should be safety graded but are not, and safety support systems like the service water pumps, fire water supply pumps and indication and recording instrumentation. Since these are not qualified with respect to seismic loads, their functional capability on demand in the case of an earthquake would be questionable.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

This issue was identified from safety reviews and represents a deviation from international practice and especially from NUSS 50-C-D [9]. Insufficient or lacking qualification of equipment important to safety with respect to extreme environmental or seismic conditions would seriously affect Levels 1 to 3 of plants' defence in depth and the main safety functions would be questionable for scenarios within the design basis envelope.

COMMENTS AND RECOMMENDATIONS:

1. The list of equipment qualified to shutdown the plant and maintain it in a safe condition for all design conditions should be reviewed and completed if necessary. Reference should be made to the codes or standards specifying the qualification requirements and conditions.

2. The adequacy of the qualification of the safety related equipment should be assessed and additional qualification should be made if found necessary.

3. Collect all related documentations, specifications and test records from design institutes, manufacturers and test laboratories. Evaluate all available information to check consistency and adequacy of the results. Compare the test conditions with the expected LOCA conditions.

4. The environmental qualification of safety related components has to be extended to all of the safety related items, inside and outside containment, and take into account the environmental conditions of all the postulated accidents and internal and external hazards.

5. The equipment which does not qualify for required design conditions should be replaced with qualified ones.

REFERENCES: [2, 3, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The plant does not intend to perform a full scale requalification of equipment. The final decision for improvements or replacements of the equipment is based on the statistical analyses of available data and reliability analyses.

The Russian experts indicated that equipment is qualified for environmental conditions, though not qualified for seismic conditions. The equipment important to safety inside the containment has passed the environmental tests for LOCA conditions, but there are no records available at the plant. Safety valves on the pressurizer, safety valves and BRU-As on the main steamlines will be replaced with new ones which pass the qualification tests. In case of a severe earthquake, the reactor protection system, which is not qualified for seismic conditions, may not be able to function. The procedure is available to manually trip the reactor early and to isolate the containment in case of an earthquake.

Kalinin Units 1 and 2:

The equipment and systems are tested in accordance with the design requirements prior to commissioning, and during operation in accordance with the schedules of the unit shutdown, maintenance and periodic inspections. The tests are performed according to the requirements of the technical specifications and regulations in compliance with the approved programmes and methodologies.

The work on the verification of the equipment qualification is being implemented at the station. The final decision on the improvement or replacement of the equipment is based on the statistical analysis of the current data and the reliability analysis.

South Ukraine Units 1 and 2:

Methods, programmes and procedures for qualification of equipment have been prepared by SNPP and approved by the Regulatory Body to be applied for requalification of components. The re-qualification of equipment according to their upgraded classification can be performed according to available resources. The initial schedule fixed by the Regulatory Body appears particularly strict.

A comprehensive programme for qualification of all safety related components is under development by Ukrainian Nation Power Production Company “Energoatom”. The first step of this programme is the listing of safety related components and its features, which should be checked by qualification.

The schedule of this programme implementation will be developed due to available resources.

The new equipment and components may be installed in safety related systems for replacement or modifications only after qualification.

REVIEW AREA/ISSUE NUMBER: General 3 (G 3)

ISSUE TITLE: Reliability analysis of safety class 1 and 2 systems

ISSUE CLARIFICATION: Reliability analyses of safety class 1 and 2 systems are necessary to confirm that the systems are as reliable as expected by the designer. It is also important to collect component reliability data during operation, to confirm the validity of the original analysis. Some plants had neither carried out reliability analyses during the construction phase nor a systematic reliability data collection had been implemented during operation. OPB-88 [4] requires the reliability data of class 1 and 2 systems, taking into account common mode failures and personnel errors. NUSS 50-C-D [9] has a similar requirement.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. A lack of the demonstration of the reliability of safety and safety related systems to function as intended would affect Levels 1 to 3 of plants' defence in depth. Consequently, one or all main safety functions can be impaired for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

A reliability analysis and a systematic collection and evaluation of component reliability data is recommended at all plants. Necessary measures should be taken if weaknesses will be identified.

REFERENCES: [2, 3, 4, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Since 1990 a computer database on equipment failures is maintained at Novovoronezh NPP according to ISCO-5 from (information system of collection integration and exchange of quality and reliability data). Information on reliability data of the operating period 1980-1989 is not included in the database. The database is also used for PSA development.

OPB-88 requires to perform reliability analysis of safety class 1 and 2 systems. Except for the core melt frequency and radioactive release frequency targets, there are no regulatory reliability targets established for the safety class 1 and 2 systems.

Kalinin Units 1 and 2:

Since 1990 a computer database of equipment is maintained at the plant. The reliability analysis will be performed using this database and focusing on equipment in which deviations have been identified.

PSA is carried out in co-operation with USNRC in the frame of the “BETA Project”.

South Ukraine Units 1 and 2:

The reliability data are part of the input data needed for analysing the deviations from re-classification of the components. Therefore, a systematic collection of data has been initiated at SNPP. The corresponding process has been anticipated for I&C components and seems to work well. It is intended to extend it to the mechanical components. A special procedure has been developed and a reliability group has been established.

The reliability analysis of these data should be developed and focused on equipment on which deviations have been identified. The PSA is based on the component reliability database.

3.2. REACTOR CORE

REVIEW AREA/ISSUE NUMBER: Reactor core 1 (RC 1)

ISSUE TITLE: Prevention of inadvertent boron dilution

ISSUE CLARIFICATION: There are several possible causes of dilution transients, which are mainly the following:

- Fast boron dilutions caused by a rapid and massive injection of pure water into the core. This could happen, for instance, when restarting a main primary pump after a period of shutdown at low residual power and therefore with no, or low level of natural circulation and with presence of clean water in the loops due to leakages from the connected auxiliary circuits or condensation processes in the steam generator. It should be taken into account that the natural circulation is blocked when the Residual Heat Removal System (RHR) is in operation. In such conditions, in case of total loss or shutdown of the primary pumps, homogenous conditions are not guaranteed in all the loops.

- dilution by the chemical and volume control system, particularly after RCP trip due to loss of off-site power, and

- dilution due to a leak in an ECCS heat exchanger.

The fast boron dilution is not detected on time. Inadvertent slow dilution is normally detected using nuclear flux measurements and boron concentration monitoring. The continuous boron concentration monitoring is based on boron meters which are not accurate enough and have a time lag between the actual and the indicated values.

Neutron flux monitoring is difficult during rector shutdown or startup phase, because neutron level in the ionization chambers is very low (10^4 to 10^3 n/cm^2·s) and cannot be recorded by the originally installed system AKNP-3 (see issue RC 3).

In order to have an improved neutron flux measurement during reactor shutdown conditions, a new neutron flux monitoring system that uses new detectors (AKNP-7-02) with higher sensitivity and improved signal processing has been developed.

The replacement of the existing boron meters with the new NAR-12 models is planned. These boron meters have an increased measuring accuracy and the ability to monitor the B-10 isotope

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Administrative procedures preventing dilution transients are not fully backed up with interlocks. A combination of human errors and monitoring equipment limitations would affect Levels 1 and 2 of plants' defence in depth. This can lead to a situation in which the main safety function controlling the power is impaired, i.e. there may be scenarios within the design basis envelope, where a power surge can affect the core.

Fast injection of pure water into the core could result in prompt criticality with large potential damages to the first barrier (fuel cladding).

COMMENTS AND RECOMMENDATIONS:

1. The probability of boron dilution transients and their consequences , especially caused by a water slug, should be assessed. Based on the results, the existing operational procedures should be improved. If necessary, additional interlocks to prevent unborated water supply should be installed (see issue AA 11).

2. With regard to accuracy and sensitivity problems of the presently installed monitoring equipment, the planned replacement of existing boron meters is supported.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The plant has procedures for prevention of boron dilution during power operation, shutdown, refuelling, spent fuel transportation and storage, and procedures for specification of valve states of systems not containing boric acid and systems containing boric acid. When primary pressure is falling below 6 kgf/cm^2, the procedure “Actions of Preventing Water Ingress into the Primary Circuit, Cooling Pond, Refuelling Pond during Unit 5 Outage” is initiated.

Determination of boric acid concentration by laboratory method is carried out once per shift during power operation and twice per shift during refuelling.

On-line automatic boron meters are available in the following sampling points: reactor, pressurizer, loops, water from makeup pumps and water to special water treatment. The AKNP-7-02 system has been implemented already.

The shutdown concentration of boric acid at Unit 5 is increased from 16.0 g/kg to 16.3 g/kg.

Kalinin Units 1 and 2:

Prevention of uncontrolled boric acid dilution in KNPP during refueling, transportation and storage of spent fuel is assured by implementing organizational and technical measures and preventive actions of personnel. The actions to be taken in case of uncontrolled boron dilution are stated in the '‘Procedure for Operating Personnel Actions in Emergencies’ and in the emergency procedures developed according to the EdF methodology.

It is planned to replace the existing NAR-B boron meters with NAR-12 meters and replace the AKNP-3 neutron flux monitoring system (detectors) with the AKNP-7-02 system (detectors).

South Ukraine Units 1 and 2:

No inadvertent boron dilution events have happened so far at all three units including the standard model 320 of Unit 3.

The plant has procedures for preventing boron dilution during refuelling, transportation and storage of spent fuels. The procedures are revised every three years. The procedure for preventing boron dilution during refuelling is initiated for implementation after the trip of main coolant pumps (MCP).

The main points include:

- closure of isolation valves on the discharge lines of pure condensate pumps
- closure of valves on the primary makeup and letdown lines
- drainage of the dearator of the makeup system
- isolation of the bypass filters across the MCPs (filters to be cleaned, washed with pure water)
- isolation of the water line to the filter on the primary letdown line
- isolation of the boron storage tank (40g boron concentration) from pure water
- isolation of service water to the MCPS
- isolation of the intermediate cooling water system during MCP disassembly, etc.

The switches for the relevant driven motors are tagged both at control room and at field, and the hand wheels for the valves are locked.

One chapter in the Unit 3 emergency operating procedures deals with the operator actions in case of an inadvertent boron dilution. The procedures are being revised to an updated version. The same efforts on Units 1 and 2 are in progress.

The SNPP intends to replace the existing boron meter with the NAR-12 boron meter which provides better precision, reliability and response time and is capable of measuring the boron-10 concentration with an alarm signal. This replacement will be implemented at Unit 3 by the end of the 1997. The same replacement will be made for Units 1 and 2 in the future. The AKNP-7-02 system has already been implemented.

REVIEW AREA/ISSUE NUMBER: Reactor Core 2 (RC 2)

ISSUE TITLE: Control rod insertion reliability/Excessive water gaps between fuel assemblies

ISSUE CLARIFICATION: An increased drop time of control rods exceeding the maximum design value of 4 seconds has been observed at operating 'small series' WWER-1000 units of South Ukraine Unit 2, and Kalinin Units 1 and 2.

Investigations of root causes are being made in The Russian Federation and the Ukraine (OKB Gidropress, RRC Kurchatov Institute, NIIAR (Scientific Research Institute of Nuclear Reactors, Dimitrovgrad) and other institutes). There are many factors which result in the increased drop time. As the direct cause for the increased drop time excessive friction between the control rods and their guide tubes in the fuel assembly due to deformation has been identified [12, 13].

The safety concerns are related to the structural deformation of fuel assemblies affecting the reliable insertion of control rods and leading to a water gap larger than the nominal gap between fuel assemblies which will cause a higher local power density. Several events have also occurred at WWER-1000/320 units as well as at some western nuclear power plants based on similar causes.

The following compensatory and interim measures have been taken to justify continuing operation in the short term:

- if excessive rod drop times are observed at full coolant flow rate, operation with three or two reactor coolant pumps at correspondingly reduced power is permitted, provided that the measured drop time of any rod does not exceed 4 seconds. If the transfer to operation with three or two coolant pumps is not successful, then the unit has to be shut down;
- control rod drop times are measured at least once every 3 months. If any control rod drop time is more than 4 seconds, the next test is carried out within a month;
- in order to minimize the potential rod insertion problems, fuel assemblies which have been used for 2 years are not inserted into the control rod locations, but are replaced by new fuel assemblies with nearly the same physical characteristics;
- before loading of fuel assemblies into the core, they are tested on stands for verification of free control rod movement. The deviations of lifting and lowering forces from normal values should not exceed ±3 kg. The central instrument thimbles are measured by means of a specially designed calibre; and
- the position of the upper internal structure (protective tube unit) was readjusted and moved upward for several millimetres to reduce the excess axial load exerted on the fuel assemblies and to alleviate the deformation of guide tubes.

Where implemented these measures have helped to avoid increased rod insertion times. However, the experience to verify the design modifications is not yet sufficient.

A new design of the control rod, with approximately 30% greater weight to shorten the drop time, and a new fuel assembly design with a modified top nozzle and softer springs are being tested in WWER-1000 reactors.

The treatment of excessive water gaps associated with fuel assembly bow beyond that included in the design basis or methodology common to many WWER-1000 reactors is being addressed in the "Generalized Methodology" which is presently developed under the umbrella of OKB Gidropress and to be submitted for licensing. The main advantage of this methodology is to provide a generic licensing case by adding to the design basis a large amount of plant data related to fuel bow which has been collected for the WWER-1000 reactors to exclude the necessity to perform specific reload safety justifications currently performed at each unit with excessive gaps. Fuel bow caused by irradiation creep of the guide tubes under excessive axial loads during operation is currently understood as the root cause of the safety concerns.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The issue was identified from operational experience and represents a deviation from NUSS and the Russian standard PBJa-89 [5]. These require that the core be maintained such that, during normal operation and design basis accidents, its structural stability is preserved and no deformation occurs which could either impede effective operation of the reactivity control and emergency shutdown systems or which would prevent efficient cooling of the fuel. Proceeding on the assumption that the loads exerted on the fuel assemblies may exceed the limits which the affected assemblies can withstand, this issue indicates a weakness of the core design, i.e. Level 1 of plants' defence in depth is affected. In the case of transients, the main safety function controlling power might be impaired with respect to rod drop time and shutdown margin; in addition, the flow channels may also be affected (Levels 2 and 3 of defence).

COMMENTS AND RECOMMENDATIONS:

1. An independent review is recommended on the analyses of accidents where fast rod insertion is important.

2. Further investigations to define finally all possible root causes should be continued and the accident analysis for this event needs to be reviewed and extended.

3. Periodic monitoring of the possible further degradation of rod insertion should be ensured. Enough reactivity margin should be maintained, considering the possible further degradation in the 3 month operation time interval.

4. After relieving the axial forces through the repositioning of the protective tube unit, it should be verified by at least three years of operational experience that excessive axial force is the main contributor to assembly bowing [12].

5. Russian design and research organizations are encouraged to continue investigating the underlying damage mechanism which leads to permanent assembly bowing after two to three years of operation.

6. If the final solution to excessive axial forces is a new top nozzle design with softer springs, its verification in normal operation is encouraged. The disappearance of the problems will indicate that the main contributors to the root cause have been properly addressed [12].

REFERENCES: [2, 3, 5, 12, 13, 50, 51]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP Unit 5 so far has not experienced problems with the delayed dropping of control rods. The fuel assembly has an outside shroud of 1.5 mm thickness which is different from other WWER-1000 units. Unit 5 originally designed for two years' fuel cycle has been shifted to three years' fuel cycle with 4.4% enrichment fuel for refuelling. The burnup of discharged fuel is 40 MWd/kgU.

Kalinin Units 1 and 2*:*

The problem of control rod insertion time is regarded as solved. A Gosatomnadzor's Permit to operate at 100% power is obtained. The compensatory measures implemented are as follows:

- The position of the control rod protection tube unit (upper internal structures) was readjusted for Units 1 and 2 to reduce the load exerted on the fuel assemblies contributing to fuel assembly deformation.
- At both units the control rod drive shafts were drilled to reduce hydraulic resistance.
- The control rods and their shafts were replaced by heavier ones at Unit 1. The same replacement will also be gradually implemented at Unit 2.
- The control rod position sensors have been replaced by the advanced ones (DPL).
- Fuel assemblies with softer springs in their nozzles are used.
- The control rod insertion (drop) time is tested every three months.
- Fuel assemblies to be placed under the control rods during refuelling are chosen by taking into consideration their irradiation history.
- The curvature of the control rod guide tubes along the length is measured to maintain the maximum local power density of the fuel rods within design limits.

Distortion of neutron flux (power density) in the location of increased water gaps is taken into account in operation in the form of limitations imposed on the value of allowable volumetric power density and does not lead to restrictions of permissible reactor power.

South Ukraine Units 1 and 2:

The SNPP Unit 1 so far has not experienced problems with the delayed dropping of control rods, but Unit 2 has the problem.

The plant has taken series of compensatory measures:

- A routine test of control drop time is made every three months for all three units according to the regulatory requirement.
- The control rod cluster is not be located in a fuel assembly which has been operated for three cycles during the refuelling outage.
- The curvature of the selected control rod guide tubes is measured along the length, in order to ensure that peak power density of fuel elements is maintained within the design limits.
- The protective tube unit of Unit 2 was repositioned in order to reduce the excessive load on the fuel assemblies and therefore the extent of their deformations.
- For Units 1 and 2 the control rod drive shafts will be drilled in order to reduce the hydraulic resistance. This modification was carried out already at Unit 3 (model 320).
- During the 1996 outage period, the control rod drives at Units 1 and 2 were replaced by more reliable and more powerful ones which will allow to use heavier control rods.

REVIEW AREA/ISSUE NUMBER: Reactor Core 3 (RC 3)

ISSUE TITLE: Subcriticality monitoring during reactor shutdown conditions

ISSUE CLARIFICATION: The subcriticality margin has to be sufficient for all reactor shutdown conditions, and adequate information on this margin has to be provided to operators. In WWER-1000 'small series' reactors, the subcriticality is monitored by means of boric acid concentration and neutron flux measurement.

Neutron flux monitoring is difficult during the reactor shutdown or startup phase, because the neutron level in the ionization chambers is very low (10^4 to 10^3 n/cm^2.s) and cannot be reliable recorded by the originally installed equipment (AKNP-3). The monitoring is based on hearing the frequency of click signals. Reactivity or subcriticality control can not be performed online.

As already mentioned in issue RC 1, at SNPP and NNPP the new neutron flux monitoring system (AKNP-7-02) that uses new detectors with higher sensitivity and improved signal processing has been implemented. An improved signal processing and display system for subcriticality monitoring is under preparation.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power

RANKING OF ISSUE: I (KNPP only)

JUSTIFICATION OF RANKING:

The issue has been identified based on operational experience. Insufficient subcriticality monitoring during shutdown conditions is a deviation from international practice and may affect the first two levels of plants' defence in depth.

COMMENTS AND RECOMMENDATIONS:

The installation of an improved neutron flux monitoring system partly solves the issue and is supported for Kalinin NPP (KNPP) as well. Furthermore, the installation of an integrated reactivity and subcriticality monitoring system for shutdown conditions is recommended.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP has implemented several measures to improve the subcriticality monitoring during reactor shutdown conditions:

- The original flux monitoring system (NFMS) was replaced at Unit 5 by a new system AKNP-7-02, featuring improved sensitivity, reliability and performance.
- The original boron meter, NAR-B was replaced at Unit 5 by a new one which provides better precision reliability and response time and is capable of measuring the boron-10 concentration with an alarm signal.
- During refuelling, six neutron detectors (fission chambers) are temporarily inserted in the outer space of the reactor core, within the reactor pressure vessel, to monitor the reactivity change. They are able to monitor the neutron flux level of 10^{-7}% to 10^{-8}%. The refuelling is carried out in such a way that considerable contribution of the source neutrons (from the fuel

assemblies located close to detectors) is excluded, and high sensitivity of detector pulse intensity to the change of core neutron multiplication is ensured.

- Reactivity meters has been installed at Unit 5. However, they only work at power range, or used at control rod calibration test.

An on-line subcriticality monitoring system is not considered to be installed at Unit 5.

Kalinin Units 1 and 2:

The following measures are implemented or planned to improve the subcriticality monitoring during reactor shutdown conditions:

- At both units six neutron flux detectors are temporarily installed into the outer space of the reactor core inside the reactor pressure vessel. They are intended to monitor the neutron flux at the power level of 10^{-7} % through 10^{-8} % during refuelling.

- The reactivity meters have been installed at both units. However, they operate only within the power range and are used to calibrate the control rods.

- It is planned for 1999 to begin replacement of the existing neutron flux monitoring system AKNP-3 by the newer one AKNP-7-02 with the higher sensitivity, reliability and performance.

- It is further planned to replace the boron-meter NAR-B by the more modern NAR-12 with less delay and B-10 measurement.

South Ukraine Units 1 and 2:

The SNPP has implemented and planned several measures to improve the subcriticality monitoring during reactor shutdown conditions:

- The original neutron flux monitoring system (NFMS) was replaced at Units 1 and 2 by a new system AKNP-7-02, featuring improved sensitivity, reliability and performance.

- The original boron meter, NAR-B will be replaced at Units 1 and 2 by a new one which provides better precision, reliability and response time and is capable of measuring the boron-10 concentration with an alarm signal. The number of sampling locations are to be increased.

- During refuelling, six neutron detectors are temporarily inserted in the outer space of the reactor core, within the reactor pressure vessel, to monitor the reactivity change. They are able to monitor the neutron flux level of 10^{-7}% to 10^{-8}%.

- Reactivity meters has been installed at Unit 3. However, they only work at power range, or used at control rod calibration test.

An on-line subcriticality monitoring system is considered to be installed at Units 1, 2 and 3. The equipment will be provided by the Russian Institute of Instrument Engineering.

3.3. COMPONENT INTEGRITY

REVIEW AREA/ISSUE NUMBER: Component Integrity 1 (CI 1)

ISSUE TITLE: RPV embrittlement and its monitoring

ISSUE CLARIFICATION: The fast neutron (E > 0.5 MeV) flux at the WWER-1000 reactor pressure vessel wall is in the same range as for the western PWR vessels of the same vintage. The calculated maximum end-of-life (40 years) fast neutron (E > 0.5 MeV) fluence is about 5.7×10^{19} n/cm^2. However, at most of these plants, the irradiation embrittlement could progress faster than anticipated by the code prediction. The concern is related to the high Ni concentration (up to 1.9 wt.%) in vessel beltline area welds. The code prediction formula may not be therefore conservative for the evaluation of the critical brittle fracture temperature T_k.

Surveillance specimen, made from base, weld and heat affected zone metal, should provide for monitoring of mechanical properties and, T_k temperature shifts due to irradiation and thermal ageing. The containers with these specimens are placed on the top of the thermal shield shell, where irradiation conditions (temperature and neutron spectrum) are significantly different from those at the inner surface of the reactor pressure vessel wall. Additionally, the specimen irradiation temperature is not precisely known and the variation of neutron fluences of specimens taken out from one set does not allow determination of representative T_k values in line with requirements of applicable Russian codes. Thus, the data from the current surveillance programmes are not valid for monitoring the reactor pressure vessel wall irradiation embrittlement and therefore cannot support the end of life material toughness estimations needed for reactor pressure vessel integrity assessment. Specimen containers will be located in positions representative of vessel wall conditions at Temelin plant. This measure is not directly applicable to the 'small series', but reflects the necessary systematic reconsideration of the surveillance locations in order to provide representative irradiation conditions for future designs.

The limited number of relevant data and results from R&D programmes cannot provide sufficient knowledge to generate neither an alternative data set nor valid correction factors. Therefore, complementary investigations on the irradiation conditions of the surveillance specimens have to be performed to define corrective and/or alternative procedures to generate useful material data for supporting the reactor pressure vessel integrity assessment.

The original PTS analysis for RPV integrity assessment, in particular the selection of transients, thermal hydraulic analysis and structural analysis including fracture mechanics assessment is outdated.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

The issue was identified as a deviation from applicable standards (OPB-88 [4], PNAE G-7-008-89 [14]) and from operational experience. Adequate monitoring to detect the degradation is not provided affecting Level 2 of defence. This may increase the probability of a vessel failure. This is a BDBA scenario, which would result in an inability to cool the fuel and confine the radioactive material and consequently in loss of all barriers with unacceptable consequences.

COMMENTS AND RECOMMENDATIONS:

1. An early implementation of flux reduction measures should be considered at WWER-1000 plants.

2. The owners of the WWER-1000 reactor pressure vessels should accelerate the establishment of a common view on how to assess the current surveillance programmes. Therefore, complementary investigations on the irradiation conditions of the surveillance specimens have to be performed (temperature measurements and neutron spectrum determination) to define corrective and/or alternative procedures. Further irradiations should be performed taking advantage of the available archive materials and unirradiated surveillance specimens. For plants under construction, a modification of the surveillance programme is considered.

   TACIS projects (SRR2/95 & R206/96) as well as the EC/DGXII WGCS activity plan are addressing these tasks extensively. The results of these experiments and studies should be taken into account in improvement programmes.

3. The plant staff should take any appropriate measure to save archive as well as irradiated specimens. The current withdrawal schedule of surveillance capsules remains questionable. Thorough review and justification of the current surveillance programme practices should be provided prior to further withdrawal and testing of specimens.

4. Qualified NDT methods should be used for the reactor vessel in-service inspection, using state of the art ultrasonic methods. Special attention should be given to the core weld.

5. The reactor pressure vessel integrity assessment with respect to pressurized thermal shock events should be reviewed and if necessary re-evaluated. The operational experience, implemented and planned modifications should be taken into account when performing the analyses. The results should be reflected in the operational procedures and should guide the corrective measures (e.g. heat-up of the ECCS water).

REFERENCES: [2, 4, 14, 15, 16, 17, 18]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The RPV is believed to have the lowest Ni content out of the WWER-1000 plants. The vessel is of rather similar design as other 320 model vessels. There is a specific ISI programme in place. UT from inside is not carried out. The integrity assessment is obsolete. A bypass of the ECCS heat exchanger was implemented with the objective to reduce the PTS loads.

Kalinin Units 1 and 2:

The Kalinin RPVs have relatively high Ni content in the beltline welds (~1.7 ut%). Surveillance testing is carried out at Kurchatov Institute, who also defines now the surveillance specimen containers removal schedule (modified). The first results indicate irradiation embrittlement coefficient A_F=29 (considered preliminary value due to low fluence accumulated) which is substantially higher than the code value, A_F=20.

The RPV is inspected by UT from outside using a newly developed manipulator “ASK-132”. 100% of the beltline region is inspected each year.

In 1994, PTS analysis was carried out to justify that it is not necessary to heat up the ECCS tanks till 2003 (the ECCS is heated up anyway). The analysis considers only limited scope of transients and some of the assumptions (and methods) and may lead to non-conservative results. This could be of particular importance towards the end of the plant design life due to the high Ni content.

Actions have been initiated to introduce low leakage core loading.

South Ukraine Units 1 and 2:

Some modifications to the surveillance programme have been implemented, reliable data are, however, not available at the plant. A TACIS project on Operational Surveillance of RPV to develop and establish surveillance procedures and an irradiation embrittlement surveillance programme was started in June 1995 and completed in June 1997. The surveillance programme of each unit is different.

A more precise sample-testing programme has been developed. The RPVs embrittlement database is planned to be established in co-operation with Russian specialists.

The RPV integrity assessment is not available at the plant, some of the related data available are obsolete. It is considered to implement flux monitoring in the reactor cavity.

REVIEW AREA/ISSUE NUMBER: Component Integrity 2 (CI 2)

ISSUE TITLE: Non-destructive testing

ISSUE CLARIFICATION: The non-destructive testing (NDT) for reactor coolant system in-service inspection is carried out according to the individual Member States' requirements, which are in principle based on former Soviet Union Standards. Defect-reject manufacturing approach is used rather than defect-follow approach, which is capable of a timely detection of the degradation. Different techniques and tools are used. Some deficiencies have been revealed, related to vessel inspection from outside, testing of underclad area, and testing of steam generator collectors and tubing. There is also restricted accessibility and inspectability of some vessel welds, vessel head, vessel head penetrations, piping welds, steam generator shell welds, and specific piping nozzles. Furthermore, recent results of a programme similar to PISC indicate insufficient reliability of NDT methods, tools and personnel, as in the case of the PISC programme results. There are no qualification requirements for methods, personnel and equipment established at present.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **III**

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards (OPB-88 [4], PNAE G-7-008-89 [14], NUSS) and from the operational experience. The design of some sections does not allow for inspection by NDT methods, which affects Level 1 of defence. The approach adopted is not adequate to detect degradation in time. Reliable in-service inspection is a key provision (Level 2 of defence) required to preserve the integrity of the second barrier. It can not be considered sufficient without adequate qualification requirements. Undetected defects can result in primary circuit failures, which significantly increase the challenge of the safety function cooling the fuel. In case of a vessel break, both safety functions are lost, leading to unacceptable consequences.

COMMENTS AND RECOMMENDATIONS:

1. Defect-follow predictive approach should be developed and implemented for in-service inspection.

2. The NDT methods, tools and personnel should be qualified on national basis through performance demonstrations on specimens with real type defects. National requirements for such qualification should be established. A related methodology has been recently published by the IAEA [19].

3. Development of NDT qualification requirements has been requested by regulatory bodies in Czech Republic, Hungary and Slovakia. In the frame of PHARE programme, some test specimen were manufactured and these specimen could be also used in the Russian Federation and Ukraine.

REFERENCES: [2, 4, 14, 15, 17, 19]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The ISI is carried out in line with respective Russian requirements.

Defect follow approach is used. A component (pipeline) is replaced or repaired when a defect is identified exceeding rejection criteria established. The allowable defects are not repaired based on analysis and monitored during ISI in addition to the normal annual NDT schedule.

Six NDT methods are used at NNPP including eddy current testing for the SG tubing.

All inspectors directly involved in performance of NDT are trained and qualified in compliance with Russian codes and standards. The results obtained are recorded in logbooks, no computerized system is available. There are no plans to implement ISI qualification at present.

Kalinin Units 1 and 2:

The ISI is carried out in line with respective Russian requirements. Defect follow approach is used. New techniques are being implemented (e.g, ECT for SG tubing (planned 100% in four years) and condenser tubing (100% each year), RPV UT from outside (new manipulator, see issue CI 1).

Implementation of, e.g. RPV UT inspection from inside, and ISI qualification are considered useful, but no definite plans exist at the moment due to funding limitations (it should be noted that such resource consuming tasks are to be rather addressed on a regional level by, e.g. the utility than by individual plants).

South Ukraine Units 1 and 2:

The ISI is carried out in line with respective Russian requirements. Some minor improvements were implemented, others are planned. There are no plans to implement RPV inspection from inside. There are no plans to implement ISI qualification at present.

A facility for primary circuit welded joints control by non-destructive methods (TACIS-93) has already procured. Prior to introduction of the modernized ultrasonic surveillance system for welded joints, a list of zones at enhanced risk is prepared.

Another TACIS project on the Modernization of the SK-187 Manipulator is going to be finished at the end of 1998.

Further improvements regarding positioning accuracy and scanning mechanism of the RPV ISI manipulator are also considered as well as preparation of RPV samples with artificial defects. Introduction of risk based ISI approaches is under discussion.

REVIEW AREA/ISSUE NUMBER: Component Integrity 3 (CI 3)

ISSUE TITLE: Primary pipe whip restraints

ISSUE CLARIFICATION: The primary coolant main circulation piping is made of carbon steel clad with austenitic stainless steel and contains dissimilar welds. The primary piping layout of 'small series' NPPs, as compared to the standard model, differs mainly due the isolation valves in primary circuit. Pipe whip restraints, designed as a welded steel "cage" structure, are located next to each main circulation line elbow similarly to the standard model. Detailed information on the related design considerations and adequacy of the pipe whip restraints is not available at the plants.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: **II**

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards (OPB-88 [4], NUSS Design Code 50-C-D [9]) and international practice. The design provisions to mitigate the consequences of a main circulation line break are not appropriate, thus affecting Levels 1 and 2 of plants' defence in depth. The dynamic effects associated with a main circulation line double ended guillotine break could lead in the worst case to the damage of two steam generators which was not considered in the original design.

COMMENTS AND RECOMMENDATIONS:

1. The pipe whip restraints adequacy and related design considerations should be re-assessed or an alternative approach such as LBB concept to demonstrate that ruptures can be prevented should be used. Updated seismic loading should be considered (see also EH 1).

2. Unit specific LBB analysis could be carried out taking benefit of the already existing studies and programmes on standard model units. Updated seismic loads as well as adequate layout, operating loads and material property description should be used.

3. Leak detection system, developed to meet the LBB requirements, as well as the necessary in-service inspection improvements need to be carefully evaluated.

REFERENCES: [2, 4, 9, 15, 20, 21, 22]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

It is planned to apply the LBB concept in connection with the seismic upgrading.

Kalinin Units 1 and 2:

It is considered to apply the LBB concept in the future, utilizing to the extent possible the results obtained through the development of the LBB case for standard model WWER-1000/320.

South Ukraine Units 1 and 2:

A technical task for implementing the LBB concept has been developed.

REVIEW AREA/ISSUE NUMBER: Component Integrity 4 (CI 4)

ISSUE TITLE: Steam generator collector integrity

ISSUE CLARIFICATION: In the period from late 1986 to 1991, cracks have been revealed in the ligaments between tube holes in the collectors of 24 steam generators at 6 NPPs of WWER-1000 units. The operating time before detection of this damage has varied between 7000 and 60,000 hours for the affected steam generators. These cracks have mainly been detected in the cold collectors. However, indications have also been reported for the hot collector. The material of the collectors is carbon steel, cladded from the primary side. The collector damage was caused by a combination of high residual stresses due to manufacturing technology used, environment assisted cracking in the area of crevices due to violation of secondary water chemistry requirements (pH in particular) and localised corrosion damage and subsequent crack growth due to poor collector material quality (non-metallic inclusions).

Further, the secondary water chemistry has to take into account the material composition of the secondary circuit which results in controversial requirements and compromise solutions. This could lead also to accelerated degradation of steam generator components. Repeated chloride ingress into the secondary circuit through damaged condenser tubes was also observed. The existing monitoring of water chemistry, of primary to secondary leakages and the prevention of component degradation is not sufficient to prevent violation of the design limits of safe operation.

Compensatory and interim measures have been implemented at all SGs of operating plants which have been effective except for one failure which occurred at the Balakovo NPP in 1995. Further careful observation of SG integrity is necessary. Detailed information on compensatory measures and design modifications is provided in [23, 24].

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

This issue was identified from operational experience. The integrity of this barrier is seriously degraded due to design and manufacturing aspects, reflecting a major weakness of Level 1 of plants' defence in depth. The related monitoring is not sufficient, affecting Level 2 of defence. A large failure of the collector, if not properly addressed, could lead to a bypass of the containment and a degradation of the main safety function cooling the fuel, and potentially to its complete loss in the long term and serious consequences with respect to radioactive release (see issues CI 6, S 2, S 9, AA 7 and OP 2).

COMMENTS AND RECOMMENDATIONS:

1. Implementation of a primary to secondary leak detection system to allow reactor shutdown if a leak rate beyond 5 l/h occurs should be considered (I&C 7).

2. Regular inspections of the collector using optimized NDT should be given high priority. A programme to monitor steam generator degradation should be developed and implemented.

3. A common database from the chemistry records, inspection results and operational experience should be established among WWER-1000 owners.

4. Consideration should be given to replacement of copper containing alloys in the secondary circuit in order to establish a higher pH (9) water chemistry and try to eliminate chloride ingress from condenser cooling water.

5. Modification of secondary water chemistry with automatic pH monitoring and local sampling inside the steam generator and the monitoring of condenser tubing leaktightness should be implemented (I&C 10).

REFERENCES: [2, 15, 23, 24, 47]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Following detection of cracks in one steam generator collector, all 4 steam generators were replaced in 1989 with new ones with some corrective measures implemented, additional heat treatment of collectors was performed in 1996. Since the replacement, stricter secondary water chemistry control has been implemented. The collectors are inspected in line with recommendations of the designer. The eddy-current testing of SG-collectors is performed annually. No cracks were found since the replacement.

Kalinin Units 1 and 2:

No collector cracking was found to date and the plant still operates with original steam generators with several corrective measures implemented (such as additional heat treatment, modification of blowdown system). The collectors are inspected in line with recommendations of the designer. Automatic water chemistry monitoring and control have been implemented.

It should be noted that the plant gives high attention to water chemistry and associated aspects (e.g. relatively high pH in the range of 8.8-9.2, 100% ECT of condenser tubing each year). The detection of leaks from primary to secondary circuits is based on a combination of automatic on-line (alarm) and off-line (quantification) methods.

South Ukraine Units 1 and 2:

At this plant, the collector damage was found for the first time by detection of primary to secondary leakage. The steam generators had to be replaced twice. The latest steam generators put in service are of the most modern design (1 at Unit 1, 4 at Unit 2). The inspection of collectors is carried out in line with recommendations of the designer. The plant was said to have a particularly poor secondary water chemistry control. The situation can be summarized as follows:

1. SNPP meets the requirements of the main designer codes and standards (OKB GP) in part of collector metal control.

2. New norms have been introduced for the secondary side water chemistry.

3. An agreement has been concluded for supply of LPH manufactured of high alloy steel to raise pH.

4. In frame of TACIS Programme, a number of activities are being carried out aimed at introduction of morpholin conditioning mode.

REVIEW AREA/ISSUE NUMBER: Component Integrity 5 (CI 5)

ISSUE TITLE: Steam generator tube integrity

ISSUE CLARIFICATION: Defective steam generator tubes, with cracks penetrating a certain part of the wall thickness, are usually plugged or sleeved, based on criteria developed using the results from burst testing. The original in-service inspection method for WWER-1000 steam generators is based on the aquarium method by monitoring of bubbles by a camera inside the collector while the secondary side is drained and pressurized by gas. This method does not reveal tube degradation until there is a through-wall crack and does not allow for a predictive defect follow in-service inspection. Some plants are permitted to operate with small tube leaks. The technically justified basis for the leakage limits is not available.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified from operational experience. Non-appropriate plugging criteria and leakage limits affecting Level 2 of defence could increase the frequency of steam generator tube rupture which would question the safety function confining the radioactive material in case of multiple tube rupture, as a BDBA. The off-site radiological releases may exceed the limits. If an additional failure occurs, e.g. BRU-A stuck open, then a major loss of primary water inventory could question the safety function cooling the fuel.

COMMENTS AND RECOMMENDATIONS:

1. State of the art non-destructive methods should be implemented along with a predictive in-service inspection approach.

2. Justified plugging criteria have to be developed.

REFERENCES: [2, 15, 23]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Steam generators were replaced in 1989. The tubing is inspected using the aquarium method and since 1996, eddy current testing (ECT) is being used for volumetric inspection. It is planned to complete 100% ECT inspection after 4 years. The routine volume for ECT will be defined according to the results. Since 1989 till 1996, 3 tubes were plugged as a result of the inspections carried out. The indications found were classified as manufacturing defects. Justified plugging criteria based on experiments and calculation are being developed at present by OKB Gidropress. Preliminary results suggest that 70% wall thickness defects may be appropriate according to the plant staff.

Leakage limits are based on secondary water radioactivity with respect to both leak rate (5 1/h) and radioactivity (I^{131}).

Kalinin Units 1 and 2:

Justified plugging criteria based on experiments and calculation are being developed at present by OKB Gidropress. At present, provisional plugging on 80% of tube wall thinning is recommended generally, at KNPP 70% is applied. Leakage limits are based on secondary water radioactivity with both leak rate of 5 l/h and I^{131} concentration.

South Ukraine Units 1 and 2:

The inspection of the tubes was performed until recently only by aquarium method on all 3 units. Thus the corresponding current plugging criteria is based on through wall air leaking defects. The corresponding primary/secondary leakage limits are less than 1 1/h by leakage flow and less than 2.10^{-8} Ci/l by iodine activity for each SG unit.

Due to repeated replacements, the SGs have not yet been operating for a long time: 28 000/37 500 h, for Unit 1, 32 000/37 700 h for Unit 2 and 46 600 for Unit 3. Thus, the number of plugged tubes is low (10/20 tubes from 11 000 for one SG).

The plant experts have indicated no intention to reconsider the secondary water chemistry monitoring.

ECT was recently used to inspect rows 85-110 on all 4 steam generators at Unit 2 (where problems occurred at Balakovo NPP) and steam generators No. 3 on Unit 1.

During the outage 1997 at Unit 2, an eddy-current test has been conducted at SG 1-4 with application of Intercontrol equipment according to OKB GP recommendations and regulatory requirements, the same activities are planned for Units 1and 3.

REVIEW AREA/ISSUE NUMBER: Component integrity 6 (CI 6)

ISSUE TITLE: Steam and feedwater piping integrity

ISSUE CLARIFICATION: There are two main factors contributing to the safety concern on steam and feedwater piping integrity. Time related degradation mechanisms (corrosion, erosion-corrosion, etc.) may affect the piping since the material combination of the secondary circuit required compromise solution of water chemistry. The steam lines are designed for steam load (operation) and for cold water load (hydrotesting) but not for hot water load. Possible water hammer due to ingress of hot water from primary circuit was not taken into account.

Further concern is connected to the piping layout (see issue IH 7). For the 'small series' plants, the piping layout outside the containment is similar to WWER-440 plants with rather limited physical separation and no special restraints or supports. The assessment of consequences of high energy secondary piping breaks inside and outside the containment is not available except for the containment penetrations which were designed considering dynamic effects associated with piping breaks.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **III**

JUSTIFICATION OF RANKING:

This issue was identified from current standards. Weaknesses in the design of steam and feedwater piping affect Levels 1 and 2 of plants' defence in depth. The performance of the related safety functions is questionable in DBA scenarios (steam generator collector break up to equivalent diameter of 100 mm according to the updated TOB) or lost in a BDBA case (steam generator collector break larger than an equivalent diameter of 100 mm, see issue CI 4), leading to a bypass of the containment and loss of cooling in the long term.

COMMENTS AND RECOMMENDATIONS:

1. The investigation of the capability of steam piping and its supports to withstand hot water load (water hammer), needs to be completed.

2. Based on the results of the analysis of the piping systems, the concerned supports should be modified if necessary and either pipe whip restraints should be added or a concept similar to LBB should be applied.

3. Consideration should be given to the replacement of sections of secondary piping with high thinning rate with materials with higher degradation resistance. A thorough qualification of the replacement materials should be considered.

4. Improvement of water chemistry monitoring and control should be implemented in order to prevent degradation and extend the lifetime of components concerned. An optimal but expensive solution could be to establish a higher pH (9) water chemistry in the secondary circuit. For this purpose, copper containing alloys in the secondary circuit (condenser tubing) would have to be replaced. This could further result in an improvement of leaktightness, i.e. eliminate ingress of condenser cooling water and reduce steam generator components degradation.

5. A programme to monitor degradation due to secondary water chemistry should be developed and implemented along with appropriate NDT methods.

REFERENCES: [2, 15]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The piping concerned is inspected with a frequency of 2-6 years, including the evaluation of corrosion and erosion damage. There are no other plans in place to address this issue.

Kalinin Units 1 and 2:

Steam and feedwater piping is inspected in line with the requirements of the applicable standards and the recommendations of the plant designer. It is considered to apply in the future to this piping a concept similar to LBB but there are no fixed plans yet.

The operational experience accumulated to date is rather good. The plant maintains strict water chemistry, see also CI 4. Some sections of secondary piping and equipment were replaced (level control, check values). EOP for the case of large primary to secondary leaks should ensure that there is no SG overfill.

South Ukraine Units 1 and 2:

There are plans to replace low pressure reheaters for stainless steel.

Indicators to measure temperature changes on the mean steam pipes of Units 1 and 2 were installed (Unit 1: YuAT-292-39-31, Unit 2: YuAT-292-39-32 mod.1).

REVIEW AREA/ISSUE NUMBER: Component integrity 7 (CI 7)

ISSUE TITLE: Structural integrity related monitoring

ISSUE CLARIFICATION: Accurate monitoring and control of water chemistry is important to maintain integrity of components and piping. This may be of a particular importance for the secondary circuit as it is directly related with an identified degradation mechanism or its prevention (see issues CI 4, CI 5, CI 6, I&C 10).

Leak detection systems of the primary boundary are required for various objectives:

- Component specific leak detection systems are systematically required for Class 1 bolted flanges (RPV main flange, CRDM housings, steam generator collector cover). Operating experience may ask for complementary improvements when the sealing system and/or its leak detection system is found not satisfactory. The CRDM is of particular concern.

- Improving the performance of the general leak detection systems to detect possible leaks in pressure retaining components and piping. Beside the general concern regarding the tidiness of the pressure boundary, the implementation of the leak before break concept requires diversification and redundancy of these systems in order to improve the detection reliability (see issues CI 2, I&C 7).

- Monitoring primary to secondary leaks is of concern in general. The identified degradation of the steam generator collector (see issue CI 4) calls for specific measures (see issue I&C 7).

Monitoring of loading conditions shall provide for the verification of specified loads (e.g. global loop displacement). Unspecified loads such as thermal stratification, dynamic effects, vibration as well as significant operating transients can be assessed by implementation of specific monitoring means (see issue I&C 7) to enhance the reliability of the pressure boundary and support the leak before break concept (see issue CI 2).

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. Inadequate monitoring related to the reactor coolant pressure boundary integrity affects Level 2 of plants' defence in depth to control abnormal operation and to detect failures. The main safety function can be impaired because of lack of an early warning when the integrity of the reactor coolant pressure boundary is threatened. This scenario is possible during normal operational conditions with degraded components.

COMMENTS AND RECOMMENDATIONS:

1. The primary and secondary water chemistry monitoring systems should be redesigned in such a way that negative impacts on material degradation mechanisms (e.g. corrosion, stress corrosion, corrosion - erosion) by violation of the water quality specification are avoided. Optimization of the locations and chemical species to be of major importance should be identified by material experts.

2. The optimization of the reliability of the tidiness of the CRDM housing flange should be addressed by both redesign and / or replacement of the gaskets as well as by ensuring a continuous leakage monitoring.

3. The installation of global leak detection systems shall provide for improved pressure boundary surveillance and would be required for the implementation of the leak before break concept.

4. Load monitoring systems shall provide for the validation of the significant transients as well as for local loads which were not included in the initial design. In particular, requirements for supporting the leak before break concept should be carefully addressed.

REFERENCES: [2], [15]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The following inspection and diagnostic means are available at Novovoronezh Unit 5 to monitor the primary circuit integrity:

- leak detection at the reactor main flange, control rod drive flanges, reactor temperature monitoring nozzle flanges, SG collector and manhole flanges on primary and secondary sides, pressurizer manhole flanges, ECCS hydroaccumulator manhole flanges, main loop isolation valve flanges and some valve flanges on pressurizer injection line,

- vibroacoustic examination of the reactor, reactor coolant pump, steam generator, main circulation circuit, pump bearings, electric motors, fans (KSA-10 instrumentation),

- acoustic leak control of pressurizer safety valve,

- moisture detection in the bellows and in the reactor pit,

- leak control of control rod drive flanges,

- measurement of humidity, temperature, pressure, hydrogen concentration inside the containment,

- water detection in floor drains of the lower containment plate,

- water and oil detection in reactor coolant pump motor body.

Kalinin Units 1 and 2:

At Unit 1, system KAZMER from KFKI in Hungary was put in trial operation. The system consists of three subsystems:

- acoustic emission based leak detection (sensors on RPV head, one on each gate isolation valve, i.e. four in total, and one on each steam generator blowdown line, eight in total),

- vibration monitoring for MCPs and FWPs,

- reactor noise diagnostics.

Implementation of a similar system to Unit 2 is under way in the framework of TACIS.

Further, it is planned to implement fatigue monitoring system (FADIS, modernized FAMOS). Related design work was completed but the implementation itself (TACIS) is being delayed.

In addition to the above, the plant also has monitoring of humidity, temperature, pressure and radioactivity in the containment (original design).

Regarding the primary to secondary leak monitoring, see issue CI 4.

South Ukraine Units 1 and 2:

NAEK “Energoatom” is developing a national programme for implementation of an integrated diagnostic system. A schedule has been developed for implementation of diagnostic systems in the NPPs of Ukraine. The following systems are planned to be implemented under this programme at SNPP Units 1, 2:

1. Control rod drives diagnostic system,

2. Reactor vessel, primary tanks and pipelines diagnostic system,

3. RCPs diagnostic system,

4. Valves diagnostic system,

5. Reactor internals diagnostic system.

This work is being carried out as an all-industry programme.

The following activities are currently carried out at site:

1. The water chemistry norms have been revised.

2. Morpholine treatment of the secondary circuit is being introduced under TACIS U1.02.95B Project.

3. According to TACIS U1.02.95A, a guideline is being developed on the secondary circuit treatment which will include procedures and methods for chemistry monitoring. According to the same project, additional instruments for water chemistry, automatic and laboratory monitoring, will be supplied to SNPP.

4. It is planned to start in 1999 the work under TACIS Project a three years programme for creation of an expert diagnostic system of chemistry treatment in primary and secondary circuits.

5. The transfer from nickel gaskets to ones of swollen graphite, that are sealing the upper block neutron instrumentation channels, has been implemented.

6. The Units 1,2 LPH -3,4 piping has been replaced by one of high alloy steel.

7. Upgrading of reactor upper block leak detection system is envisaged. An agreement with the Central Design and Technology Institute is under consideration.

8. In order to eliminate intakes of raw water that deteriorate the secondary side chemistry, a protective paint has been applied to the tubesheets and the piping of the turbine condensers.

9. Replacement of the HPH piping by the one manufactured of high alloy steel is in progress at Units 1, 2.

3.4. SYSTEMS

REVIEW AREA/ISSUE NUMBER: System 1 (S 1**)**

ISSUE TITLE: Primary circuit cold overpressure protection

ISSUE CLARIFICATION: When the reactor is in the cold shutdown state, there may be a risk of overpressure in the primary circuit due to:

- loss of residual heat removal (RHR) systems,
- imbalance between the charging and discharging rates of the primary side makeup circuit,
- spurious actuation of HP ECCS or primary circuit makeup pumps,
- spurious accumulators discharge.

There is a potential risk for overpressurization especially when the primary circuit is in a solid state, e.g. during hydraulic and tightness tests.

During the plant cool down, the primary circuit makeup pumps remain in operation until the primary circuit is depressurized and the main coolant pumps (MCPs) are switched off. The pressure is decreased and cool down is carried out through the primary side heat removal circuit, which has two safety valves set to 16 kg/cm^2 (TH43 S01, S02/model 302, model 338 and ARPK 1,2/model 187, respectively).

The LP ECCS is connected to the primary circuit during the whole shutdown period.

To prevent the risk of overpressurization during that stage, administrative measures are taken:

- the accumulator lines are isolated and the pressure in the accumulators is decreased to 30 kg/cm^2,
- creation of a nitrogen blanket in the PRZ when the primary circuit is close to the critical temperature,
- the high pressure injection lines are isolated and the respective electric circuits of valves and pump drives are disconnected from the grid.

The two safety valves on the RHR line are designed to avoid overpressurization of the LP ECCS after connecting to the primary circuit. Their total flow is 600 t/h corresponding to a spurious start of one HP injection pump and of one makeup pump for each valve. However, during hydrotesting the pressure is raised to 33 kg/cm^2. Above 16 kg/cm^2 the LP RHR system with the two safety valves is isolated (automatically by interlock) and the primary circuit is at water solid condition. Therefore, the two safety valves on the RHR line do not constitute automatic protection of the primary circuit against overpressurization under cold conditions. Since the pressure is raised at a rather slow rate, special organizational measures are taken to avoid overpressurization, which includes procedures and operators to switch off the pumps from control room as well as to disconnect the power.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Cold overpressurization transients might challenge the RPV integrity (see issue CI 1). The procedure to avoid overpressurization can be lost by a single human error which would affect Level 2 of plants' defence in depth to control abnormal operation.

COMMENTS AND RECOMMENDATIONS:

1. A special interlock for automatic stopping of the makeup pumps in case of the primary pressure reaching 35 bars and primary temperature being less than 100°C should be implemented in the short term.

2. In addition, alarm signals of cold overpressure conditions and high PRZ level should be implemented to support the proposed operator intervention.

3. In order to meet the whole content of the safety issue, an additional device is recommended to protect the primary circuit against overpressurization during cold shutdown.

   A study needs to be performed to define the design requirements with respect to all possible causes of overpressurization particularly when the primary circuit is in a solid state.

REFERENCES: [2, 3, 25]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

RHR safety valves were replaced by new ones. It was noted that organizational measures were implemented, development of other measures is underway.

The NSSS designer OKB Gidropress is developing an automatic solution which should protect the reactor, but the results are not available yet.

Kalinin Units 1 and 2*:*

There are administrative measures proposed to avoid an inadvertent operation of the HP safety injection and boration pumps or a faulty injection from the ECCS accumulators when the primary circuit temperature is lower than 100°C:

- Administrative measures to avoid inadvertent startup of high pressure safety injection pumps and boron injection pumps in shutdown
- closure and disconnection of the isolating valves in the ECCS accumulator injection lines
- pressure decrease in the accumulators by dumping the nitrogen to a pressure <30 bar.

South Ukraine Units 1 and 2:

There are administrative measures proposed to avoid an inadvertent operation of the HP safety injection and boration pumps or a faulty injection from the ECCS accumulators when the primary circuit temperature is lower than 100°C:

- disconnection of power supply to the HP ECCS pumps
- closure and disconnection of the isolating valves in the ECCS accumulator injection lines
- pressure decrease in the accumulators by dumping the nitrogen to a pressure <30 bar.

REVIEW AREA/ISSUE NUMBER: Systems 2 (S 2)

ISSUE TITLE: Mitigation of a steam generator primary collector break

ISSUE CLARIFICATION: Cracks revealed in the ligaments between tube holes of primary steam generator collectors (see issue CI 4) may develop into large breaks where primary water may bypass the containment. Since the SG relief and safety valves are not qualified for the flow of water-steam mixture, they may fail to reclose after initial opening and then primary coolant could be released directly to the atmosphere (see S 9). In addition, the ECCS water injected into the primary circuit may be lost as well, instead of being collected in the containment sumps. Further, the injection of ECCS water tends to keep up the pressure in the RCS, thus counteracting the actions aimed at RCS depressurization.

There are design differences compared to the standard model 320. First, the primary circuit of 'small series' WWER-1000 plants is equipped with loop isolation valves which are emergency power supplied. Second, for NPPs of WWER-1000 model 302 and model 338, the steam generator relief valves (BRU-A) located upstream from the MSIV can be isolated. At NNPP (WWER-1000/187) isolatable BRU-As are located downstream from the MSIV on the main steam header.

However, accident analyses should be performed to ensure that large coolant leakages to secondary side can be managed within the DB envelope even without primary side isolation of the affected SG. Therefore, the strategy for accident management could be based on a short term depressurization of the primary circuit, fast cooling down via the steam dump stations, and the isolation of the damaged steam generator.

Countermeasures to address the issue were implemented or are planned to be implemented. Currently procedures are being developed but they have not been verified.

Some WWER-440 plants (e.g. Loviisa and Paks) where the system design is similar to the Novovoronezh Unit 5 have been required to be upgraded so that they can cope with a SG collector break which is followed by a postulated failure to isolate the SG from the primary side and a failure of a SG safety valve to close after initial pressure relief. The main improvements are connection of HP safety injection lines to the pressurised auxiliary spray line, in order to provide fast and reliable depressurization, and an additional 1500 m^3 borated water tank which ensures supply of ECC water until the primary and secondary circuits have been cooled down and depressurized to ambient pressure.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Countermeasures to mitigate the consequences of large steam generator collector breaks which would affect Level 4 of plants' defence in depth need to be fully developed and verified to ensure the main safety functions.

COMMENTS AND RECOMMENDATIONS:

1. Member States operating WWERs broadly agree on the need to strengthen plants' defence in depth to cope with primary to secondary leaks (PRISE) as DBAs. An IAEA guidance document was developed to assist Member States in implementation of national approaches for PRISE treatment as a DBA [47].

2. Further accident analyses should be performed for postulated scenarios of SG collector failures where isolation of the damaged SG fails and opens a direct release to the atmosphere (see issue AA7). Depending on the results, the safety system and design and the emergency operating procedures should be upgraded and operating staff training should be conducted to minimize the adverse consequences.

3. The existing design differences compared to the standard model are appropriate to facilitate mitigative measures (loop isolation valves, isolatable BRU-A).

   However, the operability of the steam generator relief and/or safety valves (see issue S 9), of the isolation valves before BRU-A, and the integrity of the main steam lines should be evaluated for SG collector breaks with a leakage area of 100 mm equivalent diameter and even larger with respect to the event-specific loads (see issue S 9).

   When the accident management approach is based on an isolation of the damaged SG from the primary circuit, consideration should be given to reliability aspects of these loop isolation valves (operability with respect to pressure differences).

4. Based on the results of the thermohydraulic analyses, the radiological consequences of such scenarios need to be assessed.

REFERENCES: [2, 3, 47]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Based on the OKB Gidropress analyses results, operating procedures were developed.

Measures are planned to replace the existing BRU-A valves with new ones qualified for water-steam flow within TACIS. The eddy - current testing of SG collector is performed annually.

Kalinin Units 1 and 2:

No SG replacements to date. Since 1987 special upgrading measures have been implemented to improve water chemistry conditions e.g. modification of the SG blow down system.

Additional calculations of accidents accompanied by rupture of Dn-100 SG header/collector in the framework of “Emergency Operating Procedures” were performed according to the EdF methodology.

On the basis of these calculations an optimal strategy of accident control startup safety systems was developed.

South Ukraine Units 1 and 2:

An emergency instruction for the event “Large coolant leakage from the primary to secondary circuit” has been developed and is part of the operating instruction for accident management of SNPP.

This emergency instruction is based on the results of the accident analysis of a primary to secondary leak with a 100 mm equivalent diameter break size, taking into account the design features of the ‘small series’ WWER-1000 NPPs. Further analysis will be performed to identify those scenarios of collector failures which could lead to severe consequences. Based on the results of the calculation to be performed within the safety assessment report development, the existing procedures will be upgraded.

REVIEW AREA/ISSUE NUMBER: Systems 3 (S 3)

ISSUE TITLE: Reactor coolant pump seal cooling system

ISSUE CLARIFICATION: Injection of sealing water to the reactor coolant pump (RCP) seals is provided by the primary circuit makeup system which is emergency power supplied (WWER-1000/302, 338). In case of loss of sealing water supply from the makeup pumps the injection of sealing water is backed up by an autonomous cooling circuit, which is designed to cool the lower bearing of the primary pump. Therefore, the autonomous cooling circuit avoids the primary hot water from reaching the primary pump seals. The autonomous cooling circuit which is circulated by a special impeller in the RCP and backed up by an emergency pump with diesel generator power supply is able to take water from the primary circuit and to inject it into the bearing after its cooling when the RCP is not running.

In case of loss of off-site power and a failure of one diesel generator, or in case the emergency pump of the autonomous cooling circuit fails to start, there is a possibility of damage of the primary pump seals.

In case of LOCA the sealing and/or cooling water supply is lost as a consequence of the containment isolation signal. In the worst case of station blackout the cooling the RCP seals is totally lost. The integrity of the RCP seals can be affected resulting in a SB LOCA without HP injection being available.

On the other hand, the RCP manufacturer has paid special attention to this problem. So, the pump seals material is silicate graphite and the RCP themselves are equipped with special heat shields. According to the Russian experts, a test implemented by the manufacturer showed that the seals are capable of remaining tight after 24 hours without sealing and cooling water supply.

There are design differences such as:

- emergency power supplied makeup system which decreases the probability of a total loss of sealing water supply,
- primary circuit isolation valves (failure of one or two RCP cooling circuits), provide a lower probability for RCP seals damages.

However, as the reports on the conditions and results of the seal qualification tests required by the NUSS Code on Design were not made available to date, it is still considered as an open question.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

Inadequate provisions in the design of the main coolant pump to keep the seal tight affect the first Level of plants' defence in depth. A loss of seal injection flow may damage the seal and lead to a SB LOCA. The protection of the second barrier may be affected. Compared to the standard model the issue is of less concern.

COMMENTS AND RECOMMENDATIONS:

1. To resolve the issue, information on the seal qualification tests and their results should be assessed.

   If in case of loss of cooling and sealing water supply (station blackout conditions) the RCP seals integrity cannot be demonstrated, the proposed emergency response procedures to be developed based on the respective accident analyses for total loss of electrical power should consider necessary compensatory measures to maintain the primary circuit integrity (see issue AA13).

2. Consideration should be given to the design of the RCP in WWER-1000/187 with respect to the applicability of a.m. test results (no autonomous circuit).

REFERENCES: [2, 3, 9, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

At Unit 5 main coolant pumps GZN-195 are installed which differ in their features from GZN-195M MCPs. The sealing water is supplied by makeup pumps which could be powered by diesels. In case the makeup pumps are lost, cooling water could be provided by a "passive" system, taking water from MCP discharge, cooled in 2 heat exchangers and supplied to the seals through a check valve. The MCP seals itself in this regime. The MCP could be operated in this regime for 30 minutes and if the normal cooling water is still not available, it has to be shut off. When the MCP is shut off, the supply and drain of the sealing water are isolated and the MCP can stay in this condition for an extended period of time without any problem (such condition is used e.g. during hydrotesting).

Kalinin Units 1 and 2:

Taking into account former restrictive requirements for cooling and sealing water supply to the MCP, the behaviour of the MCP seals in case of long-term loss of sealing water supply has been evaluated. Modifications on the RCP heat shield have been performed.

As stated by the counterpart, the MCP seal will not be impaired in case sealing water supply interruption for a minimum of 48 hours according to tests performed by the MCP manufacturer (test report was presented for examination).

Even in the case of loss of sealing and cooling water supply, the seals are capable of remaining tight for a defined time period.

South Ukraine Units 1 and 2:

The sealing and cooling water supply to the main coolant pumps (MCP) could be interrupted:

- in case of spurious closure of one or more containment isolation valves in the water supply lines,
- in the event of LOCA by the corresponding containment isolation logic (pressure or temperature increase inside the containment),
- in the event of a loss of off-site power supply and a failure of the emergency diesel generators.

It was noted by the counterpart, that at Units 1 and 2 the primary circuit makeup pumps are backed up by the existing three emergency diesel generators (2nd category of emergency power supply). In case of loss of off-site power supply, the makeup pumps, the cooling circuit pumps and the MCP intermediate cooling circuit pumps are supplied from these emergency diesel generators.

When the intermediate cooling circuit pump fails to start (e.g. loss of off-site power supply and a failure of the emergency diesel generators) or the cooling water supply to the intermediate cooling heat exchanger is interrupted, the cooling water supply to the MCP seals can be restored via an additional line from the primary circuit makeup pumps (connection between the sealing water supply line and the intermediate cooling circuit).

Taking into account former restrictive requirements for cooling and sealing water supply to the MCP, the behaviour of the MCP seals in case of long-term total loss of their cooling has been evaluated. According to a test performed by the MCP manufacturer, the MCP seal remains tight even without sealing and cooling water supply of sealing water for a minimum of 24 hours. Although detailed information on the test conditions and results were not available during the meeting, it was stated by the representative of the Kurchatov Institute, that the tests have been performed under nominal RCS conditions and considering the MCP run out time.

With reference to the plant documentation, the MCP thermal shield has been modified (status of SNPP design safety issues). Other measures, such as the modification of the anti-reverse device and the installation of an automated vibration control system, have been completed to improve the reliability of the MCP.

REVIEW AREA/ISSUE NUMBER: Systems 4 (S 4)

ISSUE TITLE: Pressurizer safety valves' qualification for water flow

ISSUE CLARIFICATION: Protection of the primary circuit against overpressure during incidents is provided by the pressurizer (PRZ) safety valves. The ‘small series’ WWER-1000 units are equipped with three pilot operated valves with 50% steam blow-off capacity each. The first safety valve acts as the control safety valve, the others as working safety valves (different set points).

Application of the single failure criterion to these safety valves in connection with a pressure transient results in a postulated failure to close after operation with steam (loss of leaktightness). The risk of safety valve failure to close is high if the valve would release water or steam-water mixture during this transient.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Lack of qualification of PRZ safety valves for water or steam-water mixture discharge would seriously affect Level 3 of plants’ defence in depth. Inappropriate design with respect to possible water loads could lead to a stuck-open of the PRZ safety valves and could make pressure transients in some cases run into SB LOCAs. Long term core cooling might be impaired by increased challenge of the safety systems.

COMMENTS AND RECOMMENDATIONS:

1. In order to resolve the issue, it is necessary to confirm that the safety valves have been tested to carry water. Information on the qualification tests and their results should be made available and assessed.

2. If adequate qualification cannot be confirmed, a replacement of these safety valves should be considered. In connection with a possible valve replacement it would be worthwhile to provide a signal which indicates valve position.

3. Consideration should be given to perform corresponding fluid-dynamic and mechanical strength analyses of the whole PRZ safety valve system, taking into account the specific ATWS and primary bleed requirements (Interface to safety issues AA 9, AA 12).

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The installed pressurizer safety valves (Bopp-Reuter) do not have corresponding qualification documents with regard to steam-water mixture and water flow operability. The plant is planning to replace these valves by valves which are qualified for steam, steam-water mixture and water flow (Bopp - Reuter UF50024-100). No activities were started yet.

Kalinin Units 1 and 2:

Decision on the modifications of the existing PRZ safety valves has been taken. However, PSAs performed for KNPP as well as for NNPP show a low contribution to risk for core melt initiated by the stuck open of the PRZ safety valves.

South Ukraine Units 1 and 2:

Units 1 and 2 are equipped with three pilot operated PRZ safety valves with 180 t/h steam blow-off capacity each (Type Bopp & Reuther, Type SiH 3112). The performance of qualification tests with support of the manufacturer is planned within the TACIS programme. The proposed measure covers the requirements related to the possibility of steam-water blowdown in case of ATWS and primary side "bleed" procedures to be performed in the frame of BDBA management.

REVIEW AREA/ISSUE NUMBER: Systems 5 (S 5)

ISSUE TITLE: ECCS sump screen blocking

ISSUE CLARIFICATION: Contrary to the standard model WWER-1000/320, there are three independent containment sumps in WWER-1000 models 187, 302 and 338. The openings of the containment bottom plate are covered with screens (wire mesh) which are intended to prevent debris penetration to the suction of the ECCS and containment spray system.

The primary and secondary system equipment and pipelines inside the containment are covered with fibrous thermal insulation. The thermal insulation used inside the containment and the limited area of the screen above the sumps form a combination that raises a safety concern regarding the possibility of maintaining ECCS circulation after a medium or large LOCA. Operational experience based on recent events in Sweden and in the USA have demonstrated that even a relatively small amount of similar fibres can effectively block a large screen area. In addition, tests at Zaporozhe NPP [26] have demonstrated that a small amount of fibrous material can plug the ECCS heat exchangers.

The sump screen blocking effect is dependent on many factors such as the type of insulation, size and arrangement of sump screens and the material transportation mechanism to the sump.

As mentioned above, there are three independent containment sumps. Beyond that, the ECCS water storages are located in separate tanks outside the containment. Switch over of the LP injection and containment spray pumps to the sump suction line takes place after emptying the available ECCS water storages (preliminary 1500 m^3).

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

This issue was identified on the basis of international operating experience. The insufficient design of screens and filters inside the sumps and of thermal insulation of equipment and pipelines inside the containment affects Level 1 of defence to prevent abnormal operation and failures. Under LOCA conditions, a common mode failure by clogging the sump screens and/or the ECCS heat exchangers may occur. The risk of loosing ECCS recirculation seriously affects Level 3 of defence to control accidents within the design basis. The main safety function is thus questionable in the extreme situation of affecting all three containment sumps for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. The proposals made by Russian design organizations for measures to ensure coolant recirculation through the containment sump screens in connection with LOCA should be implemented efficiently to reduce the risk to a tolerable level. The sump screen performance should be confirmed by tests that simulate the actual flow conditions (density of fibres in coolant flow rate).

2. Since for all three plants the replacement of the thermal insulation is planned, the behaviour of new insulation material on sump screens should be thoroughly tested under comparable ‘small series’ WWER-1000 conditions.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

For the short term phase, the following compensatory measure is planned to be implemented:

- Installation of a connection line between the ECCS heat exchangers and the spent fuel pool cooling heat exchangers. For the long term cooling of the core, a switch-over from the ECCS heat exchanger to the spent fuel pool heat exchanges is done manually.

The following measure has been implemented and introduced in EOPs:

- A special procedure to keep one of the three emergency core cooling and spray system channels in "clean" conditions, i.e. 15-30 minutes after actuation of these systems, one channel is switched to a standby mode and switched back into operation even when the two operating ECCS heat exchangers are degraded or blocked by the insulation material.

The existing sump screens have been modified in order to decrease the risk of sump screen blocking. However, further operational tests to evaluate the upgraded screen construction at Unit 5 have not been performed to date.

In order to improve the ECCS reliability for long-term core cooling it is planned to replace the existing thermal insulation. In the short term phase containment sump filters will be installed at Unit 5 which are similar to those of Loviisa plant.

Kalinin Units 1 and 2:

Organizational and technical measures to prevent or limit blockage of filters by thermal insulation are described in “Technological regulations on safe operation” and in “Emergency Operating Procedures.”

Activities are undertaken in designing new filters in the framework of the TACIS Program.

Testing of new thermal insulation for equipment located in the containment are being carried out.

South Ukraine Units 1 and 2:

For the short term phase, the following compensatory measures are planned to be implemented (LTISP, i.4.5.):

- Installation of a connection line between the ECCS heat exchangers and the spent fuel pool cooling heat exchangers. For the long term cooling of the core, a switch-over from the ECCS heat exchanger to the spent fuel pool heat exchangers is done manually.

Remarks: At SNPP Units 1 and 2, the two existing spent fuel pool cooling heat exchangers and the three ECCS heat exchangers are located in different compartments.

- Implementation of a special procedure to keep one of the three ECCS channels in "clean" conditions, i.e. 15–30 minutes after actuation of the ECCS, one channel is switched to a standby mode and switched back into operation even when the two operating ECCS heat exchangers are degraded or blocked by the insulation material.

The existing sump screens have been modified in order to decrease the risk of sump screen blocking. However, further operational tests to evaluate the upgraded screen construction at SNPP Units 1 and 2 have not been performed to date.

In order to improve the ECCS reliability for long-term core cooling it is planned to replace the existing thermal insulation by thermal insulation developed in the research institute "Energomash".

REVIEW AREA/ISSUE NUMBER: Systems 6 (S 6)

ISSUE TITLE: ECCS suction line integrity

ISSUE CLARIFICATION: The ‘small series’ WWER-1000 plants are equipped with ECCS and containment spray systems that have a similar design basis and similar basic configuration as in western PWRs. These systems have 3 × 100% redundancy. Contrary to the standard model, there are three storage tanks located outside the containment with a total borated water capacity of 1500 m^3.

Three containment sumps are located in the reactor building at the containment bottom level. Each of the LP safety and containment spray system trains has a separate suction line from the corresponding water storage tank (physically separated from the common ECCS pump room) and, via the ECCS heat exchanger from the 3 containment sumps. Each suction line from the sumps is equipped with a containment isolation valve. The distance from the containment bottom to the first isolation valve in the LP ECCS sump suction line is 1 to 1.5 m only; the rupture probability for a passive component considering the existing NDT monitoring is considered to be very rare.

A postulated passive single failure in one of the three suction lines between the containment sump and the isolation valve during LB LOCA would become a source of a flooding scenario in the common ECCS room disabling all ECCS trains due to flooding or worse ambient conditions. Such a failure may lead to a bypass of the containment and in the long term to severe core damage.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The vulnerability of the ECCS suction lines to single failures affects Level 3 of plants’ defence in depth to control accidents within the DB. In these LOCA scenarios, the main safety function cooling the fuel would be impaired in extreme situations. However, the probability of an undetected leak before the accident or of a leak occurring during the recirculation phase is considered very low, since the suction lines are operated under conditions of low pressure and temperature and there are three independent trains in comparison with the standard model 320.

COMMENTS AND RECOMMENDATIONS:

1. Adequate periodical tests, inspections and QA programmes as well as material non-destructive control should be applied.

2. Potential solutions for elimination of the consequences of vital piping section failure should be studied and implemented in the long term, if found necessary and feasible (see issue S 16).

3. An evaluation of the consequences of breaks of on the ECCS when it is used during cold shutdown with respect to necessary operator intervention should be carried out. Depending on the results of these analyses, the need for complementary measures should be defined.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

There are three emergency boron solution tanks of 582 m^3 each located outside the containment. There is a low pressure injection pump and a sprinkler pump connected to each tank. In addition, there is one concentrated boron solution tank of 150 m^3 connected to three high pressure injection pumps. Connecting pipes between high pressure injection pumps and the three boron solution tanks were installed. Further modification in the ECCS system are still planned. There is an in-service inspection programme in place of ECCS tanks (100% NDT once in 6 years) and piping 100% NDT once in 4 years) examination.

It was noted that measures to address the issue were implemented and further measures are planned.

Kalinin Units 1 and 2:

Non-destructive testing of the vital piping sections are performed as an adequate resolution of this concern. In addition, a special programme for rest lifetime evaluation of the ECCS suction line is under preparation

South Ukraine Units 1 and 2:

Annual periodical inspections of the ECCS equipment and the metal state of the pipes are performed. In addition, a special programme for rest lifetime evaluation of the ECCS suction line is under preparation in the frame of the TACIS programme.

It was noted by the counterpart that, for the time being, the SNPP does not consider any further upgrading measures.

REVIEW AREA/ISSUE NUMBER: Systems 7 (S 7)

ISSUE TITLE: ECCS heat exchanger integrity

ISSUE CLARIFICATION: Residual heat is removed in shutdown and emergency conditions through heat exchangers of the low pressure safety injection systems. At SNPP, these are cooled by the essential service water system with three independent semi-open cooling circuits. Heat is removed by cooling towers with natural air circulation. For NNPP the heat sink for essential service water is the cooling pond of Unit 5 (trains 1 and 2) or discharge circulation water channels of the WWER-440 units in operation. At KNPP the ECCS heat exchangers are cooled by a closed loop intermediate cooling system. The heat from the intermediate cooling circuit is removed by the service water system to spray ponds/lake.

The ponds are isolated against the soil, but it is unavoidable that airborne dirt and/or biological matter enters the heat exchanger, resulting in fouling and consequent degradation of cooling. If there is a blockage of the heat exchanger which would reduce the heat transfer, the pressure difference over the separation plate between inlet and outlet chambers of cooling water could damage the heat exchanger.

For NNPP and SNPP, the ECCS and spent fuel pool cooling system heat exchangers form part of the second containment barrier. If the primary coolant pressure is higher than the essential service water system pressure (RHR line in operation), primary coolant could be released to the atmosphere through the essential service water system, thus bypassing the containment. If the ECCS pressure is lower than the essential service water system pressure (stand-by mode or during LOCA), a leak in the heat exchanger would cause boron dilution in the affected ECCS.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified from operating experience for the standard model at Khmelnitzki NPP. The vulnerability of the ECCS heat exchangers to be damaged by several mechanisms affects Level 3 of plants' defence in depth to control accidents within the design basis. Diagnostics means are not available at all units to detect, at an early stage, any damage of the ECCS heat exchangers which affects Level 2 of plants' defence in depth to control abnormal operation and to detect failures. Depending on the conceivable scenarios within the DBA envelope, each main safety function can be impaired by diluting the primary coolant, by losing some primary water inventory, or by bypassing the third barrier.

COMMENTS AND RECOMMENDATIONS:

1. Definition of reliable means to monitor permanently the fouling of the heat exchangers cooled by the service water system and means to clean up them when it is necessary. If necessary, corrective measures to protect the heat exchangers from fouling and other damage mechanisms, e.g. by using chemical reagents in the essential service water system, should be implemented.

2. Means to improve the periodical inspections and examinations to be performed during unit outages (visual and ultrasonic tests of the ECCS heat exchanger integrity).

3. The reliability of activity release monitoring at the outlet of the ECCS heat exchangers should be analysed in order to identify if improvements are necessary.

4. The probability and the consequences of a dilution transient caused by an ECCS heat exchanger leak should be assessed. If necessary, compensatory measures with respect to boron concentration monitoring to be considered.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The ECCS heat exchangers are cooled by the service water system taken from Unit 5 cooling pond or Units 3 and 4 channel. The service water is not continuously chemically treated but the heat exchangers are periodically flushed by chemically treated water. To date, there have been no problems observed with heat exchangers fouling. There is continuous radioactivity monitoring downstream of the heat exchangers and if the radioactivity exceeds the set point, the heat exchangers are isolated on both primary and secondary sides. ECCS heat exchangers are connected only when the primary pressure is down to atmospheric pressure. This is ensured by organizational measures.

Kalinin Units 1 and 2:

This issue is not applicable to KNPP Units 1 and 2. Heat removal from the ECCS heat exchangers is provided by a closed-loop intermediate cooling circuit.

South Ukraine Units 1 and 2:

A reconstruction of all ECCS heat exchangers have already been performed.

A project for backing up of two or three ECCS heat exchangers by spent fuel pool cooling exchangers has been carried out and the corresponding measures will be realized during the next outages. Furthermore, it is foreseen to modify the installed boron meters, or to replace them by NAR-12 type boron meters which provide more precise control of possible boron dilution within the ECCS.

REVIEW AREA/ISSUE NUMBER: Systems 8 (S 8)

ISSUE TITLE: Power operated valves on the ECCS injection lines

ISSUE CLARIFICATION: The motor operated valves on the high pressure boron injection and high pressure and low pressure injection pump discharge lines are closed during normal operation. Since these valves are located inside the containment and they must be opened under accident conditions, their reliability needs to be very high.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The use of power operated isolation valves rather than check valves on the ECCS lines inside the containment is a departure from international practice.

COMMENTS AND RECOMMENDATIONS:

1. The possibility of installing normally closed isolation valves on the ECCS lines outside containment should be investigated. Measures should be taken after the investigation, if appropriate.

2. The system solution related to this issue should be reviewed (supply of compressed air, qualification of valves) and if applicable, compensatory measures developed and implemented.

REFERENCE: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

At the NNPP Unit 5, the isolation valves on the injection lines of high pressure and low pressure injection system are motor-operated. One isolation valve is installed outside the containment and two isolation valves are inside the containment. These valves are backed up electrically by the diesel generators.

During normal operation, the valves are closed. The valves will automatically open, actuated by an ECCS injection signal.

The isolation design of the ECCS injection line at Unit 5 is different from both standard model 320 and the other ‘small series’ models 302, 338.

Kalinin Units 1 and 2:

Pneumatic actuation valves of TJ11/12/13 S04, S05 (BABCOCK) types are installed at the high pressure emergency injection lines. These valves are of normal closing type and in case of air loss the springs close the valve.

Pneumatic actuation valves of TH11/12/13 S06, S07 (BABCOCK) types are installed at the low pressure emergency core cooling lines. These valves are of normal closing type and in case of air loss the springs close the valve.

The pneumatic actuation valves are supplied with air via pressurized air supply system which is a safety system.

The valves are closed at normal operation. The valves are located inside the containment.

South Ukraine Units 1 and 2:

The high pressure safety injection system valves TJ10/20/30 S04, S05 and the low pressure safety injection valves TH10/20/30 S06, S07 are of pneumatic type (Babcock).

These valves are located inside the containment and closed during normal operation of the plant. As stated in the plant response, this arrangement is required by the national safety regulations, because the check valves downstream of the isolation valves are not considered as isolating devices to protect the low pressure lines of the safety injection systems connected to the primary circuit.

The pneumatic drives used for the ECCS isolation are of different type, in order to secure the fail-safe direction of the corresponding valves. In case of a failure of the pressure air supply to the pneumatic valves of the high pressure injection system (TJ10/20/30 S04, S05), these valves will be actuated in an open position by spring force. In case of a failure of the pressure air supply to the pneumatic drives of the low pressure injection system (TH10/20/30 S06, S07), these valves will remain in the actual position, i.e. closed during normal operation of the unit to protect the low pressure lines or opened in case of an ECCS request to supply emergency water.

The majority of valves used for the containment isolation, in case of a failure of the pressure air supply will be actuated in a closed position by spring force in order to prevent radioactivity release.

Technical solutions to bring the ECCS isolation valve arrangement in line with western practice (i.e. to change the normal position or to install the normally closed valves outside the containment) are not applicable to SNPP Units 1 and 2 due to regulation requirements (loss of primary circuit integrity, containment bypass risk).

REVIEW AREA/ISSUE NUMBER: Systems 9 (S 9)

ISSUE TITLE: Steam generator safety and relief valves' qualification for water flow

ISSUE CLARIFICATION: The overpressure protection of each SG is carried out by one relief valve (BRU-A) and by two safety valves installed before the main steam fast isolation valve (MSIV). In case of a primary to secondary leak, the primary circuit water can quickly fill the SG and the steam line up to the BRU-A. The lack of qualification of relief and safety valves to operate with water or water-steam mixtures can then lead to their failure to reclose after opening. In such an event, there is a potential for containment bypass leakage discharging the primary water radioactivity. Long term cooling may be endangered in case the steam dump to the atmosphere cannot be isolated.

There is a design difference as opposed to the standard model 320. For NPPs of WWER-1000/302 and 338 type, the steam generator relief valves (BRU-A) located upstream from the MSIV can be isolated. At NNPP (WWER-1000/187) isolatable BRU-As are located downstream from the MSIV on the main steam header. Information on qualification tests performed by the SG safety and relief valve manufacturers are not available.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **II**

JUSTIFICATION OF RANKING:

A lack of qualification of relief and safety valves for water flow affects Level 3 of defence, since this represents insufficient design of a safety system. In a DBA scenario of a large primary to secondary leak (see issue CI 6), the cooling could be impaired only if the BRU-A is not reliably isolatable.

COMMENTS AND RECOMMENDATIONS:

1. In order to resolve the issue, information on the qualification tests for the SG safety and relief valves and their results should be assessed. Considering the foreseen replacement of the SG safety valves in order to meet OTT-87 requirements, the corresponding qualification requirements should be addressed.

2. In addition, consideration should be given to the operability of BRU-A and BRU-A isolation valves to operate with water or steam-water mixtures.

3. Accident management procedures and compensatory measures to reduce the probability of the need to open non-isolatable safety valves should be considered (NNPP).

REFERENCES: [2, 3, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Documentation to certify qualification of the valves is not available at the plant. However, the plant staff considers that the valves are capable to be used for water flow due to their design. In any case it is intended to replace the safety valves with ones qualified for steam, steam water mixture and water flow (TACIS).

Kalinin Units 1 and 2:

It is planned to replace the SG safety valves with the ones qualified for steam, steam water mixture and water flow (TACIS).The electrically actuated regulating valve of 973-500-es manufactured by

Chekhov Company is installed before the fast acting relief valve. This regulating valve is designed for operation in steam and water and capable to disconnect the fast acting relief valve in case it failed to close.

South Ukraine Units 1 and 2:

It was stated that the BRU-A is able to operate on steam-water flow. However, information on qualification tests results were not available.

The installed SG safety valves are not qualified according to the new classification requirements for NPP equipment. Therefore their replacement is planned within the TACIS programme. It is assumed that the new safety valves will be qualified for water and steam-water flow.

In relation to the third recommendation, special measures and procedures to reduce the consequences of a large coolant leakage from the primary to the secondary circuit will be elaborated after performance of the corresponding accident analysis (see issue S 2).

REVIEW AREA/ISSUE NUMBER: Systems 10 (S 10)

ISSUE TITLE: Steam generator safety valves' performance at low pressure

ISSUE CLARIFICATION: For NPPs of WWER-1000 models 302 and 338, two types of valves are installed on each steam generator line upstream the MSIV; one isolatable dump valve to atmosphere (BRU-A) and two pilot controlled safety valves. At NNPP (WWER-1000/187) the two safety valves are installed on each steam generator line upstream the MSIV; 4 isolatable BRU-As are located downstream the MSIV on the main steam header (two per MSH half side). The safety valves have the possibility of adjusting the secondary pressure from 84 bars to 30 bars. The relief valve is operable from 74 bars to 1 bar.

The BRU-A is supplied from the uninterruptable emergency power supply system (1st category), and the isolation valve upstream the BRU-A from emergency power supply system (2nd category).

In case of loss of main heat sink (turbine condenser) and/or LOOP with one or two BRU-A unavailable, the remaining relief stations provide sufficient flow capacity for secondary side heat removal during normal operation and cooling down until the ECCS residual heat removal line is put into operation (900 t/h each).

However, for NNPP the possibility of isolation of all BRU-A's from the SG's in transients is of concern due to the high probability of spurious closure of MSIV. An extended scope of manual actions to be performed obstruct a re-opening of these MSIV. The existing SG safety valves do not allow decay heat removal via steam release to atmosphere at low pressure (below 30 bar) so that residual heat removal and primary circuit cooldown by steam dumping to atmosphere could be delayed or interrupted.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

Decay heat removal by feeding the steam generators and relieving at the SG safety valves is the only safe procedure to cool the fuel by the secondary side. Inadequate design of the SG safety valves may affect Level 3 of defence to control accidents within the design basis.

COMMENTS AND RECOMMENDATIONS:

1. Considering the necessary replacement of the existing SG safety valves in order to meet OTT-87 requirements, the corresponding qualification requirements with respect to low pressure performance should be addressed (model 187). For the WWER-1000 models 302 and 338 this issue is of less concern (BDBA conditions, station blackout).

   However, if a decision is taken to install new safety valves operable at low pressure, consideration should be given to the risk of spurious opening of the new safety valves during normal operation of the plant.

2. Even if for the WWER-1000/302 and 338 the present situation seems to be acceptable from the design configuration point of view, consideration should be given to possible failure of more than one BRU-A caused by failures in the electrical supply or I&C control system or due to worse ambient conditions in case of secondary side leakages. If necessary, compensatory measures should be taken immediately.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The safety valves are planned to be replaced by qualified ones for water and steam mixture flow, their performance at low pressure will be ensured in the preparation of technical specifications for the new valves.

Kalinin Units 1 and 2:

There are two types of safety valves at the SG lines of KNPP: the first one is the fast acting relief valve (with release to atmosphere) which is shut off by the regulating valve. There are also two safety valves controlled at the MCR and the ECR. The safety valves at the Unit 2 are planned to be replaced with the new ones designed for operation in two-phase media (TACIS Program). The issue of changing the parameters of SG safety valves (provision of steam dump at pressure below 30 bars) shall be resolved based on the Level 1 PSA results.

South Ukraine Units 1 and 2:

According to the document “Status of SNPP design safety issues” the existing steam generator safety valves will be replaced (TACIS programme). In the technical specification for the new valves their performance at low pressure will be considered.

REVIEW AREA/ISSUE NUMBER: Systems 11 (S 11)

ISSUE TITLE: Steam generator level control valves

ISSUE CLARIFICATION: The original SG level control valves (full-power control valve, startup control valve) have the drawback of erosion damage due to high pressure differential across the valve. Moreover, SG level control valves have a slow response and this negatively affects the possibility of maintaining the SG level during transients. The improper functioning of level control valves also has lead to new initiating events in the form of transients.

During plant operation, modifications and changes of the control valves have improved the situation to some extent. PSA studies have indicated the importance of reliable SG level control.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: **I**

JUSTIFICATION OF RANKING:

The steam generator level control valves used are below the quality level of the state of the art equipment. This represents a deviation from international practice.

COMMENTS AND RECOMMENDATIONS:

1. The SG level control valves replacement is supported.
2. Based on operation experience, consideration should be given to improve the reliability of the SG level measurement and control circuits.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A device to prevent the erosion damage of the valve is installed at the existing SG level control valve at NNPP Unit 5. The response time of the valves is improved from 120 s to 55 s, and this meets the requirements of SG level control in transients. There is no problem with level control valves in startup modes. However, there is a problem in valves control electronic circuit due to the single channel design.

Currently, NNPP is working on the replacement of SG level control valves with the valves in conformity with the qualification requirements for NPP equipment, with increased reliability and less power consumption. The carbon steel pipelines of level control valves used for startup and shutdown were replaced by austenitic steel ones. Multi-channel valve control circuit will also be realized in the Automatic Turbine Control System-500 which is being implemented at Unit 5.

Kalinin Units 1 and 2:

It is planned to replace the original level control valves with newly designed ones within the framework of the TACIS Program.

South Ukraine Units 1 and 2:

In accordance with the “Status of SNPP design safety issues”, it is planned to replace the existing full-power control valves by new one which comply with new classification requirements for NPP equipment.

The new valves will provide a full scale control range and adequate stroke time (level control during main feedwater and auxiliary feedwater supply to the SG).

In addition, a new autonomous SG water level control system will be installed.

REVIEW AREA/ISSUE NUMBER: Systems 12 (S 12)

ISSUE TITLE: Ventilation system of control rooms

ISSUE CLARIFICATION: The ventilation of the main control room (MCR) and the emergency control room (ECR) should be designed so that penetration of radioactive or toxic substances to those rooms in emergency conditions can be prevented.

As opposed to the standard model 320, the ventilation systems of the two control rooms are separate from each other. The fresh air intake to these two systems is from a common point on the roof of the turbine hall. At present, the ventilation systems are not capable of filtering the intake air in case of outside radioactive releases. Consequently, there is the potential hazard of breathing the contaminated air in the main control rooms if a serious accident occurs. The main air ducts for MCR and ECR air supply are not provided with smoke detectors, the signal of which would be used to cut off the ventilation system.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE : **II**

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards: OPB-88 [4], NUSS Code on Design [9] and INSAG-3 [7].

Inadequate design of the main air ducts of the control rooms affects Level 1 of defence. In emergency situations, all main safety functions may be impaired or questionable in beyond DBA conditions, because the habitability of the control rooms cannot be ensured.

COMMENTS AND RECOMMENDATIONS:

Backfitting of the ventilation system of WWER-1000 main control rooms.

REFERENCES: [2, 3, 4, 7, 9, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

In case of outside radioactive releases, the ventilation systems will be manually closed and air recirculation system will be used.

The ventilation systems, including their devices, air ducts, ventilators and supporting structures, are not designed to withstand seismic effects.

Kalinin Units 1 and 2:

At present, the ventilations systems at Units 1 and 2 are not capable of filtering the intake air in case of radioactive releases from outside.

The issue of autonomous life support system for the MCR and the ECR has been investigated, but there isn’t any room for its installation. The air for the MCR and the ECR is taken at locations of different height and distance from each other.

There is a project for the installation of smoke detectors.

In case of external radioactive releases the ventilation system will be switched off by operating personnel and the air recirculation system will be used.

Conditioners will function in recirculating mode in the MCR and the ECR.

Ventilators (conditioners) of the MCR and the ECR are supplied by electrical power from different bus bars powered by diesel-generators.

South Ukraine Units 1 and 2:

At present, the ventilations systems at Units 1 and 2 are not capable of filtering the intake air in case of radioactive releases from outside. In relation to fire protection requirements, it should be noted that the air ducts to the control rooms are not equipped with smoke detectors.

REVIEW AREA/ISSUE NUMBER: Systems 13 (S 13)

ISSUE TITLE: Hydrogen removal system

ISSUE CLARIFICATION: In the WWER-1000 NPP design, the hydrogen removal system was not considered for use during LOCA (design basis accidents) and severe accidents. Only on-line hydrogen concentration monitoring is carried out inside the containment in four location points (SG compartments, valve and piping compartment and containment dome).

The concentration of hydrogen generated during LOCA (DBA) or severe accidents inside the containment can reach a detonation level which could damage the containment.

MAIN SAFETY FUNCTIONS AFFECTED: Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue has been found to be a deviation from OPB-88 [4] and INSAG-3 [7]. Insufficient hydrogen removal systems for use during DBA scenarios and severe accidents may seriously affect Level 4 of defence, i.e. to avert damage from the third barrier. The safety function to confine the radioactive material may be impaired in LOCA scenarios or questionable in case of beyond DBA emergency situations.

COMMENTS AND RECOMMENDATIONS:

1. Study hydrogen generation and accumulation effects within the containment and take measures based on the results of these studies.
2. Implement H_2 monitoring and removal systems inside the containment The measures proposed to control hydrogen should be introduced as soon as realistically possible in all WWER-1000 units as a matter of urgency.

REFERENCES: [2, 3, 4, 7, 9, 25]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Unit 5 does not have a containment hydrogen removal system. As a compensatory measure, a design modification was developed for discharge of hydrogen-containing medium from the containment to the hydrogen burning system in the primary makeup system. The existing two-train hydrogen burning system is used to remove the hydrogen in the dearator of the makeup system. Two fans, one for each train, are to be connected from the containment to the inlet of the hydrogen burning system. The modified system does not meet the requirements of safety systems, because of its compensatory nature.

For the hydrogen removal from the containment in case of beyond design basis accidents, design and research work is being considered. The calculation of hydrogen generation and distribution within the containment during a reactor core melt accident will be made in co-operation with Siemens company.

Kalinin Units 1 and 2:

The installation of a hydrogen monitoring and removal system is planned within the TACIS programme. The analysis of design basis accidents indicated that a hazardous level of hydrogen concentration might be achieved after 30 days if specific measures would not be taken. A tender was held to motivate technical specifications of the system controlling and removing hydrogen. While conducting the tender, preliminary analysis of hydrogen generation and its penetration and propagation in containment during core melt accidents has been performed. Calculations of the number and locations of passive recombiners of hydrogen supplied by Siemens to provide hydrogen safety during such accidents have been made.

South Ukraine Units 1 and 2:

The implementation of a hydrogen monitoring and removal system is planned to be implemented at SNPP Units 1-3, starting with Unit 3, the standard model 320 unit. At present, corresponding investigations and design activities are being performed.

With reference to the long term safety improvement activities, the following compensatory measures are envisaged before installation of the hydrogen removal system:

- installation of gas analysers for hydrogen concentration monitoring
- hydrogen dilution by operation of the containment ventilation system and additional exhaust systems
- project to release hydrogen-containing medium from the containment to the vent stack by the normal hydrogen igniting system
- hydrogen dilution by periodic operation of the containment spray system.

REVIEW AREA/ISSUE NUMBER: Systems 14 (S 14)

ISSUE TITLE: Boron injection system capability (NNPP)

ISSUE CLARIFICATION: The original design of ‘small series’ WWER-1000 units is not equipped with a high boron injection system for shutting the reactor down during transients and accidents. For KNPP and SNPP a high boron injection system has been designed and installed during the erection and commissioning phase of the plants. The system configuration and water storages are comparable to the design of standard model 320 units which have a three train high boron injection system with PT-6 pumps (160 bar, 6.3 m^3/h).

However, small design differences (e.g. a common boron solution tank for all three trains at SNPP Unit 1 and KNPP Unit 1) exist. The NNPP (model 187) on the other hand does not have a high pressure boron injection system. The emergency power backed-up makeup pumps cannot be accepted as safety classified due to the lack of physical separation and functional isolation.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

Lacking high boron injection system for the ‘small series’ WWER-1000 model 187 unit seriously affects Level 3 of plant’s defence in depth to control accidents within the design basis. The main safety function is questionable for scenarios within the design basis envelope or disabled beyond design basis.

COMMENTS AND RECOMMENDATIONS:

1. At NNPP a safety classified boron injection system for backing-up the primary circuit makeup system is not available. This situation is not acceptable and compensatory measures comparable to the WWER-1000/338 design should be realized in the short term.

2. At SNPP and KNPP the sufficiency of the high boron injection systems with respect to multiple steam line break and ATWS requirements should be verified by corresponding accident analyses (taking into account the more stringent recriticality requirements in the Russian normative documents).

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP Unit 5 does not have a high pressure boron injection system. However, three makeup pumps backed up by the emergency diesel generators deliver high concentrated boric acid (40 g/kg) from the storage tank to the primary system with a flow rate of 60 m^3/h. The makeup pumps are located in a common compartment without physical separation and seismic qualification, and are not considered as safety systems. Connecting pipes have been installed between boron solution tanks of V=750 m^3 and suction lines of makeup pumps.

According to the “Concept of safety upgrading for operating WWER-1000 Power Units”, it is planned within TACIS programme to install three additional high pressure emergency boron injection trains similar to the ones at WWER-1000/320 Units.

Kalinin Units 1 and 2:

An additional high pressure boron injection system exists. The system consists of three PT-6 pumps (160 bar, 6.3 m^3/h) supplied from the emergency diesel generators. High concentrated boric acid 40 g/kg) is stored in one common storage tank (150 m^3, Unit 1) and in three separated storage tanks (3×100 m^3 for Unit 2), respectively. The designer for the reactors of 338 type OKB Gidropress has performed the analysis for the possibility of boric acid supply from ECCS into the primary circuit in case of LOCAs with leak diameter less than 100 mm (actually the analysis was performed for the leak diameters of 50 mm, 80 mm and 100 mm). The Kurchatov Institute has performed spectrometric analysis of leaks on steam pipelines in the secondary circuit.

South Ukraine Units 1 and 2:

An additional high pressure boron injection system has already been implemented in Units 1 and 2 before commissioning. The system consists of three PT-6 pumps (160 bar, 6.3 m^3/h) supplied from the emergency diesel generators. High concentrated boric acid 40 g/kg) is stored in one common storage tank (150 m^3, Unit 1) and in three separated storage tanks (3×100 m^3 for Unit 2) respectively. The boron injection lines are connected to the injection lines of HP ECCS downstream the containment isolation valves.

As proposed in the LTSIP, i.3.4.7, accident analyses will be performed to prove the sufficiency of the existing boron injection systems in case of secondary pipe ruptures.

REVIEW AREA/ISSUE NUMBER: Systems 15 (S 15)

ISSUE TITLE: Feedwater supply vulnerability

ISSUE CLARIFICATION: Emergency feedwater supply vulnerability has been found 'small series' WWER-1000 units.

At Novovoronezh Unit 5 (model 187), the existing emergency feedwater system with three pumps is used to supply feedwater to four steam generators in emergency conditions and in startup and shutdown conditions as well. The three pumps are located in one compartment in the turbine hall. The pumps are separated by walls which can not provide real separation in case of flooding or fire. Due to the lack of seismic qualification, the emergency feedwater system can be lost in case of internal and external hazards. The absence of additional systems for SG emergency feedwater supply leads to a high risk of severe accidents due to common cause failures of the pumps.

A second concern is the lack of functional isolation between the emergency feedwater system trains (supply lines to the SG via headers, connections to the dearators).

Moreover, at NNPP there is no separate EFW piping directly connected to the SGs. At present the respective SG nozzles are plugged and the EFW piping is connected to the main feedwater lines in the turbine hall (see issue IH 7).

At South Ukraine Units 1 and 2 (models 302 and 338), two auxiliary feedwater pumps are located inside the turbine hall for startup and shutdown conditions. The emergency feedwater system pumps are located separate from the auxiliary feedwater system outside the turbine hall. However, the three emergency feedwater pumps are not completely separated from each other and could be lost due to common causes, such as flooding in case of a service water line rupture.

The situation at KNPP is comparable to SNPP, e.g. no sufficient physical separation and functional isolation between the trains, no additional auxiliary feedwater system.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

The vulnerability in feedwater supply seriously affects Level 3 of plant's defence in depth to control accidents within the design basis. The main safety function is questionable for scenarios within the design basis envelope or disabled beyond design basis.

COMMENTS AND RECOMMENDATIONS:

1. Compensatory measures, depending on the plant in question, should be considered in the short term to prevent common cause failure of all available SG supply sources. In the long term, an additional emergency feedwater system should be provided (NNPP). The equipment of additional emergency feedwater system should be qualified for operation in conditions of high humidity.

2. Further analyses including Level 1 PSA with respect to the potential risk of common cause failures should be performed for all units. The results of these analyses would permit decisions to be taken on further upgrading measures, e.g. emergency power back up for the auxiliary feedwater pumps (SNPP, KNPP).

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Installation of additional emergency feedwater system with three trains which meets the safety requirements is included in the safety improvement programme "Safety Improvement Concept for WWER-1000 Operating Unit with B-187 Reactors, Item 23". Each train contains a feedwater pump, a demineralized water tank of 500 m^3, control and stop valves and piping. Each of the two pumps is connected to the header, supplying water to two steam generator. The third pump can supply water to any of the four SGs. The pumps will be seismic resistant.

After the installation of new emergency feedwater system, the existing three pumps in the turbine hall are to be used as auxiliary feedwater pumps to maintain normal water in SGs during transients and to supply feedwater to SGs during startup and shutdown.

Kalinin Units 1 and 2:

The emergency feedwater system is qualified in accordance with the on-site seismic requirements. There are three emergency back-up chemically pure water tanks (1000 m^3) at each unit. This volume is sufficient to provide the residual heat removal for more than 24 hours. The system of feed water supply for the steam generator main header from the de-aerator via auxiliary feed water pumps is designed to provide the water supply in case of the emergency water supply failure. The intakes of the emergency feed water tanks are connected to the D-7 de-aerator and emergency back-up chemically pure water tanks. The emergency feed water system is located in separate compartments. This location does not exclude the possibility of RL system partial failure to perform as expected in case of multiple pipeline ruptures and flooding of all the compartments.

South Ukraine Units 1 and 2:

The emergency feedwater system is qualified in accordance with the on-site seismic requirements. After a first review of the system, it seems that the emergency feedwater supply lines are not endangered by a main feedwater or main steam line break, but needs to be validated (see issue IH 7). No actions are planned at the moment.

REVIEW AREA/ISSUE NUMBER: Systems 16 (S 16)

ISSUE TITLE: Physical separation and functional isolation of the ECCS

ISSUE CLARIFICATION: Some basic principles, such as physical separation and functional isolation between redundant systems important to safety, were not fully applied due to the lack of proper rules and standards during the design phase of 'small series' WWER-1000 NPPs.

In particular, most of the equipment of the three ECCS trains, such as the boron injection pumps, the emergency core cooling pumps and containment spray pumps are located in one room below the containment floor separated by fire walls. In case of any component failure such as a leak in the fluid retaining boundary or any valve spindle seal or pump packing, the complete ECCS would be at risk of failure due to adverse ambient conditions. There is a potential risk of internal flooding which could originate from ECCS pipe rupture as well as from a rupture of the technical water system pipelines located in this area.

Furthermore, a large amount of combustible material (reactor coolant pump lubrication) is located in a room separated by fire resistant walls and doors from the ECCS compartment.

With respect to functional isolation requirements, it should be noted that specific design features, such as common storage tanks for HP ECCS, common suction lines of redundant ECCS trains, common injection lines for boron injection and HP ECCS (for KNPP and SNPP) and the single letdown line for primary side residual heat removal, do not comply with current design principles as required in national and international rules applied for systems evaluation.

At NNPP, the ECCS residual heat removal line can be blocked by a single failure of 1 of 2 valves to open, and both valves are located inside the containment and cannot be opened manually in case of SB LOCA.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

Lack of physical separation and functional isolation of the most important components of the ECCS may seriously affect Level 3 of plants' defence in depth to control accidents within the design basis. In the event of a common cause failure the main safety function is questionable or can even be lost in the worst case.

There are two possible scenarios to be considered:

1. In case of an initiating event such as flooding by technical water or fire in the ECCS compartment, all equipment needed for normal cold shutdown might be lost due to a common internal cause. Without the primary side heat removal path, safe plant conditions can only be provided for a limited time depending on the amount of feedwater available for secondary side heat removal.

   The risk of internal flooding leading to common cause failure of the ECCS is considerably reduced in KNPP only, because there is an intermediate cooling circuit with a volume of less than 100 m^3. This amount of water is not sufficient to short-circuit the pumps as they are installed on plinths of 0.5 m height.

2. In case of LOCA (<10-4/year) with a passive single failure in any pipe in the ECCS compartment to be assumed in the long-term phase (<10-2/year) and an operator failure to isolate the ruptured piping (<10-2/year), the complete ECCS might be at remote risk of failure due to adverse ambient conditions.

COMMENTS AND RECOMMENDATIONS:

1. The design weaknesses and their specific features at all three plants are well known and should be addressed immediately in corresponding modernization programmes. Based on walkdown results, the risk potential for internal hazards (fire, flooding or HEPB) should be assessed and decreased as much as possible by implementing adequate compensatory measures. Realization of measures to ensure physical separation of the redundant trains of the ECCS should be a priority. It is recommended to perform further studies whether or not the main safety function would be provided in the long-term post accident period.

2. The qualification status of electrical and I&C equipment located in the ECCS compartment should be reconsidered with regard to worse ambient conditions (high temperature and humidity) to be assumed after loss of integrity of the ECCS fluid retaining boundary.

3. The results of PSA studies completed recently at NNPP and KNPP would permit decisions to be taken on further upgrading or compensatory measures to improve the reliability of the ECCS. Consideration should be give to the aspects of physical separation and functional isolation of the ECCS support systems (cooling water supply, electrical and I&C systems).

4. At NNPP, the single residual heat removal line with two isolation valves to be opened for primary side heat removal is an issue of high concern and should be solved in the short term.

REFERENCES: [2, 3, 4, 7, 9, 10]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

It was noted that no action has been considered.

Kalinin Units 1 and 2:

More strict metal inspection, more severe requirements for equipment service, operation and testing of ECCS systems located in one compartment and separated by fire walls are provided. To ensure physical separation of the ECCS, the plant also plans to close the passages of 2×3 m^2 in the fire walls by leaktight doors.

The analysis of fire and flooding hazards was performed for the ECCS compartment (NA-G255). On the basis of this analysis it was recommended to make fire protective coating of metal structures in ECCS premises in order to prevent common cause failure. Analysis of internal flooding risk (see IH 5) has been carried out in the framework of the "BETA" project.

South Ukraine Units 1 and 2:

The plant has taken compensatory measures on making on-the-spot inspection and clarifying the status and condition of the electrical and I&C equipment. The plant also plans to install partitions in the A001 compartment to ensure physical separation of the ECCS. It is also planned to analyse a possible common cause ECCS failure during a primary circuit leak in the PSA and to analyse DBA and BDBA taking into account the probability of ECCS failure.

REVIEW AREA/ISSUE NUMBER: Systems 17 (S 17)

ISSUE TITLE: Limited boric acid storage for HP injection

ISSUE CLARIFICATION: In the event of LOCA the HP and LP emergency core cooling pumps, as well as the boron injection pumps (WWER-1000/302, 338) are started automatically by corresponding safeguard actuation signals. In the first stage, the HP injection pumps and boric acid pumps are connected to a storage tank for high concentration boric acid (Unit 1 of SNPP and KNPP: 1×150 m^3 or Unit 2 of SNPP and KNPP: 3×75 m^3).

After supplying the high concentration boric acid inventories to compensate coolant leakages and to provide sufficient subcriticality for safe shutdown of the plant, the HP injection pumps are switched over to the ECCS storage tanks (3×500 m^3), which feed the LP ECCS and containment spray pumps.

After injection of the available ECCS boric acid inventories, the primary circuit should be sufficiently de-energized by secondary side cooldown to provide the operating conditions for the LP core cooling pumps. Otherwise, long-term post-LOCA core cooling could be interrupted due to the limited boric acid inventories for HP injection.

In contrast to the WWER-1000 standard model 320, the HP safety injection pumps cannot be operated in a closed-loop circuit via the containment sump.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Limited boric acid storage for HP injection affects Level 3 of plants' defence in depth to control accidents within DB. The main safety function is impaired for DBAs. However, taking into account the reliable water storages on the secondary side, the long-term core cooling could be impaired in case of failure combinations or operator errors.

COMMENTS AND RECOMMENDATIONS:

1. It is recommended that safety analyses be performed taking into account the specific requirements for long-term core cooling in case of SB LOCA. Consideration should be given to possible failure combinations and time margins for manual actions. With regard to necessary operator actions to be performed during the accident, alarm signalization with high reliability would decrease the probability of operator failures.

2. The results of these analyses would permit decisions to be taken on necessary upgrading measures.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP is considering the connection of the HP safety injection pumps to the containment sumps in the actions to be taken to cope with the transients related to the pressurized thermal shock.

Kalinin Units 1 and 2:

OKB Gidropress has investigated the possibility of boric acid supply into the primary circuit in case of LOCAs with leak diameter less than 100 mm (actually the analysis was performed for the diameters of 50 mm, 80 mm and 100 mm).

The analysis of results made it possible to define personnel actions while using the boron injection system to provide effective reactor transition into the safe state.

No design deficiencies in the boron injection system (TJ) preventing it were detected.

South Ukraine Units 1 and 2

The safety issue was not addressed in the long term safety improvement programme for Units 1 and 2 at SNPP.

3.5. INSTRUMENTATION AND CONTROL

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 1 (I&C 1)

ISSUE TITLE: I&C reliability

ISSUE CLARIFICATION: The I&C equipment of 'small series' WWER-1000 units are based on a technology that is known to present reliability problems. The failure modes found, including relay contact oxidation and low insulation resistance of wiring and terminals, are typical of the technology. Operational experience has shown that the I&C failure rate is relatively high and can cause power reduction.

The I&C system design does not include:

- an analysis of the impact of reactor I&C systems and unit I&C systems on possible system failures

- an analysis of the reliability of hardware and software and of the system as a whole.

The I&C systems were not fully designed to provide automatic and/or automated diagnosis of operating states and conditions of I&C hardware. Only software was provided with self-diagnostic capability.

Without major efforts in maintenance, the I&C reliability may have a serious impact on safety. As the equipment becomes older, the amount of maintenance required to keep an acceptable status of I&C reliability will increase remarkably.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **II**

JUSTIFICATION OF RANKING:

This issue affects plants' defence in depth and may have a direct or indirect impact on deviations from normal operation (Level 1 of defence), on bringing back the installation to normal operating conditions (Level 2 of defence) and on the capability of engineered design features to prevent the evolution of deviations into more severe accidents (Level 3 of defence). One or more main safety functions can be impaired due to the insufficient reliability of the I&C system. The issue may cause initiating events during normal operation and can aggravate the abnormal conditions.

COMMENTS AND RECOMMENDATIONS:

1. I&C upgrades should be continued.

2. The programme planned for I&C equipment upgrading should give priority to those involved in the emergency protection and safety actuation system. I&C equipment involved in these system for which high failure rates, limited capability for further repairs, spare part unavailability, obsolete technology and whose seismic and environmental qualification can not be demonstrated should be replaced with up to date (modern) technology featuring high reliability for operation in NPP safety and safety related systems, qualified and with self diagnostic characteristic.

3. A preventive maintenance programme based on equipment failure rates should be performed in order to replace obsolete parts at the end of their lifetime or to extend the design life of I&C equipment.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The plant has a comprehensive system for reporting and tracking equipment failures. Each plant department maintains an equipment failure log which identifies which instrument failed. The cause of the failure is analysed and the measures to prevent future failures are taken.

If an instrument channel is an input to the plant computer, it is monitored for being within range, for loss of power supply and for excessive oscillation. If any of these occurs, there is a visual and audible alarm in the main control room.

In addition to many process control and protection instruments the following systems/instruments have been added or replaced:

- the ex-core nuclear instrumentation system and accelerated unit unloading system
- the power supply panel for control and protection system
- the control system for two turbines
- the control system for two generators
- the control rod drive mechanism system
- the in-core nuclear instrumentation system
- the replacement of automatic control system of “Cascade-I” by “Cascade-II”
- the bearing vibration monitoring system for turbines, generators, brush-free thyristor exciters and turbine driven main feedwater pumps.
- the automatic startup and protection of diesel generators.

The following additions/replacements are planned:

- the power unloading and limiting system
- the automatic power controller
- the reactor scram system
- the automatic process control system
- an upgrade of the plant computer
- the steam generator water level control system (emergency mode)
- the monitoring equipment for reactor cooling pump vibration.

Kalinin Units 1 and 2:

Instrumentation and control equipment operated at the KNPP was developed in the beginning of the 80s. During the operation the equipment condition is assessed against the index reflecting the duration of the equipment operation before its first failure. The analysis of the I&C equipment failure frequency shows that the index demonstrating the average time of the equipment operation before its failure remains rather high. Based on the results of the I&C condition assessment, measures aimed at the reduction of the failure frequency are developed. Among these measures the following have been completed: replacement of the secondary equipment KSU with the KP-140 equipment; use of “Sapphire” sensors instead of DME sensors; and replacement of RPU relays with RP-21 relays.

The lack of I&C and sensors equipment with the diagnostic aids is compensated to a certain extent by the periodic equipment inspection performed by the operating staff (at least twice during a shift).

South Ukraine Units 1 and 2:

In addition to many process control and protection instruments the following systems/instruments have been added or replaced:

- the ex-core nuclear instrumentation system for both units
- the turbine control system for both units
- the main generator monitoring system for Unit 2
- the control rod drive mechanism power system on Unit 1
- rod position indication on Unit 1
- the in-core nuclear instrumentation system on Unit 2
- phase 1 of the installation of the Westinghouse unit information system is nearly complete on Unit 1, phase 2 will be completed in 1997
- the boron concentration meters on both units
- the plant computer for Unit 1 has been upgraded
- the equipment which trips the plant on high seismic activity on both units
- reactivity meters for both units
- a digital system of thermal process control has been introduced at Units 1 and 2.

The following additions/replacements are planned:

- the main generator monitoring for Unit 1
- the control rod drive mechanism power system on Unit 2
- the rod position indication system on Unit 2
- the in-core nuclear instrumentation system for Unit 1
- the Westinghouse information system for Unit 2
- an upgrade of the plant computer for Unit 2
- the steam generator water level control system for both units.

A methodology of maintenance and check of I&C devices, process alarm devices, protection and interlocking means is available at the plant. Besides, step-by-step programmes are developed for post-maintenance and integrated testing of safety systems, safety significant and normal operations systems.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 2 (I&C 2)

ISSUE TITLE: Safety system actuation design

ISSUE CLARIFICATION: The safety system actuation consists of 3 independent trains that correspond to the trains of the technological safety systems. Within each train, four channel sensor redundancy is used to develop the logic for initiating the safety system equipment. The logic circuits are designed with a high degree of fault tolerance to prevent failures from defeating the safety actuations. Fault detection is used in both the analogue sensor circuits and the actuation logic. Remote control of the safety equipment is provided from the main control room and the emergency control room. Appropriate priority is established between automatic and manual controls.

A safety concern here is that the ECCS actuation circuits are exclusively based on an energize-to-actuate principle. Hardware reliability issues will have a direct impact on safety in the area. In western practice both energize-to-actuate and de-energize-to-actuate principles are used depending on a case by case analysis to assure the necessary safety level

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The issue is a deviation from current practice. IAEA safety guide 50-SG-D3 [27] indicates that as a supplement to the basic requirements for selection and use of reliable equipment, it is desirable to design the safety actuation system so that most probable modes of failure should increase probability of spurious action rather than the likelihood of an unsafe fault. When the probability of safe action is high after failure this is called fail-safe design. Loss of power to the safety actuation system may be one of the cases to be consider as probable failure.

COMMENTS AND RECOMMENDATIONS:

1. In case of a major replacement of the safety actuation system an analysis of the reliability of the existing design approach should be assessed and compared against de-energize to actuate design to see if any improvements could be made.

REFERENCES: [2, 3, 9, 26, 27]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The reactor protection system is actuated on loss of power supply, and the other safety systems are designed by a principle of non-actuation in case of loss of power supply. The probability of loss of power supply is low since all the safety actuation systems are backed up by batteries.

The plant has no plan to address this safety issue.

Kalinin Units 1 and 2:

Reactor protection system is actuated on the signal of loss of power. The actuation of the safety systems is based on the principle of energizing. However, the power loss probability is not very high, as all safety systems are backed up with the batteries.

The plant finds it reasonable to maintain the current scheme decisions.

South Ukraine Units 1 and 2:

A principle of initiation by “blackout” has been realized for emergency protection systems (control rod insertion into the core and suppression of the reaction).

A scheme 2 out of 3 and 2 out of 4 is realized for initiation at technological protection systems and interlocks. This scheme conception provides reliability of the operation safety with coefficient $10^{-4} \div 10^{-6}$.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 3 (I&C 3)

ISSUE TITLE: Automatic reactor protection for power distribution and DNB

ISSUE CLARIFICATION: The ex-core neutron flux monitoring system feeds reactor scram commands to the reactor protection system. The scram is actuated at high neutron flux and low reactor period. Although separate upper and lower flux signals are measured in the power range, there is no automatic protection for power distribution. The operator is required to react to peaking factor problems.

However, operator errors, or an unforeseen combination of events, could lead to an adverse power distribution. The situation could be aggravated if the use of a low leakage core design reduces margins or if the plants are used in the load follow operation where the power distribution changes are significant and frequent. Low leakage core designs are required to protect the RPV wall (see issue CI 1).

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

For base-load operation, the reactor is not protected against an adverse power distribution. This is a departure from recognized international practices.

COMMENTS AND RECOMMENDATIONS:

1. If core design analysis and operating experience show that automatic protection for adverse power distribution and DNB during base and future load follow mode is needed, then actual reactor power control design should be revised to include automatic protection for adverse power distribution and DNB.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The ex-core nuclear instrumentation AKNP-03 has been modernized by installing a new system AKNP-7-02. A new in-core monitoring system (type SVRK-05, 06-01) was installed and put into operation.

The operating crew will use the flux differences from the nuclear instrumentation system and/or the peaking factors from the in-core instrumentation system to detect power peaking problems and respond accordingly. Since the plan was designed to operate at a base load mode, the automatic reactor protection for power distribution and DNB was not considered. The plant has been carrying out the work to establish an operator advise system which can continuously predict the xenon oscillations and provide advice for action needed. The plant has also planned to implement an automatic actuation of the power limiting system in case of DNB and adverse power distribution.

Kalinin Units 1 and 2:

The base-load operation experience has shown that the margin for departure from nucleate boiling is sufficient for safe reactor operation. If a more detailed analysis of the problem shows later the necessity for automatic reactor protection to reduce the margin for departure from nucleate boiling, the current scheme of reactor power control may be reviewed in order to introduce the indicated protection.

South Ukraine Units 1 and 2:

The nuclear instrumentation system has been modernized by installing a new system on both units.

The in-core nuclear instrumentation system was modernized by installing a new system on Unit 2.

The in-core nuclear instrumentation system was modernized by installing a new system on Unit 1 in 1996.

The operating crew will use the flux differences from the nuclear instrumentation system and/or the peaking factors from the in-core instrumentation system to detect power peaking problems and respond accordingly.

Further optimization of information delivery to operators will be realized with introduction of process computer (I&C) from “Westron”.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 4 (I&C 4)

ISSUE TITLE: Human engineering of control rooms

ISSUE CLARIFICATION: The 'small series' WWER-1000 control rooms provide the controls and indications necessary for the operator to carry out actions required during normal and shutdown operations of the plant. It follows the classical division by subsystems with an "active mimic" diagram of each subsystem being shown with controls for pumps and valves in their appropriate functional position on the diagram. This type of organization has led to operational problems, most notably the Three Mile Island accident in the USA, because the operator's attention is focused on a specific item and he tends to disregard the interactions between the subsystems. An action that is taken at one point to solve one problem may create other problems in related subsystems.

There are further deficiencies related to human engineering design if a comparison is made with the most modern international practices. Indicators of different types of process measurements, for example flow and pressure, are not distinguishable, except by the engraved legend. Control switches for pumps, valves, circuit breakers, etc., all have handles with the same shape. These switches have red and green indicating lights showing the operational status of the associated component. The brilliance of these lights varies greatly, and, in some cases, it is difficult to determine which lamp is lit. There is an increased likelihood that operators will make an error in assessing the equipment status. Indicators that provide data that is important to the operator's evaluation of the safety state of the plant are not differentiated from those used for normal operations. Some of the most valuable space on the control panel, that which is directly in front of the operator, is used for infrequent activities related to plant startup and surveillance testing.

In summary, the design of the information display in the control rooms does not give the operator a rapid overview of information regarding the current state of plant and reactor safety as a whole. It also has deficiencies that increase the incidence of human error.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **II**

JUSTIFICATION OF RANKING:

The issue represents a deviation from OPB-88 [4], IAEA 50-SG-D3 [27] and IAEA 50-SG-D8 [28]. The deficiencies in the design of the main control room and the emergency control room affect Levels 1 to 4 of plants' defence in depth. Due to human errors, one or more main safety functions can be impaired. These situations are possible under normal and anticipated operating conditions as well as under DBA and BDBA conditions.

COMMENTS AND RECOMMENDATIONS:

1. A structured, in-depth design review of the control room, similar to those done for western plants, should be conducted. The objective of this review would be to identify the specific deficiencies of the control room design to establish a modification plan to improve the safety situation. The design review should be conducted as soon as possible. For best results, it should be done in connection with the considerations of the emergency operating procedures.

2. The recommendations of the control room design review should be prioritized according to their importance to safety, and interim modifications should be made to the control room as soon as practical. More extensive modifications to the control room could be included in the I&C reconstruction programme.

3. A Safety Parameter Display System should be added to provide the operator with the information needed to assess the critical safety functions of the plant. This safety parameter display system will be a necessary step in establishing symptom based emergency operating procedures. A systematic review of the accident monitoring should be conducted to identify the parameters necessary to monitor an accident and to verify that the instrument ranges are adequate to cover all plant states. This review should also confirm that the sensor channels for the accident monitoring instrumentation are sufficiently independent from the environmental conditions that result from the accidents. This review should be done in conjunction with the proposed control room design review.

REFERENCES: [2, 3, 4, 27, 28]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The plant has no plan on this issue.

Kalinin Units 1 and 2:

The following measures for improving human engineering of control rooms have been implemented at Kalinin Units 1 and 2:

1. Control room ceiling has been changed in order to perform lighting improvement
2. Control panels have been repainted, which lead to decreasing “patch of light” effect
3. Number of safety systems on the panels have been repainted in different colours
4. Mimic diagram was remarked
5. Currently, the indication display is being replaced with light diodes being installed instead of incandescent lamps.

The following measures are being implemented:

1. Safety parameter display system (SPDS)

The SPDS will provide reactor operator with sufficient information to control safety functions during accident. At present, system, architecture and functions are identified; software, including sizes, algorithms, database have been developed; prototype has been developed; and tests for Unit 2 have been carried out. The plant plans to supply equipment within TACIS programme and to commission full scope SPDS for Unit 2.

2. Symptom based EOPs are being developed in co-operation with EdF. At present, procedures for Unit 1 have been written and prepared for validation.

South Ukraine Units 1 and 2:

The plant plans to implement a safety parameter display system.

Ergonomic requirements have been developed by Main Designer of Ukrainian process computer (I&C) for design of Main Control Rooms. According to the plans for the units’ equipment upgrading, the modifications of the MCR switchboards, panels and mimic diagrams are being gradually implemented. Replacement of gate valves, pump sets, safety valves is being performed in order to facilitate the personnel orientation. Warning tags at the measuring instruments have been replaced by new ones, etc. The process is going on in frame of introduction of “Westron” instrumentation.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 5 (I&C 5)

ISSUE TITLE: Reactor protection system redundancy

ISSUE CLARIFICATION: Emergency protection for reactors of WWER-1000 'small series' units include two independent sets of neutron flux monitoring. Each set consists of three independent measuring channels which includes signal conditioners, comparators and trip signal generators. Trip signals from the three channels vote 2 out of 3 to produce a reactor trip from each set. One of the two protection sets in trip condition will open reactor trip breakers and trip the reactor.

The technological part of reactor protection (e.g. trip signals from pressure, temperature, level, etc.) consists only of one protection set. This set consists of three independent channels for each parameter in separate cabinets, though in one room. The three channels voted two out of three combined with a logic one out of two in such a way that single failure of any element in the set will not affect the generation of a scram output signal. This type of design is different from WWER-1000/320 units and even from WWER-440/213 units. The Russian regulatory document "Nuclear Safety Rules of Nuclear Power Plant Reactor Facilities (PBYa RU AS-89, section 2.3.2.9 requires at least two independent sets for all variables which are required to prevent postulated initiating events. In addition, this type of design makes the test of the protection set during operation impossible.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power

RANKING OF ISSUE: **III**

JUSTIFICATION OF RANKING:

The lack of redundancy and independence in the protection set which contains the initiation and actuation devices for technological parameters may lead to the emergency shutdown of the reactor being questionable. Moreover, testing of a single set is not possible during reactor operation. This issue is a violation of national standards and international practice.

COMMENTS AND RECOMMENDATIONS:

1. The architecture of the emergency protection system of the 'small series' WWER-1000 units, including redundancy and independence of initiating channels, voting logic and actuating features for the technological parameters, should be analysed in depth to find whether faults on this single set may lead to reactor emergency shutdown being questionable for PIEs which are prevented by technological parameters.

2. The specific plant information for the 'small series' WWER-1000 units for reactor protection should be the basis for the analysis indicated above.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

No action has been planned.

Kalinin Units 1 and 2:

The lack of redundancy in the set of emergency protection, which contains the initiation and actuation devices for technological parameters does not lead to the violation of the single failure criterion. However, it does not allow to perform a complete testing of the set during reactor power operation.

No measures are planned.

South Ukraine Units 1 and 2:

A 2-set system of reactor protection has already been implemented at Unit 1. This work was performed in co-operation with “SKODA”. The equipment for Unit 2 has been manufactured and is planned to be implemented during the outage in 1998.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 6 (I&C 6)

ISSUE TITLE: Condition monitoring for the mechanical equipment

ISSUE CLARIFICATION: Diagnostic systems are needed to provide the operators with an early warning of mechanical equipment degradation, in order to avoid termination of safe operation as a consequence of a sudden failure. Also, condition monitoring systems could be used to confirm good current status, thus avoiding unnecessary opening of the equipment. This monitoring should be carried out with respect to vibration, displacement, position and condition for the mechanical equipment important to safety.

The original design does not provide for an adequate diagnostic system to monitor the equipment of 'small series' WWER-1000 units.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The condition monitoring system is not adequate and accurate enough to fulfil its function. This is a departure from recognized international practice.

COMMENTS AND RECOMMENDATIONS:

Condition monitoring systems for mechanical equipment should include an analysis of sensor location and sensitivities, required data acquisition performance requirements (discrimination of raw data) and automatic and manual startup of data acquisition. The acquisition system should be automatically started in the event of an alert to provide an easy interpretation of results. In addition real time and on demand data display and storage should be provided.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Condition monitoring for the mechanical components of primary and secondary circuits of Unit 5 is realized by means of different types of sensors: temperature, vibration, shaft, displacement, etc. The output signals of sensors correspondingly go to the individual instruments, computer complex URAN-V, and special I&C systems such as:

- the bearing vibration monitoring of turbines, generators, turbine drive feedwater pumps and brush-free thyristor exciter (VVK-331)
- the vibration monitoring of primary components (KSA-10)
- the temperature monitoring of main generators and thyristor exciters (A-701-03, ASKR).

The existing monitoring systems, VVK-331, A-701-03 and relevant I&C parts will be dismantled and replaced by a new automatic process control system ASUT-500M which enables monitoring the bearing vibration of turbines, generators, turbine driven feedwater pumps and thyristor exciter, the temperature of main generators and thyristor exciters, and shaft displacement of turbogenerators.

In the modification of URAN-V, additional measurement channels will be installed to monitor the condition of mechanical components.

Kalinin Units 1 and 2:

The pilot diagnostic systems are operated at the Unit 1. Two types of systems, reactor diagnostic system "KARD" and vibration diagnostics system "ARGUS", are used to monitor the conditions of mechanical equipment. The "KARD" system monitors the vibration of the reactor core components. This system is currently not operable and needs to be repaired. The "ARGUS" system periodically performs the following functions:

- Monitor the vibration parameters of turbine generators, main circulation pumps, and turbine feed water pumps during operation; for monitoring of turbine generators and main circulation pumps, the "ARGUS" system uses the signal sources from the original vibration monitoring system which has limited capabilities.

- Monitor the vibration parameters of the aforementioned equipment in the transient modes;

- Perform the automated diagnostics function for the turbine generators with involvement of the expertise system.

It is planned to improve the condition monitoring for mechanical equipment by modifying the turbine generator vibration control system ("KOMPASS" of Unit 2). The "KOMPASS" system is used to monitor the equipment mechanical conditions and the vibration parameters of the turbine generators and main circulation pumps; for main circulation pumps, the "KOMPASS" system uses the signal source from the original vibration control system which has limited capabilities. In addition, the NPP vibration monitoring system is implemented on the basis of "KOMPASS" system in order to perform periodical vibration tests for all the rest revolving mechanisms including safety related ones. The "KOMPASS" system is planned to be upgraded within the scope of TACIS project. The contract has already been signed.

South Ukraine Units 1 and 2:

The main generator monitoring system has been replaced on Unit 2 and is planned to be replaced on Unit 1. The existing main turbine monitoring system monitors vibration, bearing temperatures and shaft displacement. There are plans to replace the systems with more modern systems on both units. Reactor coolant pump bearing temperature and vibration are monitored. Main boiler feed pump vibration, bearing temperature and shaft displacement are monitored.

The plans for the Technical Support Centre include extensive equipment monitoring capability to predict possible failures of equipment.

The rotor vibration control programme is being realized at turbine - generator sets 1 and 2 (implemented at Unit 3 already).

Automatic chemistry monitoring is planned to be improved within TACIS Programme.

The further development of this strategy is determined by the national programme of Ukraine.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 7 (I&C 7)

ISSUE TITLE: Primary circuit diagnostic systems

ISSUE CLARIFICATION: Diagnostic systems are needed to provide the operators with an early warning of a situation where the integrity of the reactor coolant pressure boundary is threatened.

The original design of 'small series' WWER-1000 units does not provide for adequate diagnostic systems to monitor the reactor coolant pressure boundary integrity. For example, the existing means to detect a primary-to- secondary leakage in steam generators is not sufficient to monitor any violation of the design limits of safe operation.

In the reactor vessel head, the CRDMs, instrumentation, etc. are attached to the head penetrations through bolted joints (flanges). Each joint is sealed by two parallel sealing rings (Ni) and the leak detection is based on the collection of the leakage water between these two sealing rings. The leak detection system is not tested or inspected periodically. The humidity monitoring system in the upper reactor block is not sensitive enough to detect leaks in the bolted joints.

Another example is the lack of monitoring and assessing the unspecified loads. In penetrations, nozzles and in certain piping, high thermal loads relevant for fatigue analysis have been expected and in many cases treated with specific design. It was not possible to specify loading due to e.g. stratification at the design stage. Penetrations and nozzles are usually high stress concentration areas with specific design features (e.g. thermal sleeves, wall thickness reduction, dissimilar welds) and residual stresses. NDT and thorough integrity assessment is required but difficult to achieve in many occasions. It is common practice, according to reference codes and standards, to implement a monitoring system to ensure the required integrity is maintained for the concerned components.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. Inadequate monitoring of the reactor coolant boundary integrity affects Level 2 of plants' defence in depth to control abnormal operation and to detect failures. The main safety function can be impaired because of a lack of an early warning when the integrity of the reactor coolant pressure boundary is threatened. This scenario is possible during normal operational conditions with degradations of components.

COMMENTS AND RECOMMENDATIONS:

1. Plans to develop and introduce diagnostic systems for leak detection, vibration monitoring and work monitoring are supported and should be implemented as soon as possible.

2. Outer surface temperature monitoring at critical sections and recording of the results for fatigue, as proposed by Russian designers is endorsed. Transfer functions for the inside surface should also be determined.

   It is not clear if all the potential locations are covered by the proposed and/or implemented measures. The measures should be reviewed for completeness and adequacy.

3. The leaktightness of the nozzles of CRDs of the instrumentation should be upgraded by replacing sealing rings with new ones (Dukovany is using expanded graphite rings).

4. For the leak detection system under consideration by the plants the guidance of the international standards IEC 1250 for the design of this system and the instrumentation should be used. The

guidance of this standard is in accordance with requirements IAEA 50-SG-D13 “Reactor coolant and associated systems in NPP”.

5. For loose part monitoring in primary circuit under consideration by the plants, the guidance of the system design provided by international standards e.g. USNRC RG 1-133 should be used.

6. The reliability of the vessel head monitoring system should be improved, e.g. by using new humidity detectors.

REFERENCES: [2, 3, 4, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The following are some of the primary diagnostic systems under consideration which will replace the existing systems or to be added as new systems:

(1) A vibration diagnostic system
(2) A loose parts monitoring and identification
(3) A primary leak detection and identification system
(4) An MCP status control system
(5) An acoustics diagnostic system.

Kalinin Units 1 and 2:

At KNPP, elastic seals of containment connections, type TK-10, and TK-13, is tested by the leak-detector TI-1-14.

The laboratory LTD carries out diagnostic of equipment integrity:

At Unit 1, the KARD, ARGUS and ALMOS systems are used. The KARD system monitors the vibration of the core and reactor internals and the coolant flow inside core; analyses primary circuit hydrodynamic processes according to the signals of pressure pulse detectors; and find the statistic interconnection of the observed process. The ARGUS system carries out MCP diagnostic on the basis of vibration characteristic analysis.

At Unit 2, the KOMPASS system, which has a vibration diagnostic expert subsystem, monitors the MCPs. The system is to be expanded in the framework of TACIS program, and will include the rotor vibration monitoring. It is also planned to install a primary circuit diagnostic system at Unit 2 in the framework of TACIS projects, as follows:

- The KOMPASS vibrodiagnostic system as mentioned above
- A leak detection system for primary equipment (including reactor cover)
- A detection system for out of core components
- A diagnostic system by neutron noise and vibration of heat mechanic equipment (specifications are being developed).

South Ukraine Units 1 and 2:

Various organizations are developing plans for a number of systems to diagnose primary system status. The following are some of the primary diagnostic systems under consideration:

(1) A vibration diagnostic system
(2) A loose parts monitoring and identification
(3) A primary leak detection and identification system
(4) An MCP status control system
(5) An acoustic diagnostics system.

The plans for the Technical Support Centre include a very sophisticated method of monitoring component status. These include a system which provides the current status of the reactor, the safety systems and other major plant equipment in an easily readable form. A system which combines current component parameters and creates a trend which can be used to predict when a design limit will be exceeded is envisioned.

As the above systems (and others) become more well defined, they will be evaluated for implementation at the SNPP.

A programme for monitoring of nozzle thermal loads has begun with some of the equipment already in service.

The main coolant pump vibration system is already in service.

The further improvement of the primary circuit monitoring will be performed within all-Ukraine TACIS programme.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 8 (I&C 8)

ISSUE TITLE: Accident monitoring instrumentation

ISSUE CLARIFICATION: During and following an accident, appropriate parameters and system functions are monitored in order to enable the operator to cope with the event sequence. The operator must have sufficient information available to: (1) determine the course of an accident (2) make decisions concerning appropriate manual actions; and (3) assist in determining what actions, if any, are needed to execute the plant emergency plan. To supplement these actions and improve the plant operation under emergency conditions it could be necessary to assist the operators with display systems making the information easy to understand and providing aids in procedures utilization.

The TMI-2 accident reinforced the need to supply the NPP operators with pressure, temperature, radiation and humidity measurements that have a measuring scale beyond the normal operating range. In case of a design basis accident or a beyond design basis accident these measurements have to provide reliable information of the conditions inside the reactor pressure vessel and the containment.

The accident monitoring instrumentation in 'small series' WWER-1000 units is not adequate. For example, the reactor pressure vessel level indication is currently not provided, and the level can be estimated only by indirect means. A further example is that the effluent from the ventilation duct is not properly monitored in terms of radioactivity. The information obtained on the effluent from ventilation ducts during accidents is not reliable and accurate and this can lead to an over-irradiation of plant personnel and inhabitants in the vicinity.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. A lack of precise and reliable information provided to the operator will increase the failure rates of operators, especially in the highly stressed circumstances. The means for accident management are not adequate and this affects Levels 3 and 4 of plants' defence in depth. One or more main safety functions can be impaired or even questioned, because of insufficient monitoring for scenarios within the DB envelope and beyond.

COMMENTS AND RECOMMENDATIONS:

1. Guidelines for the accident instrumentation should be developed. The guidelines should include definition of plant variables whose indication is required by plant operators in the control rooms to take pre-planned manual action to accomplish plant safety status. Guidance for accident monitoring instrumentation is contained generally in IAEA 50-SG-D8 [28]. Detailed guidance on design requirements (qualification, ranges, reliability etc.) can be found in international standards e.g. IEC 911, USNRC Guide 1.97 Rev.3, KTA 3502. Accident monitoring instrumentation should be supported by accident management procedures.

2. In designing new instrumentation for RPV level measurement specifically, experience from similar Western NPPs should be taken into account.

3. An effluent monitoring system which meets the OPB-88 [4] requirements should be implemented.

REFERENCES: [2, 3, 4, 9, 26, 28]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The design of the NNPP Unit 5 did not consider severe accidents. Although there are sufficient amount of monitoring instruments for the containment with range margins, the analysis of proper functioning of these instruments under severe accidents has not yet been done.

A design of "black box" to be used for accident conditions is being considered to be introduced in the future to Unit 5.

The development of a number of sensors working under beyond design basis accident conditions is in progress.

Kalinin Units 1 and 2:

The analysis of the deviations of accident monitoring instrumentation from the current regulations has been performed at KNPP. A long-term programme and a set of compensatory measures have been worked out. The long-term programme includes, for example:

- the SPDS
- steam monitoring in the reactor upper plenum
- gamma radiation level measurement in case of the accident at high humidity and high temperature conditions
- determination of measuring ranges and environmental conditions for accident monitoring instrumentation from PSA results.

South Ukraine Units 1 and 2:

The plant does not have this instrumentation but plans to install it. Improved stack vent monitoring instrumentation has been installed.

The Technical Task for the Crisis Centre is developed with regard to criteria of that type.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 9 (I&C 9)

ISSUE TITLE: Technical support centre

ISSUE CLARIFICATION: Recent international practice has been to design NPPs with a room where current plant data and status are compiled for display to enable technical experts to support the operators during the management of an accident. This room is separate from the control room. The WWER-1000 plants do not have technical support centres.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: **II**

JUSTIFICATION OF RANKING:

There is no technical support centre for WWER-1000 NPPs, which can affect the control of abnormal operation and management of accidents. This affects Levels 3 and 4 of plants' defence in depth. The main safety functions can be impaired or even questioned for scenarios within the DB envelope and beyond.

COMMENTS AND RECOMMENDATIONS:

This measure should be implemented in parallel with the upgrading of the control room designs (see issue I&C 4).

REFERENCES:

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Unit 5 has an emergency centre (or crisis centre) but does not have a technical support centre to provide technical support to the operators.

The emergency centre is equipped with computer which can predict the accident progress and the radiation level at different locations inside and outside the plant, a filtered ventilation system and telephone communication.

Kalinin Units 1 and 2:

The local crisis centre (LCC) is located underground about 100 meters away from the MCR. The LCC has two functions:

1. Technical support for the MCR

It provides on-line display of operating parameters of Units 1 and 2. A safety parameter display system is under development. It is expected that this function will be fully activated in 1999 to be used by the plant technical support group.

2. Emergency planning

It provides radiological data display in the course of accident, a communication system with Rosenergoatom centre and local town centre, and software system which predicts the radiological consequence of the accident.

Within the framework of the International Nuclear Safety Programme (INSP), the co-operation between the specialists of the KNPP, joint-stock society "Consist" and PNNL USA is organized in accordance with the contract N 307807-A-RO (task 2) "Local Crisis Centre for KNPP". At present the implementation of this contract is in the final stage. For a completed emergency preparedness system, KNPP plans to develop the co-operation of INSP in the following directions:

(a) Methodological provision of LCC

In particular, the procedures of the LCC actions in emergency situations, similar to the current procedures at American NPPs (Severe Accident Management Guidelines - SAMG), are to be developed.

(b) Engineering implementation of LCC in accordance with "the Technical assignment at KNPP LCC".

South Ukraine Units 1 and 2:

An on-site technical support centre exists inside the security fence. It is located in an underground cellar which is equipped with a filtered ventilation system and telephone communication. A project is in progress which may ultimately supply this centre with access to a site wide computerized data collection and distribution system. This system will provide the capability to monitor design limits, safety parameters, safety critical functions and radiation levels within the plant. It is planned that this system will also have the capability to process information so that it can provide trending of parameters and thus provide the capability to predict future developments in the event scenario. It is also planned that this information will be supplied to off-site technical support centres both locally and at Kiev and ultimately linked to an international data system.

It is planned to add the following sensors to the plant for the purpose of supplying data to the technical support centre:

(1) 402 new process sensors (pressure, temperature, flow, etc.) per unit
(2) 66 new radiation monitoring sensors
(3) 9 new meteorological sensors.

These new sensors will be used to supplement the data from existing plant sensors.

At the present time, an availability is provided at SNPP to extract information from Unit 3 and supply it to the NPP Information Computer System. It is planned to further develop that system for Units 1, 2 and, as the result, to supply this information to the Crisis Centre in Kiev.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 10 (I&C 10)

ISSUE TITLE: Water chemistry control and monitoring equipment (primary and secondary)

ISSUE CLARIFICATION: An accurate and preferably on-line chemical monitoring system is important to enable the operator to respond in time to deviations in the primary and secondary coolant water-chemical condition indices. The specified water chemical conditions must be continuously maintained to avoid corrosion problems in the main equipment.

The chemical monitoring system presently used is more than 10 years old and a great deal of maintenance effort is required to ensure reliable and accurate results. It is also increasingly difficult to obtain the necessary spare parts.

The issue has been identified from operating experience.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The chemical monitoring system is essential to keep coolant parameters within prescribed limits. If these limits are exceeded, the integrity of physical barriers can be endangered. This is a departure from international practice.

COMMENTS AND RECOMMENDATIONS:

1. The water chemistry control and monitoring equipment should be up graded.

REFERENCES: [2, 3, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Water chemistry control and monitoring of the primary coolant is carried out by means of laboratory instrument analysis, and of the secondary coolant by automatic devices and recorders. The on-line boron meters are available in the primary circuit.

At present, contract has been signed on the automatic water chemistry control and monitoring of the primary coolant and the computerised water chemistry monitoring of primary and secondary water.

Kalinin Units 1 and 2:

The chemical control of the primary circuit coolant is performed periodically by means of the laboratory analysis and constantly by boron meters.

Secondary circuit water chemistry control is partially carried out by the automatic equipment and self-recording devices. The results of measurements are shown on the computer of the chemical shop shift supervisor. At present a new device for sample preparation is being introduced, which will improve the quality of the measurements.

South Ukraine Units 1 and 2:

Both units have condensate polishing systems. During normal operations Unit 1 has approximately 20% of its condensate flow going through the filter and the condensate polishers. The figure for Unit 2 is 50%. The condensate polishing systems are currently operated only manually. Control systems are being installed to enable the systems to be operated in automatic. In addition to the local indications of pH, sodium content and oxygen content, the chemistry laboratory has terminals which are connected to the plant computer. This provides access to dozens of measurements of pH and conductivity throughout the secondary side.

Contracts have been signed with western suppliers to provide the following new equipment:

(1) A new system to detect sodium in the condenser
(2) A new system to detect oxygen in the condenser
(3) A new system to detect oxygen in the main feedwater system
(4) A new system to detect sodium in the steam generator blowdown
(5) A new system to detect organic matter and chlorine in demineralized water.

A similar level of modernization is planned for the primary circuit.

Additionally, it is planned to transfer within this TACIS Programme from ammonium water treatment to morpholine.

REVIEW AREA/ISSUE NUMBER: Instrumentation and Control 11 (I&C 11)

ISSUE TITLE: Separation of the primary circuit instrumentation taps to I&C detectors

ISSUE CLARIFICATION: Some redundant instrumentation use common tapping points for connection to process. The parameters measured by this instrumentation may be used in important control safety systems and in the protection system. For example, the pulse lines of the safety actuation system and the reactor protection system are not adequately separated. For some parameters, as the pressure above the reactor core, two of the three impulse lines which feed the three channels for activating the safety systems are also used to feed the two channels of the reactor protection system. Failure of the common tap will cause failure of all instruments connected to it and may result in actuation or non-actuation of one channel of the protection system. It is a deviation from the Russian Safety rule "General safety regulations for NPPs (OPB-88)".

A proposal was made to increase the number of taps and to arrange the actuation logic. The new arrangement of WWER-1000 pulse lines operates on a train-by-train basis for control safety system and emergency protection, for which four additional pipe sleeves are required to be made on the reactor vessel I&C nozzles and to use four available pipe sleeves (one for each hot leg loop) to measure SG pressure differential. With this arrangement, the failure of one pulse line will result only in the activation failure of one train out of three for the safety systems and one train out of two for the emergency protection systems (without safety system actuation and EP actuation). A practice of welding in additional pipe sleeves on reactor vessel I&C nozzles has been developed.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The damage of the not physically separate redundant instrument taps caused by an external or internal hazard may, in some cases, disable the functioning of part of the reactor protection system, i.e. the safety functions may be impaired for scenarios within the design basis (DB) envelope.

COMMENTS AND RECOMMENDATIONS:

1. In accordance with plant representatives and also from the documentation presented in the meeting it appears that common tapping points for redundant instrumentation important to safety systems are not used.

2. This issue should be investigated in depth at the plants.

REFERENCES: [2, 3, 46]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5*:*

No action has been taken.

Kalinin Units 1 and 2:

The KNPP design provided six independent pulse lines from the reactor vessel I&C nozzle for pressure measurement. Three independent lines (one line for each channel) are used for reactor protection signals. The other three pulse lines are used for the ECCS signals, each of which connected to three sensors for one train of the safety system. The configuration does not violate the Russian regulatory requirements.

The KNPP design on primary circuit instrumentation taps is different from the WWER-1000/320 model.

South Ukraine Units 1 and 2*:*

The channels will be separated where safety is affected.

3.6. ELECTRICAL POWER SUPPLY

REVIEW AREA/ISSUE NUMBER: Electrical Power Supply 1 (El 1)

ISSUE TITLE: Diesel generator reliability

ISSUE CLARIFICATION: Each of the emergency power buses are supplied from a 6 kV station service busses via two breakers. In the case of loss of off-site power, on-site power supply is provided by three diesel powered emergency supply trains, electrically and physically separated but located in the same building, and started automatically at $U \leq 0.25\ U_N$, 2 s delay.

Each diesel generator unit has an autonomous cooling, lubrication and compressed air system and is backed up with a fuel storage for 72 hours (SNPP) and 200 hours (NNPP) operation. The compressed air system consists of two compressed air bottles for four accelerated starts of the diesel engine and is automatically restored by two compressors.

Functional tests of the diesel generators are performed monthly, twice with run-up to synchronisation speed without loading, and once time with initiation of the loading sequence of the related emergency power consumers. In addition, a parallel operation of the diesel generator with the grid at nominal power is performed yearly.

However, the diesel generator system has some deficiencies. The DG building is neither designed for shock wave impact nor for an aircraft fall load. The DGs are in a single design. The heat and ventilation system for DGs does not meet the Russian fire protection standard VCN 01-87, and some I&C lines do not comply with the Russian standard OTT-87.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The lack of an analysis of station blackout represents a deviation from international practice.

COMMENTS AND RECOMMENDATIONS:

1. Based on the PSA results, which should also address the frequency of loss of off-site power, the probability of a station blackout, exceeding a duration corresponding to the start of core damage, should be evaluated on each site. If the result is higher than expected by the designer, there will be a safety concern and measures will have to be taken to resolve the issue.

2. The necessity of an additional reserve emergency power source is site dependent. Therefore, each plant should evaluate the real impact on safety of an additional emergency power source and take the corresponding appropriate decisions. If an additional source is needed, then it should be protected against the remaining potential common cause failures of the already existing sources, i.e. location in a separate building with sufficient distance to the existing one and functional separation with respect to the support systems and equipment diversity.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP Unit 5 will consider the measures proposed by the SNPP to increase the diesel generator reliability.

Kalinin Units 1 and 2:

The feasibility study of reconstruction of diesel generator station was completed. The main points are:

- Technical measures to exclude the impact of wave effect on the existing diesel generator buildings

- Implementation of the current requirements for emergency diesel generator fire protection (e.g. use of fire-resistant cables, collection and pumping back of oil leaks, etc.) and of the OTT-87 requirements for diesel generator pipelines and valves.

South Ukraine Units 1 and 2:

Test experience has shown a sufficient reliability of the existing emergency diesel generators. Modifications of the diesel generator lubrication system will be performed, replacing the electrical heaters of the oil system with a hot-water heating system, in order to minimise the potential risk of fire hazards inside in diesel generator building. The diesel generators of Units 1 and 2 are located in a common building.

Modification of air supply and fire alarm systems has been performed in the planned outage 1997. Additional analysis is planned to be carried out as part of the Safety Analysis Report.

REVIEW AREA/ISSUE NUMBER: Electrical Power Supply 2 (El 2)

ISSUE TITLE: Protection signals for emergency diesel generators

ISSUE CLARIFICATION: Protection signals are used to trip the diesel generator when the respective parameter setpoints are reached in order to avoid heavy damage to the diesel generator. As each single signal can stop the diesel generator, a conflicting situation occurs. On the one hand, the diesel generator is intended to provide energy for the safety system in order to prevent severe accidents. On the other hand emergency power supply may be interrupted by its own protective circuits. The benefit of the adopted solution, however, is characterized by the fact that heavy damage to the diesel engine can be avoided and continuation of diesel operation after a prompt and short repair will be possible.

At present, all trip signals for diesel generator protection are established in a single chain configuration. Besides the electrical winding protections, these are the technical protections of oil pressure and rotation speed. Rotation speed control is performed by diverse — mechanical and electronic — devices. Each failure either within the sensor or the related circuit may lead to a trip of the diesel generator. Single failures of sensors which are exposed to strong mechanical vibration and other influences of the diesel motor are not uncommon. The remaining technological protective circuits provide corresponding alarm signals only, i.e. cannot lead to emergency trip of the diesel.

In current western approaches the technological trip signals are generated in a two out of three configuration leading to a significantly higher reliability of the emergency diesel generator system, because a single failure of one sensor does not cause a trip signal.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This weakness has been identified from operational experience.

The design of the protection signals required to trip the diesel generator does not include sufficient provisions against spurious trip of the diesel. This is a departure from recognized international practice.

COMMENTS AND RECOMMENDATIONS:

It is recommended that the influence of the existing protection circuits on the reliability of the diesel generator be evaluated within the planned PSA studies. Based on the results, decision should be taken on the modification of the existing logic scheme for the technological diesel generator protection.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP has made a technical proposal on changes of the diesel protections to alarm signals. It was noted that this proposal will avoid excessive actuation of the diesel protection.

Kalinin Units 1 and 2:

Based on the technical and economical calculation of Units 1 and 2 reconstruction programme, it is planned to remove some technological protections which generate diesel generator trip. The technical specifications require that certain protection signals for tripping diesel generator be removed. At present, “oil temperature above 88ºC” and “single phase circuit operation” have been switched off. Two out of three configurations for generating trip signals are not used for the time being. Further exclusion of other trip signals are under consideration.

South Ukraine Units 1 and 2:

The safety issue is not addressed in the Long Term Safety Improvement Programme.

The probability of initiation of a spurious signal in the DG trip control schemes will be analysed in the safety analysis report.

REVIEW AREA/ISSUE NUMBER: Electrical Power Supply 3 (El 3)

ISSUE TITLE: On-site power supply for incident and accident management

ISSUE CLARIFICATION: Power supply by diesel generators is provided to safety systems that are necessary to cope with design basis accidents. However, the scope of systems with diesel backed power supply is very limited in comparison with the common international practice, and does not cover many systems that would reduce the severe accident risk by facilitating management of anticipated incidents.

Examples of safety relevant systems without diesel backed power supply, but may be different from plant to plant, are the following:

- auxiliary feedwater system
- cooling system for control rod drives
- radiation control panel
- telephones for communication between control room and the plant
- pumps for filling diesel generator fuel tanks (tanks have fuel for 8 hours of operation) and
- DC distribution system in turbine hall.

All of the above systems would be needed for proper management of incidents that entail complete loss of off-site power supply and necessitate plant cooldown to cold shutdown state. Specifically, the makeup system of the primary circuit would be needed for depressurization and for main coolant pump seal injection (even though the seals are less vulnerable than in western PWR types and withstand without failure a loss of seal injection for at least several hours). Availability of normal makeup would also speed up boration of the primary circuit.

The operation of auxiliary feedwater system would eliminate thermal shocks to the SGs by preventing unnecessary startup of the emergency feedwater system.

Power supply for the above mentioned systems cannot be taken from the existing diesel generators because their capacity is exhausted by the existing consumers.

This weakness has been identified from operating experience with WWER-1000/320 plants, notably the accident at Kozloduy Units 5 and 6 in September 1992 (fires and short circuits in electrical systems).

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Insufficient diesel backed power supply for the management of emergencies affects Levels 3 and 4 of plant's defence in depth to control severe plant conditions. The main safety functions can be questioned in scenarios beyond DBA.

COMMENTS AND RECOMMENDATIONS:

1. An additional diesel per unit should be considered to extend the scope of systems with diesel backed power supply.

2. A generic study should be made to decide which systems need backup power after loss of off-site power supplies.

REFERENCES: [3, 23, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

In case of loss of off-site power supply, the consumers of three independent channels of safely systems and safety important consumers are supplied with electric power from three diesel generators.

In case of failure of all three diesel generators, limited consumers of one channel of safety system can be supplied with electrical power from the NNPP Units 3 & 4 (WWER-440/230) normal power source which is independent of NNPP Unit 5.

An additional diesel generator is not being considered to be installed, similar to that which was implemented at Zaporozhe NPP and shared by two units. However, the NNPP plans to purchase a mobile diesel generator set with necessary cable connections in the framework of the TACIS programme. Also included is a mobile feedwater pump supplying water to the steam generators.

Uninterruptible power supply units of ABP-1500 type of the first generation were replaced with the ABP-1500 units of the second generation in three safety systems, due to their service life expiration.

Kalinin Units 1 and 2:

There is no additional diesel generator at KNPP. However, the Terms of Reference has been developed for its design development intended to perform the following functions:

- Supply power to the consumers not participating in emergency shutdown operations (e.g. turbine oil lubricating pumps, turbine shaft turn-over mechanism, etc.)

- Supply power to the unit DC control console in case of loss of power of 6kV bus bar.

South Ukraine Units 1 and 2:

At present, three emergency diesel generators are installed per Unit 1 and 2. As stated by the plant experts, pre-investigations on the installation of two additional diesel generators for reliable station service of both units have already been started. However, corresponding detail feasibility studies will be performed after the re-evaluation of the current station in the framework of a probabilistic safety analysis.

REVIEW AREA/ISSUE NUMBER: Electrical Power Supply 4 (El 4)

ISSUE TITLE: Emergency battery discharge time

ISSUE CLARIFICATION: Batteries are the ultimate energy source in the power plant and a high reliability and adequate capacity of this device is therefore a prime goal.

The three redundant batteries for the uninterruptable power supply of safety consumers (1st category) have to provide power supply to significant 220 V DC consumers and the corresponding DC/AC converters during the startup of the diesel generators (15 s). However, in case of any start delay or a failure of the emergency diesel generators, it is necessary to provide extended battery discharge capacity in order to increase the time margins for necessary corrective measures.

The international trend goes towards an extension of the battery discharge time in order to better cope with accident management and station blackout requirements. In case of a station blackout event, the battery is the ultimate energy source of the unit. A higher battery capacity maintains vital I&C systems in operation and illuminates the main control room. This would enable monitoring of essential plant parameters and safety significant motor operated valves would remain manoeuvrable. Therefore, the reactor can be controlled and can be kept in a safe condition in performing accident management actions (e.g. feed and bleed). The extended battery discharge time leads to larger time margins for operators to decide on further actions.

The batteries for normal operational and emergency electrical I&C systems for 'small series' WWER-1000 units are designed for a discharge time of not less than 30 min. The real discharge time of the emergency batteries can be assumed to be in a range of 1 h but corresponding discharge time tests have not been performed to date.

A further concern is the lack of battery circuit monitor. Therefore, the possible galvanic interruptions within the battery circuitry will not be automatically recognized, as long as the chargers are in operation.

In addition, the batteries are inadequately isolated from the concrete floor and cannot withstand seismic loads. An earthquake could lead to a loss of the batteries and consequently to a loss of the non-interruptible power supply.

These weaknesses have been identified during safety reviews.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

Insufficient supply by batteries in emergency situations seriously affects Level 4 of plants' defence in depth to control severe plant conditions. The safety functions may be lost in beyond DBA scenarios.

COMMENTS AND RECOMMENDATIONS:

1. Increase the emergency battery discharge time to at least one hour. It is also suggested to increase the capacity of those batteries connected to the plant process computers.

2. The battery monitoring strategy should be improved by installing automatic battery circuit monitoring equipment.

3. The batteries should be seismically protected.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

It was noted by the plant experts that the batteries are not seismically resistant. In case of failure of all three diesel generators, category 1 consumers can be supplied with electric power from the NNPP Units 3 and 4 (see issue El 4).

Within the next three years, it is planned to install seismic - resistant batteries VARTA type with 15% increased capacity.

Kalinin Units 1 and 2:

To provide reliable power supply for the safety system components, KNPP is equipped with DC 220 V batteries of SN-648 type (manufactured in Yugoslavia). The discharge time of these batteries is 30 minutes. These batteries, due to their lifetime expiration, are planned to be replaced with the new ones of VARTA AB type with a discharge time of one hour.

South Ukraine Units 1 and 2:

A programme has been developed for the replacement of the emergency batteries with VARTA type ones at Unit 1. The new batteries will provide sufficient capacities of not less than 30 min. to meet the first recommendation. Consideration should be given to provide the new batteries with automatic control equipment to monitor short circuits, cell cut-off and the charge statues of the batteries. With reference to the third recommendation it was stated by the plant expert, that the battery racks will be upgraded with regard to new seismic requirements.

REVIEW AREA/ISSUE NUMBER: Electrical Power Supply 5 (El 5)

ISSUE TITLE: Ground faults in DC circuits

ISSUE CLARIFICATION: At present there is no automatic detection of ground faults in DC circuits due to the lack of reliable tools. The occurrence of ground faults is recognized by the operating staff by emergency alarm initiated by the monitoring system. However, the detection of the failed or effected bus has to be done manually.

In order to localize the earthfault, one electrical consumer after another must be manually transferred from the main bus to an auxiliary bus until the earthfault alarm disappears. This may be a lengthy process.

Ground faults have to be detected within two hours. If a ground fault cannot be eliminated within 8 hours, a decision has to be taken as to whether the unit has to be switched off and cooled down (depending on the affected DC consumers).

A single earthfault does not affect the operability of the respective DC system, as it is operated fully unearthed under normal operating circumstances. Problems may occur first, where a second earthfault appears within the same DC system. Various combinations of double earthfaults can be distinguished leading to different consequences, e.g.:

- Double earthfaults in the same potential (negative or positive pole) do not influence operability, but they significantly complicate the earthfault search process as the number of possible combinations is highly increased.

- Double earthfaults in different potentials (negative and positive pole) constitute a short circuit, that may be cleared by blowing the respective fuses or tripping circuit breakers with the loss of one or two consumers. This case is covered by the single failure criterion as the affected train is considered to be not operable as a consequence.

- Double earthfaults affecting negative and positive potential simultaneously and when one of them occurs at an appropriate position within a control circuit where safety loads like an injection pump can be erroneously activated even if the process does not require their operation. Automatic trip by the protection system or manual tripping from the main control room is enabled. Clearing of this situation can only be achieved by manual de-energization of the respective DC bus, or by interrupting power supply from the 6 kV essential buses.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This weakness has been identified from operating experience at WWER-1000 and WWER-440 plants. It is a departure from recognized international practice.

Under worst case conditions, double earthfaults may not only lead to the loss of one train but may also introduce risks by forcing safety systems erroneously into operation.

COMMENTS AND RECOMMENDATIONS:

1. Analyse the possible worst case conditions of double earthfaults.

2. Investigate the possible consequences under consideration of a PSA approach, taking operating records into account.

3. Investigate options for fast or automated earthfault localisation.

4. Identify the weak points or components within the DC system which deliver the highest contribution to earthfaults and consider exchanging related equipment against more reliable devices.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The safety issue is applicable to Unit 5, but no action has been planned.

Kalinin Units 1 and 2:

This issue exists at KNPP, but no action has been taken.

South Ukraine Units 1 and 2:

The development of improved methods and tools for ground fault detection in the DC circuits is under progress.

3.7. CONTAINMENT

REVIEW AREA/ISSUE NUMBER: Containment 1 (Cont. 1)

ISSUE TITLE: Containment bypass

ISSUE CLARIFICATION:

In general, all the systems connected to the primary system are designed to bear the pressure of this system during normal operation and, in case of pipe break outside the containment building, can be isolated to avoid a loss of water from the primary circuit. But, there are other systems that in case of a single failure can be connected with the primary circuit and they are not designed for bearing this pressure. In these cases, the pipes located outside the containment building could break resulting in a complete bypass of the containment.

Rupture of a heat exchanger of the intermediate cooling circuit has been postulated to lead to a two phase flow discharge to the intermediate closed cooling circuit at a high flow rate (90 t/h). The integrity of this intermediate closed cooling circuit cannot be ensured under such conditions. In the worst case, the rupture in the intermediate closed cooling circuit could occur outside the containment, resulting in a complete bypass of the containment. This would result in direct releases of radioactivity from the primary circuit and would endanger decay heat removal in the long term, since the primary water inventory could be lost.

It should be noted that according to Russian specialists the existing relief capacity of intermediate cooling circuit is capable of handling of breaks in the heat exchangers up to a equivalent diameter of 80-100 mm. Further, the intermediate system is equipped with temperature, radioactivity and level monitoring and the signals are provided to the main control room.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

A design weakness in the overpressure protection of the intermediate cooling circuit (Level 1 of defence) could result in the damage of a safety support system, thus affecting Level 3 of plants' defence in depth. Although such a scenario has very low probability, the consequences could be a loss of primary coolant and a bypass of the containment of some amount of radioactivity contained in the primary coolant. This means that both safety functions can be impaired for scenarios beyond the design basis (DB) envelope.

COMMENTS AND RECOMMENDATIONS:

1. The design of the overpressure protection of the intermediate cooling circuit should be reviewed (including the postulated assumptions on break size) and if necessary, a design modification should be developed and implemented.

2. Within the framework of a review of the safety analysis report (TOB), attention should be paid to other LOCAs which could bypass the containment. All the lines going through the containment should be checked and the possibility of isolating them from the primary circuit to avoid bypass of the containment should be analysed.

REFERENCES: [2, 3, 5]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The intermediate circuit which penetrates the containment is equipped with safety valves inside the containment, level tank open to the containment, 3 fast acting motor-operated isolation valves (1 inside and 2 outside the containment), temperature monitoring and level monitoring in the system level tank (with indication to the control room). In case of a failure in the heat exchanger, the operator is able to isolate the containment penetration. There are, however, no instructions available for the operator. Containment cable penetrations are being replaced by penetrations of ELOX type.

Systems needed during reactor shutdown have manually-operated isolation valves. These valves are locked before the plant startup and organizational measures are in place to verify their status. Other penetration valves are either electrically operated (dia.>150 mm) or pneumatic (dia.<150 mm). Pneumatic valves are opened by compressed air.

Work was performed with respect to the containment prestressing system by replacing tendons.

The plant is considering further analysis on this issue.

Kalinin Units 1 and 2:

Reconstructed design of the main circulation pump cooling system lines has been implemented, which prevents inadmissible pressure increase in the system during primary circuit leaks. Main circulation pump cooling system lines at the Unit 1 have three localizing devices (one - inside, two - outside), and the Unit 2, two devices (one - inside, the other - outside).

It is planned to perform a complex analysis of the containment bypass, taking into account one localizing device rupture (requirements of PNAE G-10-007-89).

Existing cable penetrations not compliant with current safety requirements were replaced by those of ELOX type.

South Ukraine Units 1 and 2:

The rupture of the heat exchanger of the intermediate cooling circuit of the reactor coolant pump and the primary circuit let down flow after-cooler in the Units 1 and 2 is of concern. If the tubes of one of these heat exchanger break a discharge of water from the primary circuit outside the containment building could occur in the worst case.

The issue is not addressed in the LTSIP. However, the issue will be addressed in the SAR, as required by the Ukrainian regulatory guide.

3.8. INTERNAL HAZARDS

REVIEW AREA/ISSUE NUMBER: Internal Hazards 1 (IH 1)

ISSUE TITLE: Systematic fire hazards analysis

ISSUE CLARIFICATION: In a NPP, there are many components that can be a source of fire. These are generally related to electric equipment which can dissipate heat, and combustible materials such as oil and hydrogen.

All these equipment or combustible materials are normally located or could be placed in the different areas of the plant and could become a source of fire affecting equipment with the subsequent loss of safety systems.

A systematic fire hazards analysis needs to be performed to identify the different areas where fires can start and cause damage to safety systems in case of fire spreading. A fire hazard analysis also needs to be performed to verify the required fire resistance of the fire compartment boundaries, the requirements for the fire extinguishing systems and other features necessary to fulfill the fire protection requirements.

The first step in such an analysis should be based on plant walkdowns and on expert judgement in order to identify the areas in which a fire could affect the safety due to the equipment placed inside or placed in other areas reached by the fire. The existing situation should be compared with current national regulations, NUSS Safety Guide 50-SG-D2 [29] and IAEA-TECDOC-778 [30]. The analysis has to consider the effects of fire in the operation of fire detection and extinguishing systems located in the different areas of the plant.

As a result of this analysis, measures may need to be taken to prevent, detect and extinguish fires and to mitigate their effects on safety systems.

For NPPs in operation, such analyses should be performed periodically. A systematic fire hazards analysis, as specified by NUSS standards [9], could improve and optimize fire safety, thus reducing the risk of damage, and subsequent loss of safety important systems. This kind of systematic analysis has not been carried out for the 'small series' WWER-1000 plants.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. Level 1 of defence provides the main basis for protection against external and internal hazards including fire. The results of a systematic fire hazard analysis would show the extent to which defence in depth and the main safety functions of the plant can be impaired.

COMMENTS AND RECOMMENDATIONS:

1. A systematic fire hazards analysis is strongly recommended to be performed for each area of every plant. This should help to identify the weak points of the fire barriers intended to separate redundant trains, and to justify the acceptability of redundant train separation. The existing situation should be compared with the current national regulations, NUSS 50-SG-D2 [29] and IAEA-TECDOC-778 [30].

2. The secondary effects of fires and of the operation of fire extinguishing systems should be evaluated in the fire hazards analysis.

REFERENCES: [2, 3, 9, 29, 30, 31]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A fire hazard analysis based on experts judgement was carried out in the eighties and as a result of the analysis a number of measures were implemented, such as cables being covered with fire resistant coating, firewalls being created and fire doors replaced, staircase ventilation installed, high pressure water cannons installed in the turbine hall, an emergency control room constructed and plastic floor cover interrupted by steel inserts.

Systematic fire hazard and flooding analysis is planned to be performed together with the PSA study in the frame of the co-operation with Switzerland.

Plant and Atomenergoproject specialists were trained in systematic fire hazard analysis in the USA and will carry out the analysis together with a specialized Russian institute.

Kalinin Units 1 and 2:

Fire hazards analysis of the ECCS A-62550 compartments has been fulfilled. As a result, recommendations on fire protective coating of metal components of platforms in these rooms have been developed with the objective to prevent loss of all three trains of the emergency system due to the common cause.

An analysis of scenarios of fire initiation and spreading in safety important systems compartments has been fulfilled in the framework of “BETA” project. A probabilistic analysis of core damage due to fire has been carried out, but the results are considered as preliminary and need to be defined more precisely.

South Ukraine Units 1 and 2:

This kind of analysis has not been performed at Units 1 and 2 but it will be carried out in the second phase of the US DOE project to assess the safety of Unit 1.

Prior to the comprehensive analysis, a list of potentially hazardous areas and equipment which will be given special attention is developed as a compensatory measure.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 2 (IH 2)

ISSUE TITLE: Fire prevention

ISSUE CLARIFICATION: In accordance with NUSS Safety Guide 50-SG-D2 [29], an adequate degree of fire protection is required to be provided in NPPs. This should be achieved by a defence in depth concept in the design with three principal objectives. The first objective in this concept is preventing fires from starting, i.e. fire prevention.

Safety reviews of WWER NPPs have identified several weaknesses, which, in many cases are deviations from the NUSS Safety Guide. The main concern for 'small series' WWER-1000 NPPs is that redundant equipment, components and cable trains of safety important systems, are, in some areas, located without sufficient physical separation and are not protected against fire spreading (see issue S 16). This results from deficiencies such as:

- lack of qualified fire doors in fire barriers,
- redundant cable trains run too close to each other,
- lack of qualification of penetrations, and
- lack of fire resistance of overlayers covering the cables.

A fire could thus lead to the loss of more than one redundancy of safety important systems.

A specific concern in this respect is related to the cable spreading rooms. The cable spreading rooms under the MCR and ECR contain substantial quantities of safety system control cables which penetrate the ceiling into these control rooms. The segregation of cables belonging to redundant safety trains is inadequate. This is a serious weakness, since a fire in one of these areas could potentially lead to the loss of control over all three safety systems from the affected control room. It is not clear whether a fire affecting cabling in one control room would affect the functioning of the remaining control room.

Another concern is related to inadequate protection against oil fires. The pieces of equipment which are filled with oil are not always provided with bottom trays to collect oil in case of a leakage. The flange connections for oil piping do not have gland fixtures and casings. The check valves on the air ducts leading to the oil tank room were designed without spark protection. In the oil tank room, the heated instrumentation is not screened and the fire door is not designed against the pressure from an oil explosion. All these are deviations from the Russian standard VSN-01-87 [32] and NUSS 50-SG-D2 [29]. A specific concern related to oil fires is the oil lubrication of reactor coolant pumps.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: III

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from NUSS 50-SG-D2 [29]. Insufficient protection against common mode failures due to fire would seriously affect Level 3 of defence to control accidents within the design basis. The main safety functions can be questioned depending on the loss of redundant trains during DBA scenarios.

COMMENTS AND RECOMMENDATIONS:

1. Based on the results of the fire hazards analysis, the location and types of fire doors should be defined.

2. If overlayers are used for covering cables, their performance should be demonstrated prior to their use. In the case of overlayers already in place, their effectiveness after ageing should be checked.

3. The penetrations and connections between fire areas should be inspected to ensure adequate separation of the areas from one another.

4. Analysis is required to demonstrate that a fire affecting cabling in one control room would neither affect the functioning of the remaining control room nor prevent control of safe shutdown systems.

5. Probabilistic analysis should be carried out to quantify the core damage frequency resulting from fires in the spreading rooms.

6. Reactor coolant pumps of modified design should be installed at all WWER-1000 plants, or the lubrication system should be modified to avoid risk of major oil fire.

7. If feasible, separate the four 6 kV main distribution boards into two parts, by improving or additionally installing fire barriers with an appropriate fire resistance class. The ventilation system for these two parts must be treated likewise as to avoid spreading of fire or the negative influence of the combustibles.

8. An alternative approach to solve this problem would be by connecting 3 independent feeder cables to the three safety 6 kV busses. These feeders may be energized e.g. from the local 6 kV system which is not dependent on the units own supplies.

9. All the measures related to fire prevention should be implemented after the performance of the systematic fire hazards analysis.

REFERENCES: [2, 3, 25, 26, 29, 32, 33]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

During the NNPP Unit 5 design and construction period, there were no specific requirements regarding fire safety enforced in the former Soviet Union. These were developed and implemented only following a fire during the startup of the Zaporozhe plant. Some measures were implemented at the Novovoronezh plant additionally, such as covering of the cables with a fire retardant coating, installation of fire doors. The equipment of the MCP oil lubrication system is equipped with trays and located in a compartment with fire doors, fire dampers in the ventilation systems and curbs.

In order to compensate for the lack of fire prevention consideration in the plant design, a number of organizational measures was also implemented, related mainly to inspections, e.g. of the cable coating, etc. The quantity of combustible material in the plant appears to be small and well controlled.

Kalinin Units 1 and 2:

There are wooden fire doors (impregnated) with fire resistance of 1.5 hours installed in line with the design requirements in compartments housing safety or safety important systems, namely:

- safety systems compartments
- control rooms e.g. MCR, ECR, etc.
- safety important systems fire compartments boundaries adjoining to normal operation zones.

For cable corridors fire metal doors have been installed (fire resistance 1.5 hours).

The existing doors have been improved (seals, catches, upright bolts, and self-closing devices). According to the requirements of current Russian standards (VSN 01-87) it is necessary to replace wooden fire doors by metal fire doors with fire resistance equal to 1.5 hours.

All cables in the main building are covered with fire retardant coating and they meet respective fire protection requirements. Fire prevention “belts” are installed in the cable ducts, there are fire resistant penetrations. There are some situations, when 2 cable trains are installed on the top of each other. In such cases, a fire resistant duct (1.5 hours) protects one of the cable trains. The cables in containment are not covered with fire retardant coating at present. Design documentation for application of fire retardant composition “Flammoplast” to these cables have been developed.

Fire and thermal resistance of all 6 kV and 0.4 kV cables is regularly examined.

MCP oil system flange connections are covered with casings, the trays under equipment are in place. Oil from trays and casings is being gathered by a special system. The MCP oil system compartments are equipped with automatic fire detection and extinguishing systems. Considering these measures and based on the experience accumulated to date, the plant does not plan to replace the MCPs with water lubricated ones.

South Ukraine Units 1 and 2:

To remove combustible materials from the areas in which safety equipment are placed is a good practice. This is the reason to eliminate the oil lubrication system of the reactor coolant pumps. The control of all the different combustible located in the plant and works with risk of fire is another prevention measure.

There is no decision about changing the reactor coolant pumps or modifying their lubrication system in order to avoid fires due to oil, but both possibilities are going to be analysed. A system of fire detectors has been placed inside the reactor building.

The Kharkov Institute (KI) is going to analyse the effect of fire in cables in one control room and the impact on the functioning of the remaining control room in relationship with the safety. The conclusions of this study will be sent to all NPPs to take them into account.

The study to separate the four 6 kV main distribution boards into two parts will be carried out by KI during 1997, improving or additionally installing fire detection systems, suitable fire protection systems and fire barriers with an appropriate fire resistance class. Other possible alternatives will be taken into account during the development of the study.

A probabilistic safety analysis to quantify the core damage frequency resulting from fires in the cable spreading rooms will be carried out by the plant during the next two years based on the results of issue IH 1.

A strict control of possible combustibles inside the plant should be performed.

In 1995 an integrated programme of fire safety improvement has been adopted in Ukraine. But before that time the SNPP maintained the boiler’s facility in due order.

The plant has coated the tables with Polistop-Polyplast paste (1986-89) instead of originally applied OPK paste (1982-83). Later, the fire protection belts of Kamum paste have been replaced by Polystop packets.

The codes and standards are available for in-service inspection of the cables. Analysis for heat resistance have also been conducted with corresponding corrective measures.

1. There is a contract for supply of 350 qualified fire resistant doors.

2. The power cables penetrations have been replaced at their access through containment to RCP by penetrations of Elax type and “Garumilion”.

3. A replacement of control cable penetrations through containment by Elax penetrations was performed for safety systems and safety significant systems (control rods, safety valves, pressurizer, RCCs, temperature control, etc.).

4. As stated in design documentation, the containment cables’ lifetime is 20 to 30 years with regard to their type.

5. All flanges at Units 1,2 are equipped with casings against probable oil leak.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 3 (IH 3)

ISSUE TITLE: Fire detection and extinguishing

ISSUE CLARIFICATION: In accordance with NUSS Safety Guide 50-SG-D2 [29], an adequate degree of fire protection is required to be provided in NPPs. This should be achieved by a defence in depth concept in the design with three principal objectives. The second objective in the concept is detecting and extinguishing quickly those fires which do start, thus limiting the damage, i.e. fire detection and extinguishing.

Safety reviews of WWER NPPs [26] have identified some weaknesses in this area, which represent deviations from the relevant NUSS Standards or from applicable national regulations or standards.

A concern is the functional capability of the fire detection and alarm system which has to be designed and qualified for DBA conditions and internal and external hazards to ensure its capability to detect a fire or provide the alarm in case of such abnormal conditions. The equipment in the fire detection and alarm system was designed according to conventional industrial standards without the capability to resist earthquakes, or other abnormal conditions characterized by mechanical, thermal, chemical and other effects which might arise as a consequence of design basis accidents. In case of such abnormal conditions, the system may not be able to detect a fire or provide the alarm. This is not in compliance with OPB-88 [4] and NUSS 50-SG-D2 [29].

The fire water supply system for 'small series' WWER-1000 unit is different from that of WWER-1000 model 320. This system has six pumps which take water from two tanks of 2000 m^3 each. Four of these are electrical pumps and two are diesel driven. They operate in parallel to maintain the pressure in the general collector that supplies water to all the different points of the system. Due to the difference between the fire water supply system of 'small series' units and the WWER-1000 model 320, an analysis of the redundancy of the water supply system should be made.

In the MCR, ECR and other I&C rooms equipped with electric and electronic apparatus and having an area more than 20 m^2, there are no fixed automatic gas systems for fire extinguishing. This is not in compliance either with the requirements of the Russian standard VSN-01-87 [32] or NUSS 50-SG-D2 [29]. For fire water supply systems, the redundancy of water supply systems should be ensured and the source of water for fire extinguishing systems inside the containment should be safety graded.

Another concern is related to the sources of water supply for fire extinguishing systems within the containment. The water supply for these systems comes from the non-safety related rather than from the safety related service water system. This is a deviation from OPB-88, VSN-01-87 and NUSS 50-SG-D2. One concern is the lack of fire dampers in ventilation ducts. This would fail to isolate the affected fire compartment to prevent the spread of fire, heat or smoke to other fire compartments.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from the current standards. The equipment of the fire detection and alarm system is not qualified to specific environmental conditions including internal and external hazards (Level 1 of defence). In addition, systems important to safety are not sufficiently protected against fire and this would affect Level 3 of defence. Under certain scenarios within the DB envelope the fire equipment will fail to operate such that the main safety functions can be impaired.

COMMENTS AND RECOMMENDATIONS:

1. The fire detection and alarm system should be qualified for DBA conditions and internal and external hazards.

2. The redundancy of the water supply system should be ensured.

3. The source of water supply for fire extinguishing systems inside the containment should be safety graded.

REFERENCES: [2, 3, 4, 29, 32]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The fire detection system has been replaced/upgraded with a more advanced and reliable system. The compartments with 0.4 kV and 6 kV distribution boards, control panels, electronic equipment and generally compartments larger than 20 m^2 were equipped with a fire detection system. Cables were covered with fire retardant coating. An independent high pressure fire fighting system was installed in the turbine hall. An automatic air overpressure system has been installed on the staircases and other personnel evacuation routes in the reactor building. Fire fighting systems are redundant and are switched on in parallel automatically. Both water and power supply are redundant.

At present, work is underway to provide automatic makeup of fire fighting tanks, installation of water jets for turbine bearings, separation of turbine main oil tanks in boxes, installation of a fire detection system in the fresh fuel storage room, installation of a ventilation system for cable ducts, installation of an automatic air overpressure system on the staircases and other personnel evacuation routes in the turbine building.

Kalinin Units 1 and 2:

There are only resistance thermometers used to detect fires inside the containment. Smoke detectors as required by applicable standards are not installed in the cable runs. It was stated that the reason is that the Russian industry does not produce smoke detectors qualified for the containment environment.

There is a high pressure common fire water pipe at the plant, which provides fire water for the whole plant including safety systems and service water for auxiliary buildings. The fire water is supplied to a reservoir from the lake Udomlia by 2 lines. There are 2 groups of pumps in the pump station: service (2 working and 1 emergency) and fire (1 working, 2 emergency; one of the emergency pumps with diesel power). Two additional fire pumps are installed at Unit 2 pump station; these are emergency pumps and they are started up in the case of failure of basic fire pumps. Besides, pumps to maintain permanent pressure in water pipe are installed in Unit 2 pump station. Fire pumps control is automatic, remote or direct manual.

To improve fire extinguishing reliability, a design of separate fire water piping was developed but not implemented. In connection with the recently observed fouling of the lake water, a design modification has been developed based on the use of drinking water in the fire fighting system. It is planned to store the drinking water in reservoirs close to the pump station. Two additional pumps to maintain pressure in the fire water piping are also planned to be installed (1 working, 1 in reserve).

Cable corridors including safety systems ones and compartments with oil containing equipment were equipped with water spray fire fighting system. Each fire compartment fire water is supplied by a separate pipe isolated by an electrically driven valve with manually operated bypass except of leaktight compartment in the reactor building. Isolating startup devices, which provide water supply to reactor building leaktight compartment, are grouped to fire extinguishing distribution centre which is situated in the reactor buliding.

Working documentation for installation of automatic gas fire fighting system (control rooms, I&C compartments, radioactive materials storage compartments) has been developed.

South Ukraine Units 1 and 2:

At Units 1 and 2 the work planned for fire detection and extinguishing system modification is being carried out. This programme includes:

- development of a plan to modify fire detection and extinguishing system inside the containment building
- a project to modify the turbine hall roof fire extinguishing water pipes
- a fire alarm system installed at the compartments of the integrated switchyard KRU-6 and KRU-9 in Units 1 and 2.

Fixed automatic gas system is considered to be installed in MCR, ECR and other I&C rooms.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 4 (IH 4)

ISSUE TITLE: Mitigation of fire effects

ISSUE CLARIFICATION: In accordance with NUSS Safety Guide 50-SG-D2 [29], an adequate degree of fire protection is required to be provided in NPPs. This should be achieved by a defence in depth concept in the design with three principal objectives. The third objective in the concept is preventing the spread of those fires which have not been extinguished, thus minimizing their effect on essential plant functions, i.e. the mitigation of fire effects.

To mitigate the effects of a fire, it is necessary to ensure that the fire remains inside the area in which it started during a sufficient period of time to take additional measures to extinguish it or protect the equipment against the fire effects.

Fire doors, seals and dampers avoid the fire spreading through the doors, electrical and mechanical penetrations in the walls or ventilation ducts. Overlayers protect the components against the fire effects.

Safety reviews of WWER NPPs [26] have identified some weaknesses in this area, which represent deviations from the relevant NUSS Standards or from applicable national regulations or standards. Based on a plant walk-down, the following deficiencies have been detected:

- The fire doors are not fully in compliance with the fire protection rules, they have to remain closed and their position have to be supervised,
- The seals of penetrations and connections between fire areas do not exist or are defective,
- The cables of redundant trains are not separated,
- No adequate fire dampers in ventilation ducts are installed to prevent the spread of fire, heat or smoke to other fire compartments,
- No adequate curbs are located to avoid oil spreading.

A further concern is that the rooms with a potential fire danger and the evacuation corridors were not designed to have provisions to remove smoke in case of a fire. This would detrimentally affect the operating personnel and would lead to severe problems in the evacuation of personnel. This represents a deviation from the Russian standard VSN-01-87 [32] and from NUSS 50-SG-D2 [29].

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current standards. The equipment for the mitigation of fire effects have not been designed adequately and could fail on demand affecting Level 1 of plants' defence in depth. This may lead to main safety functions being impaired under certain scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. The existence of adequate fire dampers in ventilation ducts should be checked and additional dampers installed if necessary.

2. Based on the results of the systematic fire hazards analysis (see issue IH 1) fire doors, seals, overlayers and dampers should be built into in the different areas of the plant.

REFERENCES: [2], [3], [4], [26], [29], [32]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Fire doors were installed and are kept in the closed position.

Fire seals are installed in air ducts of e.g. main coolant pump (MCP) lubrication system compartments, the system is equipped with trays to collect leaking oil and curbs are provided in the reactor building to prevent oil spill.

Cables are coated with fire retardant coating but not separated.

The plastic floor covering is interrupted by metal sheet inserts and at these locations sprinklers are installed. Fire damages are installed in the ventilation ducts.

Kalinin Units 1 and 2:

The plant recognized the need to replace the existing wooden fire doors for metal fire doors with a required fire resistance of 1.5 hours, see also IH 2.

It is planned to install additional fire dampers in the ventilation ducts.

Measures regarding smoke removal in the evacuation corridors were implemented.

South Ukraine Units 1 and 2:

The following improvements will be carried out:

- About 350 fire doors, not in compliance with the fire protection rules, will be supplied.
- The seals of penetrations between fire areas will be done.
- The replacement of fire protection belts has been performed during the 1997 outage of Units 1 and 2.
- Adequate fire dampers will be installed.
- Adequate curbs should be placed to avoid oil spreading.
- The cables of redundant trains have been separated.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 5 (IH 5)

ISSUE TITLE: Systematic flooding analysis

ISSUE CLARIFICATION: Floods could be a common cause failure of different equipment leading to loss of main safety functions. The lack of separation in safety systems make the 'small series' WWER-1000 units vulnerable to common cause failures like flooding (see issue S 16). Therefore, a systematic flooding analysis should be done.

This analysis begins with a plant walk-down to identify:

- location of the equipment important to safety,
- paths or connections between areas with equipment important to safety (holes, drains, doors, etc.),
- systems containing water or steam which can discharge their contents into these areas,
- protective features such as sump alarms, barriers against water pass, isolation possibilities, etc.

This information would permit the determination of the highest possible water level in rooms where safety equipment are installed and the capability of the different sources of water to spray electrical and I&C equipment.

In the 'small series' WWER-1000 plants, the risks from internal flooding have not been systematically analysed.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This issue was identified as deviation from international practice. A missing flooding analysis may affect Level 3 of defence to control accidents within the design basis. Safety and support systems need to be protected adequately against internal flooding hazards to fulfill the main safety function as intended.

COMMENTS AND RECOMMENDATIONS:

1. A systematic flooding analysis should be carried out as a first step. This analysis should start from a walkdown that identifies the following:

- systems containing water or steam, including fire fighting equipment, in the building areas with systems important to safety,
- protective features (detection, retention, leak isolation possibilities),
- openings or connections between redundant safety related building sections and drains,
- the highest possible water level in rooms where safety related equipment are installed.

As a next step, a PSA should be carried out using the initiating events identified (see issue AA 10).

The effect of water spray on electrical and I&C equipment should also be addressed.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The systematic flooding analysis is not available for Unit 5. Systematic fire hazard and flooding analysis is planned to be performed together with the PSA study in the frame of the co-operation with Switzerland (see also issue IH 1).

Kalinin Units 1 and 2:

An analysis of scenarios of flooding of compartments housing systems important to safety including estimation of the highest possible water level and consequences of the flooding has been carried out in the framework of the “BETA” project. A probabilistic analysis of core damage due to flooding has been carried out, but the results available to date are preliminary and need to be defined more precisely.

A flooding analysis for steamline breaks has not been performed.

South Ukraine Units 1 and 2:

Systematic flooding analysis has not yet been carried out, but a study about circulation system pipes rupture inside turbine building has been developed. Pipe routings that could affect safety systems have been identified as service water pipes in the area of emergency feedwater pumps which are submitted to special surveillance.

The issue needs to be addressed to determine spray water effects inside containment and the circumferential building next to the reactor as well as floods into the area where emergency feedwater pumps are located.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 6 (IH 6)

ISSUE TITLE: Protection against flood for emergency electric power distribution boards

ISSUE CLARIFICATION: The 6 kV, 0.4 kV and DC emergency power distribution boards are located within different rooms of the electrical section of the turbine building. Fire extinguishing systems located in the floors above the switchgear rooms can lead to water ingress into the electrical rooms underneath when activated. Metal roof structures have been mounted to protect the switchgear cabinets from the water.

The emergency power supply to the ECCS systems and residual heat removal system may get lost in case of flooding in the rooms of emergency electric power distribution boards.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Insufficient protection of emergency power supply to safety systems against internal hazards may affect Level 3 of defence to control accidents within the design basis. The main safety functions would be impaired for certain scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. The amount of water which can penetrate into the switchgear rooms as well as possible damage to the electrical equipment should be investigated. It may become necessary to install a drainage system to remove the water in a controlled way. Water ingress shall not reduce reliability and availability of safety distribution boards.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The floor above compartments housing emergency electric power distribution boards has been designed to be waterproof, there are drains provided (no information was however provided about the functionality of the drains and its testing) and there are no penetrations through the floor. The cables to the emergency electric power distribution boards are entering these compartments from the bottom, there are no drains in this compartment.

The electrical cabinets are fitted with metal “roofs”, which serves as a dual purpose: to protect the cabinets from water and to enhance the fire detection system performance by directing the smoke to the fire detectors.

Measures to address the issue were implemented; further consideration for this issue will be necessary when the systematic flooding analysis is completed.

Kalinin Units 1 and 2:

The flooding due to actuation of the fire extinguishing systems has been analysed. The DC, 6kV and 0.4kV emergency power distribution boards are located at elevation 0.00 m of the deaerator in the turbine hall at Kalinin plant. To prevent the boards from water leaks due to actuation of the fire extinguishing system at the elevation 5.12 m, special covers were installed above the cabinets. The drainage system from the covers is not assembled yet. Only one section (valve) of the automatic fire extinguishing system can be opened at a time. The next actuated section (valve) opens only after the closure of the previously opened valve. Therefore it is considered that all three trains of ECCS power supply can not be lost due to flooding by fire water.

South Ukraine Units 1 and 2:

A drain system needs to be installed to remove the water from these rooms in a controlled way.

SNPP is intended to perform additional analysis for preventing of fire fighting water ingress to electrical equipment.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 7 (IH 7)

ISSUE TITLE: Protection against the dynamic effects of main steam and feedwater line breaks

ISSUE CLARIFICATION: The main steam and main feedwater pipes of different steam generators could affect each other in case of break inside the reactor building and in the section located between the reactor building and turbine building due to whip effect.

The rupture of one steam line could lead to a damage of other lines, e.g. feedwater lines, and of the containment penetrations.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from the NUSS standards [9], from OPB-88 [4] and from international practice. Improper design provisions to protect safety systems against dynamic effects affect Level 3 of plants' defence in depth. The safety function can be impaired for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. The consequences of secondary high energy piping breaks should be analysed and, if applicable, compensatory measures should be developed and implemented.

REFERENCES: [2, 3, 4, 9]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The assessment of consequences of high energy secondary pipe breaks inside and outside the containment was not carried out except for the containment penetrations.

At the NNPP Unit 5 there is no separate piping of the emergency feedwater feeding directly steam generators. The emergency feedwater piping is connected to the normal feedwater piping in the turbine hall where the three emergency feedwater pumps are located. The emergency feedwater could be lost due to a common cause failure. The original steam generators PGV-1000 did not have nozzles for the emergency feedwater. The replaced steam generators PGV-1000M have nozzles but they are plugged at the moment. Preparation of the design for backfitting of emergency feedwater lines connected directly to the steam generators is going on. This will also include construction of a separate building to accommodate 3 storage tanks for demineralized water (500 m^3 each) and a pump station. The design of the modification is prepared in line with OPB-88 (this also applies to all other modifications carried out at present). The progress is, however, rather slow due to lack of funds. Upon completion, the original emergency feedwater pumps are intended to be used as auxiliary feedwater pumps.

Kalinin Units 1 and 2:

The emergency feedwater pipelines are located separately and independent from the steam and main feedwater piping. Therefore dynamic impacts on the emergency feedwater supply system in case of breaks in the steam and main feedwater piping can not occur.

Regarding the concerns associated with the main steam and feedwater lines integrity, see also the issue CI 6.

South Ukraine Units 1 and 2:

A detailed analysis about whip effect between main steam and main feedwater pipes was not carried out (see also issue CI 6).

SNPP does not have any concern for dynamic impact on emergency feedwater pipelines resulted from steam and feedwater pipelines as there is a large distance between them.

Shock absorbers have been installed at such pipelines (Japan manufacturers). Calculations proved operability of the pipelines at seismic events up to 6 grades.

REVIEW AREA/ISSUE NUMBER: Internal Hazards 8 (IH 8)

ISSUE TITLE: Polar crane interlocking

ISSUE CLARIFICATION: The polar crane in the reactor building lacks adequate interlocks at the WWER-1000 plants. Interlocking is required to prevent the simultaneous transport of heavy loads over the reactor and spent fuel pool and to avoid a possible decoupling of the crane forks and hooks which would lead to heavy loads being dropped upon the different components located inside the reactor building and under the polar crane.

The drop of a heavy load on to the reactor or spent fuel pool could lead to a damage of the spent fuel or to a loss of the cooling capabilities and to a consequential release of radioactive materials.

This weakness has been identified from operational experience and safety reviews and is a deviation from international practice.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Insufficient design provisions to protect the reactor and the spent fuel pool from damage of dropped loads affect Level 1 of plants' defence in depth to prevent abnormal operation and failure. The main safety functions can be impaired for relevant scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

The Russian proposal to provide the polar crane with interlocking properly addresses the issue. In this connection, plant specific studies should be made to find and establish transport routes and methods that minimize the adverse effects of dropped loads.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

At Unit 5, there are no automatic interlocks which would prevent the transport of heavy loads over the reactor and spent fuel pool. The prevention is ensured by thorough organizational measures, which include procedures for the crane operator, limitation of access of personnel to the containment during heavy load transport, preparation of detailed transport routes for each outage involving heavy load transport.

Transport routes are marked on the containment floor. Following the event at South Ukraine plant, the electric drive coupling/decoupling the load was removed and this operation has to be done manually. Load decoupling during transport is not possible. It should be noted, that in a number of other PWR plants automatic interlocks are also not provided.

Kalinin Units 1 and 2:

The design of the 400/80/10/5t polar crane does not include automatic interlocks which would prevent the load transportation above the reactor and the spent fuel pool. Administrative measures are therefore in place. At present a reconstruction of the polar crane is considered, which foresees the introduction of automatic interlocks which would prevent the movements of load trolleys 400 t and 160 t in the zone above the reactor and spent fuel pool.

South Ukraine Units 1 and 2:

The decoupling of the load during transportation happened at Unit 2 due to human error. However, this event did no have consequences.

To ensure protection against erroneous operator action during load transportation two push buttons are located in the polar crane control panel which need to be simultaneously pushed to decouple the load. These push buttons are physically separated and placed in the opposite sites of the panel.

Procedures have been implemented to establish the routing of each load above the elevation +38.1 m inside the containment building to minimize the damage in case of load dropping.

A study is being carried out to automize this process by the Kharkov Institute and to avoid human errors.

SNPP is replacing the equipment that had exceeded its reservers.

3.9. EXTERNAL HAZARDS

REVIEW AREA/ISSUE NUMBER: External Hazards 1 (EH 1)

ISSUE TITLE: Seismic design

ISSUE CLARIFICATION: The seismic design basis (i.e. seismic input parameters) is generally not in accordance with current international practice. The as built status of NPP structures, components and distribution systems have to be evaluated against the site specific seismic loads.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Deficiencies in the seismic design basis would affect Level 1 of defence. All safety functions may be impaired or questioned by seismic events exceeding the improper seismic design basis for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. The seismic design basis (i.e. seismic input parameters) should be evaluated in accordance with current international practice and in any case with a minimum acceleration value of 0.1g associated with an appropriate design response spectra. Analyse, if necessary, the safety related systems, structures and components, including piping systems and supports, using the new seismic design basis.

2. The 'as built' status of the NPP structures, components and distribution systems should be assessed.

3. Upgrading of NPP structures, components and distribution systems should be made, if required.

4. Phenomena associated with earthquakes and which may induce permanent ground deformation should be evaluated for the impact on plant safety.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Seismic effects were not considered in the design. The effort to re-evaluate the site seismicity were initiated recently but are progressing slowly due to lack of funding. The objective of the re-evaluation is also to demonstrate that the site seismicity is lower than that used globally in this region following the Spitak earthquake in Armenia. It is planned to perform seismic analysis of the plant buildings and equipment in particular for NSSS after the site evaluation is complete. Application of LBB concept is considered for the primary circuit for the future and in this would include seismic upgrading if necessary. It was stated, that any plant modification performed at present is designed against OPB-88, i.e. considering seismic effects. AEP staff noted that seismic analysis of the containment was successfully carried out after the plant startup (up to a level of MSK 7).

Kalinin Units 1 and 2:

The buildings and structures of the plant were designed considering seismic effects. The results of a review performed by Atomenergoproekt in 1994-96 confirmed the seismic input parameters for the NPP site taken into account at the design stage.

The following seismic conditions were considered for the NPP site:

- 5th grade of MSK-64 scale (for Maximum Design Basis Accident with occurrence once every 1000 years);
- 4th grade of MSK-64 scale (for Design Basis Accident with occurrence once every 100 years).

The accelerogram of the earthquake, which occurred on 4 February 1977 in Nish, Romania, was recommended to estimate the Vrancea area influence. Analogous accelerograms of the data registered by the seismic station "Tashkent" on 16 July 1966 were recommended for estimation of the possible local earthquakes.

The activities to further verify the seismic characteristics of the plant site are underway and should be completed in 1999.

The analysis with acceleration of min. 0.1g was not performed since it is not required by the applicable Russian codes and standards (the current IAEA requirement is considered applicable to new plants only). Actual estimated horizontal acceleration of 0.04g (provisional, to be finalized in 1999) is used in the analysis.

Based on the results obtained, no corrective measures are planned to be implemented at the moment.

South Ukraine Units 1 and 2:

According to the report "Measures to ensure seismic resistance of SNPP Unit 1," 11-038D9, civil structures including reactor building, turbine and dearator building, water chemical treatment building, boron tanks building were analysed with respect to maximum design earthquake loads (6 deg. MSK, once in 10000 years). Accelerograms were developed for various plant elevations. The ground response spectrum used was based on accelerograms of the Karpatian earthquake in city Nis, 4.3.1977. The stack was analysed with respect to an earthquake of 5 deg. MSK. Systems important to safety were also analysed and where necessary, dampers installed.

In 1989 the site was investigated by Ukraine Academy of Science Institute of Geophysics resulting that Units 1 and 2 are located in the 5th degree area with zero degree growth. The maximum seismic hazards is equal to 6 degrees and all the different seismic structures have been designed with these basis.

At present, re-evaluation of structures taking into account the 1989 site data is planned in the frame of the preparation of the SAR.

REVIEW AREA/ISSUE NUMBER: External Hazards 2 (EH 2)

ISSUE TITLE: Analyses of plant specific natural external conditions

ISSUE CLARIFICATION: In accordance with NUSS 50-C-S [34], proposed sites are required to be adequately investigated with respect to all the characteristics that could affect safety in relation to design basis natural events. A site specific assessment is the first step to reach a decision regarding a particular event (see also issue EH 1). A systematic site specific assessment of this nature is not evident for WWER-1000 nuclear power plants.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING: The lack of an adequate investigation of the nuclear power plant site with respect to natural events is a deviation from current international practice.

COMMENTS AND RECOMMENDATIONS:

1. A site specific assessment should be made with respect to the design basis natural events.

2. A probabilistic analysis could be utilized to assess the potential hazard.

REFERENCE: [2, 3, 34]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The natural external conditions were taken into account during the design of the plant in the same way as for a conventional industrial facility (non-nuclear).

The design has considered the following conditions:

- Actual observed maximum wind speed 34 min/s.
- Maximum number of days during a year with strong winds (equal to and over 15 min/s) 30 days.
- Calculated wind speed possible once in 10,000 years (recurrence of 0.01%) 56 m/s.
- Average annual air temperature 5.7 °C extreme values from 41°C to 30°C.
- Daily maximum precipitation with recurrence of 1 % (repeated once in 100 years) 120 mm.
- Standard snow weight on horizontal ground surface 100 kgf/m2.

Tornadoes were not considered in the design, heavy rain and the river Don flood were analysed and they do not affect safety. Loss of off-site power due to cable ice overload is not considered a problem due to the diversity of power supply.

Kalinin Units 1 and 2:

External natural impacts were estimated during the NPP design development in the same way as for non-nuclear industrial facilities. The following characteristics were considered:

- Maximum wind velocity (observed): 28 min/s (blow).
- Maximum number of the windy days (stable, strong wind of larger than 15 min/s): 24.
- Maximum wind velocity (calculated) with repetition of once every 10 000 years (0.01% of probability): 42 min/s
- Average air temperature: 3.7ºC, absolute maximum air temperature: 34.4ºC (observed); absolute minimum air temperature: -42.3ºC (observed); calculated maximum air temperature with repetition of once every 1000 years: 39ºC (0.01% of probability), calculated minimum air temperature with repetition of once every 10 000 years: -58ºC (99.99% of probability).
- Calculated atmospheric precipitation daily maximum (repetition of once every 100 years): 80 mm, observed atmospheric precipitation daily maximum (repetition of once every 100 years): 102.2 mm.
- Average snow weight: 1kPa (100 kg/m^2).
- Basic tornado features:
  - annual probability of tornado occurrence: 3.6 10^7/reactor year
  - calculated tornado intensity: 1.75 (F-scale)
  - air rotation maximum velocity: 55 min/s
  - tornado forward motion velocity: 14 min/s
  - pressure difference (between centre and periphery): 37kPa.

The area of the NPP site location was considered as tornado safe since the tornado intensity is only 1.75 of F-scale, which equals to the impact of the air wave of 0.015 kg/cm^2 (0.0015 MPa) strength (it can cause only damage of window glass).

Resistance of structures to wind and snow load was reportedly evaluated with safety factors 2.5 resp. 2.0.

South Ukraine Units 1 and 2:

The SAR establishes that the different buildings and structures placed in the plant site are designed for wind overload of 95 kg/m^2 (120 km/h) and snow overload of 50 kg/cm^2. All plant structures were designed to withstand extreme atmospheric phenomena. Failure of water reservoir dam is also considered. In the frame of the SAR development, it is still planned to reassess the plant site natural conditions.

REVIEW AREA/ISSUE NUMBER: External Hazards 3 (EH 3)

ISSUE TITLE: Man-induced external events

ISSUE CLARIFICATION: Although the reactor building has a structural containment, other safety related buildings of WWER-1000 plants may be vulnerable to external man-induced events which generate extreme blast and impact type loading. It is important to assess the potential for such loading to the NPP through identification of sources in the site vicinity (e.g. airports, arsenals, pipelines, transportation routes, petrochemical facilities, etc.). The lack of such an assessment represents a deviation from the Safety Guide NUSS 50-SG-D5 [35].

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Insufficient protection of the safety systems may affect Levels 1 and 3 of defence. The safety functions could be impaired by man-induced external event for scenarios within the DB envelope.

COMMENTS AND RECOMMENDATIONS:

1. A source map for man-induced events in the site vicinity should be prepared.

2. Screening distances from the sources should be calculated using conservative assumptions. For sources/events which are not screened out, annual frequency of events should be calculated and compared with accepted "screening probability values". For sources/events which still remain as potential threat, a refined analysis should be carried out.

3. Depending on the results of the analysis, the structural and systems safety of the plant should be verified. If there are inadequacies, an upgrading programme should be initiated.

4. Strict administrative measures should be implemented to maintain the sources and events frequencies under control.

REFERENCES: [2, 3, 35]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The original design did not take into consideration man induced external events but additional analysis was carried out for the sources of risk both on-site (compressed gas, tanks, etc.) and off-site. There is a source map and analysis available now, considering (the numbers in brackets indicate the source distance/radius of effect in km) explosion on a river boat (1.6/.38), railway (11/1.5), road transportation (27/8), gas pipeline (12/.91). There is a military airport in Voronezh at a distance of 40 km, there are no civil aviation routes nearby (there are no small private planes allowed in the Russian Federation). The prohibited zone for flying is a cylinder of 15 km diameter with no limitation on its height.

There are no explosive chemical facilities, including toxic and corrosive in the nearest vicinity of the Plant site, therefore the possibility of their effect was not considered. The following are classified as potential explosion risk sources in the Plant site: diesel oil storage, acetylene generating plant, underground diesel fuel tank for emergency diesel generator station.

According to the assessment performed, all the above explosives are located at allowable distances from buildings and structures belonging to safety category 1 and designed for blast shock wave of 30kPa.

Kalinin Units 1 and 2:

There are no industrial facilities (storages, mines, open casts, drilling facilities, etc.) which can influence the NPP operation (within 30 km). There are no gas and oil pipelines in the region of the NPP site. There is no gas supply in the plant town.

The railway Bologoe-Sonkovo is at 3 km distance from the NPP and a potential source of external explosive danger. The effect of a maximum possible accident at the railway amounts to 2-5 kPa which is bellow the limiting bearing capability 4-6 kPa of the civil structures of the NPP (the weakest element).

From the hypothetical explosion of the whole stock of the ammunition stored on the territory of the nearby military object the peak pressure at the plant site was evaluated to be 0.529 kPa, i.e. less than pressure at which window glass breaks. The seismic effect at the territory of the NPP is not more than 2-3 degrees MSK.

The nearest place from KNPP where chlorine is stored is the waste water cleaning facility which consists of two cleaning systems. The total amount of chlorine stored is five containers of 800 litres each at pressure 6 bar. The distance from the NPP is 1.8 km. The sanitary protection zone for the chlorine facilities with container storage is 300 m.

In 1993 a specialized organization GosNII "Aeronavigatsija", Moscow performed the "Analysis of the influence of aviation situation on the safety of KNPP". The nearest airport is situated at 60 km distance from the NPP, the nearest local flight corridor is at a distance of 5 km from the NPP. The results of probability analyses of aeroplane crashes at the buildings of Kalinin NPP are the following:

- for aeroplanes with the weight of less than 10 t: 0.2 × 10-7,
- for aeroplanes with the weight of 10 - 20 t: 1.6 × 10-7,
- for aeroplanes with the weight of more than 20 t: 22.4 × 10-7.

It is possible to reduce this probability further to the level of 1 × 10-7 through organizational measures.

In 1996 NIAEP performed the "Evaluation of hazards from internal sources of explosives" which includes the following:

- an analysis of potential sources of explosive hazard situated at the NPP site has been carried out,
- air pressure wave parameters on the buildings of the first category according to the rules and regulations AP-5,6,
- evaluation of the building load bearing capacity,
- identification of critical sources of explosion was carried out and corrective measure proposed.

The critical source of explosive hazard was defined to be the installation of 12 pieces of hydrogen receivers 20 m^3 each. It is planned to replace the installation at a distance at least 340 m from the nearest 1st category building. It is planned to develop the design in 1999.

South Ukraine Units 1 and 2:

The different buildings should be assessed to external man-induced events which generate extreme "blast" and "impact" type loading or its personnel affected by toxic produces. According to the plant experts, explosives are stored 4 km from the plant at the construction site of a reservoir next to the river. The storage is, however, located behind a hill. The distance to the nearest railway is 3 km and the nearest gas pipeline is placed 8 km from the plant.

There are no chemical industries near to the plant site or fluvial transportation. A military airport is located at 40 km distance from the plant site. One important high road passes as close as 0.7 km the plant site and all kinds of materials can be transported through it.

The analysis of the man-induced external events was done within the initial SAR and additional analysis is planned to be carried out in frame of the final SAR. An automated NPP access control system has been installed and a programme for the physical protection has been developed.

A source map for man-induced events in the site vicinity as well as compensatory administrative measures still need to be developed.

3.10. ACCIDENT ANALYSIS

REVIEW AREA/ISSUE NUMBER: Accident Analysis 1 (AA 1)

ISSUE TITLE: Scope and methodology of accident analysis

ISSUE CLARIFICATION: According to the IAEA NUSS code on design, accident analysis shall be performed to ensure that the overall plant design is capable of meeting prescribed and acceptable limits for radiation doses and releases set by the regulatory body for each plant condition category. The operating organization needs additional analyses for personnel training to cope with accidents, for the preparation of emergency operation procedures, and for protection and signal settings.

Design basis events are chosen in the deterministic method of the safety assessment to encompass a range of related possible initiating events which could challenge the safety of the plant. These events form the basis for sizing and selecting safety systems. Analysis is made to show that the response of the plant and its safety systems to abnormal transients and accidents considering single equipment or human failure satisfies predetermined specifications both in damage to the barriers and in doses to the population.

A list of initiating events to be analysed, and some recommendations on how to perform them, are included in the Russian NTD documents TS TOB RU-87 and TS TOB AS-85 (Typical Content of Technical Justification) [36]. Concerns regarding accident analysis in the existing TOBs are with respect to the accident spectrum, the assumptions to be used, the acceptance criteria, the quality of analysis and computer code validation. A new regulatory document which defines the typical content of safety analysis report (SAR), similar to RG 1.70, was issued in 1996 by Russian Regulatory body. Currently, it only applies to new nuclear power plant.

The need for accident analyses for safety related modifications has been recognized, and the related calculations have been performed. However, these analyses have not been reviewed with respect to their completeness.

The issue was identified from safety reviews.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The lack of a complete set of analyses of design basis accidents and properly used methodologies increases the likelihood that transients and accidents could progress in severity and result in significant radiological releases which affects Level 3 of defence and makes it questionable whether the plant can safely cope with accidents with different probabilities of occurrence. Consequently, all main safety functions can be impaired and may not perform as demanded.

COMMENTS AND RECOMMENDATIONS:

1. A comprehensive safety analysis should be performed. First, the criteria should be established to select and classify the accidents to be analysed. Bounding cases, in accordance with selected safety analysis methodologies (rules), could be used to select the accidents to be calculated. The analyses should be performed and the results should be presented in a systematic manner, in order to permit an independent review.

2. The regulatory organization is encouraged to prepare a guide in which at least the following items related to anticipated operational transient and postulated design basis accident analyses of pressurized water reactors (PWR) are presented: events to be analysed, methods of calculation, assumptions to be used in the analyses and acceptance criteria for results.

3. Co-operation with the utility organization is encouraged, since it is always beneficial when a regulatory guide is being prepared.

REFERENCES: [2, 3, 36, 37]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Three types of the accident analysis are available or will be available at the NNPP Unit 5 :

1. Design basis accident analysis within the scope of plant safety analysis report (SAR).

2. Beyond design basis accident of ten initiating events (see issue AA 9).

3. Accident analysis within the scope of probabilistic safety analysis (PSA) (see issue AA 10).

The scope of design basis accident analysis in the plant safety analysis report is in conformity with the Russian regulatory documents "Typical Content of Technical Justification of Nuclear Power Plant Safety" (TS TOB AS-85) and "Typical Content of Technical Justification of Reactor Facility Safety" (TS TOB RU-87).

Presently the activities on first and second items are completed.

Level 1 PSA performed within the SWISRUS project was completed in March 1997.

Based on the PSA results the decision will be made on the required additional work on accident analysis.

Kalinin Units 1 and 2:

The design based accident analysis has been committed in the scope of the safety report according to TS TOB AS-85 document.

The IAEA guidelines on the accident analyses of WWER NPPs [37] are used currently to renew the DBA analysis.

It is planned to work out a list of the DBAs including justification by operational experience, regulatory requirements and PSA insights of chosen scenarios of their progression.

The DBA analysis will be updated on this basis.

South Ukraine Units 1 and 2:

A regulatory guide "Requirements on the content of safety analysis report for the existing WWER NPPs in the Ukraine" was approved by the Nuclear Regulatory Administration (NRA) and Goskomatom of the Ukraine and put into force on 27 November 1995. Two chapters (Chapters 4 and 5) in the guide contain the requirements on the scope and methodology for performing analyses of design basis accidents (DBA) and beyond design basis accidents.

Each unit of WWER NPPs operating and constructing in the Ukraine is required to prepare the safety analysis report (SAR) including the accident analysis, according to the Ukrainian regulatory guide.

The requirements for the accident analysis include:

- classification of initiating events
- a list of initiating events
- selection of initial conditions
- boundary conditions
- calculation methods
- format of presenting the results, etc.

The accident analyses included in the existing TOB (version 1991) of the SNPP Units 1 and 2 do not meet the requirements of the new regulatory guide.

Unit 1 is decided to be the reference Unit to prepare the SAR for the ‘small series’ units, and then the SAR will be adapted to the Unit 2, taking into account the design differences.

REVIEW AREA/NUMBER OF ISSUE: Accident Analysis 2 (AA 2)

ISSUE TITLE: QA of plant data used in accident analysis

ISSUE CLARIFICATION: The starting point of all accident analyses is a reliable plant database such as: geometrical data, material properties, physical and thermohydraulic data including boundary conditions of plant operational status. Every accident analysis needs a plant model or a detailed model of a specific part which must be constructed on the basis of valid data. This database is subject to quality assurance programme.

The experience of WWER-1000 plant owners shows that it is sometimes very difficult to obtain reliable and verifiable data on the plant construction. It is, however, mandatory to subject the data to a quality assurance procedure, when performing accident analyses. Therefore, it is essential to collect and verify the necessary data. Several deficiencies have been identified during safety reviews.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The lack of accurate and current plant information as the basis for accident analyses can lead to erroneous conclusions as to accident sequence and consequences. This could result in impairment of the preventive and mitigative capabilities of the plant.

COMMENTS AND RECOMMENDATIONS:

1. A QA procedure must be put in place to control the collection, documentation and verification of all data used in accident analyses.

2. Data should be collected from as many sources as possible. Cross-checking of data from different sources is necessary to eliminate unreliable or faulty data. Eventually, as-built measurements should be taken on-site during a plant outage. Set points and response times from I&C systems can also be verified on-site.

3. All data collected should be thoroughly documented in a database.

4. The data must also be related to a specified plant status in time, so that all plant modifications can be traced back in time.

5. Every single data in this database must be independently verified. A QA procedure must be put in place to assure an independent verification of the data and to control any modifications to the database.

6. An external audit of the QA procedure and the database should be carried out.

REFERENCES: [2, 3, 37]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A conservative approach was used in the preparation of plant data and in the updating of computer codes. Documents for quality control are being developed for carrying out specific calculation tasks, in co-ordination with operator and designers.

In the case of independent experts' review, relevant IAEA documents are used.

Within the framework of TACIS programme, Gosatomnadzor has completed the activity for preparation of plant data using a stringent QA programme. In this case, the Balokovo NPP was selected as a reference plant. The QA programme includes three major steps:

1. Listing the necessary data to be collected, with the consultation of western experts involved in using of the particular computer codes.

2. Filling out the list by scientific and technical centre staff, sending it to plant for careful check, and approval from plant chief engineer.

3. Approval by Gosatomnadzor after checking.

The same approach is going to be used to prepare the plant data of NNPP Unit 5 based on the previous work in the framework of SWISSRUS project.

Kalinin Units 1 and 2:

To assure QA of plant data when performing safety assessment, the following measures have been taken:

- the design was initially developed on the basis of the adopted conservative database

- at the stage of assembling, component testing, unit commissioning and startup, the database was verified and updated

- during KNPP operation parameters of components are monitored and continuously compared with the parameters of the original database

- in case the normal operating conditions are violated, the relevant information are assessed by the designer and taken into consideration in the database, if necessary, e.g. while the initial database is conservative in nature, the current database reflects the real plant status based on QA activities.

- initial data to be used by the designer for safety assessment are agreed with KNPP experts.

South Ukraine Units 1 and 2:

The Nuclear Regulatory Administration (NRA) has the requirements regarding the QA programme, which has to be established for every safety-related activity.

The quality of the accident analysis for the Unit 1 will be ensured by a QA programme which is to be established before the work. The data collection, documentation and verification are to be controlled.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 3 (AA 3)

ISSUE TITLE: Computer code and plant model validation

ISSUE CLARIFICATION: A computer code is based on a model of the reality the code should describe. A comparison of the code predictions with experimental data gives an indication to which extent the code is validated, i.e. the model sufficiently reflects the reality. The general principles are given in the NUSS Guide 50-SG-D11 (under revision) and IAEA-EBP-WWER-01 [37]. In addition to the code validation, it is essential that each code user is qualified to use that code. The qualification has to be based on successful analysis of test cases, such as separate effects tests and integral system tests. A number of International Standards Problems have been used to validate codes and at the same time to qualify organisations to use these codes.

The main computer codes used for accident analyses (TRAP code package: DYNAMIKA, TETCH-M, KANAL) have been developed in the Russian Federation. According to the Russian experts, these codes have been validated and verified and are currently under review by the Russian regulatory authority Gosatomnadzor for final certification.

Additional analyses, mainly for comparative purposes, are now being performed with western codes (RELAP-5 [versions 2.5 and 3.1], CATHARE, ATHLET). However, the applicability of the models included in these codes for WWER-1000 has not been fully confirmed.

The calculation of severe accidents is in an early stage of development. The use of western codes (MARCH, MELCORE) will require adaptation and verification.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

Without careful validation of computer codes for the intended application, the code predictions cannot be considered reliable. This could result in impairment of the prevention and mitigative capabilities of the plant.

COMMENTS AND RECOMMENDATIONS:

1. All accident analyses should be performed following established procedures for
   - code development
   - code use and
   - model construction.
2. Code developers should provide frozen code versions with adequate code documentation and user guidelines.
3. The input data reliability, plant model used and the procedure to run the code are the responsibility of the code user, who must use properly reviewed QA procedures for this work.
4. Any organization performing accident analyses for WWER plants should present a report addressing the plant models used and information on the validation of codes, as well as information on the qualification of the code users. This report should address the applicability of the code and plant model for a given type of accident. It should also assess the accuracy by comparison with separate effect test and integral test facility and by performing sensitivity studies.

REFERENCES: [2, 3, 37]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Two categories of computer codes are to be used in the accident analysis. The Russian design codes have been validated by OKB Gidropress and submitted to Gosatomnadzor of the Russian Federation for review and approval. The western computer code are to be validated via the standard sample problems programme for all WWER NPPs and OECD matrix verification programme.

Some western codes, i.e. RELAP, MELCORE, ATHLET, CATHARE, ESCADRE, are already available for use in the Russian Federation.

Kalinin Units 1 and 2:

The main tool to perform DBA calculations is the TRAP code package. Efforts are ongoing to qualify TRAP for best estimate type calculations (see also NNPP Unit 5).

South Ukraine Units 1 and 2:

The regulatory guide "Requirements on the content of safety analysis report for the existing WWER NPPs in the Ukraine" in its Section 2.5 requires that the information regarding the validation of calculation methods and computer codes used in the safety analysis be presented in the safety analysis report.

Two categories of computer codes are to be used in the accident analysis. The Russian design codes have been validated by OKB Gidropress and submitted to Gosatomnadzor of the Russian Federation for review and certification. Ukraine will be informed regarding the status of the validation and approval for the use of computer codes in the Russian Federation. The western computer codes are to be validated via the Ukrainian participation in the standard sample problems programme for all WWER NPPs and OECD matrix verification programme.

Currently, the validation of computer code on the stress calculation of pre-stressed containment is progressing in Ukraine.

Some western codes, i.e. RELAP, CONTAIN, MELCORE etc. are already available for use in the Ukraine.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 4 (AA 4)

ISSUE TITLE: Availability of accident analysis results for supporting plant operation

ISSUE CLARIFICATION: The results of accident and transient analyses should be used to support plant operation in the areas such as setting limiting values for different operational parameters, preparation of operating instructions and emergency operating procedures (EOP) and operator training. It is important to train the operators with regard to realistic and expected characteristics of an accident event and to provide high quality EOPs to mitigate accidents which may occur.

The accident analysis reports available at the 'small series' WWER-1000 plants present the results of conservative licensing analyses and do not meet the needs mentioned above. The emergency operating procedures currently used at the plants do not always have basis supported by the best-estimated calculations.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confirming the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The lack of realistic information to the operators can affect the correct actions to be taken, which does not satisfy international practice for training and EOP development.

COMMENTS AND RECOMMENDATIONS:

1. The scope of the accidents analysed needs to be expanded, the calculation time should be extended, the quality of analysis work must be improved, and realistic (best estimate) modelling should be used.

2. The plant staff should be encouraged to utilize more analyses to support their work.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The NNPP has a plan to develop symptom-based emergency operating procedures (EOP) for Unit 5. For this purpose, best-estimate calculations will be made by using up-to-date computer codes RELAP 5 and MELCORE. The accident scenarios to be used for developing EOP will be selected based on the Level 1 PSA results (SWISRUS project). The results of these analyses will also be used for training to enable the operator to better understand the accident scenarios and the expected results of the prescribed operator actions.

Kalinin Units 1 and 2:

In the framework of developing a set of EOPs based on EdF methodology and for the purpose of computational justification of procedures realistic assessments of the processes were conducted. These results are used by the operating personnel when getting prepared for the work under these EOPs.

South Ukraine Units 1 and 2:

The SNPP has contracted to RRC Kurchatov Institute, OKB Gidropress and Energorisk to perform some best estimate analyses. The results of these analyses will also be used in training to enable the operator to better understand the accident scenarios and the expected results of the prescribed operator actions.

Energorisk of the Ukraine is performing the Level 1 PSA for the Unit 1. The RELAP 5 and CONTAIN codes are to be used to make best-estimate calculations, since the analyses done in the TOB are based on the conservative assumptions and cannot be used for supporting operating procedures.

RRC Kurchatov Institute and OKB Gidropress provided a report “Guidance on the control of beyond design basis accidents for the Units 1 and 2”. Ten initiating events recommended for consideration by the former Soviet Union Ministry of Power in 1991 for the beyond design basis accidents were included in the report. The Guidance provides the symptoms and operator actions for each of the initiating events. It was stated by Russian expert that the report was developed based on the calculations of initiating events. The results have not been used by the plant to develop emergency operating procedures.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 5 (AA 5)

ISSUE TITLE: Main steam line break analysis

ISSUE CLARIFICATION: The results of accident analyses of the main steam line break (MSLB) showed that reactor returns to criticality after scram. The analysis was made by OKB Gidropress using two assumptions: mixing in downcomer and no mixing in downcomer. In the latter case, it was assumed that cold water from the defective loop enters and covers one quarter of the core. This resulted in an increase of the power level to 43%. The worst case with respect to the cooling of the defective loop is when the single failure criterion is applied such that either the main coolant pump in the defective loop will not stop or such that the feedwater supply to the defective steam generator cannot be stopped.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This issue is a deviation from the national rules. According to PBJa-89 [5], a recriticality of the reactor after a scram is not allowed under any conditions (an exception to this requirement must be approved by the regulatory authority in case the deviation does not lead to the violation of permissible limits). However, according to western standards recriticality itself is allowed if it does not lead to an unacceptable damage of fuel elements.

COMMENTS AND RECOMMENDATIONS:

1. A deviation from the regulatory requirement is obvious and measures should be taken depending on the requirements of the regulatory body.

2. Considering that four main steam lines and four main feedwater lines run closely to each other outside the containment and that each two main steam lines run closely for about 30 m inside the containment, more than one steam line break scenario should be analysed in case it cannot be demonstrated that such a scenario is excluded by design.

REFERENCES: [2, 3, 5]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The main steam line break (MSLB) analysis was performed for the standard WWER-1000/320 unit. The plant does not plan to perform an updated specific calculation for the Unit 5, considering that the results of MSLB analysis for WWER-1000/320 unit can be applied to the NNPP Unit 5.

Kalinin Units 1 and 2:

The existing MSLB calculations in the TOP presents the results of DBA calculations performed using conservative assumptions (increased level of reactor initial power: 107%, most unfavourable combination of the core neutron-physical characteristics, additional failure, etc.).

Recalculations for KNPP Units 1 and 2 should use the real plant characteristics (in particular, supply of boric acid solution by piston pumps, increased worth of emergency protection, etc.). It is expected to achieve considerably reduced power level increase. The issue of two MSLBs is being addressed by design measures.

South Ukraine Units 1 and 2:

The results of analysis of the main steam line break contained in the TOB of Unit 1 showed that a return to power of 43% could occur at 50 s after the initiating event.

The main steam line break accident is to be recalculated in the course of safety analysis report preparation.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 6 (AA 6)

ISSUE TITLE: Overcooling transients related to pressurized thermal shock

ISSUE CLARIFICATION: The analysis of overcooling transients related to pressurized thermal shock (PTS) is needed to assess the risk of a reactor pressure vessel brittle fracture. However, an analysis of this nature was not found in any of the 'small series' WWER-1000 plant TOBs (Safety Analysis Reports).

If overcooling transients of this nature occurred under circumstances where the reactor pressure vessel has a high fluence value and where the primary pressure is high, there is a danger that cold water may cause a thermal shock in the pressure vessel downcomer welding and, as a consequence, the pressure vessel integrity would be endangered because of embrittlement. It is therefore necessary that the procedures based on the results of overcooling transient analyses are provided (see issue CI 1).

This issue was identified from operational experience. In 1985, an overcooling transient occurred in the Zaporozhe NPP [26]. The initiating event was a fire in a transformer, which caused a loss of off-site power to Units 1 and 2. The primary coolant temperature decreased from 285°C to 160°C within 15 minutes. The risk of a reactor pressure vessel brittle fracture in the case of 'small series' WWER-1000 reactors can be fully estimated only if overcooling transients analysis related to PTS have been performed.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The lack of an analysis of overcooling transients related to PTS would affect Level 3 of plants' defence in depth to control accidents within the DB. Therefore, the impact of overcooling transients related to PTS on the RPV integrity cannot be evaluated.

COMMENTS AND RECOMMENDATIONS:

1. The plants should have a comprehensive set of overcooling transient analyses. This would enable a better understanding of those situations which could lead to a brittle fracture of the pressure vessel. Examples of initiating events which may lead to an overcooling transient are:
    - stuck open pressurizer safety valve which later closes
    - small break LOCA
    - secondary side leakage including steamline break
    - primary to secondary side leakage and
    - feedwater system malfunction.

2. The plant operators should obtain an accurate summary of the overcooling transient analyses results, so that this information can be incorporated into the operator training.

3. Based on the results of the analyses, specific emergency operating procedures should be prepared for the prevention of such transients and for the mitigation of their consequences.

4. The IAEA-EBP-WWER-08 report "Guidelines on Pressurized Thermal Shock Analysis for WWER NPPs" [17] may be considered as a guidance for analysis.

REFERENCES: [2, 3, 10, 17, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

At Unit 5, the list of overcooling transients related to PTS, based on the list of design basis accidents and beyond design basis accidents, have not been developed.

For the purpose of future extension of the reactor vessel service life, the design modifications were developed and are in implementation phase by the plant for:

- bypass of emergency cooling-down heat exchangers
- water heat-up in boric acid storage tanks above 55°C
- water heat-up in accumulators above 55°C.

Kalinin Units 1 and 2:

Based on the PTS analyses the main designer of reactor pressure vessel has provided information to the plant that it is not necessary to take any additional measures to improve the situation with PTS problems before 2002 year. However, water in boric acid storage tanks is heated to 55°C now.

An analysis of the situation which can lead to brittle fracture of primary circuit equipment has been performed in the framework of EOPs development using the EdF methodology. These results have been taken into account during development of EOPs. Measures taken to prevent brittle fracture of primary circuit equipment include permanent monitoring of permissible pressure to temperature ratio for primary circuit as well as recovering measures in case brittle fracture conditions arise.

South Ukraine Units 1 and 2:

The overcooling transients is required to be analysed by the Ukrainian regulatory guide. The list of overcooling transients related to pressurized thermal shock should be developed, based on the list of design basis accidents and beyond design basis accidents. The plant indicated that the five examples of initiating events recommended in IAEA-EBP-WWER-05 [10] will be included in the analysis of overcooling transients.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 7 (AA 7)

ISSUE TITLE: Steam generator collector rupture analysis

ISSUE CLARIFICATION: The steam generators (PGV-1000) of the 'small series' WWER-1000 nuclear power plants in operation have developed cracks in the primary collectors. The observed maximum crack length from the secondary side is about 1000 mm, and a larger critical crack length could be expected.

The maximum leakage path for a PGV-1000 steam generator is calculated by the Russian designer to be about 100 mm equivalent diameter. Analysis of SG collector rupture with a break size of 100 mm equivalent (Gosatomnadzor has not approved this value yet because of having not received a justification report) diameter has been performed for 'small series' WWER-1000 units by OKB Gidropress. The analysis indicates that operator actions should be taken at 15 min. after initiation of the event, i.e. open BRU-A, venting of primary circuit, open safety valves on the pressurizer etc., in order to terminate break flow and to avoid the excess loss of water inventory.

The accident was also analysed by RRC Kurchatov Institute with the BRU-A stuck open. However, the possible consequential break of steam lines, which are not designed for hot water loading, was not taken into account in this analysis.

In the original WWER design, a failure of the SG collector was not considered a DBA scenario. The consideration of larger primary to secondary leaks as DBAs is based on operational experience from WWER-440/213 NPPs (Rovno accident) and WWER-1000 NPPs as discussed above.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

A comprehensive analysis of a large SG collector rupture is not available, and this affects Level 3 of plants' defence in depth to control accidents within the DB. A large steam generator collector rupture accident can impair both main safety functions in case the BRU-A valve fails to close. This could lead to a bypass containment leakage to the environment via a BRU-A valve, and the long term cooling of the core would be endangered by losing the primary water inventory via a BRU-A valve.

COMMENTS AND RECOMMENDATIONS:

1. The national approach on primary to secondary leak treatment is still under development and implementation.

2. The results of existing steam generator collector rupture analyses should be reviewed and the need for safety improving measures or design modification should be considered.

3. A comprehensive study of the steam generator collector rupture accident should be performed. This should take into account the possible collapse of the main steam lines and the justification of the break size. The single failure criterion should be applied to ensure that the worst case is being considered.

REFERENCES: [2, 3, 47, 49]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:
The analysis of SG collector rupture for Unit 5 was performed before 1991 according to the "Guide on means and methods to control the beyond design basis accidents for NNPP Unit 5". The need for re-evaluating and upgrading the previous analysis will be decided by the result of Level 1 PSA of SWISRUS project.

Kalinin Units 1 and 2:

The thermohydraulic calculations of 100 mm equivalent diameter leaks from primary to secondary circuit (with conservative assumptions) have been performed. These calculations are presented in the TOB AS document (Technical Justification of Safety of KNPP Units 1 and 2).

The thermohydraulic calculations of primary to secondary leaks, including break size of 100 mm equivalent diameter, with and without actuating of safety systems, have been carried out in the framework of developing a set of emergency operating procedures using the EdF methodology.

The analysis has shown that in case of such accident the design solutions implemented at Units 1 and 2 of KNPP, including the main isolation valve and check valves located before BRU-A, allow to avoid impermissible discharges to the environment when operators act properly.

South Ukraine Units 1 and 2:

Analysis of SG collector rupture with a break size of 100 mm equivalent diameter has been performed by OKB Gidropress and the result was presented in the TOB (1991 version) of the SNPP Units 1 and 2. The analysis indicates that operator action should be taken at 15 min after initiation of the event, i.e. open BRU-A, venting of primary circuit, open safety valves on the pressurizer, etc., in order to terminate break flow and to avoid the excess loss of water inventory. However, OPB-88 requires no operator action within 15-30 min after initiation of an event.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 8 (AA 8)

ISSUE TITLE: Accidents under low power and shutdown (LPS) conditions

ISSUE CLARIFICATION: When a reactor is shutdown for maintenance and refuelling, some safety systems are switched off or isolated. Moreover, a great number of operator actions are required under this situation for different purposes. From the safety point of view, there are less barriers and protective means to prevent an event from developing into an accident.

Accidents which take place during low power and shutdown conditions (LPS) have been under extensive study all over the world for several years. Results have shown that the risk of an accident initiation during the shutdown and refuelling phase is high. Important contributors to the risk are boron dilution, loss of residual heat removal with the reactor cooling system in reduced inventory conditions, loss of primary coolant, loss of off-site power, fires and human errors.

Analyses of the shutdown and refuelling conditions are not available for ‘small series’ WWER-1000 reactors. However, a positive feature as compared with most other PWRs is that during the maintenance of steam generators, there is no need to decrease the water level in the reactor pressure vessel to the loop level.

OKB Gidropress recommends that two cold water sources be made available during shutdown conditions. According to the Russian experts, this recommendation has been implemented at the Russian and Ukrainian plants.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The lack of an analysis of accidents under LPS conditions affects Level 1 of plants’ defence in depth. All the main safety functions can be impaired as seen from generic observations of PSA studies made for different plant types world-wide. The issue contributes to the risk of losing barriers and main safety functions at the low power and shutdown conditions, if proper procedures are not implemented.

COMMENTS AND RECOMMENDATIONS:

1. A study related to accidents during low power and shutdown (LPS) conditions should be initiated, and measures should be taken to solve the resulting plant specific vulnerabilities.

2. Technical specifications addressing the administrative and equipment requirements for LPS conditions and emergency operating procedures should either be revised or developed.

REFERENCES: [2, 3, 38]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Analysis of accidents during shutdown conditions for Unit 5 has not been considered by the plant. Up to now, no definitive list of initiating events exists. Several accidents during shutdown conditions were already analysed, for example, a dropping of the container with spent fuel assemblies. A procedure to prevent boron dilution during the refuelling, transportation and storage of the spent fuel exists, which was developed on an engineering analysis of possible diluted or pure water sources (see issue RC 1).

Kalinin Units 1 and 2:

Same status as Novovoronezh Unit 5.

South Ukraine Units 1 and 2:

Analysis of accidents during shutdown conditions for the SNPP Units 1, 2 has not been considered by the plant. Up to now, no definitive list of initiating events exists. Several accidents during shutdown conditions were already analysed, for example, a dropping of the container with spent fuel assemblies. A procedure to prevent boron dilution during the refuelling, transportation and storage of the spent fuel exists, which was developed on an engineering analysis of possible diluted or pure water sources (see issue RC 1).

REVIEW AREA/ISSUE NUMBER: Accident Analysis 9 (AA 9)

ISSUE TITLE: Severe accidents

ISSUE CLARIFICATION: Current practice is to perform the analysis of very low likelihood accidents but more severe than those considered explicitly in the scope of DBA and even beyond DBA. Severe accidents may cause such plant deterioration that proper core cooling cannot be maintained and fuel damage occurs. These severe accidents have a potential for major radiological consequences if radioactivity released from the fuel was not adequately confined.

OPB-88 [4] requires the analyses of severe accidents for WWER-1000 NPPs. These analyses are used to identify existing weaknesses in the measures for prevention and mitigation. The analyses so far done for the 'small series' WWER-1000 units do not have a systematic approach and required quality, and are incomplete in scope.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This issue represents a deviation from the national standards and from international practice. The lack of the analyses of severe accidents would affect the appropriate actions to be taken during the course of an accident.

COMMENTS AND RECOMMENDATIONS:

1. The work on severe accident analyses should be accelerated. The quality of the analysis should be ensured to reach an internationally acceptable level.

2. Guidance on the performance of severe accidents' analysis should be developed.

REFERENCES: [2, 3, 4]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The former Soviet Union Ministry of Power recommended in 1991 ten initiating events to be analysed for consideration of the beyond design basis accidents (BDBA). The ten initiating evens are the following:

- SBLOCA with failures of HPI and LPI,
- MBLOCA with failures of HPI and LPI,
- LBLOCA with failures of HPI and LPI,
- PRISE with BRU-A and SG safety valves stuck open,
- RHR pipe rupture with failure of isolation valves,
- LBLOCA with failure of closure of ventilation system (duct diameter 400 mm),
- total loss of electrical power,
- main steam line break at the non-isolable part with failure of RHR system,
- rupture of SG collector with failure of HPI and EFW,
- failures of heat removal from secondary side and of LPI.

The calculations were done on the above initiating events, and based on the calculations, a document "Guide for methods and means of beyond design basis accident management for Novovoronezh WWER-1000 Unit 5" has been implemented.

In addition, within the framework of SWISRUS project, Level 1 of PSA was completed in June 1998. The need for extending the list of initiating evens for BDBA will be analysed.

Kalinin Units 1 and 2:

Same status as Novovoronezh Unit 5. Within the framework of the "BETA" project, a Level 1 PSA is completed in December 1998. The need for extending the list of initiating events for BDBAs will be analysed.

South Ukraine Units 1 and 2:

The report "Guidance on the control of beyond design basis accidents for the SNPP Units 1 and 2" provided by the RRC Kurchatov Institute and OKB Gidropress contains ten initiating events with event symptoms and operator actions to be taken, as mentioned for NNPP Unit 5.

The report has not been reviewed and approved by the Nuclear Regulatory Administration (NRA). Therefore, the results have not been used by the plant to develop emergency operating procedures.

The Ukrainian regulatory guide "Requirements on the content of safety analysis report for the existing WWER NPPs in the Ukraine" in its Chapter 5 requires the analysis of beyond design basis accidents (BDBA). A list of initiating events is recommended in the appendix to Chapter 5:

- total loss of electrical power,
- SBLOCA with failure of HPI,
- SBLOCA with failures of HPI and LPI,
- SBLOCA with total loss of electrical power,
- MBLOCA with failure of HPI,
- MBLOCA with failures of HPI and LPI,
- LBLOCA with failure of HPI,
- LBLOCA with failures of HPI and LPI,
- LBLOCA with failure of containment spray system,
- ATWS,
- total loss of feedwater,
- PRISE (SG cover lifting) with BRU-A stuck open,
- main steam line break,
- LOCA with failure of bubbler condenser (WWER-440).

Units 1 and 2 are required to submit safety analysis reports (SARs) according to the Ukrainian regulatory guide to the NRA to get the license for operation. The work on the analysis of BDBA for Units 1 and 2 have not been started because of financial problems.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 10 (AA 10)

ISSUE TITLE: Probabilistic safety assessment (PSA)

ISSUE CLARIFICATION: PSA is an important tool which evaluates all the different aspects (technical and human) in the assessment of plant safety. PSA may be used to rank the importance of the different aspects of the plant in terms of nuclear safety. Specifically, PSA results are an important base for the assessment of the measures directed to upgrading the safety. The final results of the PSA are not available at 'small series' WWER-1000 NPPs.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current international practice.

COMMENTS AND RECOMMENDATIONS:

1. A Level 1 PSA should be performed as a minimum for all 'small series' WWER-1000 NPPs.

2. An exchange of information on different studies should be arranged between the organizations conducting these studies. Such an exchange should start before completion of the PSAs, e.g. for a comparison of the reliability models and equipment failure data to be used in each study. It would also be worthwhile to invite independent experts who have experience with similar studies to conduct a peer review of the analysis methods. Such a review would be useful for the various steps during the study, as well as after obtaining the final results.

REFERENCES: [2, 3, 45]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

In 1995 Atomenergoproject has made a PSA of the Unit 5. However, due to the insufficiency of thermal hydraulic calculations the results could not be accepted.

In March 1997 the activities on Level 1 PSA was completed within the framework of phase 1 of the SWISRUS project [45], and the activities on PSA with external events and "Living PSA" will be continued.

Kalinin Units 1 and 2:

It was reported that Level 1 PSA had to be completed in December 1998 within the framework of the "BETA" project under USNRC support. External events (earthquakes, fire, floods, etc.) would be included later.

South Ukraine Units 1 and 2:

Unit 1 PSA is being performed by Energorisk of the Ukraine under a contract from the SNPP. The scope of the PSA included Level 1 plus radioactivity release. The work will be finished at the end of 1996. However, it is postponed for one year because of financial problems. So far, 35% of the work volume has been completed, including reliability database (collected from Unit 1), system analysis, input deck for RELAP 5 and CONTAIN codes, and 2 event trees. After grouping, 18 initial events were decided.

The PSA results from Unit 1 are planned to be adapted to Unit 2.

The Ukrainian regulatory guide in its Chapter 6 requires the performance of PSA for each unit, as a part of the safety analysis report.

The long term safety improvement programme includes the performance of PSA Units 1 and 2. The PSA is considered as a complementary method to the deterministic analysis.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 11 (AA 11)

ISSUE TITLE: Boron dilution accidents

ISSUE CLARIFICATION: Inadvertent boron dilution accidents may happen during power operation and under low power and shutdown conditions as well.

Examples of initiating events which may result in a boron dilution accident during power operations are:

- small break LOCA and
- Steam generator tube rupture (secondary to primary side leakage).

In small break LOCA events, the water level in the reactor pressure vessel may decrease below the hot leg elevation. As a consequence, steam begins to flow to the steam generator and condenses there. However, steam contains practically no boron and, consequently, the boron concentration in the cold leg loop seals begins to decrease. Depending on the total duration of this situation, the boron concentration in the loop seal may decrease to a very low value. If, for some reason, this water plug with the very low boron concentration begins to flow towards the core and enters the core without any major mixing on the way, a large power increase may occur.

During an SGTR event, the primary pressure might decrease to a value lower than the ruptured steam generator pressure (depending on the course of events and operator actions), thus reversing the break flow. In this case, water with low boron concentration would flow from the ruptured steam generator to the primary circuit and further to the core. The situation would be further aggravated if the feedwater isolation were to fail.

Examples of initiating events which may result in a boron dilution accident at low power or shutdown conditions are:

- Low rate boron dilutions due to a leak into the primary circuit or due to human errors.
- Fast boron dilutions caused by a rapid and massive injection of pure water into the core. This could happen, for instance, when restarting a main primary pump after a period of shutdown at low residual power and therefore with no, or low level of, natural circulation and with presence of clear water in the loops due to leakages from the connected auxiliary circuits or condensation processes in the steam generator. It should be taken into account that the natural circulation is blocked when the Residual Heat Removal System (RHR) is in operation. In such conditions, in case of total loss or shutdown of the primary pumps, homogenous conditions are not guaranteed in all the loops.

These specific accident scenarios have not been found in the TOB reports of 'small series' WWER-1000 NPPs.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

This issue was identified as a deviation from current international practice. The potential consequences of this type of event could be severe. Fast injection of pure water into the core could result in prompt criticality in some reactors with large potential damages to the first barrier (fuel cladding) in a situation where the third barrier (containment) might be open. However, with adequate emergency and operating procedures, the probability of such events is very low.

COMMENTS AND RECOMMENDATIONS:

1. Boron dilution analysis should be made to the extent as needed to understand the limits of dilution rate and amount which affect fuel integrity.

2. Since early studies of boron dilution events indicate the potential consequences could be severe, operating and emergency procedures should be written to prevent such accidents.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The TOB contains analysis of boron dilution accidents under startup, power operation, shutdown and outage conditions. The results showed that sufficient time is available to enable the operators identifying the cause and taking corrective actions.

Procedures are available for preventing of uncontrolled boron dilution during plant operation at power, shutdown, refuelling, spent fuel handling and storage. These procedures specify the status of the valves for the systems containing boric acid and those without it. When the primary pressure reaches <6 kgf/cm^2, specific actions of “Preventing water ingress into the primary circuit, cooling pond and refuelling pond without neutron absorbers during Unit 5 outage” are implemented.

Kalinin Units 1 and 2:

The TOB contains the assessment of accidents involving the decrease of boric acid in the primary during startup, operation, hot and cold shutdown conditions. The results show that sufficient time is available to enable the personnel to identify the accident and to take corrective actions.

Existing operating procedures prevent uncontrolled decrease of boric acid during operation, shutdown, refuelling, spent fuel transportation and storage. Procedures have been developed, but not yet approved, according to the EdF methodology to prevent uncontrolled decrease of boric acid in emergencies. It is necessary to investigate the applicability of systematic accident analysis performed for NNPP and Lianyungang project, including fast boron dilution by clean water slugs for KNPP.

South Ukraine Units 1 and 2:

A systematic analysis of the boron dilution accidents has not yet been planned for the Units 1 and 2.

The plant has procedures for preventing boron dilution during refuelling, transportation and storage of spent fuels. However, procedures for other shutdown conditions have not been well developed.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 12 (AA 12)

ISSUE TITLE: Anticipated transients without scram (ATWS)

ISSUE CLARIFICATION: Anticipated Transients Without Scram (ATWS) are defined as accidents initiated by anticipated transients, which are assumed to proceed without scram. If the automatic reactor trip fails during these transients, it could have adverse effects on the integrity of physical barriers in a reactor.

According to current practice, ATWS are analysed for PWRs in order to demonstrate their defence in depth capabilities to cope with these transients.

International practice considers the analysis of ATWS for a variety of initiating events such as loss of feedwater, loss of load, turbine trip, loss of condenser vacuum, loss of off-site power, closure of main steamline isolation valves, uncontrolled boron dilution, inadvertent control rod withdrawal, etc. ATWS analyses are performed in general by using best-estimate tools to determine the preventive (e.g. a diverse scram system) or mitigative measures (e.g. initiation of turbine trip and emergency feedwater supply) which need to be implemented for strengthening plants' defence in depth.

It was also recognized from operating experience in PWR plants, such as Salem Unit 2 where there was a failure to scram, that ATWS are possible.

ATWS analysis results are not available at 'small series' WWER-1000 plant, and no preventive or mitigative measures have been implemented at the operating units. This is a deviation from international practice.

The issue was identified from western operational experience and from a safety review of WWER-1000 units.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

Incomplete or lacking ATWS analysis or their unavailability at plants would make it impossible to understand which primary safety function(s) would be affected and which corrective measure need to be implemented to cope with ATWS. This issue affects Level 3 of plant's defence in depth to control of accidents within the design basis.

COMMENTS AND RECOMMENDATIONS:

1. An extensive study of ATWS accidents should start as soon as possible.

2. Based on the results of the analysis, preventive and mitigative measures for ATWS should be prepared and implemented.

REFERENCES: [2, 3, 39]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

ATWS analysis has not yet been performed for Unit 5.

Analysis of ATWS will be included in the probabilistic safety analysis Phase II within the framework of SWISRUS project.

Kalinin Units 1 and 2:

Currently, there are no specific regulatory requirements in the Russian Federation to perform ATWS analyses. For KNPP preliminary ATWS assessments were made within the framework of EOP development based on EdF methodology and the Level 1 PSA for KNPP Unit 1 ("BETA" project).

South Ukraine Units 1 and 2:

The Ukrainian regulatory guide in its Chapter 4 requires the analysis of anticipated accidents without scram (ATWS). The following initiating events should be considered:

- loss of feedwater,
- loss of off-site power,
- loss of AC power,
- turbine trip and,
- inadvertent closure of fast-acting isolation valves.

ATWS analysis is not available at SNPP Units 1 and 2.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 13 (AA 13)

ISSUE TITLE: Total loss of electrical power

ISSUE CLARIFICATION: The total loss of the electrical power supplies, which constitutes a beyond design basis accident, results in a situation where:

- residual heat cannot be removed, since the steam generator emergency feedwater systems and the steam relief valves are dependent on electrical power supplies,
- the plant systems cannot be controlled after the batteries have discharged, and
- there is potential risk of a primary break caused by rupturing the primary pump seals.

Consequently, it is necessary to determine the:

- means of recovering a power source,
- means of refilling the SG,
- means of supplying I&C parameters to enable plant systems control,
- means of maintaining the operation of BRU-A after the batteries have discharged,
- means to maintain integrity of the primary circuit and,
- means of injecting borated water into the primary circuit.

The means and procedures based on the respective accident analyses to cope with a station blackout (power supply to minimum vital instrumentation, means of supplying the SGs, means of using the BRU-A, means of replenishing feedwater tanks, etc.) are not all available at 'small series' WWER-1000 units.

MAIN SAFETY FUNCTIONS AFFECTED: Controlling the power
Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: II

JUSTIFICATION OF RANKING:

The lack of accident analyses to determine compensatory measures in the case of a station blackout affects Level 4 of plants' defence in depth. The main safety functions will be questioned for beyond DBA scenarios.

COMMENTS AND RECOMMENDATIONS:

1. An analysis should be made to have a clear understanding of timing and consequences of the total loss of electrical power and to develop compensatory measures to cope with this beyond design basis accident. The following points should be investigated:
   - means to recover electrical power for needed equipment
   - means to continue decay heat removal by the secondary side
   - means to maintain primary circuit integrity
   - means to inject borated water

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The Unit 5 has three diesel generators located in one building, but separated by walls. Total loss of electrical power is one of the ten initiating events in the list of beyond design basis accidents (BDBA) recommended by the former Soviet Union Ministry of Power in 1991 to be analysed (see issue AA 9). The analysis was done at that time. Re-evaluation of this BDBA may be considered depending on the results of Level 1 PSA (SWISRUS project). The NNPP planned to purchase a mobile diesel generator set with necessary cable connections in the framework of TACIS programme. Also included is a mobile feedwater pump supplying water to the steam generators.

A power supply connection with NNPP Units 3 and 4 may also improve the situation.

Kalinin Units 1 and 2:

The thermohydraulic calculations for EOP development using the EdF methodology show that in case of such accident (total loss of electrical power without diesel-generator actuating) the technical means available are not able to bring a unit into a safe state. In order to overcome such accident it is necessary to implement additional technical facilities ensuring core heat removal (such as feedwater pump supplying SGs in case of total loss of electric power, etc.) which should use an additional source of electric power.

At the same time the design solutions implemented at Units 1 and 2 KNPP allow to ensure electrical power supplies from diesel-generators of one unit to equipment of safety systems of another unit. Besides, construction of transmission line from “Sukharevo” substation (another grid) has been planned. This allows to ensure independent electrical power supplies for safety systems facilities of Units 1 and 2.

South Ukraine Units 1 and 2:

The Ukrainian regulatory guide in its Chapter 5 requires the analysis of total loss of electrical power as one of the beyond design basis accidents.

Such an analysis is not available either in the TOB or at the plant. The six diesel generators of SNPP Units 1 and 2, each having three diesel generators, are located in one building with six separate rooms. This layout may cause common failures by zone internal hazardous events.

REVIEW AREA/ISSUE NUMBER: Accident Analysis 14 (AA 14)

ISSUE TITLE: Total loss of heat sink

ISSUE CLARIFICATION: In the event of a loss of heat sink, e.g. the essential service water system cooling function in the case of an external event or common mode failure, the safety systems with which it is possible to control the water inventory and remove the decay heat would be unavailable.

A systematic analysis of total loss of heat sink has not yet been completed for the 'small series' WWER-1000 units.

The means and procedures to cope with a total loss of heat sink are not all available at 'small series' WWER-1000 units.

MAIN SAFETY FUNCTIONS AFFECTED: Cooling the fuel
Confining the radioactive material

RANKING OF ISSUE: I

JUSTIFICATION OF RANKING:

The lack of compensatory measures affects Level 4 of plants' defence in depth. Missing analysis does not allow for the judgement of the impact of total loss of heat sink on the main safety functions.

COMMENTS AND RECOMMENDATIONS:

An analysis should be made to have a clear understanding of timing and consequences of the loss of the service water system and to develop compensatory measures to cope with this beyond design basis accident.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

At NNPP Unit 5, heat sink for the essential service water system is the cooling pond (for trains 1 and 2) and discharge canals of circulation water of NNPP Unit 4.

Total loss of heat sink can be caused either by failure of the low pressure injection pumps or failure of essential service water or total loss of electrical power (see issue AA 13). Some of above initial events were analysed based on the list of beyond design basis accidents (BDBA) recommended by the former Soviet Union Ministry of Power in 1991 (see issue AA 9). Further analysis of total loss of heat removal was performed in the Level 1 PSA (SWISRUS project). The symptom-based emergency procedures are planned to be prepared based on the PSA results.

The NNPP planned to purchase a mobile diesel generator set with necessary cable connections in the framework of TACIS programme. Also included is a mobile feedwater pump supplying water to the steam generators.

Kalinin Units 1 and 2:

Heat from the essential service water system is removed into natural lakes Udomlia and Pes'vo which cannot be lost. Even under conditions of total destruction of dam on the river S'ezh which is solely flowing out from these lakes, the water level would not fall below permissible level of 154 m Abs which is required for normal operation of the pump facility of NPP's coolant systems. Correspondent natural level mark of the bottom of the river prevents from further decrease of water level.

Some of the initial events leading to total loss of heat sink have been analysed on the basis of the BDBA list which is recommended by the Ministry of Power Engineering of the former USSR in 1991 (see AA 9). Further analysis of total loss of heat sink will be included into the "BETA" project. The symptom-based EOPs are assumed to be worked out using results of the analyses.

According to preliminary PSA results, the probability of common cause failure of all cooling system channels is low.

The corresponding EOP is planned to be developed depending on the results of a more detailed assessment.

South Ukraine Units 1 and 2:

The Ukrainian regulatory guide in its Chapter 5 requires the analysis of LOCA with failures of high pressure and low pressure injection. The loss of low pressure injection can be caused either by the failure of essential service water which is related to heat sink or by the low pressure injection pumps or else.

The analysis of total loss of heat sink is not available either in the TOB or at the plant. At the SNPP Units 1 and 2, heat from the essential service water system (3 × 100% trains) is removed by nine air cooling towers in three groups.

4. OPERATIONAL SAFETY ISSUES

4.1. OPERATING PROCEDURES

REVIEW AREA/ISSUE NUMBER: Operating Procedures 1 (Oper. Pro. 1)

ISSUE CLARIFICATION: Guidelines on writing of operating procedures have been issued for both Russian and Ukrainian NPPs. This standard contains many desirable requirements in accordance with accepted international practice. However majority of current operating procedures as well as some new ones developed for startup, shutdown and systems operation do not comply with this common procedure. The format of procedures greatly influences their use and ultimately ensures consistent and correct operation of the equipment. The process of arrangement of current operating procedures in accordance with international practice and standard guidelines is apparently delayed.

COMMENTS AND RECOMMENDATIONS:

A programme and timescale for revising of current operating procedures and arranging them in accordance with standard guidelines has to be provided, and the programme has to be implemented as planned.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The application of the standard guidelines to upgrade current operating procedures has not yet been started at Novovoronezh Unit 5 and no schedule exists for completion of their implementation. Current operating procedures on Unit 5 were stated to contain some of the desirable features of the standard guidelines but they are not presented in a user friendly way. There is a continuous process in place to upgrade the existing procedures.

Kalinin Units 1 and 2 :

Development of the KNPP operating instructions is under way in accordance with the existing plant standards. There are programmes for plant startup and shutdown designed for step by step usage. The documents were developed at the KNPP, confining specific requirements for the transformation of the operating instructions into step by step procedures. The transformation of the operating instructions into step by step procedures is performed according to the planned documentation review (once in three years).

Currently the new guidelines are being developed at the KNPP in the framework of the Operation Quality Assurance Programme (POLAS(E)). These guidelines take into account the requirements of QA Programme.

South Ukraine Units 1 and 2 :

User-friendly procedures have been developed for unit startup and shutdown to various predetermined hot and cold shutdown states. They consist of step-by-step format and require sign-off for each step. Additionally, an alarm response procedure has been prepared and is under review. Nearly 100 new testing procedures are developed for the units 1 and 2.They encompass post maintenance tests of the safety relevant systems as well as the tests of protection system and interlocks. Under the frame of TACIS project development of new operating procedures is launched.

REVIEW AREA/ISSUE NUMBER: Operating Procedures 2 (Oper. Pro. 2)

ISSUE TITLE: Emergency operating procedures

ISSUE CLARIFICATION: In the original approach, the emergency operating procedures (EOPs) were event oriented and the list of events considered was developed in accordance with the safety analysis report including DBA. Alarm setpoints, symptoms and possible causes were included only to varying degrees in the appropriate operating procedures. The generic weak point of the event oriented approach is that the operating personnel in the control room have to first identify the event, then select a proper event in the procedure, and finally perform actions in compliance with it. In case of difficulties in identifying the event or in selecting the inadequate procedure, the actions required might not be carried out in time and the event might proceed to more serious consequences. In most cases, the event oriented procedures are "one way" oriented towards success. This approach does not take unforeseen events into account. Symptom based procedures are needed for support, especially in situations which are not absolutely clear. Besides effectively promoting the management of complicated accident scenarios, there is a need to cover as widely as possible the prevention of sequences leading to core melts.

COMMENTS AND RECOMMENDATIONS:

1. The development, validation and implementation of symptom based EOPs is supported. It is necessary to establish a schedule and to select the approach used for this work. Supporting calculations are necessary which are based on the best estimate approach. The development of separate alarm response procedures should also be considered to provide information such as: cause of alarm, alarm setpoints and corrective actions required.

2. The experience on the development of symptom based EOPs for the reference plants Balakovo and Zaporozhe in the framework of the US assistance (Lisbon initiative) should be utilised for the other WWER-1000 power plants.

3. Consideration should be given to the elaboration of the management procedures for beyond design basis accidents (BDBA).

REFERENCES: [2, 3, 40]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The current emergency operating procedures address eighteen design basis events. Guidelines exists for the ten most severe beyond design basis accidents. A probabilistic safety analysis is currently being completed. It is planned to produce symptom based emergency operating procedures which are based on the scenarios developed from this probabilistic safety analysis. The schedule is to complete these symptom based procedures by 2001. They will address beyond design basis accidents as well as design basis accidents. A multifunctional simulator will be used for validation of the new symptom based procedures, since the main simulator is not certified for validation.

Separate alarm response procedures are not currently available, but responses to most alarms were said to be embedded in the current operating procedures.

Kalinin Units 1 and 2:

Currently a set of event-based instructions for DBA and BDBA is used at the KNPP. A set of symptom based step by step emergency procedures on the basis of EdF methodology has been developed to replace the existing event-based instructions. Validation of procedures and personnel training, together with EdF, is under way.

South Ukraine Units 1 and 2:

Symptom Based Emergency Operating Procedures(EOP) are not available currently at the SNPP. The development of step by step symptom oriented EOPs are planned under the TACIS programme. They will be adapted for use at SNPP initially on Unit 3 and ultimately on Units 1 and 2. The current event based emergency operating procedures are mainly used for training and the operators are expected to deal with emergencies from memory, although they are expected to comply with the current emergency operating procedures. Development of Alarm response procedures has been prescribed in the Order 1 for 1997.

REVIEW AREA/ISSUE NUMBER: Operating Procedures 3 (Oper. Pro. 3)

ISSUE TITLE: Limits and conditions

ISSUE CLARIFICATION: At the nuclear power plants in the Russian Federation and Ukraine the limits and conditions are incorporated in the Technical Specifications (Technological Reglament).However at some plants, there is a lack of either the detailed bases (justifications) or the explanations for quantitative values determining both the physical parameters and the requirements for the operability of safety trains when checking and cross-checking the compliance with limits and conditions. They also lack the basis for quantitative values related to the actions of the operating personnel in case of violations of a limiting condition. In cases when the justifications of parameters, requirements and operator's actions are available, their presentation is not clear enough.

COMMENTS AND RECOMMENDATIONS:

The justification of safety limits and limiting conditions of operation should be developed and included in the Technical Specifications on the basis of analyses of accidents and transients, reliability analyses and operating experience.

REFERENCES: [2, 3, 41]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The limits and conditions since 1990 have been combined in one document called the "Technological Reglament for NNPP Unit 5 Safe Operation". This document was last issued in 1994. That issue reflected operating experience and modifications to safety standards. Continuous updates have been carried out since that time, as necessary, to reflect changing conditions, such as analysis of accidents and transients, reliability and operating experience. Operating documents are modified to ensure consistency with the 'reglaments'. Copies of the reglaments are supplied to operating work stations and department specialists to ensure ease of access to these documents.

Kalinin Units 1 and 2:

The new Process Procedures TRB-1000-4 were issued in 1998, in which safety limits were defined and validated on the basis of transients and accident analyses. The Procedures are being adapted to the specific design features of KNPP model 338.

South Ukraine Units 1 and 2:

The limits and conditions for Units 1 and 2 were initially compiled in one document, called the Technical Regulations for Plant Safety, in 1990. The limits and conditions for Unit 3 are currently being upgraded with the assistance of the OKB Gidropress Institute in Moscow. It is planned that the Unit 1 and 2 limits and conditions will be upgraded once the Unit 3 programme is complete. These limits and conditions only take into account normal operating conditions, but have considered operating experience. The limits and conditions are readily available to the staff who require to use them from the technical archives and they are reflected in the Technical Regulations available in the shift office. It was stated that the technical substantiation, which justifies the limits and conditions, is adequate.

4.2. MANAGEMENT

REVIEW AREA/ISSUE NUMBER: Management 1 (Man. 1)

ISSUE TITLE: Need for safety culture improvements

ISSUE CLARIFICATION: Safety culture embodies a top to bottom approach for plant operation from a safety perspective, as detailed in INSAG-4 [42]. The role of safety culture is already internationally recognised as an important element of operational safety, although the implementation of the programmes related to safety culture can be improved at many plants in the world. Many of the elements of safety culture are already in place at the WWER plants. However, systematic activity and well-defined programmes are not established at most plants in order to effectively communicate to all the plant personnel the principles of safety culture which include the role of procedure usage, a self-critical attitude, and an attitude that refuses to accept second best in accomplishing safety goals.

COMMENTS AND RECOMMENDATIONS:

1. Incorporate safety culture in training and qualification programmes for the prevention of incidents. This should include acquaintance with rules and regulations, vigilance in performance of duties and active feedback from plant operational events.

2. Develop an effective programme to detect weaknesses of the proficiency of personnel and to encourage promoting safety culture, including management vigilance to monitoring and analysing personal errors without using punishments.

3. Plants are encouraged to benefit more of the IAEA ASCOT services.

REFERENCES: [2, 3, 42]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Rosenergoatom, which is the utility management body, has issued a declaration that safety is the first priority of the organization. This is part of a policy statement made in 1995, that addresses the need to improve safety culture. Other documents have also been issued on safety culture at the corporate level including the concept of safety culture, a programme on safety culture enhancement and a directive on safety culture assessment. Novovoronezh Unit 5 is committed to introducing a comprehensive programme consistent with corporate directives. The plant manager has issued a policy statement on safety culture. Various measures have already been introduced, including incorporating safety culture into training programmes. Recognising good performance in this area is being considered for future implementation.

Kalinin NPP Units 1 and 2:

The main principles set out by the INSAG-4 document are included in the Statute of the KNPP and documentation on personnel training.

Special safety culture reports were prepared at the plant in 1997 and 1998, in which safety culture criteria were defined. To control safety culture level at the plant “Days of Safety” are held.

South Ukraine Units 1 and 2:

In 1993 plant management issued a declaration that defined the basic principles governing the way in which the plant should be operated. These principles include a statement that safety is the responsibility of the plant staff, rather than the regulator, and that safety must be an individual requirement of all plant personnel. These principles are amplified in the Standard Regulations for the plant, which include statements such as: any mistake can cause an accident; safety is a higher priority than production; reporting of errors will not be punished; personnel will be promoted and rewarded for a high level of job performance; management will support initiatives to improve safety; and maintenance should be carried out before equipment failure.

REVIEW AREA/ISSUE NUMBER: Management 2 (Man. 2)

ISSUE TITLE: Experience feedback

ISSUE CLARIFICATION: Despite evident progress in this area which has been achieved in both countries, the Russian Federation and Ukraine, there is still a need to improve the evaluation system with respect to the feedback from operating experience and compliance of national accident analysis with IAEA standards. The root cause analysis methodology is only partially applied and the communication channels for corrective actions are, in many cases, not clearly identified.

COMMENTS AND RECOMMENDATIONS:

1. The experience feedback should ensure that all the plant departments are informed of all the lessons learned from operating events.

2. Direct communication channels should be established and used efficiently between plants of similar type.

3. The procedure and the specification criteria for an accident analysis and event reporting should be revised in accordance with the IAEA NUSS documents.

REFERENCE: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Both internal and external operating experiences are taken into account by the plant. VNIIAES, the technical support centre located in Moscow, is involved in the review of external operating experience. This data consists of information related to incidents, good practices and results of plant inspections. It is received from national and international sources. Both WANO and IAEA sources are used. Important information is translated, where necessary, and sent to the plants for their consideration. A feedback form is used to determine the degree of implementation of this information by the plants. Information is also received directly by the plant from other similar units. Internal operating experience is recorded and investigated at the plant based on regulatory requirements and is reported externally, as required. Exceeding limits and conditions is one of the requirements for these investigations. Lower level events are investigated at shop level. Near miss accidents are also investigated on NNPP Unit 5. Root cause analysis is used for these investigations. Plant personnel are trained at VNIIAES in operating experience management including root cause analysis.

Kalinin Units 1 and 2:

Much attention is paid at the KNPP to study and use the internal and external operating experience. The main information sources are Concern Rosenergoatom, VNIIAES, WANO and IAEA. A system of working with information reports exists at the plant. The guidelines for handling the information and the operating experience feedback (1.2 - ON.03.03.02) were developed in the framework of the QA Programme (POKAS(E)), according to which the analysis to determine the necessity of actions to prevent similar events is carried out.

South Ukraine Units 1 and 2:

SNPP has a structured operations experience feedback programme, utilising both internal and external experience. The process is co-ordinated by the plant Chief Engineer, with the Deputy Chief Engineer, Nuclear Safety, taking prime responsibility. National regulations exist for reporting criteria to the regulatory body, and the utility has clear internal reporting criteria. National standards have been developed for incidents and accident analysis but they are not consistent with IAEA standards. The SNPP procedures for operational experience feedback specify the classes of incidents that must be investigated by an investigation commission or a departmental investigation. Investigation commissions are usually headed by the Deputy Chief Engineer. According to these procedures, the resulting report must include a root cause analysis and corrective actions, using the ASSET methodology.

At present there is no procedure for the tracking or evaluation of the effectiveness of corrective actions. However, a draft agreement has been prepared for the supply of an event actions tracking database.

REVIEW AREA/ISSUE NUMBER: Management 3 (Man. 3)

ISSUE TITLE: Quality assurance programme

ISSUE CLARIFICATION: The objective of quality assurance programmes is to support consistent and safe nuclear plant operation. In specific, the programmes should ensure verification and maintenance of high equipment quality. In case of WWER plants, such programmes were not supplied by the equipment vendors and the main designer. Therefore, the development of these programmes remains with the operating organizations including an outlay for an implementation of the programme. After the implementation stage, an independent assessment of programme effectiveness should also be considered.

The ASSET missions to the WWER-1000/320 nuclear power plants have found that the majority of failures that caused safety related events were initiated by equipment failures. These again were fairly evenly distributed between the failures of mechanical and electrical equipment and, to a slightly lower extent, of I&C equipment. The main problem in this conjunction is the inadequate quality standards of the original equipment.

COMMENTS AND RECOMMENDATIONS:

A quality assurance programme should be developed and implemented. As the starting point for this work the IAEA Safety Codes, respective Guides and Technical Reports on quality assurance are available.

REFERENCES: [2, 3, 44]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The policy of Rosenergoatom is to have a quality assurance programme applied to all elements of the power plant life cycle. The basic document of first level has been issued and it contains information on generic requirements such as information, support, monitoring and feedback. A set of 33 documents of second level have been prepared by Rosenergoatom and NNPP will prepare plant specific documents of that level based on these model documents. At present one has been issued in the maintenance area and eleven more will be issued this year. A programme to complete plant specific documents of second level will be determined by NNPP by the end of 1997. No programme has been determined to complete the full QA programme, including the documents of third level that must be prepared. Some plant personnel have been introduced to the QA management system having been used in UK.

Kalinin Units 1 and 2 :

In accordance with the Rosenergoatom policy, a QA Programme (POKAS(E)) for units in operation, as well as a QA Programme (POKAS (E)) for Unit 3 under construction were developed at the KNPP in 1994. The QA Programme (POKAS(E)) for units in operation calls for the development of new Level 2 and 3 documents (or modification of existing ones) in 14 activities. At present 28 documents have been developed, particularly on:

- Modernization,
- Operation management,
- Maintenance,
- Metrology,
- Emergency preparedness.

The documents are developed by the NPP personnel on the basis of standard papers developed by Rosenergoatom, as well as on the basis of independent experience. After the document is approved, it is validated and put into effect along with its introduction to the personnel. The effort to issue Levels 2 and 3 documents is ongoing. If necessary, the basic procedures of the lower level are developed.

South Ukraine Units 1 and 2:

The basic principle of quality assurance (QA) at the plant is that it is the responsibility of the people doing the work. QA consists of 3 steps that are: quality is assured by the performer, checked by the foreman and confirmed by the supervisor. Although, at present, there is no standard for quality assurance at the national level, all the nuclear plants in the Ukraine are co-operating in the production of a comprehensive set of QA procedures, based on ISO 9001. The implementation of the QA system at the SNPP is under way under the TACIS- 95 programme, concentrating on 5 areas that are considered vital to their organisation. They are maintenance, material procurement, operation, training and fire protection. There was stated to be good management backing of the QA programme, but resources are limited at the station and only 4 people are assigned to the core QA group.

REVIEW AREA/ISSUE NUMBER: Management 4 (Man. 4)

ISSUE TITLE: Data and document management

ISSUE CLARIFICATION: The records and data are not stored such that they are easily retrievable and can be of use. Consequently, they cannot be used for the configuration management to support different plant activities such as maintenance, surveillance tests and backfitting. At some plants, there is a particular need for the improvement of the record system related to training and qualification of staff. This deficiency can degrade human performance and result in many further deficiencies.

COMMENTS AND RECOMMENDATIONS:

1. The plants should establish a programme for improving the configuration management to ensure that the records and data are easily accessible and retrievable, preferably with the use of electronic data processing.

2. A central system should be established for all training records to ensure that all staff have received appropriate training.

3. Consideration should be given to protecting the documents more effectively against fire.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Data and document management are based on two documents. The first controls the circulation of documents and the other controls their distribution. There are multiple levels of document storage at the plant. There is a local level at the workplace where the control of documents is the work group's responsibility. The second level is divisional where originals are stored in a fire protected environment. The third level is a central plant archive that is also protected against fire. The original design documents are stored in another building off site. All documents are periodically revised, based on a schedule. Documentation was stated to provide assurance that the configuration is controlled. Plant personnel are required not to deviate from procedures.

Kalinin NPP Units 1 and 2:

Documentation and information management is similar to that at other NPPs under the umbrella of Concern Rosenergoatom. The documentation at the KNPP was updated in 1994 prior to the mission of the ASSET group to evaluate safety related events. At present, computerized DB is widely used at the plant.

South Ukraine Units 1 and 2:

All data and documents are currently stored on paper which is ageing and becoming difficult to read. Documents cannot easily be found, resulting in reports taking excessive preparation time. Documents are stored at several locations in the plant and are not always well protected against fire damage. As part of the TACIS programme, a computerised documentation management system is being provided and will be ready to start optical scanning of documents in June 1997.

Training records are kept with each individual having an individual qualification card with a backup record keeping system.

No comprehensive configuration management system appears to be in place to ensure that all plant documentation accurately reflects the status of the plant at all times and is consistent with the design configuration.

4.3. PLANT OPERATIONS

REVIEW AREA/ISSUE NUMBER: Plant Operations 1 (Plant Oper. 1)

ISSUE TITLE: Philosophy on use of procedures

ISSUE CLARIFICATION: Despite the promulgation of an adherence to strictly comply with operating procedures at the top management level the plant operating practise is largely based on operator knowledge, training and educational background. Procedures are used to varying degrees and in some cases used only for operator training (such as the EOPs). This is due in part to the fact that many current operating procedures are not written in step by step format. Even in the cases when some operating procedures are written in step-by-step format numerous individual actions refer to the other operating manuals. The current internationally accepted philosophy recognised that the highest level of safety is ensured by having highly trained and qualified operators consistently using and following well written procedures, regardless of whether operators are graduate engineers. The similar conclusions in respect of use of EOPs are based on the absence of step-by-step EOPs and the fact that EOPs are not used during events as well as the lack of separate alarm response procedures.

COMMENTS AND RECOMMENDATIONS:

1. Senior plant management should insist on implementation of strong following operating procedures in the operational practise. This has to be considered as an element of the safety culture. These requirements and their importance should be conveyed to all levels of management and the work force.

2. Each department should evaluate their existing procedures against the new requirements for the use of procedures and identify and correct deficiencies that prevent the use of procedures in accordance with the requirements. Ownership of procedures should be clearly established with the work force. A feedback process should be set up with the involvement of the 'owners' to improve procedures.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit:

The plant policy is that personnel must comply with the content of procedures unless there is a clear indication that the procedure is inappropriate for the current operating state. However, they are not required to have the procedure open in front of them and to be following it on a step by step basis. This is due in part to the fact that many current operating procedures are not written in a step by step format. Plant management is planning to change to step by step normal operating procedures. A draft standard for these procedures is currently being reviewed by plant management. As discussed in The 'Operations Procedures 2' issue, there are plans to prepare step by step symptom based emergency operating procedures in the future, once the probabilistic safety analysis is completed.

Kalinin Units 1 and 2:

There are strict plant requirements to have the operating documents up front and readily available while performing any operating actions to correlate specific actions with the documents. Most of these documents were developed in a step by step format; they should be signed by the operator after completion of each action.

South Ukraine Units 1 and 2:

Management policy requires plant personnel to comply with procedures. The normal operating procedures, surveillance test procedures and emergency operating procedures are all carried out from memory by the operating staff. This practice is accepted by management because of the high level of education and training of the control room staff and also because of the difficulty in using the current procedures due to their non user-friendly format. Startup and shutdown procedures were recently issued in May 1996 and they have a step-by-step format which requires sign-off on completion of each item. However the individual items in these plans refer to the operating manuals, which are carried out from memory. However the alarm response procedure has been prepared in draft format for Units 1 and 2, to enable the operators to easily determine the appropriate response to alarms. The renovation of operating procedures is launched under the TACIS-97 programme.

REVIEW AREA/ISSUE NUMBER: Plant Operations 2 (Plant Oper. 2)

ISSUE TITLE: Surveillance programme

ISSUE CLARIFICATION: The positive changes are noticed in the surveillance policy at the nuclear power plants under review as compared to the earlier reviews of the standard model WWERs. However there are some concerns in respect of justification of the contents of the surveillance programme, tests' frequency ,use of the tests' results to optimise the surveillance programme. The balanced surveillance programme has to ensure availability of the safety systems to perform their safety functions and to avoid early wear out as a result of over testing.

COMMENTS AND RECOMMENDATIONS:

1. Test intervals should be considered carefully in order to ensure the functional capability of equipment and to avoid unnecessary tests which could result in the decreasing of the equipment availability.

2. The organization of the complete surveillance test programme should be unified so that all surveillance tests results are recorded, verified and analysed. A complete surveillance programme will help management to maintain the required level of safety systems availability.

REFERENCES: [2, 3, 26]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A set of comprehensive surveillance tests were stated to exist which are based on regulatory requirements. Step by step procedures are used which fulfil or exceed regulatory requirements. Ageing, pressure and thermal cycling, radiation damage, equipment history and operating experience were all stated to be considered when determining tests and test frequencies. Over testing is avoided by reviewing and analysing test results and changing test frequencies as required. Overview of testing is provided by the safety laboratory which ensures independent review of test results. Test results are trended by this group so that drifting within allowable error bands can be identified.

Kalinin Units 1 and 2 :

The main control activities are performed in the shops having the appropriate equipment. Technical support groups are established in the shops; these groups annually develop control programmes and monitor programmes' implementation. Information on the data, addressed in the requirements of design documents and the Safety Analyses Reports, is collected and stored.

South Ukraine Units 1 and 2:

The surveillance programme for safety system testing consists of the following elements: scheduled testing, post maintenance testing and surveillance testing after component failure. The programme is controlled by one group at the plant, called the Equipment Adjusting Department, which ensures that these tests are co-ordinated and controlled.. The test programme is based on the requirements of the Technical Regulations. The results are recorded indicating the Technical Regulation reference, the criteria for success and the actual test results. Long term trending of the test parameters is carried out so that maintenance can be planned before a parameter drifts outside the acceptable limits. However a testing frequency is fixed and no analysis is carried out, or planned, to determine if the optimum test frequency is being used. Additionally no review is planned to determine the effects of plant ageing on the optimal test frequency.

The new automatic control system of the operational parameters, which is in the implementation at the Unit 2 will enable to check the parameters of the equipment. The appropriate database enables to discover degradation of the safety related systems to launch preventive measures.

REVIEW AREA/ISSUE NUMBER: Plant Operations 3 (Plant Oper. 3)

ISSUE TITLE: Communication system

ISSUE CLARIFICATION: The existing communication system includes different communication means .Some of them are out of date to provide sufficient effectiveness even in normal operating conditions. The effectiveness of the systems in emergency and severe accident conditions is questionable.

COMMENTS AND RECOMMENDATIONS:

1. Upgraded communication systems should be installed at all plants.

2. Procedures should be established or existing procedures reviewed to improve the communication between plant activities and control room to ensure that the operator is aware of the status of the plant.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

The current communications system at the plant consists of a combination of public address systems, telephones and radios. A paging system will be installed in the future. Both the radio and the public address systems require upgrading and possible options, including a satellite system, are being evaluated. No recent incidents at the plant have been caused by the current communication deficiencies, but plants with similar communications systems have experienced difficulties. The potential impact of the present radio system on electronic circuits in the main control room has not been evaluated.

Kalinin Units 1 and 2:

There are various types of communication systems at the plant, including a telephone system, a public address system and a radio system. The authorized personnel have a paging type communication. The communication system existing at the plant was considered acceptable.

South Ukraine Units 1 and 2:

The communication system includes dial telephones, a direct prompt system, which enables single button communication with key personnel from the control rooms, a partially effective loud speaker system and the use of mobile telephones to contact operators in areas where the loud speaker system is not effective. During emergencies, the off-site communication is carried out by the plant shift supervisor using the normal off-site telephone system. There is a site-wide computer network under consideration which would result in the ability to transmit plant data directly to the emergency control centre and to off-site centres. Although plans are in place to start this system using unit 3 data, there are no firm plans to complete this system on Units 1 and 2. There are no other plans to upgrade the communications system.

4.4. RADIATION PROTECTION

REVIEW AREA/ISSUE NUMBER: Radiation Protection 1 (Rad. Prot. 1)

ISSUE TITLE: Radiation protection and monitoring

ISSUE CLARIFICATION: Radiation protection regulations should be kept up to date with the international recommendations and the strict implementation of proper measures required by all personnel. The radiation protection practices should be directed towards minimising radiation doses in accordance with the ALARA principles according to the ICRP-60 [43] recommendations. The NPPs under review comply with these recommendations to various degrees. The administrative individual dose limit in 1996 was 30 mSv at the Novovoronezh NPP and 48 mSv at the South Ukrainian NPP. The radiation monitoring instrumentation originally designed and supplied needs upgrading to cover the whole range of parameters including accidental conditions.

COMMENTS AND RECOMMENDATIONS:

Review the radiation protection including regulations and dose limits, implementation of basic principles, personal dosimetry, instrumentation for radiation monitoring, environmental monitoring and radiation monitoring and protection during emergencies.

REFERENCES: [2, 3, 26, 43]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A document has been issued by the Russian government which was said to embody the requirements of ICRP60. The plant plans to fulfil all of it's main requirements by 2000. In 1996 the administrative individual dose limit was 30 mSv and nobody exceeded this limit. In 1997 the target is 20 mSv. NNPP does not have a specific ALARA programme, but elements of such a programme exist. Maintenance of radiation instrumentation is expensive and time consuming due to ageing of equipment, but it is possible to keep sufficient instruments in service to provide an effective radiation protection programme. At present there is no automatic remote monitoring of atmospheric releases. This is carried out by a mobile laboratory. Measures to install a remote system are being considered under the TACIS programme.

Kalinin Units 1 and 2:

Special measures are developed at the plant to comply with the requirements of the new State Radiation Protection Standard NRB-96. According to the requirements, the maximum acceptable collective dose is established. The annual individual maximum acceptable dose of 20 mSv will be approved in 1999. Due to the special status of the area around the plant, there are strict requirements on the radiation releases. Careful monitoring of radiation releases is provided. A special monitoring system ASKRO is being implemented to control the radiation situation around the plant. However, technical means available are obsolete and need to be upgraded.

South Ukraine Units 1 and 2:

SNPP has radiation control regulations which are consistent with national regulations which were last updated in 1997 (NRBU-97). The principles of ALARA were understood and were considered when writing the plant specific radiation protection instructions. However, some elements of a comprehensive ALARA programme are not in place. For example, the individual dose target for adult male workers is 48 mSv per year, essentially the same as the 50 mSv regulatory limit; there is no dose equalisation policy; there are no lifetime dose limits; there is no target collective dose for the plant; there are no mock-up facilities to train personnel on dose intensive work; and, as at many plants, there are difficulties in getting personnel to consistently follow radiation protection procedures.

Environmental monitoring is carried out by a plant laboratory located at the local town called Yuzhnoukrainsk. It has equipment for routine and emergency environmental monitoring. An area covered by a 30 km radius from the plant is routinely monitored. Air, water, ground water, precipitation, fish, milk, plants, and animals are sampled according to a schedule defined by a national standard for off-site monitoring.

4.5. TRAINING

REVIEW AREA/ISSUE NUMBER: Training 1 (Training 1)

ISSUE TITLE: Training programmes

ISSUE CLARIFICATION: It is widely recognised that the role of operators in WWER-1000 nuclear power plants is especially high in view of the comparatively low level of automation. In the last few years, considerable efforts have been made to improve the training of operating personnel by strengthening the training centres or departments and providing new equipment. The operating personnel sometimes are not aware of the importance of adherence to written procedures and instructions and may have difficulties in coping with the stress during rare or critical situations. The actual proficiency of personnel is not generally taken into account for developing of training programmes.

COMMENTS AND RECOMMENDATIONS:

1. Experienced staff should be identified and trained as instructors so that the training department can provide the right level of professional support to line managers and ensure that there are adequate numbers of suitably qualified and experienced personnel.

2. Enhance the consciousness of personnel to perform all tasks in adherence to written procedures or instructions, including provisional instructions, and make plant personnel aware of the consequences of violating procedures.

3. Train operating personnel, particularly shift supervisors, to handle stress during rare or critical situations, e.g. rapid plant transient or to take corrective actions following an error.

4. Event sequences dominating the risk and their prevention should be included into the training programme of operating personnel.

5. Provide more comprehensive training and qualification programmes to ensure the proficiency of plant management and shift supervisors in monitoring the proficiency of personnel and in assessing weaknesses.

6. Continue monitoring of the personnel proficiency and modify the training programmes periodically from the feedback of monitored training results.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

Training programmes are scheduled on yearly basis. Attendance is mandatory. Refresher training is included. A comprehensive maintenance training programme was initiated two years ago. Management development training is provided. The plant has no difficulty filling training posts with suitable candidates.

Kalinin Units 1 and 2 :

The KNPP has various training equipment which includes functional analytical simulator and computer based training systems. Unit 3 full-scope simulator development and commissioning is complete, while a full scope simulator for Units 1 and 2 is to be completed shortly. A system approach to training (SAT) is accepted at the plant.

Besides the training centre instructors, highly skilled plant technical staff and experts from branch research institutes and higher schools are involved in plant personnel training. A wide scope of training materials are being developed by the plant's specialized institutions and western experts (Siemens, EdF, TACIS projects, etc.). To provide effective training a new training centre building was commissioned in December 1998. Plant personnel also take up various specialized training and retraining courses (once in three years). Currently the KNPP personnel undergo training at the full scope simulator and imitators at the Novovoronezh Training Centre.

South Ukraine Units 1 and 2 :

There are lots of training facilities available at SNPP. They include different compact simulators, computerized training systems. However majority of them are oriented to the Unit 3. The development of full scope simulator for Unit 1 was launched in 1997 in co-operation with GSE Systems (USA) and financed by DOE. The project is in progress and the commissioning of the simulator is scheduled for May 2001.

A national concept of personnel training and qualification is being developed in the Ukraine with the assistance of western experts. The programmes are continuing to be developed at SNPP which are consistent with the national concept. The training centre's programmes include both initial and continuing training and the training needs of operators, maintainers and supervisory management staff are addressed.

Experienced staff have been recruited as instructors and are trained in instructing techniques. It was stated that there is no difficulty in recruiting sufficient qualified staff.

Continuous training of maintenance personnel is carried out, at the rate of about 8 courses a year, and initial training consists mainly of on the job training, enhanced by some basic training courses in radiation and industrial safety. However only very limited facilities currently exist on site for mock-up training of maintenance staff..

Training is provided for supervisors and managers on a continuing basis and includes such subjects as safety culture, management techniques, leadership, accident analysis and nuclear and radiation safety. These training sessions take place on a monthly basis.

Feedback is sought from supervisors following training sessions to ensure that training is appropriate and training is modified as necessary.

4.6. EMERGENCY PLANNING

REVIEW AREA/ISSUE NUMBER: Emergency Planning 1 (Emerg. Plan 1)

ISSUE TITLE: Emergency preparedness

ISSUE CLARIFICATION: On-site and off-site emergency response centres are existing or are under final stage of construction at the majority of the NPPs. The centres should be adequately equipped with data transfer equipment and reliable communications. Relevant data on all units at a site should be available in these centres, including technical documentation and on-line information on safety related physical parameters and equipment status. There are advanced plans in respect of equipment of these centres with adequate hardware and software. However, the emergency centres have not been uniformly organised and, practical realisation of these plans are encountering essential difficulties .

COMMENTS AND RECOMMENDATIONS:

1. Accomplish construction of the emergency centres, equip them with appropriate procedures and documentation, and carry out the necessary drills and exercises.

REFERENCES: [2, 3]

PLANT SPECIFIC STATUS:

Novovoronezh Unit 5:

A programme is in place to establish a series of emergency centres to replace or complement those already in existence. A new emergency centre is being constructed on site with a construction completion date in June 1997. An emergency management centre already exists in the town and there are plans to build another centre outside Novovoronezh. A utility level emergency management centre is also being created in Moscow.

Kalinin Units 1 and 2:

The local emergency centre has been established at the plant. It is equipped with both a Safety Parameter Displaying System and a Radiation Situation Prediction System. This emergency centre is linked to the external emergency centre in Rosenergoatom, Moscow, where a generic Kalinin Unit 2 simulator is installed to simulate potential emergency situations.

South Ukraine Units 1 and 2:

An on-site emergency response centre exists inside the security fence. The main off-site centre is at Kiev and was said to be a well equipped facility. Training of the management staff and tests of the communication system are carried out on a weekly basis. A project is in progress which may ultimately supply this centre with access to a site-wide computerised data collection and distribution system. It is also planned that this information will be supplied to off-site emergency control centres both locally and at Kiev and ultimately linked to an international data system. At present the whole scheme is in the concept stage and it is unclear how much of it will be completed and what the time scale is for completion.

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ABBREVIATIONS

AC	alternating current
AF	auxiliary feedwater
AFWS	auxiliary feedwater system
AGNES	Advanced General and New Evaluation of Safety of Paks NPP
ASCOT	Assessment of Safety Culture in Operation Teams (IAEA)
ASSET	Assessment of Safety Significant Events Team
ATWS	anticipated transient without scram
BDBA	beyond design basis accident
BRU-A	steam dump valve to the atmosphere
BRU-K	steam dump valve to turbine condenser
BRU-TK	steam dump valve to technological (process) condenser
BWST	borated water storage tank
CCF	common cause failure
CDF	core damage frequency
CIS	Commonwealth of Independent States
CR	control rod
CRD	control rod drive
CRDM	control rod drive mechanism
CVCS	chemical and volume control system
DB	design basis
DBA	design basis accident
DC	direct current
DEGB	double ended guillotine break
DG	diesel generator
DHRS	decay heat removal system
DNB	departure from nucleate boiling
EC	European Commission
ECCS	emergency core cooling system
ECR	emergency control room
EFW	emergency feedwater
EFWS	emergency feedwater system
EOP	emergency operational procedure
ESF	engineered safety features
ESFAS	emergency safety feature actuation system
ESWS	essential service water system
EFWS	emergency feedwater system

FWS	feedwater system
GAN	Russian Nuclear Regulatory Authority
GANU	Nuclear Regulatory Authority of Ukraine
GRS	Gesellschaft für Anlagen und Reaktorsicherheit mbH
HPIS	high pressure injection system
ICCS	intermediate component cooling system
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
IFBA	in-fuel burnable absorber
INSAG	International Nuclear Safety Advisory Group (IAEA)
IPERS	International Peer Review Service (IAEA)
IPSN	Institut de Protection et de Sûreté Nucléaire
ISI	in-service inspection
ISP	International Standards Problem
LBB	leak before break
LB LOCA	large break LOCA
LCC	local crisis centre
LOCA	loss of coolant accident
LOFA	loss of flow accident
LPI	low pressure injection
LPIS	low pressure injection system
LPS	low power and shutdown conditions
LTSIP	long term safety improvement programme
MCP	main coolant pump
MCR	main control room
MFWC	main feedwater collector
MFWS	main feedwater system
MIV	main isolation valve
MOV	motor operated valve
MSC	main steam collector
MSIV	main steam isolation valve
MSK	Medvedev Sponheuer Karnik (scale of seismic intensity)
MSLB	main steamline break
MSLIV	main steamline isolation valve
NDT	non-destructive testing
NPP	nuclear power plant
NUSS	Nuclear Safety Standards of the IAEA

OBE	Operation basis earthquake
OECD	Organisation for Economic Co-operation and Development
OL&C	operational limits and conditions
OSART	Operational Safety Review Teams (IAEA)
PC	personal computer
PISC	Programme for inspection of steel components
PORV	power operated relief valve
PRISE	primary to secondary system leakage
PRZ	pressurizer
PSA	probabilistic safety analysis
PSAR	preliminary safety analysis report
PTS	pressurized thermal shock
PWR	pressurized water reactor
QA	quality assurance
RCP	reactor coolant pump
RCS	reactor coolant system
R&D	research and development
RHR	residual heat removal
RIA	reactivity initiated accident
RPS	reactor protection system
RPV	reactor pressure vessel
RTS	reactor trip system
SAR	safety analysis report
SB LOCA	small break LOCA
SC	safety class
SDHR	secondary decay heat removal
SDHRS	secondary decay heat removal system
SG	steam generator
SGTR	steam generator tube rupture
SPAES	Russian regulations determining radiation limits for NPPs
SPDS	safety parameter display system
SPND	self-powered neutron detector
SRM	safety review mission
SV	safety valve
SWS	service water system
TECHSPECS	technical specifications
TOB	Russian equivalent of PSAR

TMI	Three Mile Island
TC	Technical Co-operation (IAEA)
USNRC	US Nuclear Regulatory Commission
WANO	World Association of Nuclear Operators
WWER	Water moderated, water cooled energy reactor

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