NATIONALNA ELEKTRICHESKA KOMPANIA

KOZLODUY NUCLEAR POWER PLANT

UNITS 1-4

Status of Safety Assessment Activities

Revision II

1999

SUMMARY

This paper presents the results of the status of safety assessment activities carried out by the Kozloduy Nuclear Power Plant (KNPP) in order to evaluate the current status of the safety of its reactor units 1-4.

The steam supply system of this units is based of the reactor WWER-440/ B-230, which is a PWR of Russian design developed according to the safety standards in force in USSR in late 60-s. Now a days 10 reactor units of this type are in operation in four NPPs. Despite of efforts of the different plants to implement safety improvements measures during first 10-15 years of operation of this type of reactor its major safety problems were not eliminated and were a subject of international concern..

The systematic evaluation of the deficiencies of the original design of this type of reactors have been initiated by IAEA in the beginning of 1990 and brought to developing a comprehensive list of safety problems which required urgent implementation of safety measures in all plants.

To solve this problems in 1991 KNPP initiated implementation of so called "short term" safety improvement program, developed with the help of WANO under agreement with Bulgarian Nuclear Safety Authority (BNSA) and consortium RISKAUDIT. The program was based on a stage approach and was foreseen to be implemented by tree stages in very tight time schedule in order to achieve significant and rapid improvements of the level of safety in operation of the units.

The Short Term Program was implemented between the years 1991 and 1997 thanks of the strong safety commitment of NEK and KNPP staff and the broad international cooperation and financial support. Important part of resources were supplied under PHARE program of CEC, EBRD grant agreement and EDF support.

In parallel a special assessment process started in 1995 in order to evaluate the level of the safety, achieved by Short Term Program, according to current safety standards and to define the measures, which should be implemented by the Utility to complete the process of improving the safety in future operation of the units.

The plant current safety level analysis has been performed using IAEA analytical methodology according to 50-SG-O12 standard "Periodic safety review of operational nuclear power plants".The approach and criteria for acceptable safety level definition, developed by IAEA and presented in INSAG-8 "A common basis for judging the safety of nuclear power plants built to earlier standards", Vienna, 1995, have been used for analysis performance.

On the basis of this analysis a set of activities has been developed, which ensures further plant operation with the necessary safety level. One of them is updating of Safety Assessment Report of the plant and this activity is briefly presented too.

The process of safety evaluation and the results were a subject of international assessment. In parallel two studies were performed during 1998, one by Siemens and the second one by EdF. Both studies gave a positive evaluation of the approach and methodology used. In addition some extension of the safety substantiation basis and of the scope of the measures were proposed which are considered by the utility in the framework of the ongoing updating of the program.

1. GENERAL DESCRIPTION OF KOZLODUY NUCLEAR POWER PLANT

Kozloduy NPP construction program consists of three stages :

1. Two VVER 440 MW units with V-230 type reactors were commissioned respectively in July 1974 and August 1975;
2. Two VVER 440 MW units with V-230 type reactors were commissioned respectively in December 1980 and January 1982;
3. Two VVER 1000 MW units with V-320 type reactors. The first one was commissioned in September 1988 and the second one in December 1993.

All units are PWR of Russian design and manufacture. Kozloduy NPP is an independent branch within the National Electricity Company of Bulgaria.

The main operational data for units 1-4 are presented in table bellow.

Parameter	Unit 1	Unit 2	Unit 3	Unit 4
Beginning of construction	April, 1970	April, 1970	October, 1973	October 1973
First criticality	30.06.1974	22.08.1975	04.12.1980	25.04.1982
Connection to the national grid	14.07.1974	24.08.1975	16.12.1980	17.05.1982
Turbines start up: - first turbine - second turbine	 30.07.1974 02.08.1974	 24.08.1975 26.09.1975	 17.12.1980 11.01.1981	 17.05.1982 23.05.1982
Full power	25.10.1974	07.11.1975	27.01.1981	17.06.1982
Number of effective fuel cycles, including 1998 year	20	20	15	14
Full power days	5814,76	6076,00	4687,64	4493,31
Capacity factor	61,89%	64,27%	53,09%	48,47%
Energy generated until end of 1996 (kWh)	57 748 654 517	59 587 649 849	48 088 355 754	45 396 443 296

The operational experience of WWER-440 Units at Kozloduy site is more than 81 reactor years till now.

2. MAIN INFORMATION ON KNPP UNITS DESIGN

2.1. General description

WWER-440 reactors or so called water-water power reactors are an example of one of the first types of nuclear power reactors, designed in former USSR and commissioned for the first time at Novovoronej NPP, in 1964. Currently 27 Units with WWER-440 type reactors are in operation and 10 of these Units are equipped with B-230 reactor type and 17 with B-213 type.

The following positive features are common for all of these Units:

- good reactor core design with moderate power density;
- great margins of the reactor core thermal-technical reliability;
- casing fuel assemblies;
- control rods neutron absorber type;
- control rods assemblies drives rack type;
- stable extent distribution of heat removal;
- negative reactivity factors for all operational modes;
- high reactor reliability concerning possible deviations from the nominal ones of the main operational parameters;
- coolant stable natural circulation with possibility for heat removal of up to 9% of the reactor thermal power in case of accidents;
- high degree of core heat removal redundancy due to existence of six main circulation loops;
- boron control and regulation;
- absence of reactor vessel nozzles lower than the coolant inlet nozzles;
- large amount of water above the core (the distance between the inlet nozzles axis and the top of the core is 1470 mm);
- horizontal steam generators with large amount of secondary side water with possibility for removal of the residual heat from the primary side for more than 4-5 hours in case of station total black out, when both normal and auxiliary feedwater to the SGs are lost;

- possibility for long term operation at lower power levels with less than six loops in operation (up to 3) without reactor trip until a planned outage;
- application of stainless steel materials for reactor coolant pumps, main circulation pipelines, SGs' headers, valves and pipelines of the primary pressure control system which allows to prove Leak Before Brake conception application for this Units.

Some of these positive features were used as a basis for development of a new generation design for nuclear power plants with so called inherent safety.

2.2 B-230 main design safety principles

The design of B-230 type reactors was developed at late 60-s and was based on the conventional standards, codes, rules and regulations valid at that time in former USSR.
The main safety consideration at that time was based on the assumption that it was possible to exclude from design basis accidents list ruptures of primary circuit pipelines with large diameters if there was good in-service inspection control of the main pipelines and if these pipes had been produced to the highest quality standards. On the basis of this assumption it was decided to consider as a maximum design basis accident (DBA) a LOCA(Loss Of Coolant Accident) with 32 mm diameter.

Rupture of pipelines with larger diameters, as well as reactor pressure vessel rupture, main coolant pump body rupture or any other large equipment ruptures were considered as very low probable (or hypothetical).

The localization system was also designed taking into account the assumed DBA. In accordance to the design in case of design basis accidents all generated steam-air mixture will be localized in the SGs and MCPs hermetic compartment and in case of beyond DBA the mixture will be released to the atmosphere through a specially designed for this purpose relief valves.

The performed analyses showed that in case of steam-air mixture release to the atmosphere the values of the radiation doses and concentration of radioactive materials will not exceed the allowed values in case of accidents for the public.

The principle of non-possible instantaneous rupture of materials was taken in consideration in the design process for high pressure fluid systems and equipment. Non-brittle austenite stainless steel was used for production of the primary circuit pipelines. Reactor pressure vessels were produced using high tough chromium-molibdenum steel with high radiation and thermal resistance for all operational modes.

3. IMPROVEMENTS OF THE SAFETY OF THE UNITS IN THE FRAMEWORK OF SHORT TERM PROGRAM

It is obvious that the first four units of Kozloduy NPP that are in operation for more than 15 years belong to the plants built to earlier standards which, according to the current safety practice need to be checked for compliance with the recent safety standards in order to eliminate the old design deficiencies.

A systematic analysis of units compliance with the current safety requirements and internationally accepted standards and practices started in 1990 and have been initiated by IAEA's SRM and ASSET expert missions. The IAEA teams evaluated the status of the plant based on modern standards and made a lot of recommendations for the safety improvement of units 1 to 4. In response to these missions, the Council of Ministers took decision to ensure the implementation of a special Program for safety improvement of units 1-4. Through an open discussion with regulators the plant adapted a stage-by-stage approach for technical measures implementation during prolonged outages.

The process of plant safety evaluation and qualifying the adequacy of the proposed improvements has been a subject of international concern. This was the basis for the Commission of European Community (CEC) with the agreement of the Bulgarian Government to organize the following:

* Technical and financial assistance for Kozloduy NPP;
* BNSA assistance by the establishment of a Consortium from expert institutes and regulatory authorities from member countries of the European Community. The participating members of the Consortium are as follows:
  - AIB - Vincotte Nuclear (AVN), Belgium;
  - Gesellschaft Anlage und Reaktorsicherheit (GRS), Germany;
  - Institute de Protection et de Surete Nucleaire (IPSN), France
  - UK Atomic Energy Authority (AEA Technology), R&D

A "Short-term program" is developed including implementation of the safety measures for the first three stages. The program was based mainly on experts evaluations and covered proposals and recommendations received by IAEA, WANO and the plant itself. Measures were reviewed in depth by Bulgarian Nuclear Safety Authority (BNSA) together above mentioned consortium of regulatory bodies named RISKAUDIT.

The main goal of the Program is stage by stage upgrading of reliability and safety of units 1 to 4 operation by means of:

- providing the functional operability of unit systems and equipment especially to those important for the nuclear and radiation safety.
- upgrading of reliability and stability of the safety systems in compliance with the single failure and common cause failure criteria. (including human errors)
- increasing the reliability of the three main barriers containing the radioactive material (fuel cladding, confinement, primary circuit.).

Special attention was paid on the operational aspects of the program and on the improvement of the operational and safety culture.

The program was implemented successfully between 1991 and 1997 in the framework of prolonged units outages. Detailed information for the results of implementation of the measures is presented on this forum in a separate paper.

4. ASSESSMENT OF UNITS SAFETY LEVEL AFTER IMPLEMENTATION OF SHORT TERM PROGRAM

4.1. Definition of safety level criteria and methodology for units safety assessment

In order to define the principles and approach for assessment and implementation for safety upgrading measures after Short Term Program, in March'95 the plant presented to the BNSA a "Concept for reconstruction of Kozloduy NPP units 1÷4", directed to guarantee units 1-4 safe operation until by implementation of technical measures to comply with current regulatory requirements and safety standards. A set of requirements and criteria for original design improvement were defined:

- Increasing of the DBA to Dn 100 mm LOCA;
- Implementation of measures to ensure such a reliability of primary pipelines operation, which will allow to consider Dn>100 mm LOCA as hypothetical in order to support limiting of DBA to the Dn 100 mm LOCA;
- Improvement of the Safety systems reliability in order to meet the BNSA safety level criteria for core damage probability frequency.
- Improvement of the Localization systems and accident management measures.

In respond to this, in compliance with the current international practices, BNSA determined the following safety level criteria, that should be achieved by the measures implemented :

- core damage probability lower than 10^{-4} for reactor per year;
- accident management measurers and localizing means must allow reducing by an order the probability of significant radioactive release outside the site.

Basis for assessment of the current level of safety of the units against this criteria was found in the methodology and procedure for safety evaluation of operating NPPs developped by IAEA and presented in the safety standard INSAG-8 "A Common Basis for Judging the Safety of NPPs Built to Earlier Standards" and safety guide "Periodic Safety Review of Operational Nuclear Power Plants 50-SG-O12".

According to the above mentioned methodology the following specific studies had to be performed to evaluate the current safety level of the units :

- deterministic safety assessment;
- probabilistic safety analysis level 1;
- analysis of the operational experience of the plant.

4.2. Safety assessment analyses fulfillment

The process of safety evaluation was integrated to the efforts of the plant to developed a new safety upgrading program to continue and compliment the activities, performed in the framework of Short Term Program.

A joint team of leading Bulgarian and Russian designers institutes and organizations, together with KNPP was established to performed assessment within a period of two years. The work was carried out under specially developed QA program and after receiving licensee for the specific tasks by BNSA. All together more then 450 men-months of high quality experts time was spend for assessment.

Analysis was carried out in tree stages and in parallel a program for elimination of the problems defined was drafted as follows:

First stage - deterministic safety analysis

In the scope of the first stage an analysis of the deviations of the original plant design from the current (Bulgarian , Russian) safety standards requirements as well as from IAEA recommendations was performed. The extent to which these deviations were covered and eliminated by short term safety upgrading measures was also judged.

The safety deficits left was determined and measures were proposed for solving each of the problems or to meet acceptable safety level if it was not possible to eliminate the deviation completely.

Second stage - probabilistic safety analysis level 1

In the scope of the second stage a probabilistic safety assessment level 1 was performed for Kozloduy NPP Units 1-4 . All design internal initiators (including LOCA with Dn 100) were analyzed in the PSA project and the corespondent core damage frequencies were calculated taking into account all the measures implemented during short term program.

Third stage - final draft of new Program for Safety Upgrading of the Units 1-4

In the scope of this stage a final list of program measures was developed on the basis of the results obtained from deterministic and probabilistic safety assessment or derived from plant experience, and based on general cost-benefit type analyses also.

5. DETERMINISTIC SAFETY ANALYSIS

5.1. General presentation

In the scope of the deterministic safety assessment an analysis of the deviations of the original plant design from the current (Bulgarian , Russian) safety standards requirements as well as from IAEA recommendations was performed. The extent to which these deviations were covered and eliminated by short term safety upgrading measures was also judged.

5.2. Initial Basis

The following was used as a basis for the assessment for this analysis:

- IAEA approaches for analysis of current operational safety in accordance with the document 50-SG-O12 "Periodic Safety Review of Operational Nuclear Power Plants, Safety Series, IAEA, 1994";
- methodology for safety analysis of NPP built to old standards, in accordance with INSAG-8 and elements of IAEA draft guideline "Safety Evaluation of Operating Nuclear Power Plants, Built to Earlier Standards - A Common Basis for Judgment" - CB-5/rev2;
- IAEA recommendations on possible solutions of problems described in the report "Consultant Meeting of Safety Improvements to WWER-440/230 NPPs - WWER SC-107";
- Units technical condition which will be achieved as a result of accomplishment of short-term three-stage program and principles described in "Concept for reconstruction of Units 1-4";
- other countries' experience for similar problems resolution;
- regulatory documentation valid for Kozloduy NPP;
- general regulatory documentation for NPP in Russia;
- findings of international missions performed at Units 1-4 Kozloduy NPP;
- TECDOC-640 - basic document, including non-compliance between Units with WWER-440/230 design and requirements of documents NUSS series;
- IAEA documents TECDOC, SC, SG, RD series, developed for reactors WWER-440/230 assessment (TECDOC-659, RD-072, TECDOC-710, SG-088, SC-099, SC-082, SC-094, WWER-EBP-01, -02 and others).

The complete list of standards and other documents to be used for assessment of units safety in the phase of deterministic analysis has been agreed by BNSA. The list is presented in Appendix 1.

5.3. General Description of methodology

The methodology used has been based on the approach defined by IAEA document 50-SG-012. In this document, the recommended methods for problems resuming and activities for their resolving are presented. This approach leads to increased effectiveness of particular activities and cost minimization by using results of activities accomplished in another areas.

A system of 25 classification categories developed on the basis of safety functions is used for resuming of regulations and IAEA recommendations. The set of safety functions is developed on the basis of the last IAEA document "WWER-SC-107", in which items related to Units type WWER-440/230 safety have been treated.

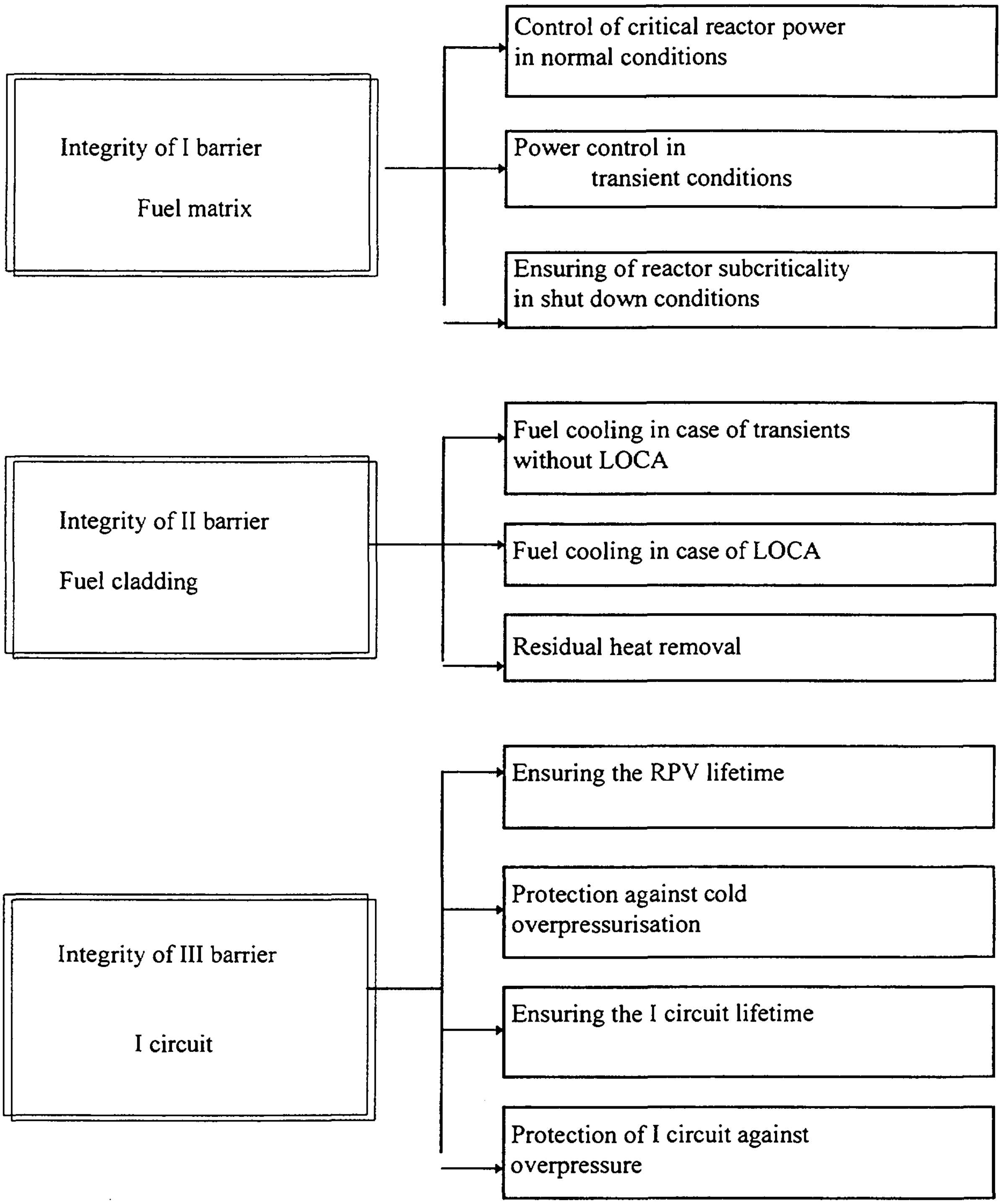
The system for categorization incorporates ensuring of protective barriers integrity which prevents radioactive products and radiation release to the environment:

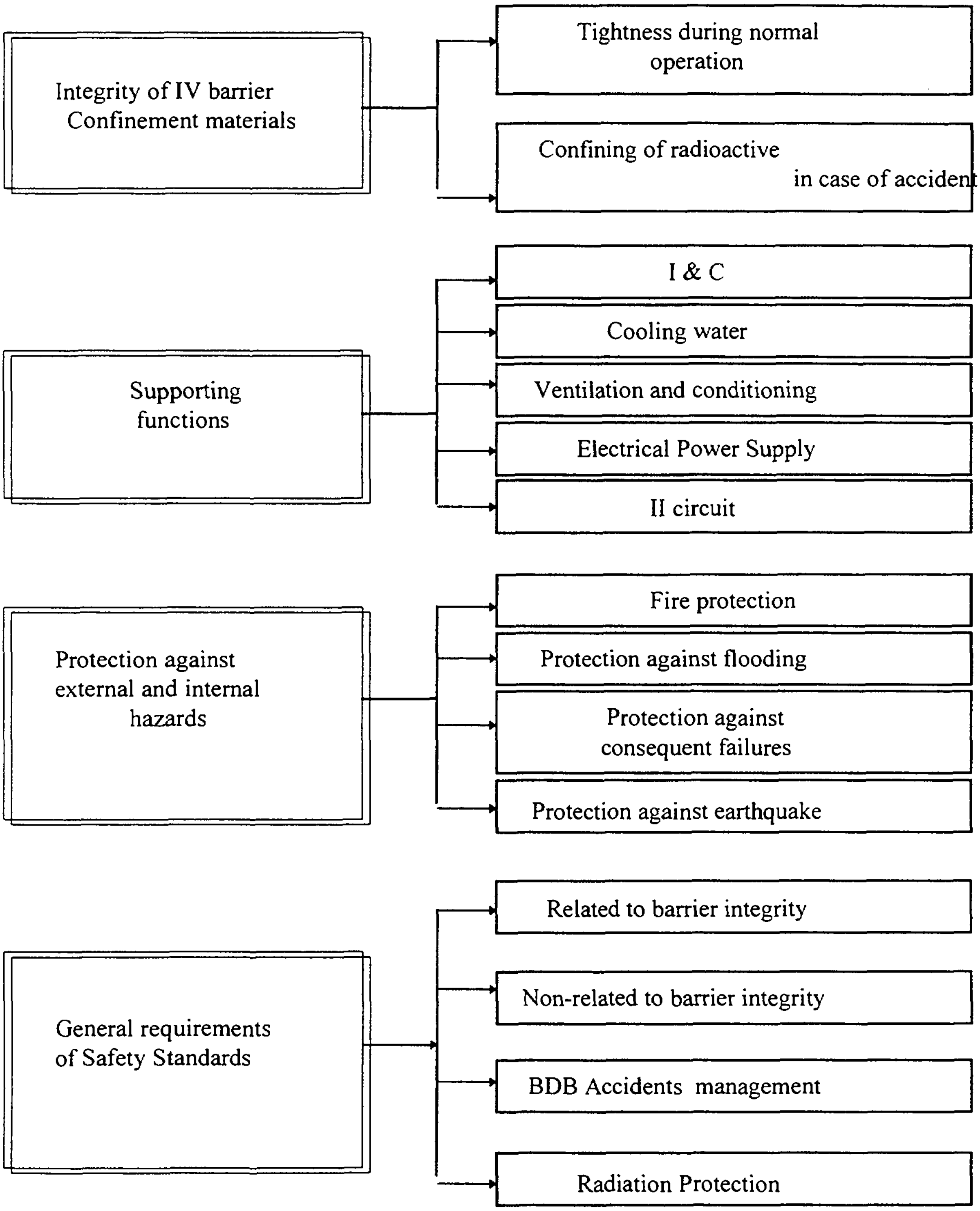
I protective barrier: fuel matrix;
II protective barrier: fuel rod wall;
III protective barrier: primary circuit boundary;
IV protective barrier: confinement boundary.

The barriers integrity has to be ensured and the following requirements have to be met:

- special requirements for necessary operation conditions (with the purpose of not exceeding the limits for particular barrier damage) specific for each barrier;
- general requirements for safety operation (violation of which can lead to facility safety decreasing), not specific for the correspondent barrier.

The list of classified categories is presented in the following figures:


Integrity of I barrier
Fuel matrix
Control of critical reactor power in normal conditions
Power control in transient conditions
Ensuring of reactor subcriticality in shut down conditions
Integrity of II barrier
Fuel cladding
Fuel cooling in case of transients without LOCA
Fuel cooling in case of LOCA
Residual heat removal
Integrity of III barrier
I circuit
Ensuring the RPV lifetime
Protection against cold overpressurisation
Ensuring the I circuit lifetime
Protection of I circuit against overpressure


Integrity of IV barrier
Confinement materials
Tightness during normal operation
Confining of radioactive in case of accident
Supporting functions
I & C
Cooling water
Ventilation and conditioning
Electrical Power Supply
II circuit
Protection against external and internal hazards
Fire protection
Protection against flooding
Protection against consequent failures
Protection against earthquake
General requirements of Safety Standards
Related to barrier integrity
Non-related to barrier integrity
BDB Accidents management
Radiation Protection

The following steps have been provided by the deterministic analysis methodology:

I. Selection of actual regulations and IAEA recommendations on safety improvement to be considered at NPP safety assessments and coordination of these documents list with BNSA.

II. Identification of basic design deviations from regulations' requirements and from IAEA recommendations and preparation of a list of deviations.

III. Classification of each deviation in accordance with the accepted classification system and grouping them by categories.

IV. Identification of non-resolved deviations for each Unit by comparison (in the frame of corresponding category) of each discrepancy with the activities already performed at NPP Units or scheduled for accomplishment at the end of three-stage program in 1997.

V. Evaluation of influence of each non-resolved deviation of real Unit on safety.

VI. Identification of residual safety deficiency for each category and formulation of safety problems.

VII. Evaluation of influence of each residual safety deficiency on safety.

VIII. Development of list of activities to resolve the problems identified.

IX. Activities description with technical requirements for implementation, effectiveness assessment, and cost estimation.

The methodology for assessment was agreed by BNSA.

5.4. Analysis results

All foreseen steps of the analysis were fulfilled and the results of conclusions for meeting the specific safety requirements are presented in Appendix 2 divided by the safety categories.

Each category has been analyzed taking into account already accomplished activities and, as a result, the residual safety deficiency has been found. It has been found that approximately 70% of deviations from current safety standards requirements and IAEA recommendations have been resolved completely or effectively (without residual deficiency) by activities accomplished in the frame of short term program for Units safety improvement. Therefore, some of the safety categories have not got a residual deficiency.

A new set of safety problems have been formulated. They have to be resolved to achieve complete accordance with the current requirements, taking into account principles accepted in "Concepts for safety operation of Kozloduy NPP". For each problem, the degree of its resolution by accomplished activities of short term program has been identified and also the essence of its residual deficiency.

For evaluation of current safety level achieved after the accomplishment of short-term three-stage program, the importance of problems for Units safety operation has been analyzed by experts. No problems need immediate improving measures.

The available information about possible solutions has been analyzed and activities for each problem resolution have been developed. The analysis has shown that, after activities accomplishment, it will be achieved a degree of accordance with safety requirements, that will allow to operate Units to the end of the life time of main equipment without additional restrictions.

It is important to point out some generic conclusions regarding the efficiency of the broad international process of discussions of KNPP units safety during the Short Term Program development :

- never the less that the "short term" program was developed mainly on the basis on experts judgments, IAEA and WANO recommendations and proposals it's implementation eliminated more then 70% of the deviations of the original design of the units from current safety standards;

- all very important safety issues were identified by experts opinions and received proper attention within the development and implementation of the short term program. As a result during detailed deterministic safety assessment no safety problems of highest priority were identified.

6. PROBABILISTIC SAFETY ANALYSIS

6.1. Background

Probabilistic safety analysis (PSA) Level 1 for Units 1-4 at NPP "Kozloduy" has been performed for evaluation of the total core damage frequency (CDF) of the reactor and the quantitative impact from propagation of the internal initiating events during operation on nominal power.

The development of the models for the accident frequencies and their quantitative evaluation were based on the technical documentation (drawings of the technological, electrical and control systems) and operational documentation (emergency operating instructions, normal operating instructions, Technical specifications), representing the current status of the Units during the PSA analyses.

Operational information from Kozloduy NPP has been mainly used as input data for the initiating event frequencies and component reliability data. If such data were missing, the data from the Russian NPP with WWER-440 as well as the generic data from IAEA and other published documents for PSA for reactors type WWER or PWR, have been used.
PSA Level 1 for Units 3&4 at NPP "Kozloduy" has been performed taking into account seismic influences and fires in the Unit's premises.

6.2.General description of the PSA methodology

The following documents have been used as a methodological basis for PSA analysis:

1) IAEA guidelines"Procedures for Conducting Probabilistic Safety Assessment of Nuclear Power Plants (Level 1), Safety Series N 50-P-4, IAEA, 1992".
2) Guides on human reliability analysis:

   - Human Reliability Analysis Procedure, 82106-AT-EA-RE-0001, Empresarios Agrupados, 1996.
   - NUREG/CR-1278 Handbook for Human Reliability Analysis, A. P. Swain, Guttman, 1983.
   - EPRI-NP-3583 Systematic Human Action Reliability Procedure .SHARP, Hannaman and Spurgin, 1984.

- NUREG/CR-3010, Post event Human Decision Errors, Operator Action Tree/Time Reliability Correction.
- A model of assessing Human Cognitive Reliability in PRA studies - G. W. Hannaman, A.J. Spurgin, Y. Lukic.

Development of the event trees, fault trees and quantitative assessment of the accident sequence frequencies have been performed using the computer codes IRRAS and Risk Spectrum - basic codes used for PSA in European countries and in USA. In parallel with PSA execution, the CDF assessment using both computer codes for verification and comparative analysis has been conducted.

6.3. PSA Main Results

During the PSA execution, the assessments of summarized CDF significance for all accident sequences for the current status of the Units were performed .This status corresponded to the Units equipment and operational documentation conditions.

PSA Level 1 studies for Units 1,2 and 3,4 have been performed for the main groups of internal initiating events during operation at nominal power including small and medium LOCA from the primary circuit inside of hermetic premises, leakage from the primary to the secondary circuit, loss of off-site power, loss of heat removal through the both turbines condensers, loss of main feed water, loss of service water, leakage from the feed water pipes, leakage from the steam lines in the isolated and non-isolated from the steam generators sections and events causing reactor trip without actuation of other safety systems. Fires and seismic initiators for Units3,4 PSA were included too.

Evaluated values for the total CDF per reactor for Units 1,2 and for Units 3,4 for analyzed internal initial events are very close to the recommended value for CDF (< 1.0*10-4 1/year) defined by BNSA as a safety operation criteria for Units 1- 4.

Main contributors in the total CDF for Units 1,2 were found to be the initiators with primary circuit LOCA, larger from primary to secondary circuit LOCA until 100mm including SG collector cover extraction, loss of off-site power, loss of heat removal through the condensers, main steam lines in non-isolated from SG section leakage, loss of main feed water and reactor trip.

Main contributors in the total CDF for Units 3,4 were found to be the initiators with loss of off-site power, steam generator collector leakage, steam collector rupture in the non-isolated from SG section, leakage from primary circuit and main steam header rupture.

The results obtained were judged as very positive based on the following factors:

- basic design characteristics providing high reliability for the heat removal from the reactor core during initiators with loss of coolant and transients violating the normal operation;
- big water inventory in the steam generators providing enough time for performing operator actions for accident mitigation;
- use of "feed and bleed" procedure for the control of beyond design based accident "loss of feed water to the steam generators";
- the measures implemented during the short term safety upgrading program.

As a result of the studies some new safety problems were found in addition to the set, defined by Deterministic analysis and were proposed corresponding solutions for each of them.

Remark: Currently PSA results are in the process of verification, including an independent international assessment by the IAEA experts. There were two IPERS mission held for review of PSA-1 of units 3 and 4 up to now. Several remarks and proposals were already adopted in the study. Remaining still need an additional evaluation to be performed which is foreseen by the utility in near future. Up to now the preliminary results do not show big discrepancy with the first evaluation even in the beginning they were evaluate as a very positive one.

7. OPERATIONAL EXPERIENCE ANALASIS

7.1 General presentation

Kozloduy NPP Units 1-4 operational experience analysis have been made in the framework of safety assessment process and preparation the new Program for safety upgrading of the units to supplement activities for resolving the problems, identified in the process of Units operation.

7.2. Initial basis for assessment

The analysis initial bases includes Units current status and results of short-term program for safety improvement activities performance in time period 1991-1997., results of analyses of operational events, registered at Kozloduy NPP or in other countries, operational system and equipment characteristics and evaluation of their change trends during plant further operation.

Several IAEA expert missions assessments of Kozloduy NPP have been performed during 1990-1991 for estimation of country safety level. Safety Review mission (which examined plant design and operation management) and ASSET mission (which analyzed operation results and registered operational events) have been performed. IAEA experts have given a set of recommendations, which have been used as a basis for short-term program for safety improvement development, together with BNSA requirements and plant proposals.

The program included not only specific technical activities for system modernization, but also some changes in operation management and equipment condition analysis. In particular, the efficiency of operational experience feed-back system have been upgraded, an analysis of main systems and equipment condition and their characteristics accordance to the current requirements have been performed.

The results of these activities performance have been analyzed during the follow-up IAEA missions in 1993-1994. The missions ascertained a significant improvement of plant operation safety level and efficiency of the methodology created for operation results analysis. This allows to include directly the important problems, identified by plant, into the "Program for safety upgrading ...".

7.3. Analysis Methodology

The analysis approach have been adopted from methodology for periodically safety reviews of operation NPP safety, presented in IAEA guideline 50-SG-012, and takes into consideration the following items, specific for Kozloduy NPP:

- plant safety analysis process in all its aspects has begun with an expert evaluation during short-term program preparation and development of "Program for safety upgrading ...", which is an extension of short-term program. As a result of several IAEA missions, discussions with BNSA and Consortium of Western Regulatory Bodies, in the frame of coordinated short-term program, the detailed examination of NPP operational experience is actually finished and activities necessary for compensation of discrepancies, identified by plant operational data analysis, have been accomplished;
- in the frame of "Program for safety upgrading ..." draft version development an deterministic analysis of NPP compliance to the regulations safety requirements and to IAEA documents, coordinated with BNSA, has been performed and the most part of remaining problems has been identified at this phase. Additionally, aspects for control of equipment aging, Units reliability and safety, existing system for operational experience feedback, human factor impact, have been assessed at operational experience analysis, and also another questions for plant operation, for which there is no clear requirements, but the plant experience show the need of their resolution.

The analysis, performed by Kozloduy NPP specialists, includes summarized information from:

- report on analysis of operational events and their causes;
- periodic reports on Units 1-4 operational safety reviews to the NPP Safety Council;
- IAEA missions reports for time period 1991-1997;
- recommendations for equipment condition investigation, its rest life time and proposals for procurement of main equipment for possible improvement of operational characteristics;
- available information about modernization programs performance at other NPP with the reactors WWER-440.

A review of main equipment condition, correspondence of its parameters to the current requirements and recommendations has been done during the analysis. The problems, which should be resolved to ensure Unit normal operation to the end of main equipment rest life time, have been identified, also.

7.4. Analysis Results

In general, the current plant operational experience review confirms the deterministic safety analysis conclusions for high efficiency of activities accomplished in the frame of short-term program for safety improvement. There have not been identified any new problems, related to safety, which need immediate corrective measures.

The improved methodology for analysis of operational events causes, which was applied in the frame of short-term program, has identified the problems, which resolution is not completed up to now. These problems are included in the general list of problems, identified by operational experience analysis, to integrate systems modernization activities in the frame of unitary program.

The analysis of equipment condition, mainly for auxiliary systems, has identified a group of problems, related to the necessity of components (or whole system) replacement to increase the capabilities, maintain the operational characteristics and resolve the deviations from the current level of requirements for this type of equipment.

Particular problems, which should be resolved to optimize main equipment capabilities use, have been formulated by the analysis. As a result, problems, that need technical activities performance, have been pointed. Activities for each problem resolution have been developed and their efficiency have been assessed.

8. SUMMARISING OF THE IN DEPTH SAFETY ASSESSMENTS RESULTS

Considering basic IAEA documents INSAG-3, INSAG-8, 50-SG-O12 detailed methodology of the Unit deterministic safety analysis is developed. It is agreed by BNSA. The methodology allowed to evaluate the original design deviations from current regulations requirements, the accomplished activities efficiency of short-term program for safety improvement and residual safety deficiency This deficiency should be removed or compensated by fulfillment of new, so called "Complex program...." activities, which goal is to carry out the Units to this safety level, which will allow their further operation without additional restrictions.

The successive using of present-day IAEA safety regulations, guidelines and recommendations, agreement with BNSA on the methodologies and set of regulations used, assures analysis adequacy and makes the analysis results acceptable at national and international levels.

Analysis was accomplished by special established group of experts from design-research organizations in Russia, who are the real designers of these Units, and specialists from Kozloduy NPP, who participate in Unit operation. This fact assures high professional level of the assessment. Special quality assurance program was applied. Quality assurance program and work group were licensed by BNSA.

The original design deviations from current regulations requirements and IAEA recommendations, which define safety deficiency of this design, were identified as a result of analysis.

By the activities, which are already accomplished within the framework of three-stage program for Unit safety improvement have been completely solved or have been effectively (without residual deficiency) compensated approximately 70% of the original design deviations from current safety standards requirements and IAEA recommendations.

For the residual deviations, using IAEA recommendations and in accordance with the methodology of Unit safety evaluation an analyses is fulfilled. The residual safety deficiency have been revealed and was transferred to the list of problems.

Fulfilled evaluation of the importance of the residual problems, relevant to the safety, shows that any problem is obstacle for further plant operation.

In this assessment process the initiating events, connected to the pipe ruptures diameter Dn200 and Dn500 were classified as events with low probability (hypothetical) and the spectrum of the primary circuit piping leakage was extended to Dn100, in accordance with the units reconstruction conception, presented to BNSA in 1995. Basis for this assumption are the results from the analyses carried out in Novovoronej and Cola NPP-s were the rupture probability of this size piping was shown to be less than 10^{-5} reactor/year, if the activities provided in the metal-monitoring have been observed. The justification of the conditions under which this results are applicable to the Units in Kozloduy NPP is part of the new Complex Program taking into account the already implemented conception "Leakage before the rupture", 100% ISI and means for the early leakage detecting (ALUS) from the primary circuit piping.

PSA - level 1 was developed for evaluation of safety operation, in accordance with the IAEA regulations (INSAG-8, 50-SG-O12) and the used in "Reconstruction conceptions for Units 1÷4 in NPP"Kozloduy" approach.

The following conclusions were reached:

1. There is relatively small risk of a core damage, because of the inherent safety of WWER, so because of the large design reserve.
2. Thanks to used in design positive characteristics and accomplished to the current moment activities , it has been reached an essential NPP safety increasing and the CDF for main internal initial events in case at the operation with power is very close to the criteria of less than 10^{-4} reactor/year for each unit.

Those activities, which have been defined as a result of operational experience in NPP were added to the scope of the new, Complex Program activities, according to the requirements of IAEA regulation 50-SG-O12. Analogous programs for this type NPP-s were taken into account.This fact assures the maximum efficiency of the planned actions.

9. INTERNATIONAL ASSESSMENT OF THE KNPP SAFETY EVALUATION APPROACH

The methodology for safety evaluation presented here was a subject of international evaluation during 1998 in connection with the efforts of KNPP to assure the maximum efficiency of the upgrading process and consistency of the approach with the current safety evaluation procedures promoted by IAEA.

The methodology was presented in depth on the IAEA Experts Meeting concerning future modernization of V-230 reactors in Vienna, February 1998 together with the results and adopted action plan. Conclusions of the assessments made on safety categories developed by the plant and an evaluation of the degree of the fulfillment of each of TECDOC 640 safety issues were added to the IAEA database.

A presentation on the evaluation approach was made during the seminar on V-230 reactors safety upgrading experience, jointly organized by GRS and IPSN in Berlin, November 1998. It was fond that the methodology is very close to the approach, proposed by GRS for safety evaluation of this type of reactors and corresponds to the proposals of the Main Designer.

Two parallel studies were ordered by NEK in 1998 for assessment of the approach consistency to current safety evaluation requirements and the scope of the measures proposed. The studies were contracted to Siemens and EdF independently. The results of both studies gave a high level assessment on the methodology developed and on the use of the current evaluation procedures.
In addition a number of proposals for introducing additional measures were made and the plant is in process of adopting of the important part of them.

Concerning the subject of safety substantiation for the units operation some extension of the basis was proposed in order to reduce the "gap" between the most used design requirements and the original V-230 design in the area of LOCAs. Following this, the foreseen by KNPP extension of DBA up to rupture of pipeline of Dn 100 will be developed introducing into the list of accidents considered by the design, LOCA of real pipeline Dn 200. This accidental study will be made with the purpose of evaluation of core cooling conditions and radioactive releases. Taking into account the low occurrence probability for this accident it is proposed the core cooling conditions to be determined based on a reduced degree of conservatism, subject of additional justification in the preparation phase. The results of this extension will affect the requirements to ECCS and will be included in the next issue of SAR of the units.

10. PROCESS OF UPDATING OF SAFETY ASSESSMENT REPORT

10.1. Format and content

The original design of V-230 units in KNPP includes specific safety substantiation chapter which have been developed according to the requirements in force in the country during designing phase. In the period up to year 1990 it was added by a set of accidental studies made without application of corresponding approaches systematically. During the short term program a specific studies on this subject were performed - to study the requirements for content and format of the full SSR, to establish the necessary basis for systematical application of conservative analyses etc.

In the same period a lot of analyses were performed in order to complete the substantiation basis for the operation of the units. These analyses include accident analyses and also system and reliability analyses and the scope and priorities were discussed with the regulatory body. There was also a specific study for evaluation which of the analyses are suitable for the purposes of updating of the SAR. In addition the design of all main modifications, important for safety includes a detailed safety substantiation report.

Several approaches for preparation of a SAR for units 1 - 4 were proposed up to now which include full or different partial SSR. The generic requirements for the preparation of the report were defined during development of Complex program for modernization taking into account the approach of 50-SG-O12 and agreed upon by Main Designer. On this basis in the mid of 1998 the content and the format of the safety substantiation report was clarified in details reflecting the requirements of BNSA. The main chapters in the content are presented in Appendix 2.

10.2. Current status

Currently the safety substantiation report is under developing according to the content presented in Appendix 2. The report is preparing as a common for twin units (correspondingly for unit 1 and 2 and for unit 3 and 4) uniting the information from various studies answering to the specific requirements for safety justifications.

The study was contracted to Bulgarian company "Risk Engineering" in November 1998. First draft now a dais is being reviewed by the plant and the expected term for presenting this document to

BNSA is end of July 1999. The work plan includes all together four stages including some additional accident analyses to be contracted by the plant consequently.

10.3. Planned activities

In parallel to the process of safety substantiation report preparation according to content, presented in Appendix 2 a detailed study was completed to determine the requirements for upgrading the scope of SAR in the future (long term activities). Based on this a ToR for large investigation (two years study) was prepared in May this year. The utility plan is to initiate this study after completion of current safety substantiation report, taking the advantage to apply two stage approach.

APPENDIX 1

LIST OF SAFETY STANDARDS AND OTHER DOCUMENTS USED FOR DETERMINISIC SAFETY ASSESSMENT OF UNITS 1-4

BULGARIAN REGULATIONS

#	Description	Title
1.1.	Common Health Code	Enacted on the 6th of November 1973; Amended and changed 1994;
1.2.	EC	Environmental Code
1.3.	LUAEPP	Law for Usage of Atomic Energy for Peaceful Purposes
1.4.	CUAEPP Order No. 2	The cases and procedures of reporting to the CUAEPP about operational changes, events and accidents related to nuclear and radiation safety.
1.5.	CUAEPP Order No. 3	Providing the safety for nuclear power plants during design, construction and operation.
1.6.	CUAEPP Order No. 4	Concerning the accounting, storage and transportation of nuclear materials.
1.7.	CUAEPP Order No. 5	For Licensing the Use of Nuclear Energy.
1.8.	CUAEPP Order No. 6	Concerning the criteria and requirements for training , qualification and certification of the personnel working in the field of nuclear energy.
1.9.	CUAEPP Order No. 7	Collecting, storage, processing, stocking, transportation and disposal of radioactive wastes on the Republic of Bulgaria territory.
1.10.	CUAEPP Order No. 8	Physical security
1.11.	GTE of EPP and EDG	Guidelines for Technical Operation of Electrical Power Plants and Electricity Distribution Grids
1.12.	MI Regulation No. 2	Ministry of Interior Regulation No. 2 regarding Fire Protection Construction Norms
1.13.	BNRP-92	Basic Norms for Radiological Protection
1.14.	AIE Regulation No. 1	Regulation regarding Assessments of Impact on Environment, enacted at Government Bulletin No. 10 on the 5th of February 1993;
1.15.	Regulation No. 0-35 of Ministry of Health and Ministry of Interior	Regulation regarding Manipulations with Radiological Substances and Other Sources of Ionizing Radiation (enacted at Government Bulletin No. 60 on the 2nd of August 1974)
1.16.	Act No. 93 of Government enacted on 27th of April 1995	Regulation regarding Civil Liability with Respect to Damages due to Small Quantities of Nuclear Materials as Treated by Vienna Convention

1.17.	Temporary Instruction No. 6 issued by SB on the 7th of February 1992, section 6 of MS	Temporary Instruction for Interactions among Specialized Groups of Permanent Commission for Population Protection with Respect to Radiological Accidents on the or outside of Territory of the Republic of Bulgaria (enacted by SB Protocol No. 6 on the 7th of February 1992)
1.18.	GCSO of NPP Equipment	Guidelines for Construction and Safety Operation of NPP Equipment

RUSSIAN REGULATIONS

#	Description	Title
2.1.	ВНР-ВКУ	Temporary Norms for Strength Calculations of PWR Vessel Internals
2.2.	ВСН 01-87	Design Norms for Fire Protection of NPPs
2.3.	ПБ-10-14-92	Guidelines for Construction of and Safety Operations with Cranes
2.4.	ПБЯ-06-10-91	Guidelines for Design and Operation of Accident Annunciation Systems with respect of Self-induced Chain Reaction as well as for Organizing Actions for Accident Mitigation
2.5.	ПБЯ РУ АС-89	Guidelines for Nuclear Safety of NPPs
2.6.	ОПБ-88	General Considerations Ensuring Safety of NPPs
2.7.	ПНАЕ-Г-7-008-89	Guidelines for Construction and Safety Operation of Equipment and Pipelines in NPPs
2.8.	ПНАЕ-Г-7-002-86	Strength Calculation Norms regarding Equipment and Pipelines in NPPs
2.9.	ПНАЕ-Г-5-7-006-87	Design Norms for Seismic Stability of NPPs
2.10.	ПНАЕ-Г-09-027-91	Rules for Designing of Emergency Power Supply Systems at NPPs
2.11.	ПНАЕ-Г-009-89	Welding, General Considerations
2.12.	ПНАЕ-Г-1-028-91	QA Requirements for NPPs
2.13.	ПНАЕ-Г-10-021-90	Rules for Construction and operations of isolation safety systems in NPPs
2.14.	ПНАЕ-Г-14-029-91	Safety Rules for Storage and Transportation of Nuclear Fuel in NPPs
2.15.	ПНАЕ-Г-7-010-89	Welding, Rules for Control
2.16.	ПНАЕ-Г-7-013-89	Rules for Construction and Safety Operation of Reactivity Adjusting Mechanisms
2.17.	ПНАЕ-Г-9-026-90	General Rules for Construction and safety operations of Emergency Power Supply Systems in NPPs
2.18.	ОСП 72/87	General Sanitary Considerations
2.19.	ОТТ-87	General Technical Requirement regarding Valves
2.20.	ПНАЕ-Г-5-020-90	Rules for Construction and Operation of System for Emergency Reactor Cooling and Heat Transfer to the Final Sunk
2.21.	СП АС-88/93	Sanitary Rules for Designing and Operation of NPPs

IAEA REGULATIONS

#	Description	Title
3.1.	Nuclear Safety Convention	Signed at Vienna on the 20th of September 1994; Ratification by Bulgarian Parliament obtained on the 14th of September 1995;
3.2.	S.S N 110	Safety Fundamentals: The Safety of Nuclear Installations IAEA Vienna 1993
3.3.	INSAG-3	Basis Safety Principles for Nuclear Power Plant, IAEA, Vienna 1988;
3.4.	INSAG-4	Safety Culture IAEA, Vienna 1991;
3.5.	INSAG-8	Basis for Judging the Safety of Built to Earlier Standards IAEA, Vienna 1995;
3.6.	INSAG-10	Defense in Depth in Nuclear Safety
3.7.	50-SG-012	Periodic Safety Review of Operational Power Plants
3.8.	TECDOC-640	Ranking of safety issues for WWER-440 model 230 nuclear power plants, IAEA-TECDOC-640, February 1992;
3.9.	WWER-SC-107	Safety Improvements of WWER 440/320 Nuclear Power Plants
3.10.	WWER-SC-SB-5	Safety Evaluation of Operating NPPs built to Earlier Standards. A Common Basis for Judgment Rev. 2/05.96
3.11.	IAEA-NENS-1996 WWER-164	Report of a Technical Visit Organized by the International Atomic Energy Agency to Kozloduy Nuclear Power Plant Units 1 to 4 Bulgaria, 15 to 19 January. Draft 1996-02-28
3.12.	IAEA-EBP-WWER-01	Guidelines for Accident Analyses for WWER NPPs
3.13.	IAEA-EBP-WWER-02	Technical Basis for Instrumentation and Control Design Improvements in WWER-440/320 Nuclear Power Plants, January 1996;

OTHER REGULATIONS

#	Description	Title
4.1.	Conception for Unit 1-4	Conception for Reconstruction of Units 1-4 of KNPP, March 1995;
4.2.	Letter of CUAEPP 47-01-30/12.06.95	Regarding Conception for Reconstruction of Units 1-4 of KNPP

APPENDIX 2

CONTENT

OF SAFETY SUBSTANTIATION REPORT

1. Introduction
2. Site description
3. Basic safety principles
4. Operational limits and conditions
5. System and components description
6. Management of safety during operation
7. Deterministic safety assessment
8. Probabilistic safety assessment
9. Radiation protection and rad. waste treatment
10. Comparison of the safety level with the requirements of the safety standards
11. Accident management and Emergency planing
12. Substantiation of the level of safety