KOZLODUY NPP PLC

PROGRESS REPORT

for Kozloduy NPP

Stress Tests

August 2011

TABLE OF CONTENTS

TABLE OF CONTENTS 3
LIST OF TABLES 5
LIST OF ABBREVIATIONS 6
1 EXECUTIVE SUMMARY 10
1.1 STRESS TEST AIM AND OBJECTIVES 10
1.2 SCOPE OF STRESS TESTS 11
1.2.1 Nuclear Facilities Covered by Stress Tests 11
1.2.2 Scenarios Covered by Stress Tests 12
1.2.2.1 Initiating Events 12
1.2.2.2 Loss of Safety Functions 12
1.2.2.3 Severe Accident Management 12
2 PROJECT ORGANIZATION AND MANAGEMENT 13
2.1 PROJECT INITIATION 13
2.2 MAIN ROLES OF THE PROJECT PARTICIPANTS 14
2.3 MILESTONES, DATES AND STAGES 15
2.4 METHODOLOGY FOR ANALYSIS 15
2.4.1 Determination of the Inputs Set 15
2.4.2 Methodology for Performance of Stress Tests 16
3 GENERAL DATA FOR KOZLODUY NPP 18
3.1 KOZLODUY NPP GENERAL DESCRIPTION 18
3.1.1 Site Characteristics 18
3.1.2 Major Facilities and Structures on the Site 20
3.1.3 Spent Fuel Storage Facilities – SFSF and SFDSF 21
3.1.4 Licensing Status 21
3.2 MAIN DATA FOR UNITS 3 AND 4 23
3.2.1 Key Features of Units 3 and 4 23
3.2.1.1 Type of Reactor 23
3.2.1.2 Thermal Power 24
3.2.1.3 First Criticality Date 25
3.2.1.4 Fuel Storage Pond (FSP) 25
3.2.2 Systems Ensuring or Supporting the Fundamental Safety Functions 25
3.2.2.1 Reactivity Control 27
3.2.2.2 Reactor Cooling to the Ultimate Heat Sink 29
3.2.2.3 FSP Cooling to the Ultimate Heat Sink 32
3.2.2.4 Containment Cooling to the Ultimate Heat Sink 34
3.2.2.5 Location, Time Constraints and Power Supply Sources for Heat Removal Channels 34
3.2.2.6 AC Power Supply 35
3.2.2.7 DC Power Supply Batteries 39
3.3 MAIN DATA ON UNITS 5 AND 6 40
3.3.1 Main Characteristics of Units 5 and 6 40

3.3.1.1 Reactor Type ... 40
3.3.1.2 Heat Power ... 42
3.3.1.3 First Criticality Date ... 42
3.3.1.4 Fuel Storage Pond (FSP) ... 42
3.3.2 Systems Ensuring and Maintaining Fundamental Safety Functions ... 42
3.3.2.1 Reactivity Control ... 46
3.3.2.2 Reactor Heat Removal – Ultimate Heat Sink ... 50
3.3.2.3 FSP Heat Removal by the Ultimate Heat Sink ... 56
3.3.2.4 Containment Heat Removal to the Ultimate Heat Sink ... 57
3.3.2.5 Location, Time Constraints and Power Supply Sources for Heat Removal Channels ... 59
3.3.2.6 AC-DC Power Supply. ... 60
3.3.2.7 DC Power Batteries ... 64
3.4 MAIN DATA FOR SFSF AND SFDSF ... 66
3.4.1 Main Characteristics of the Spent Nuclear Fuel Storage Facility ... 66
3.4.1.1 Location and Properties of the Site ... 66
3.4.1.2 Main Design Solutions ... 66
3.4.2 Main Characteristics of the Spent Fuel Dry Storage Facility ... 71
3.4.2.1 Location and Properties of the Site ... 72
3.4.2.2 Main Design Solutions ... 72
3.4.2.3 Systems and Equipment Description ... 72
3.4.2.4 Heat removal from SFP to the Ultimate Heat Sink ... 75
3.5 SCOPE AND MAIN RESULTS OF THE PSA ... 75
3.5.1 Scope and Main Results of the PSA of Unit 3 and 4 ... 75
3.5.2 Scope and Main Results of the PSA of Units 5 and 6 ... 76
3.5.3 Scope and Main Results of the PSA of the SFSF ... 78
4 KOZLODUY NPP STRESS TESTS CURRENT STATUS ... 79
4.1 CHRONOLOGY OF STRESS TESTS PERFORMANCE ... 79
4.2 INDIVIDUAL STRESS TESTS STATUS ... 80
4.2.1 Earthquakes ... 80
4.2.1.1 Design Basis ... 81
4.2.1.2 Safety Margins Evaluation ... 81
4.2.2 Flooding ... 81
4.2.2.1 Design Basis ... 81
4.2.2.2 Safety Margins Evaluation ... 81
4.2.3 Extreme Meteorological Conditions ... 81
4.2.3.1 Design Basis ... 82
4.2.3.2 Safety Margins Evaluation ... 82
4.2.4 Loss of Safety Function due to Any Credible IE on the Site ... 82
4.2.4.1 Design Basis ... 82
4.2.4.2 Safety Margins Evaluation ... 83
4.2.5 Severe Accident Management ... 83
5 REFERENCE DOCUMENTS ... 84

LIST OF TABLES

1 TABLE 3.2.1.2-1. Key design features of the WWER-440/B-230 24

2 TABLE 3.2.1.3-1. Historical data for Kozloduy NPP Units 3 and 4 25

3 TABLE 3.3.1.2-1 Main Design Characteristics of WWER-1000/B-320 42

4 TABLE 3.3.1.3-1 Historical data of Kozloduy NPP Units 5 and 6 42

5 TABLE 3.5.1-1 Core damage frequency 76

6 TABLE 3.5.2-1: Level 1PSA results 77

LIST OF ABBREVIATIONS

AB	Auxiliary building
AEC	Automatic excitation control
AFD	Automatic field damping
AFE	Automatic fire extinguishing
ALARA	As Low As Reasonably Achievable
ARM	Reactor power automacic control
ARS	Automatic reswitch
ASBE	Automatic switch to backup equipment
ASSGEM	Additional system for steam generator emergency makeup
ASVs	Automatic stop valves
AWTS	Active water treatment system
BBC	BB-cask, conditioned waste storage container
BPSS	Backup power supply section
BRU-A	Fast acting relief valve for steam dump to the atmosphere
PCP	Boron solution pump
COPP	Cold overpressure protection
CPS	Circulation pump station
CR	Control rods
CS	Confinement system
DCB	Direct current board
DG	Diesel generator
DGP	Diesel generator plant
DMW	Demineralized water
EBT	Evaporator bottoms tank
ECCS	Emergency core cooling system
ED	Electric device
EDG	Emergency diesel generator
EDGP	Emergency diesel generator plant
EES	Electric energy system
EMIT	Emergency makeup inventory tank
EMT	Emergency make-up tank
ENSREG	European Nuclear Safety Regulatory Group
EP	Emergency protection
EPA	Emergency protection automation
EU	European Union

FA	Fuel assembly
FE	Fuel element
FSAR	Final safety analysis report
HLAST	High-level active sorbent tank
HPP	Hydro power plant
I&C	Instrumentation and control
I&C and A	Instrumentation and control and automation
IAEA	International Atomic Energy Agency
ICI	In-core instrumentation
IS	Interlock switch
ISPS	Industrial seismic protection system
LCB	Local control board
LLA	Long lived aerosols
LLAST	Low-level active sorbent tank
LLSL	Lower Limit Switch Level
LOCA	Loss Of Coolant Accident
TLS	Triggered load system
LV	Low voltage
MCR	Main control room
MEER	Ministry of Energy and Energy Resources
MSH	Main steam header
MSIV	Main steam isolation valve
MSK-64	Medvedev-Sponheuer-Karnik (scale) dated as 1964
MSL	Primary circuit main steam line
NF	Nuclear fuel
NFMC	Neutron flux measurement channel
NI	Nuclear installation
NO	Normal operation
NPP	Nuclear power plant
NPPSS	Shift Supervisor of NPP
NRA	Nuclear Regulation Agency
OSY	Outdoor switchyard
PCHE	Pond cooldown heat exchanger
PNAE	Standards and Regulations in the Nuclear Energy
PPM	Planned preventive maintenance
PS	Power Switch
PSA	Probabilistic safety assessment
QA	Quality Assurance

RAW	Radioactive waste
RCP	Reactor coolant pump
RCPA	Reactor coolant pump automation
RLE	Review level earthquake
RM	Radiation monitoring
RNG	Radioactive noble gases
RP	Reactor plant
RPA	Relay protection and automation
RPCS	Reactor protection and control system
RRZ	Pressurizer
RS	Reactor shop
s.c.	Short circuit
s/st	Substation
SAR	Safety analysis report
SB	Sanitary building
SF	Spent fuel
SFBD	Switch failure backup device
SFDSF	Spent fuel dry storage facility
SFSF	Spent fuel storage facility
FSP	Fuel storage pond
SG	Steam generator
SHC-CSS	Software and hardware complex for control safety systems
SLA	Short lived aerosols
SPS	Shore pump station
SS	Safety system
SSE	Safe shutdown equipment
ST	Startup transformer
SU RAW	State utility "Radioactive waste"
SV	Safety valves
TC	Transport cask
TFWP	Turbo feedwater pump
TG	Turbine generator
TH	Turbine hall
TPP	Thermal power plant
UPSS	Uninterruptible power supply system
US	Universal slot
USS	Unit shift supervisor
VT	Voltage transformer

WENRA	Western European Nuclear Regulators' Association
WWER	Water-water energy reactor

1 EXECUTIVE SUMMARY

1.1 Stress Test Aim and Objectives

Following the nuclear accident at Japanese nuclear plant "Fukushima" on March 11, 2011, agenda was the necessity of urgent measures to assess the current status and preparedness of the nuclear plant at Kozloduy to ensure safety during extreme impacts. On March 23 Crisis Team was established for coordination and organization of conducting additional tests and on March 24 NRA presented to Kozloduy NPP their requirements to take preventive measures and set deadlines for their fulfillment. The regulations of the NRA were preliminary a short-term action for testing before the adoption of uniform requirements for all nuclear plants in the EU which later became known as "stress tests". Kozloduy NPP performed the prescribed measures and the report with the results of the test was submitted to the NRA on June 10, 2011. On June 01, 2011, Kozloduy NPP initiated "Performance of stress tests as targeted re-assessment of Kozloduy NPP equipment safety margins in natural disaster events that lead to severe accidents" according to NRA outgoing letter No. 47-00-77 dated 05/31/2011 to the Executive Director of Kozloduy NPP. The reassessment covers all nuclear facilities on the site - Units 5 and 6, fuel storage ponds of Units 3 and 4, spent fuel storage facility (SFSF) and spent fuel dry storage facility (SFDSF).

In accordance with [17] [18] and [20], stress test is defined as a targeted reassessment of nuclear safety margins in the light of events occurred at Fukushima NPP, as a result of the impact of extreme weather phenomena, requiring the performance of the safety functions of the plant and leading to a severe accident.

According to [17] and [18], in general, stress test consists in determining the preparedness of a nuclear power plant to respond to the consequences of the occurrence of extreme natural phenomena (as a rule, associated with the loss of a significant part of equipment and systems of NPP).

In accordance with [17] [18] [19] and [20], stress test is to summarize the response of the plant and the effectiveness of preventive measures, identify any potential weak point and cliff- edge effect, for each of the extreme situations considered. This is necessary to assess the sustainability of the applied defense in depth approach, the adequacy of existing measures for accident management and to identify potential improvements in safety, both technical and organizational (such as procedures, human resources, organization of emergency response or use of external resources). According to [17] [18] and [20], the tasks of stress test can be summarized as follows:

- determining the measures adopted in the plant design and compliance of the plant with the design requirements in terms of external impacts;

– determining the capacity of nuclear power plant to respond to beyond design basis events, i.e. assess the sustainability of the nuclear power plant and identify potential weak points;
– identify possible measures to increase the level of resistance of components and SSCs to increase the total plant resistance against extreme weather phenomena.

1.2 Scope of Stress Tests

1.2.1 Nuclear Facilities Covered by Stress Tests

The stress tests cover all nuclear facilities in operation at Kozloduy NPP site, including the units that are shutdown but whose FSPs store spent nuclear fuel, which includes the following nuclear installations:

– Unit 3, fuel storage pond (FSP);
– Unit 4, FSP;
– Unit 5, reactor;
– Unit 5, FSP;
– Unit 6, reactor;
– Unit 6, FSP;
– Spent fuel storage facility (SFSF);
– Spent fuel dry storage facility (SFDSF).

1.2.2 Scenarios Covered by Stress Tests

1.2.2.1 Initiating Events

In accordance with [20], Kozloduy NPP stress test will focus on the following initiating events:

- earthquake;
- flooding;
- other extreme weather conditions.

The selected range of events satisfies the general methodological requirements of [17] [18] and [19].

"Stress test" for each of these events is aimed at determining:

- design basis and current status of components and structures;
- determining the margins;
- identify possible preventive measures in case of respective impact.

1.2.2.2 Loss of Safety Functions

In order to determine the resistance of Kozloduy NPP to the above initiating events similar to those occurred at Fukushima NPP and in accordance with [17] and [18], the scope of the stress test includes assessment of the consequences of loss of safety functions for initiating events, possible on the site:

- loss of off-site power;
- loss of ultimate heat sink;
- combination of both.

1.2.2.3 Severe Accident Management

To confirm the protective measures adopted under the stress test, in accordance with [17] [18] [19] and [20] the following severe accidents management issues are included:

- protection measures against and management of the loss of core cooling function;
- measures to protect and management of the loss of function of spent fuel cooling in the FSP, SFDSF and SFSF;
- measures to protect and management of the containment integrity.

2 PROJECT ORGANIZATION AND MANAGEMENT

2.1 Project Initiation

The implementation of stress tests was launched on 01.06.2011, in accordance with the NRA outgoing letter No. 47-00-77 dated 05/31/2011 to the Executive Director of Kozloduy NPP.

After thorough analysis of what should be done to ensure timely and high quality performance of stress tests as required by the NRA letter [17], the management of Kozloduy NPP launched the following major strategy for the task:

- Step 1 – Preparation of a methodology for the implementation of stress tests allow implementation of the various stress tests to be performed by diferent companies which has the greate experianece in this field;
- Step 2 – Distribute the implementation of stress tests among different engineering organizations with many years' experience in implementing this type of analyses with the following objectives:
  - each of the invited performers has expertise in a specific area. Thus, maximum use can be made of potential of Bulgarian and foreign companies to achieve the highest possible quality of execution of tests;
  - stress tests distributed among various companies to use the potential of more experts to ensure maximum depth of analysis within the defined deadlines of performance of stress tests.
- Step 3 – Have progress report and summary report prepared by a company not involved in the implementation of stress tests so as to obtain another independent assessment of the results of stress tests and reports submitted by contractors.

Coordinating role in implementing the activities and review of the analysis is performed by Kozloduy NPP as the Employer and the operator responsible for the preparation of reports and their submittal to NRA.

2.2 Main Roles of the Project Participants

The main organizations involved in stress tests of Kozloduy NPP are:

- Kozloduy NPP;
- Nuclear Regulatory Agency;
- "WorleyParsons Nuclear Services", Sofia, Bulgaria;
- "Risk Engineering" JSC (RE), Sofia, Bulgaria;
- "Institute of Nuclear Research and Nuclear Energy" (INRNE) at Bulgarian Academy of Sciences (BAS), Sofia, Bulgaria;
- "ENPRO - Consult" Ltd., Sofia, Bulgaria;
- "Westinghouse Energy Systems" Ltd. (WES), Sofia, Bulgaria.
- Kozloduy NPP is the license holder and the Contracting Authority for the implementation of stress tests, and is responsible for the quality and timely execution.
- Nuclear Regulatory Agency as the national regulatory authority shall approve the progress report and summary report of the implementation of stress tests once they are endorsed by the Specialized technical committee in the Kozloduy NPP and if they comply fully with the regulatory requirements. In the process of reviewing the two reports, the NRA can return them to the Kozloduy NPP to incorporate the comments of the regulatory authority. On the basis of information submitted by the Kozloduy NPP reports, the NRA prepares the national report, which is then passed to the European Commission.

"Risk Engineering" JSC INRNE BAS, "ENPRO Consult" Ltd. and "Westinghouse Energy Systems" Ltd. (contractors) are responsible for implementing specific analysis and assessment associated with different tasks of the quality required under the contracts, the terms of reference, approved methodology and tasks completion schedule.

WorleyParsons NS is responsible for preparing the progress report of stress tests and the summary report of the performed stress tests.

Summary report will be in accordance with quality plan and the agreed methodology, regulatory requirements and international best practice. It should have a scope and depth sufficient to demonstrate the sustainability of nuclear facilities at Kozloduy NPP to withstand extreme impacts.

2.3 Milestones, Dates and Stages

The timeframe for the reports preparation is set by the EU as follows:

- preparation of a progress report on implementation of stress tests by the licensee - August 15, 2011;
- preparation of national progress report on implementation of stress tests by the regulatory authority - September 15, 2011;
- preparation of a summary report of the stress tests by the licensee - October 31, 2011;
- preparation of the national summary of the stress tests by the National Regulatory Authority - December 31, 2011;

NRA as the national regulatory authority to implement a regulatory review of reports prepared by the licensee, and to this end reports must be prepared and submitted by the licensee to NRA, according to its explicit requirements as follows:

- Progress Report - by 10/08/2011
- Summary report – by 28/10/2011

2.4 Methodology for Analysis

2.4.1 Determination of the Inputs Set

The stress test is based on an analysis of documents: instructions, procedures, technical specifications, SAR, PSA, technical decisions and their rationale available and adopted by Kozloduy NPP, and results of various studies, analyses and expertise.

Where the data from these documents are not sufficient application of engineering judgement is allowed.

As input data documents are considered, such as:

- those validated in the licensing process;
- those not validated in the licensing process, but passed through the quality assurance program of Kozloduy NPP;
- if needed, none of the above, i.e. other documents, for which an assessment of their applicability for analysis purposes is performed.

The licensing status of the documents applied must be clearly defined.

Use of documents not registered in Kozloduy NPP "stress tests" database must be justified and the documents used or parts thereof, must be included as attachments to the reports.

For each of the areas of re-evaluation of the safety margins a specific set of input data is determined.

Responsibility for determining the necessary input data for each of the areas is with the contractor to stress test. The main criteria for determining the input data for the stress test can be summarized as follows:

- documents containing information on design basis;
- documents containing data specifying changes to Kozloduy NPP design;
- documents containing data defining the current status of Kozloduy NPP;
- documents containing data from the deterministic analysis and expertise;
- documents containing data from real events related to the area studied;
- documents related to safety management or assessment.

In order to systematize the input data used for the reassessment of safety margins a common database for the project should be established. The main attributes of this database should enable the identification of documents (registration and/or archive number within the database of Kozloduy NPP) for the areas in which information contained in it is used.

2.4.2 Methodology for Performance of Stress Tests

The basic framework of methodology for conducting stress tests is given in the NRA letter [17] in response to the requirements of the European Commission presented in ENSREG Declaration [18] and WENRA specification [19], and in NRA letter [17].

Due to the strategy undertaken by the management of Kozloduy NPP for allocation of the stress tests between contractors and the need for the results of their work (interim reports, final reports, progress report and summary report) to meet the general rules for writing, using a common methodology, the following main objectives are to be performed:

- determine the scope of the analyses of individual stress tests;
- determine the style of writing so that at the end of the conduct of stress tests, the results of individual performers are consistent and the summary report can be prepared in a relatively easy manner.

To achieve these objectives, the methodology describes in detail what the stress test performer must do to conduct it. The document explains in detail and describes the steps of implementing of each stress test separately. Such steps fully satisfy the requirements specified in [17], [18] and [19] for the defined stress tests.

This document sets out some basic requirements, such as:

- performance of stress tests can be based on existing analyses and information and documents available in Kozloduy NPP;
- development of additional new analyses can be done only with the consent of the Employer;
- use of documents external for Kozloduy NPP can only be after they are attached to the texts refer to;
- description of how to present the results of stress tests, so that the summary report can be easily prepared;
- manner of presentation of results must also allow for easy retrieval of information to write an abbreviated version of the summary report;

3 GENERAL DATA FOR KOZLODUY NPP

3.1 Kozloduy NPP General Description

3.1.1 Site Characteristics

Kozloduy Nuclear Power Plant has been constructed in the north-west of Bulgaria, on the right bank of the River Danube near the town of Kozloduy. It is at 120 km by a straight line and 200 km by road from Sofia. Geographical coordinates of the site in the zoom are 43º 44′ 48.4″ north latitude and 23º 46′ 9.2″ east longitude. [1].

The site is located at the 694th km of the Danube. It is at 3.7 km to the south of the midstream of the river and the border with the Republic of Romania. It is located in the northern part of the first not flooded terrace of the river Danube.

The area within the radius of 30 km around the site includes municipalities with centers: Kozloduy, Valchedram, Khayredin, Mizia (fully) and Lom, Byala Slatina, Oryakhovo (partly). The 30 km zone also includes sparsely populated part of Romania, i.e. 12 villages. Nearby settlements are: town of Kozloduy at 5,3 km to the southwest, village Charlets at 5,5 km to the southeast, village Glozhene at 6,3 km to the southwest, town Mizia at 8,5 km to the southwest, village Butan at 10 km to the south, town of Oryakhovo at 15 km to the east of the site and Gorny Cibar village 21 km away.

The whole site area is about 3.2 km^2 and with the circulation and service water supply channels it reaches 5.2 km^2. The main buildings with their support structures for units 5 and 6 of Kozloduy NPP are located in an area of 1.2 km^2 [2].

The main electric power distribution grid of the settlements in the region is of medium voltage 20 kV. It is connected to 110 kV and 220 kV high voltage grid of the national energy circle.

Nuclear power plant site is located in Lom-Oryahovsko subregion of northern climatic region of the Danube lowland. The climate is moderate continental with cold winters and hot summers, with annual rainfall in the area of Kozloduy NPP of about 518-558 mm which is one of the lowest in the country. It is unevenly distributed throughout the year. Maximum rainfall is in May – June. Heavy rainfalls are most likely to occur during the summer period. During May - July frequency of days with thunderstorms and hail is increased.

In the close vicinity of Kozloduy NPP there are no protected and other sensitive areas.

Engineering geology of Kozloduy NPP site has been the subject of multiple research and analysis (1967 - 1992). In the last stage, in 1990-1992, studies were conducted in cooperation with the IAEA - BUL 9 / 012 "Site and Seismic Safety of Kozloduy and Belene NPPs". Based on the analysis of the bank of geological, geomorphological, geophysical, seismological and neotectonic data and additional topical research a certain geological profile of the site is generated with the lithological characteristics of all varieties. Also, phenomena associated with earthquakes were examined in terms of the definitions given in [24] - tectonic activity, potentially dangerous faults, landslides, liquefaction potential of saturated cohesionless of earth strata of ground subsidence, slope instability and others. The following main conclusions have been made:

In the area investigated there are no major fault structures with high energy potential (no evidence of "capable" faults).

Kozloduy NPP is located in the relatively most stable part of Mizia platform in terms of seismic.

In the said project levels of seismic ground of NPP Kozloduy have been defined using probabilistic and deterministic methods as follows:

Level for a period of 100 years recurrence peak ground acceleration (PGA) - 0.10g;

Level for a period of 10000 years recurrence peak ground acceleration (PGA) - 0.20g.

With seismic features defined in this manner, an analysis of all SSCs designated as seismic category I has been performed [25].Where necessary, measures have been taken (reinforcement, additional qualification or replacement) to bring the SSCs in compliance with the requirements for seismic stability.

The area is flat, with altitude ranging from +28.0 to +36,0 m altitude in the Baltic system. From the River Danube the valley and the site are protected by dike reaching absolute elevation of + 3.3 m.

The site is located in the non-flooded terrace with absolute elevation of +35.0 m. To the north it borders the Danubian Plain. South of the site of the watershed plateau slope is relatively high (100¸ 110 m), the west is about 90 m, while the east is lower and decreases to 30 m above the sea level.

In the main project when determining the elevation of Kozloduy NPP, to exclude the possibility of its flooding from the River Danube typical elevations and water levels of the river have been considered (km.687.00 abstraction in the NPP). These elevations were measured considering the natural river mode and the possibility of building Hydrohub Nikopol-Turnu Magurele. The maximum elevation of flooding likely to occur once every 10,000 years (probability p = 0.01%) is 33.50 meters and it is a precondition for building Hydrohub Nikopol-Turnu Magurele.

Overall relief of the area of the site in physico-geographic terms is typical of the Bulgarian valley of the River Danube and the Danube lowland. A variety of landscapes in the area under consideration is large in the area of up to 50 km from the Kozloduy NPP which is typical of lowland landscapes.

Roadways in the area are twofold, and belong to a paved secondary road network.

The site is connected to the national road network roads: Vratsa - Mizia - Oryakhovo and Mizia - Kozloduy - Lom. The site is not connected to the national railway network. By means of its own port transport links along Danube are ensured.

Within the 12 km zone the road density of road network is 142 km per 1000 km2..As compared against the density of road network in the 30 km zone of Kozloduy that density is lower. In Mizia it is 337 km, in Oryakhovo - 229 km, is in Hayredin - 212 km per 1000 km^2.

3.1.2 Major Facilities and Structures on the Site

On Kozloduy NPP site six units are located. Units 1 to 4 are WWER-440 and units 5 and 6 are with WWER-1000.

Master Layout Plan of the NPP has been developed in accordance with the principle of modular layout system of units at one nuclear power plant. Most recent planning of buildings and facilities significantly contribute to reducing vehicle and pedestrian and technological communications, and also reduces engineering communications.

Functionally all facilities at the NPP are divided into major and auxiliary ones. The composition of the major facilities of the unit includes: main building - the reactor building (reactor compartment) and turbine building (turbine compartment); AB and overpasses for connection to the reactor compartment of the unit, electrical equipment and emergency diesel stations, cable channels and tunnels for connection between DGP and the reactor compartment, service water and nuclear component cooling water supply facilities (spray ponds and buried pipelines for service water), service water and conventional cooling water supply facilities (circulation pump stations, cold channel and drainage (hot channel)) [1].

Auxiliary facilities to the units include: demineralized water inventory tank, a common auxiliary building; oil station; gas receivers site, nitrogen-oxygen station, administrative buildings, overpasses and process pipelines channels.

Site facilities support structures and communications include the following:

- Cold component cooling water channel (for units 5 and 6 the channel only performs functions for normal operation, whereas safety systems are cooled through spray ponds);
- Hot channel for returning the cooling water in the Danube

- General fire ring (connection available between the site fire rings possible).This ring supplies water to normal operation fire fighting systems.
- Auxiliary pipelines for steam and water transfer between the site of Units 5 and 6 and the site of Units 1 to 4 located along the process overpass constructed between the two sites
- Process overpasses for connection between Units 5 and 6 and connection with auxiliary facilities. On these overpasses pipelines and cables to connect to demineralized water inventory, diesel-oil sector, the common auxiliary building, and heating system at the site are located.

Main (system forming) grid of EES of Bulgaria is linked to three networks of distribution systems, 220 kV, 400 kV and 110 kV network. Thus, it provides reliable operation in normal and post-accident mode of operation. Between 400 kV switchyard, 220 kV switchyard and 110 kV switchyard connection has been provided through automatic transformers, which allows for field synchronization of operational and backup power [1], [2],[3] и [4].

3.1.3 Spent Fuel Storage Facilities – SFSF and SFDSF

On Kozloduy NPP site spent fuel storage facility (SFSF) and spent fuel dry storage facility (SFDSF), that store spent nuclear fuel (SNF) from WWER-440 and WWER -1000, are located.

SFSF building is located in south-western end of Kozloduy NPP. The SFSF has a pool filled with water where spent fuel assemblies containers are to be stored. The water serves both for biological shielding and for spent fuel cooling.

SFDSF site is located north-northwest of the building of the SFSF. Storage technology in Kozloduy SFDSF consists in storage of containers with air-cooling principle due to natural convection.

3.1.4 Licensing Status

Since October 2010 SU RAW received licenses to operate units 1 and 2 for five years, as facilities for radioactive waste management to be decommissioned. With these licenses the reactor operations licenses are suspended.

Kozloduy NPP has licenses to operate units 3 and 4 in "E" mode under which fuel is removed from the core and stored in the reactor pool. First and second loops are filled with protective solution for suppressing the corrosion process. Support is provided \ for the return of nuclear fuel in the core in case of accident. Under the terms of licenses spent nuclear fuel is stored only on the bottom shelf in the pools.

Kozloduy NPP holds licenses for operation of units 5 and 6 valid through 2017 for Unit 5 and 2019 for unit 6, which are respectively in their 18^{th} and 16^{th} fuel campaign. Both units operate primarily in the basic mode of rated power operation under the terms of operations licenses.

The License owners for Kozloduy NPP site nuclear facilities are:

- Unit 1 - licensee is the State Enterprise "Radioactive waste" - license valid until 17.10.2015, the Scope - a facility for radioactive waste management to be decommissioned.
- Unit 2 - licensee is the State Enterprise "Radioactive waste" - license to operate valid until 17.10.2015, the Scope - a facility for radioactive waste management to be decommissioned.
- Unit 3 - licensee is "Kozloduy NPP" PLC - operating license valid until 20.05.2014, the scope - storage of spent nuclear fuel in the reactor pool.
- Unit 4 - licensee is "Kozloduy NPP" PLC - operating license valid until 26.02.2013, the scope - storage of spent nuclear fuel in the reactor pool.
- Unit 5 - licensee is "Kozloduy NPP" PLC - operating license valid until 05.11.2017, the scope - operation of the unit under the terms of the license issued.
- Unit 6 - the licensee is "Kozloduy NPP" PLC - operating license valid until 02.10.2019, the scope - operation of the unit under the terms of the license issued.
- Spent fuel storage facility - licensee is " Kozloduy NPP " PLC - License to operate valid until 24.06.2014, the scope - handling and storage of spent nuclear fuel from the units of Kozloduy NPP in accordance with the requirements of the license issued.
- Spent fuel dry storage facility – currently the dry storage for spent fuel is under an Utilization Permit for construction phase I, issued by_the Directorate for National Construction Control in March 2011.

3.2 Main Data for Units 3 and 4

3.2.1 Key Features of Units 3 and 4

3.2.1.1 Type of Reactor

Each power unit has an installed capacity of 440 megawatts (electric) and includes a reactor system with VVER-440/230 reactor, turbine plant with two turbines K-220-44 and electric generator TVV-220-2A. [3] [4]. Two-circuit heat scheme is used.

The general characteristics of power units are as follows:

- Reactors 3 and 4 of Kozloduy NPP are water-water reactors with pressurized water. VVER-440/230 is heterogeneous water - water energy reactor with thermal neutrons, vessel type. The function of the reactor nuclear within the Nuclear steam supply system (NSSS) are provided and maintained by the controlled chain reaction of fuel decay and transformation of decay energy to heat which is transferred to the primary coolant. Chemically demineralized water with diluted boric acid in it is used as the coolant and the moderator in the primary circuit, with concentration of the boric acid changing during operation.
- The core (fuel system) of the reactor is designed to generate heat and transfer it to the surface of the fuel elements (FE). It includes 349 fuel assemblies, each consisting of 126 fuel elements. The pressure in the primary circuit is 12.5 MPa, reactor water inlet temperature is 263°C;

The primary circuit which is radioactive, is located in the secured reinforced steel hermetic structure. This consists of the reactor, steam pressurizer, six circulation loops including the steam generator, the reactor coolant pump and the circulation line with nominal diameter of 500 mm, main isolation valve (MIV) and the auxiliary equipment.

Low enrichment uranium dioxide is used as a fuel with enrichment of 3.6 % ^{235}U concentration. The primary coolant is heated at passing through the core. After that it is delivered to the steam generators where heat is transferred to the secondary circuit.

The steam generators, the main circulation lines and other equipment of the primary circuit are located in protective concrete shells (steam generators compartment).

The secondary circuit is not radioactive. It consists of steam generators generating steam, main steam lines, two turbine generators with auxiliary and support systems, equipment for deaeration, and steam generators feedwater regenerative heating and supply.

The units output the electricity generated to 400 kV grid.

Connection of the unit with the power grid is provided through the outdoor switchgear. Electricity consumers of the units are supplied from auxiliaries system.

Unit service water is supplied from Danube trough SPS and hot and cold channels system. Service water for safety systems is supplied from spray ponds.

The main design parameters and characteristics of units 3 and 4 are given in Table 3.2.1.2-1.

The steam generators (SG) are horizontal heat exchangers of the surface type and are designed to remove heat from the primary coolant and to generate dry saturated non-radioactive steam within the secondary circuit. Heat exchanging surface of the steam generators represents direct boundary between the primary and the secondary circuits, ensuring barrier against intrusion of the radioactive products to the secondary circuit.

The pressurizer (Prz) is component of the reactor primary circuit. This is a vertically located cylinder vessel with two elliptic bottoms which is designed to generate pressure in the primary circuit, to maintain pressure within the specified limits during the steady-state modes and to limit pressure deviations during transients and accident modes of the reactor via evaporation and condensation of the coolant. With this purpose heating by the electric heaters and condensing by lower temperature coolant injection to the pressurizer steam part are used.

Reactor coolant pumps (RCP) are components of the reactor primary circuit and are designed to generate circulation within the reactor primary coolant. The RCP represents vertical centrifugal single-step pumping unit of closed type.

The main circulation circuit consists of six circulation loops, including main circulation lines (MCLs) connecting the reactor with the steam generators and reactor coolant pumps (RCPs), main isolation valves and is designed to provide for coolant circulation between the reactor and the steam generators.

Units 3 and 4 have three independent channels of safety systems and numerous modifications aimed at improving safety.

3.2.1.2 Thermal Power

TABLE 3.2.1.2-1. Key design features of the WWER-440/B-230

CHARACTERISTICS	VALUE
Reactor heat power	- 1375 MW
Unit electric power	- 440 MW
Unit gross efficiency	- 32%
Primary circuit pressure	- 125 $\pm$ 2 kg/cm^2

CHARACTERISTICS	VALUE
Secondary circuit pressure (MSH)	- 45 ± 1 kg/cm^2

3.2.1.3 First Criticality Date

TABLE 3.2.1.3-1. Historical data for Kozloduy NPP Units 3 and 4

Characteristics	Unit 3	Unit 4
Start of construction	October 1973	October 1973
First criticality	December 4, 1980	April 25, 1982
Reaching 100 percent power	01/27/1981	06/17/1982
Worked fuel campaigns	22	21
Final shutdown for decommissioning	December 31, 2006	December 31, 2006

Units 3 and 4 have the status of nuclear facilities finally shut down for decommissioning. Currently irradiated fuel is stored on the lower racks of the fuel storage ponds (FSPs).

3.2.1.4 Fuel Storage Pond (FSP)

FSP is a pond with racks serving as safe storage of spent and irradiated fuel in the mode of activity and heat reducing to a level at which transportation is possible. It is the type of storage facility for storing spent fuel under water. FSP is filled with boric acid and the assemblies are arranged in racks, with a specific step of the grid. Spent assemblies stay in the FSP for at least three years in order to reduce residual heat to levels for transportation outside the reactor hall.

FSP is a concrete structure, lined with stainless steel without penetrations and openings.

In the fuel storage pond there are racks for storing assemblies and a space to contain a transport cask, called universal socket (US). The FSP has a connection with the reactor cavity through a corridor for transporting assemblies barred by hydraulic seal.

Assembly compartment with two rows of racks (upper and lower) containing cells for spent fuel extensions and cells for hermetic casks (HCs). The top rack is divided into three sections - northern, middle and southern.

The geometry of the assemblies storage racks available in the FSP (assemblies pitch) ensures sub-criticality of the fuel assembles of at least 5% even in case of filling the FSP with non-borated water solutions in all temperature conditions of the solution.

3.2.2 Systems Ensuring or Supporting the Fundamental Safety Functions

The unit is designed in accordance with the defense-in-depth concept where the sequence of barriers preventing from release of radioactive substances and ionizing radiation, with multiple features to protect these barriers from damage.

The main principle underlying the design of the unit is that in all operational states and accident conditions it fulfills the following fundamental safety functions:

- reactivity control;
- core heat removal;
- retention of radioactive substances within the established limits.

These functions are performed through normal operation equipment design technical solutions, installation of safety systems, use of effective technologies and procedures for operation and maintenance, based on recent advances in science and technology and internationally recognized operating experience.

Ensuring the safety functions:

- Equipment of all safety systems is located in seismically qualified buildings providing protection against external natural and human-induced impacts – the seismic qualification of the buildings containing equipment of safety systems is reviewed against changed seismic conditions of the site;
- Equipment of all safety systems is qualified against seismic impact. Qualification of equipment required for safe shutdown is reviewed;
- In the design of safety related systems and equipment design solutions are used, based on the passive principle of operation, the principle of failure safety and inherent self protection properties (self-control, thermal inertia and other natural processes, reactivity feedback and natural coolant circulation).
- Specific technical solutions applied in the design of safety systems are related to implementing the essential requirements of the standards documentation: multichannel structure (redundancy), physical separation and diversity.
- Multi-channel structure (redundancy) allows the safety systems for performing their functions independently of a possible failure of one channel (single failure).
- The physical separation of the channels is achieved by locating every channel in a separate room. This allows for successful operation of safety systems in case of failure of one channel due to local events (fire, explosion, high temperature, flooding, etc.).

- Diversity of physical principles in the performance of functions of safety systems applied in the design using both active (pumps, electric valves) and passive devices (check valves) to eliminate the possibility of failure of all safety systems due to station blackout (electricity, working media, etc.).
- The combination of redundancy, diversity and physical separation ensures resistance of safety systems against common cause failures, i.e. total loss of the ability of safety systems to perform their functions.

The safety systems are designed to prevent from occurrence of accidents and mitigation of their consequences. According to their functions they are divided into the following:

- Protective;
- Localizing;
- Supporting;
- Controlling.

3.2.2.1 Reactivity Control

3.2.2.1.1 Systems ensuring core sub-criticality.

Reactivity control both during reactor operation and in shutdown mode is performed by two independent principle of action:

- By moving RCPS CR by the height of the core.
- By changing the concentration of dissolved boric acid (H_3BO_3) in the primary coolant;

Reactivity control systems based on the use of RCPS CR and the boron injection in the coolant are designed in such a way that postulated accidents related to their failure do not lead to reactivity higher than the specified level and provides for:

- non-exceeding of the limits for fuel;
- primary circuit integrity;
- effective core cooling in accident mode.

3.2.2.1.1.1 Reactor Control and Protection System (RCPS)

Reactor control and protection system has the following functions:

- reactor emergency protection;
- reducing and limiting the reactor power at non-scheduled tripping of the equipment of the power unit according to predefined list and set program;
- reactor power automatic control according to the specified programs;

- monitoring of process parameters necessary for the reactor operation protection and control;
- fixing the root cause of protection actuation;
- providing the information to the operator and issuing signals to other subsystems of APCS.

Control and protection system is a reactor process system consisting of interconnected subsystems for:

- monitoring and measurement of neutron power and period of the reactor;
- chain reaction control;
- emergency suspension (attenuation) of the chain reaction.

3.2.2.1.1.2 High Pressure Boron Injection System

The primary high pressure boron injection system is designed to prevent from or limit damage to the nuclear fuel, equipment and pipelines containing radioactive products and for emergency boric acid injection at failure to reactor emergency protection system and compensation of positive reactivity, if additional makeup to primary circuit is needed when normal makeup system is not sufficient, as well as for application of primary circuit feed and bleed procedure.

The system consists of general emergency makeup tank (EMT) and three channels to supply the boric acid solution to the headers for primary circuit blowdown.

Each channel includes pumps, pipelines, valves, and I&C of the equipment.

3.2.2.1.1.3 Emergency Core Cooling System – ECCS

The emergency core cooling system is designed to prevent from or limit damage to the fuel and equipment via emergency supply of boric acid solution to the primary circuit at large loss of coolant accident and to maintain core criticality.

The system consists of emergency makeup tank (EMT) common for the three ECCS channels and spray system channel. Each channel includes a pump, spray system heat exchanger, valves, and equipment I&C.

The system consists of three fully independent channels of pressure side. The system design provides for full performance of all functions by each of the channels.

Boron solution injection in the circuit is performed through independent pipelines into the non isolating part of the cold circulation loops.

3.2.2.1.2 Systems ensuring FSP sub-criticality

The geometry of the assemblies storage racks available in the FSP (assemblies pitch) ensures sub-criticality of the fuel assembles of at least 5% even in case of filling the FSP with non-borated water solutions in all temperature conditions of the solution.

3.2.2.2 Reactor Cooling to the Ultimate Heat Sink

3.2.2.2.1 Existing channels and means to ensure reactor ultimate heat sink

The main logic of heat transfer in reactors 3 and 4 of Kozloduy NPP is from the reactor core through the primary coolant to the steam generators and then through the secondary systems:

- either to process condensers cooled with service water which ultimately leads to the River Danube (ultimate heat sink), in the absence of service water, via closed cooling circuit by spray ponds to the atmosphere,
- either by safety valves to the atmosphere.
- Heat transfer from the core to the secondary circuit of the steam generators to ensure proper geometry of the system to circulate the primary coolant, where the steam generator is located much higher than the core. This location allows for efficient natural circulation, which can take up to 10% of the reactor thermal power without the need for forced circulation of the primary coolant. Besides, water inventories inside the SGs (horizontal type) can ensure a few hours of easy removal of residual heat from the reactor after its suppression.
- Secondary circuit overpressure protection system (SG SV) is a safety system performing the emergency core cooling functions. This function can be performed regardless of exhaust of the media - water or steam. The main steam line of each SG in its non-isolable part has per two "SEBIM" pulse safety valves – control safety valve (CSV) and operating safety valve (OSV) – installed in turn.

SG emergency feedwater system (EFWS) is intended to supply water from the turbines deaerators and from SG emergency condensate tank in accident modes to supply feedwater to the steam generators with a flow-rate that will ensure reactor core residual heat removal and primary circuit cooldown in steam-water mode. The system consists of three emergency feedwater pumps, controllers, valves and I&C equipment. At loss of unit feedwater, it is possible to use a logic of cooling water supply from the adjacent unit. If this fails an additional system for emergency makeup (ASSGEM) can be used. In case of failure of cooling with service water of the technical group "A", cooldown can be provided from Fire Protection Pump Station by ASSGEM line which is a fully autonomous system described above.

As a final option for ultimate heat sink, the "Feed and Bleed" procedure can be used on the primary circuit for residual heat removal in case of accidents with total loss of feedwater. The procedure is performed through primary circuit overpressure protection system (PKKO), together with primary circuit emergency makeup. The system consists of three sets of two separate subsystems - two sets of PKKO "Sempell" and one set of PIKK-KO "Sebim".

Function of PKKO "Sempell" is to ensure primary circuit overpressure protection through discharge of fluid. PIKK-KO "SEBIM" is equipped with a device for the reactor vessel cold overpressure protection (COPD).

PIKK-KO by fluid discharge from the Prz performs the following functions:

- protect the primary circuit integrity at excessive pressure;
- reactor vessel cold overpressure protection in scheduled modes of reactor heating/cooling or in accident modes;
- when implementing "Feed and Bleed" procedure on the primary circuit for residual heat removal in case of accidents with total loss of feed water or to prevent or limit damage to the nuclear fuel.

3.2.2.2.1.1 Secondary Circuit Overpressure Protection (SG SVs)

The system is designed to prevent from steam generator pressure increase over permissible level to ensure the integrity of the structure (SG) as well as ensure emergency core cooldown at failure to the systems intended to perform this function by manual opening of the valve. This function can be performed as SG pressure over 0,3 MPa regardless of the media – steam or water. On the main stream line of each SG, in its non-isolable part, per two pulse-safety valves "SEBIM" - control safety valve (CSV) and working safety valve (WSV) performing the functions described – are installed in turn.

3.2.2.2.2 Layout and Logic of Heat Transfer

Equipment of the described heat transfer schemes is located in physically separate protective structures. The systems equipment is protected from internal influences such as fires, flooding, dependent failures through many design solutions and procedures for human actions in accident conditions. Civil structures, along with equipment of the respective systems, are seismically qualified.

3.2.2.2.3 Time constrains for Various Cooling Channels Operation

Water inventories within the SGs (horizontal type) of WWER-440 can provide for few hours of continuous removal of residual heat from the reactor after its suppression without additional makeup. It is possible to supply feedwater for SG from the adjacent unit. For this purpose emergency procedures have been developed. Regardless of the neighboring unit, total volume of 6000 m^3 of DMW is maintained on-site for units 3-4 that can be supplied different components including the SG of the respective unit for emergency cooldown. Additionally installed ASSGEM system ensures one more opportunity for supply of feedwater to the damaged unit, in condition of complete failure to supply feedwater and total loss of service water for cooling by an open or closed circuit the system can independently ensure long-term cooling of the reactor.

In the complete absence of means to supply feedwater to SGs mobile fire pump can be connected from the fire truck, which increases the unlimited time available for the scheme if access to the affected unit is possible.

3.2.2.2.4 AC Power Supply Sources and Batteries That Can Provide the Necessary Power for Each Channel

On the site of Kozloduy NPP units 3 and 4 an emergency diesel generator station (DGS) is located. The six SGs for Units 3 and 4 are located in six different rooms in a seismically qualified building. DG is a reliable source of power for the NPP and is actuated in the event of loss of off-site power supply. The emergency power supply of units 3 and 4 has three independent channels. In addition, there are technical measures for power supply from the adjacent unit in case of accidents. For this purpose operating procedures have been developed and put in force.

Additionally, units 3 and 4 of Kozloduy NPP are provided with a separate set of auxiliary diesel generators 1,3 EDG and 2,4 EDG. Auxiliary diesel generators were installed specifically for ASSGEM system and they provide power in all modes of operation of the plant, even the loss of off-site power.

As the last feature for AC power supply a mobile DG is provided which is located on the site. Operating procedures for its use are in force.

Each channel of the safety systems has batteries that are independent DC power source for category 1 consumers of uninterruptible power supply. Additionally, in the DGS for each DG there is a separate battery, which can serve as backup for the main battery of the respective safety system. At full load for each battery time for discharge not less than 2 hours is guaranteed.

3.2.2.2.5 Cooling Equipment to the Heat Removal Channels

For cooling of systems equipment operating in high temperature conditions ventilation system is provided with its equipment supplied from uninterruptible power supply sources.

3.2.2.3 FSP Cooling to the Ultimate Heat Sink

3.2.2.3.1 Heat Removal Means

Nuclear fuel from Units 3 and 4 are in the FSP-B/G of the lower row of racks. On the top row of racks extensions only are stored. To ensure the supply of boron solution and provide cooling of assemblies in FSP-B/G level of 20 ÷ 40 cm below the spillway of the FSP is maintained. Under the terms of the license, the FSP is filled with a solution of boric acid , he temperature in the FSP is maintained in the range between 20 and 500°C. The FSP cooling system includes two pond cooling pumps (PCP) in two pond cooldown heat exchangers (PCHE). These facilities form two independent channels of water cooling in FSP. Each of them has the capacity to implement the function to 100%.

Pond cooling pumps are intended to circulate the solution of FSP in the heat exchangers to cooldown the fuel storage pond.

Heat exchangers are cooled by service water from the safety systems, three-channel structure and uninterruptible power from DG as follows.

The FSP is designed in such a way that in case of leakage from the FSP cooling system pipeline, the possibility of uncovering of the fuel contained in the FSP is excluded.

Ultimate heat sink is water from the Danube and/or air through a closed cooling circuit through spray ponds by means of service water.. At loss of the unit cooling channel/channels or a significant loss of coolant from the FSP of either unit, it is possible to use schemes for water supply and cooling from adjacent unit. If cooling with service water technical group "A" fails the cooling can be provided from PPS-2 by ASSGEM line which is a fully autonomous system.

In accident conditions that can lead to loss of ultimate heat sink many technical and organizational measures are provided to control or mitigate the accident as follows:

- At loss of unit off-site power - power supply from DG of an adjacent unit, EDG of ASSGEM or, in extreme cases, from mobile DG.
- At loss of service water - supply of service water from the neighbouring unit or service water supply from PPS-2 through ASSGEM.

- Total for the site 6000 m3 DMW is maintained for units 3-4 (in 3600 m^3 tanks), which may be supplied to different components, including the FSP through the pumps, which are available in case of loss of off-site power.
- At leakage of coolant from the FSP - supply of boron or non-borated solutions from tanks existing on the site. In accidents with significant loss of water from the FSP can be ensured by group A service water or water from PPS-2.
- At loss of FSP normal cooling system – supply of solution from FSP to EMT in a closed circuit and cooling the solution in the EMT through the existing safety system.

3.2.2.3.2 Location of Heat Removal Channels

Channel of systems for heat removal from the FSP are located in the main building of the unit in the reactor hall. The ASSGEM system is located next to the unit main building.

3.2.2.3.3 Time Constraints for Operation of the Various Heat Removal Channels

There are no time limits for operation of various channels for normal heat removal. In accident cases, at loss of the normal logic for heat removal from either unit, cooling is ensured by the service water supplied by the PPS-2, which also provides 7 days of operation in fully autonomous mode. At loss of off-site power, power is supplied from the emergency DG of safety systems, for at least 4 days, mobile DG or DG of ASSGEM.

3.2.2.3.4 AC and DC Power Supply Sources for Each Channel

Electric power supply of Pond cooling pumps (PCP) is from category 2 section, power supply from DG:

Uninterruptible power is supplied from the DG at automatic start of at least one PCP in operation or in backup. All PCP pumps are supplied from the DG.

Additionally, backup electric power supply is ensured from EDG of ASSGEM, as well as from mobile DG.

3.2.2.3.5 Cooling of Equipment to the Heat Removal Channels

Long-term heat removal from the equipment of the channels is ensured through the operation of ventilation systems for the premises, the most important of these are powered from uninterruptible power supply system sections.

3.2.2.4 Containment Cooling to the Ultimate Heat Sink

3.2.2.4.1 All Existing Means for Containment Cooling to Ultimate and Alternative Heat Sink

Under normal operating conditions, the design provides for recirculating ventilation system for cooling the steam generators compartment, providing an average temperature of around 50-70^{o}C in closed areas. Ultimate heat sink of the system is the Danube River, or the atmosphere at closed cooling circuit.

Under condition of leakage from the primary or secondary circuit to the SG compartment, pressure and temperature in BPG increase. To mitigate it boric acid solution is supplied from the spray system. It is sprayed through special nozzles and the steam in BPG condenses, thus cooling the premises. The spray system is a three-channel safety system. Ultimate heat sink under this scheme is the Danube River, at open cooling circuit or the atmosphere through the spray ponds in cooling closed circuit using Group A service water system as described above.

Additionally, as an upgrade to the original design a jet vortex condenser is installed (JVC), which under conditions of leakage in the BPG, contributes to abrupt pressure drop at the start of the leak, and help cool the BPG in the long run. For severe accidents management, along with other functions, filtered ventilation system to maintain negative pressure in the SG compartment helps to remove heat from it.

3.2.2.5 Location, Time Constraints and Power Supply Sources for Heat Removal Channels

3.2.2.5.1 Heat Removal Channels Location

Equipment of the described schemes for heat transfer is located in protective structures, physically separated channel by channel. The equipment of the systems is protected against internal effects such as fires, flooding, dependent failures through a variety of design solutions and operating procedures of the personnel for the respective accident conditions. Civil structures, along with equipment from the respective systems, are seismically qualified.

3.2.2.5.2 Time Constraints for Operation of Various Heat Removal Channels.

In case of availability of power supply and service water, there is no time constraints for operation of the various heat removal channels. Given the complete absence of water for cooling in open or closed circuit and of power supply ASSGEM system can independently ensure continuous cooling of the heat removal channels. The system is described in detail above.

3.2.2.5.3 Sources for Power Supply and Equipment Cooling

At loss of unit off-site power supply, DGs start according to the design, in the absence or failure it is possible to use power from the DG of the neighboring unit, the EDG of ASSGEM or in extreme cases of mobile DG. For cooling of equipment of systems operating in high temperature ventilation system is provided with equipment which is powered from uninterruptible power supply sources.

3.2.2.6 AC Power Supply

3.2.2.6.1 Off-Site Power

Off-site power for Kozloduy NPP Units 3 and 4 is supplied through OSY 400 and 220 kV.

3.2.2.6.1.1 Reliability of the Off-Site Power

Database for events associated with loss of off-site power supply contains several events, associated with loss of power distribution grid of short duration. There is no actuation of EP of the units caused instability in the network, on the contrary there are several actuations of EP resulting from the loss of off-site power supply.

To increase reliability and reduce the likelihood of blackout of 400 kV outdoor switchyard, control has been introduced as a dual section bus system.

On outdoor switchyard between 400 kV and 220 kV systems, and between 220 kV and 110 kV systems connections are made through the automatic transformers.

At failure of the EPS, the main sources of the off-site power of Kozloduy NPP are the following:

- 1.Getting power from the neighboring EPS;
- 2.Getting power from HPP with capability of "black start";
- 3.Emergency corridors for nuclear and thermal power plants auxiliary power supply (to each facility there is at least two corridors from two independent sources).

3.2.2.6.1.2 Connection to the External Network

Kozloduy NPP is connected to EPS with transmission line of 110, 220 and 400 kV voltage systems.

The number of electricity lines of 400 kV outgoing from the NPP is 8, 2 of which are between the lines to the EPS system in Romania and 6 lines to substations in the country.

The number of outgoing 220 kV lines from the NPP is 5, from which 3 transit lines to substations in the country and 2 radial lines to shore pumping station (SPS).

The number of 110 kV lines outgoing from the NPP is 4, of which 3 are transit lines and 1 is radial line to regional substations in the country.

The NPP is connected to 17 electric wires from which 14 are transit and 3 are radial.

The connection of Units 3 and 4 with the OSY is made by 400kV through double sectioning bus system.

Between 400 kV and 220 kV systems, and between 220 kV and 110 kV systems connections are made through the automatic transformers.

3.2.2.6.2 Distribution of the Plant Auxiliary Power.

3.2.2.6.2.1 Main Cable Routes and Sections

Units 3 and 4 of Kozloduy NPP are connected with external 400kV network. The connection is via power switches (PS) which ensures supplying power to both units from the external network (220kV outdoor switchyard or 400kV outdoor switchyard), and auxiliary transformers. From these transformers 6 kV sections working power is supplied. By startup transformer 3 TP 6 kV sections are powered with backup power. Circuits and equipment are located in the civil structures and protected against internal and external hazards.

3.2.2.6.3 Main In-House Sources of Backup Power

6 kV backup power sections are interconnected and there are operational procedures for supplying power between the units. By means of reducing transformers 6 / 0, 4 kV they supply power to 0,4 kV sections, which are also interconnected and it is possible to supply power to the adjacent unit.

Additional sources of backup power are unit DGs that supply power to uninterruptible power supply sections of second category. Between diesel sections there are process connections and operating procedures, allowing for power supply from an adjacent unit. The ASSGEM has independent EDGs, which are also internal sources of uninterruptible backup power. System description is presented above.

3.2.2.6.3.1 Internal Sources Serve as the First Backup Power in Case of Loss of Off-Site Power

Unit DGs that supply uninterruptible power supply to sections of second category serve as the first source of backup power in case of loss of off-site power. On the NPP site inventories of diesel fuel and motor oil are stored. The inventories ensure operation of all the DGs for at least 4 days.

3.2.2.6.3.2 Independence, Physical Separation of Independent Sources of Structures or Distances and Their Protection against Internal or External Hazards

For Kozloduy NPP Units 3 and 4, each unit DG is located in a separate compartment of diesel generator Station (DGS-II). The compartments are separated by concrete fire resistant walls. The doors between the compartments are fire resistant and automatically sealed.

3.2.2.6.3.3 Time Constraints for Availability of This Sources and External Measures to Extend the Usage Time

At loss of unit off-site power supply of electric power is supplied from the DG of the unit or the neighboring unit, the EDG of ASSGEM or in an extreme case from mobile DG.

3.2.2.6.4 Permanently installed indoor backup power sources

3.2.2.6.4.1 Sources that Can Be Used for the Same Purposes as the Main Backup Source or for Some Limited Goals

At loss of unit off-site power supply power is supplied from the following sources of backup power - DG of the unit or the neighboring unit, the EDG of ASSGEM. Between diesel sections there are process connections and operating procedures, allowing for power supply from an adjacent unit

3.2.2.6.4.2 Independence, Physical Separation of Independent Sources of Structures or Distances and Their Protection against Internal or External Hazards

For Units 3 and 4, each unit DG is located in a separate compartment of diesel generator Station (DGS-II). The compartments are separated by concrete fire resistant walls. The doors between the compartments are fire resistant and automatically sealed.

3.2.2.6.4.3 Time Constraints for Availability of This Sources and External Measures to Extend the Usage Time

At loss of unit off-site power supply of electric power is supplied from the DG of the unit or the neighboring unit, the EDG of ASSGEM or in an extreme case from mobile DG.

3.2.2.6.5 Other Power Supply Sources

3.2.2.6.5.1 Potential Special Connections with Neighboring Units or Other Near Plants

At loss of power supply of one unit, operators are able to supply power to consumers of equipment, ensuring safe storage of fuel in FSP, from the neighbouring unit. In addition, four new DGs of both PSs of ASSGEM (per two DGs for each PS) are installed on the site to ensure performance of safety function of the system regardless of all other power supply systems.

3.2.2.6.5.2 Connectivity of Transportable Power Supply Sources to Supply Power to the Safety Systems

On site a mobile diesel generator is located. It is intended to ensure emergency power supply to the consumers of the safety systems in total and continuous loss of electric power. MDG act as backup DG, which, where necessary, can supply electric power to the consumers of one safety system channel.

The mobile DG is the last means to supply voltage of 6 kV to power category 2UPSS in cases where all other power supply sources cannot not be used. It can be connected to a section through which consumers of equipment, ensuring safe storage of fuel in FSP can be powered.

3.2.2.6.5.3 Information for Each Power Supply Source: Power, Voltage or Other Related Constraints

The rated power of each DG is 1600 kW. The mobile DG has rated power of 1100 kW.

3.2.2.6.5.4 Availability of the Source for Use

At units 3 and 4 detailed operating and emergency operating procedures for actions of personnel associated with the use of each of the power supply sources described above have been developed and put in force.

3.2.2.7 DC Power Supply Batteries

Each channel of the safety systems is equipped with batteries that are independent DC power supply source category 1 consumers of uninterruptible power supply. Additionally, in the DGS for each DG a separate battery is available AB which can back up the main battery of the respective safety system.

Batteries are an independent DC power supply source for category 1 consumers of uninterruptible power supply in station blackout mode. The DC system (DCS) is designed to supply the train for control, protection, automation and signaling.

As a result of modernization of the units new batteries with a capacity greater than the design have been installed. The batteries are located in separate areas independently, and the elements of each system (channel) are grouped in separate rooms.

3.3 Main Data on Units 5 and 6

3.3.1 Main Characteristics of Units 5 and 6

3.3.1.1 Reactor Type

Each unit has installed power of 1000 Megawatt (electrical) and includes the reactor installation with WWER-1000/V-320 reactor, the turbine-generator installation with K-1000/60-1500-2 turbine and TBB-1000-4У3 electrical generator. [1] [2] Two-circuit heat scheme is used.

General characteristics of the units are as follows:

WWER-1000/V-320 is a heterogeneous thermal neutrons water-water energy reactor, of the vessel type. The functions of the reactor within the Nuclear steam supply system (NSSS) are provided and maintained by the controlled chain reaction of fuel decay and transformation of decay energy to heat which is transferred to the primary coolant. Chemically demineralized water with diluted boric acid in it is used as the coolant and the moderator in the primary circuit, with concentration of the boric acid changing during operation.

The reactor core (fuel system) is designed to generate heat and to transfer it from the surface of the fuel elements (FE) and to the primary coolant. The core includes 163 fuel assemblies, each consisting of 312 fuel elements. The primary pressure is 15.7 MPa, reactor inlet water temperature is 288°C.

Low enrichment uranium dioxide is used as a fuel with enrichment of 4.4 % ^{235}U concentration. The primary coolant is heated at passing through the core. After that it is delivered to the steam generators where heat is transferred to the secondary circuit.

The primary circuit which is radioactive, is located in the secured hermetic structure. This consists of the heterogeneous reactor on thermal neutrons, steam pressurizer, four circulation loops including the steam generator, the reactor coolant pump and the circulation line with nominal diameter of 850 mm and the auxiliary equipment.

The reactor, the steam generators and other equipment of the primary circuit as well as spent fuel storage pond are located within the reinforced concrete containment.

Containment structure of the reactor building represents the reinforced concrete cylinder with the inner diameter of 45 m and dome roof designed to withstand internal pressure of 0. 46 MPa. Primary circuit structures, systems and pipelines as well as their related supporting and vital systems are located in the containment.

The secondary circuit is not radioactive. This consists of the steam generators generating steam, the main steam lines, one turbine generator with auxiliary and supporting systems, equipment for deaeration, regenerating heating and supply of the make-up water to the steam generators.

The units output the generated electric power to 400 kV power grid.

Connecting of the unit with the power grid is provided through the Outdoor switchyard. Consumers of the electric energy of the unit are supplied from the auxiliaries system.

Service water supply of the unit is provided from the Danube River through SPS and hot and cold channels. service water to safety systems is supplied from the spray ponds.

Main design parameters and characteristics of Unit 5 and 6 are provided in Table 3.1-1. [7]

The steam generators (SG) are horizontal heat exchangers of surface type and are designed to remove heat from the primary coolant and to generate dry saturated non-radioactive steam within the secondary circuit. Heat exchanging surface of the steam generators represent direct boundary between the primary and the secondary circuits, providing barrier against intrusion of the radioactive products to the secondary circuit.

The pressurizer (Prz) is a component of the reactor primary circuit. This is a vertically located cylinder vessel with two elliptic bottoms and is designed to create pressure in the primary circuit, to maintain pressure within the specified boundaries during the steady-state modes and to limit pressure deviations during transients and emergency modes of the reactor installation via evaporation and condensation of the coolant. With this purpose heating by the electric heaters and cooling by coolant injection to the pressurizer steam part are used.

Reactor coolant pumps (RCP) are components of the reactor primary circuit and are designed to generate circulation of the reactor primary coolant.

The RCP represents vertical centrifugal single-step pumping unit consisting of the hydraulic vessel, output part, electric motor, supports and auxiliary systems.

The main circulation circuit consists of four circulation loops, including main circulation lines (MCLs) connecting the reactor with the steam generators and reactor coolant pumps (RCPs) and is designed to provide for coolant circulation between the reactor and the steam generators.

Each loop has three pipe sections: section between the outlet nozzle of the reactor and inlet collector of the steam generator, section between the outlet collector of the steam generator and the inlet nozzle of the RCP and the section between the outlet nozzle of the RCP and inlet nozzle of the reactor.

3.3.1.2 Heat Power

TABLE 3.3.1.2-1 Main Design Characteristics of WWER-1000/B-320 .

CHARACTERISTICS	VALUE
Reactor heat power	- 3000 MW
Unit electric power	- 1000 MW
Unit efficiency factor (gross)	- 34.17%
Primary circuit pressure	- 15.7 MPa
Secondary circuit pressure (MSH)	- 6.27 MPa

3.3.1.3 First Criticality Date

TABLE 3.3.1.3-1 Historical data of Kozloduy NPP Units 5 and 6

Description	Unit 5	Unit 6
Commencement of constriction	09.07.1980	01.04.1982
Physical start-up – the first criticality	05.11.1987	29.05.1991
Reaching 100% power	21.06.1988	13.08.1992
Acceptance to operation	23.12.1988	30.12.1993

3.3.1.4 Fuel Storage Pond (FSP)

The fuel storage and refueling pond (FSP) serves for storage of spent fuel (until the residual heat pressure reaches permissible level) and for temporary storage of control rods of the reactor control; system (RCPS CRs).

The FSP is equipped with racks for fuel hermetic storage.

FSP ensures storage of the spent FAs for at least three years. [1] [2].

FSP consists of 4 parts – three sections designed directly for storage of FAs and universal slot (US) for fresh and spent fuel handling. US serves for transport operations with transport container TC for the SF, containers with fresh fuel and containers with hermetic casks.

3.3.2 Systems Ensuring and Maintaining Fundamental Safety Functions

The units have been designed in accordance with the defense-in-depth concept where the sequence of barriers prevents from release of radioactive substances and ionizing radiation, with many structures to protect these barriers from damage.

The main principle underlying the design of the unit is that in all operational states and accident conditions it fulfills the following fundamental safety functions:

- reactivity control;
- core heat removal;
- retention of radioactive substances within the established limits.

These functions are performed through normal operation equipment design technical solutions, installation of safety systems, use of effective technologies and procedures for operation and maintenance, based on recent advances in science and technology and internationally recognized operating experience.

In the design of safety related systems and equipment design solutions are used, based on the passive principle of operation, the principle of failure safety and inherent self protection properties (self-control, thermal inertia, reactivity feedback and natural coolant circulation) and other natural processes.

Specific technical solutions applied in the design of safety systems are related to implementing the essential requirements of the regulations: multichannel structure (redundancy), physical separation and diversity.

The multi-channel structure (redundancy) allows the safety systems to perform their functions independently of a possible failure of one channel (single failure).

The physical separation of the channels is achieved by locating every channel in a separate room. This feature allows for successful operation of safety systems in case of failure of one channel due to local events (fire, explosion, high temperature, flooding, etc.).

Diversity of physical principles for performance of safety functions of safety systems is applied in the design through use of both active (pumps, electric valves) and passive devices (pressure tanks, check valves) to eliminate the possibility of failure of all safety systems due to station blackout (electricity, working media, etc.).

The combination of redundancy, diversity and physical separation ensures resistance of safety systems to common cause failures, i.e. total loss of the ability of safety systems to perform their functions.

Safety systems and safety related systems and equipment perform the following safety functions:

- Prevention from unallowable reactivity change – through reactor and turbine instrumentation and control systems;
- Reactor tripping to prevent from anticipated operational occurrence leading to design basis accidents, and to limit consequences of design basis accidents - through reactor

protection and shutdown system with injection of absorbers (control rods of RCPS or boric acid);

- heat removal from the core after breaking the boundaries of the reactor coolant circuit to limit damage to the fuel elements - through the emergency cooldown systems;
- residual heat removal in certain operational states and accidents with reactor coolant circuit preserved - through planned cooldown system;
- heat removal from the safety systems up to ultimate heat sink – through the nuclear component cooling system;
- providing necessary functions to ensure the safety systems – through systems for uninterruptible power supply, maintaining climate and fire protection;
- maintaining acceptable tightness of the fuel cladding in the core - by controlling the FA at refueling, and keeping the activity of the primary circuit during operation within acceptable limits;
- maintain the integrity of the borders of the primary circuit - through the system of surveillance and maintenance as well as diagnostic systems for leak monitoring, cycles of loading and moving;
- restricting discharges of radioactive substances from the containment of the reactor system during and after the accident – through the cladding control system, discharge control system, ventilation system for beyond design basis accidents;
- limiting the exposure of personnel and the population during and after design basis accidents and selected severe accidents with release of radioactive substances from sources outside the containment of the reactor system - developed emergency plan and accident management guidelines;
- limiting the discharge of liquid and gaseous radioactive substances within the specified limits under all operating conditions - introducing a system of control of radioactive waste;
- maintaining the environmental conditions necessary for the operation of safety systems and personnel work when performing safety critical operations - through conditioning and ventilation systems;
- control of radioactive discharges in the transport and storage of spent nuclear fuel outside the core, but within the unit under all operating conditions - through the nuclear fuel handling procedures and equipment;
- removal of residual heat from the spent fuel stored outside the core, but within the unit – through fuel near-reactor storage system - TG;

- maintaining subcritical state at the fuel storage outside the core, but within the unit - through fuel near-reactor storage system - TG;
- prevent or mitigate the consequences of failure of equipment, the inoperability of which could cause safety function failure – by applying the principles of redundancy, diversity and physical separation.

Resistance to loss of power is ensured by the following design characteristics:

- Provision of 2 (for category 2 power supply) or 3 (for category 1 power supply) different power supply sources to each safety system: from auxiliary transformers, from diesel generators and batteries;
- Each unit (5 and 6) is provided with four diesel generators (3 from the original design and one installed during the modernization program) located in separate premises;

Availability of devices with passive action (gravity operation of emergency protection, hydro accumulators, relief valves that allow the most important safety systems for performing their functions even at station blackout.

Resistance to external natural and human-induced effects is ensured by the following design characteristics:

- Equipment of all safety systems is located in a seismically qualified buildings providing protection against external natural and human-induced impacts - seismic qualification of buildings containing safety systems equipment is examined for modified seismic conditions of the site;
- Qualification of equipment of all safety systems against seismic impact. Qualification of equipment required for safe shutdown is reviewed;
- Qualification of equipment of all safety systems for resistance to conditions expected in the premises in case of design basis accident;
- Location of electrical equipment of safety systems at levels that provide protection in case of flooding of the room;
- In implementing the modernization program are additional limiters installed on steam lines to prevent damage to adjacent safety systems equipment at steam line rupture.

Sustainability of safety against error operator action is based on the following general features of the design:

- Automatic actuation of systems due to deviation of critical parameters against predefined values;
- Automatic ban of ability of the operator to influence the functioning of safety systems as critical parameters reach certain values;

- Safety analyses postulate a complete lack of operator actions within the first 30 minutes of any design basis accident.

3.3.2.1 Reactivity Control

3.3.2.1.1 Systems ensuring core sub-criticality

Reactivity control both during reactor operation and in shutdown mode is performed by two independent principle of action:

- By moving RCPS CR by the height of the core.
- By changing the concentration of dissolved boric acid (H_3BO_3) in the primary coolant;

Reactivity control systems based on the use of RCPS CR and the boron injection in the coolant are designed in such a way that postulated accidents related to their failure do not lead to reactivity higher than the specified level and provides for:

- non-exceeding of the limits for fuel;
- primary circuit integrity;
- effective core cooling in accident mode.

3.3.2.1.1.1 Reactor Control and Protection System (RCPS)

Reactor control and protection system has the following functions:

- reactor emergency and preventive protection;
- reducing and limiting the reactor power at non-scheduled tripping of the equipment of the power unit according to predefined list and set program;
- reactor power automatic control according to the specified programs;
- monitoring of process parameters necessary for the reactor operation protection and control;
- fixing the root cause of protection actuation;
- providing the information to the operator and issuing signals to other subsystems of APCS.

Reactor control and protection system actuator is RCPS drives system, which serves to move the control rods and connects 61 actuators ШЕМ-3.

The emergency protection (EP) is ensured based on passive principle through de-energizing the systems maintaining the RCPS drives in the upper position, with their subsequent falling into the core under the effect of gravity.

Effectiveness of the emergency protection is selected based on the need to compensate for abrupt changes in reactivity associated with bringing the reactor to sub-critical condition of any power level, and considering the reduction in efficiency due to postulated sticking in the extreme upper position of one of the most effective RCPS CRs and use the rest for operational control and balancing the energy release field providing sub-criticality of not less than the minimum allowable (required) value equal to 0.01 Δk/k, i.e. 1%.

The design provides for movement of RCPS CR at a constant speed of 2 cm/s and drop them at the emergency protection signals within max. 4 s.

3.3.2.1.1.2 Boron Control Systems

Chemical reactivity control is through the process of boron control through changes in concentration of boric acid (H_3BO_3) in the coolant. After tripping the reactor by boron control reactivity change associated with dissolution of xenon and the water cooldown to the cold state is compensated, and also the necessary sub-criticality is provided for refueling, which is not less than 2% (with all RCPS CRs extracted).

In normal operation mode to increase the concentration of H_3BO_3 in the primary coolant primary circuit makeup/blowdown system is used by supplying boron solution with concentration of 40 g / kg of boron concentrate of Boron concentrate system.

In scheduled cooldown the primary circuit filling modes the balance of the primary circuit coolant system is maintained by Reactor Hall Boron Containing Water System, which supplies boric acid solution with concentration of not less than 16g/kg to the makeup/blowdown system.

In accident situations to increase the concentration of H_3BO_3 in the primary coolant high pressure emergency boron injection system and medium pressure emergency core cooling system are used.

In loss of coolant accidents the emergency and normal core cooling system, and ECCS passive part ensure sub-criticality of the core by injection of boron solution with concentration of 16 g/kg.

3.3.2.1.1.2.1 Makeup/Blowdown System

Makeup/blowdown system is a normal operation system. Its main functions related to the boron control are:

- Filling the primary circuit with boric acid;
- Maintaining the material balance of the coolant in the primary circuit (maintaining the primary circuit pressurizer level at the given reactor power);

- Degassing and return to the primary circuit of controlled drainage and compensation of possible small leaks from the primary circuit in normal operation and at normal operational occurrences;
- Compensation of slow changes in reactivity from burning and intoxication of the fuel by changing the concentration of boric acid in the primary circuit at startup and at load change;
- Create the necessary sub-criticality at reactor shutdown;
- Maintaining water-chemistry indicators in accordance with the standards in all modes of the unit normal operation;
- The system includes the following functional subsystems:
- Deaeration of the primary circuit makeup and blowdown water;
- Makeup units- 3 pcs.;
- Main lines for makeup -;
- Sealing water supply to RCP and the independent circuit;
- Release of water from RCP seals;
- Deaeration and supply of distillate (boron control);
- Primary circuit blowdown -.

3.3.2.1.1.2.2 Boron concentrate system

The system is designed to supply boron concentrate on the suction side of makeup pumps in the mode of increasing concentration of boric acid in the primary circuit for compensating changes in reactivity during startup and shutdown of the unit, and also for filling up of boron concentrate tank of the safety systems Medium Pressure Emergency Core Cooling System and High Pressure Primary Circuit Emergency Boron Injection System.

Boron concentrate system is constantly available for injection of boron concentrate into the primary circuit (in backup). Pumps are in backup and, if necessary, supply boron concentrate on the suction side of makeup pumps. Boron concentrate system is to be brought to backup condition before the core fueling.

The system has two tanks with effective volume of 192 m^3 each. Total volume of boron solution concentration H_3BO_3 greater than or equal to 40 g/kg during operation of the unit is maintained not less than 200m^3.

3.3.2.1.1.2.3 Reactor Hall Boron Containing Water System

The system is designed for:

- Intake of boron containing water in tanks;

- Filling primary circuit with boric acid solution;
- Supplying boron containing water to system AWTS-6 for concentration;
- Filling EMT with pump;
- Filling up of FSP with boric acid solution;

Reactor Hall Boron Containing Water System system has two tanks with effective volume of 468 m^3 each.

In the mode of bringing the reactor to a criticality one tank shall have boron solution with concentration of H_3BO_3 greater than or equal to 16 g/kg and volume of more than 250m^3 and the second tank must have free volume not less than 300m^3 for intake of the coolant blowdown at water exchange.

Before the start of operations for the units scheduled tripping in first tank there shall be more than 250m^3 of boron solution with H_3BO_3 concentration greater than or equal to 16 g/kg.

3.3.2.1.1.2.4 High Pressure Primary Circuit Emergency Boron Injection System

The system is designed for emergency supply of highly concentrated solution of boron in the primary circuit during operating conditions and emergency conditions associated with the separation of positive reactivity in the reactor core during storage of high pressure in the first loop (90-180 kgf/cm^2).

The system consists of three independent channels, each channel providing for 100 percent of the system functions. The main elements of each channel are:

- 15 m3 tank with boric acid (H_3BO_3) with a concentration $\geq$ 40 g/kg ;
- Pump high pressure for boric acid injection with throughput 6.0 t/h and nominal discharge pressure of 15.69 MPa;
- pipelines and valves.

3.3.2.1.2 Systems Ensuring FSP Sub-Criticality

3.3.2.1.2.1 Fuel Storage Pond

FSP ensures sub-criticality of the spent fuel is not less than 5% by:

- reliable isolation of the storage cell throughout the life of the FSP;
- effective use of heterogeneous sinks not losing their properties in the process of the operation. The cells for FAs are made of borated stainless steel. Absorbing capacity (at burning B^{10} and reducing the wall thickness due to corrosion) of the racks cells is maintained throughout its life;

- using homogeneous sink in the water of the FSP with boric acid solution with concentration of 16g/kg.

FSP level and water-chemistry of maintained by FSP water treatment system AWTS-4.

At fuel storage the level is maintained at elevation +28.8 m. The axis of the exhaust pipe is located at elevation +28.1 m, and the heads of the assemblies are at elevation +25.9 m. Pressure pipes practically enter the floor of fuel compartments and are equipped with a device for siphon passive failure at the overflow height.

In the water level reload mode elevation of +36.2 m is maintained and an additional volume of water is needed that is provided by FSP draining and filling system AWTS-4.

3.3.2.1.2.2 Fuel Storage Pond Water Treatment System

Fuel Storage Pond Water Treatment system is common for units 5 and 6 and is designed for:

- Draining and filling of the FSP.
- Purification of H_3BO_3 solution of FSP.
- Purification of H_3BO_3 solution of ponds for emergency makeup.
- Adoption of low concentration H_3BO_3 from Reactor Hall Boron Containing Water System for concentrating.
- Storage and purification of boric acid solutions of Units 5, 6 and AB-3.

The system consists of two main parts:

The first part: System for intake, storage, correction of the concentration and transportation of the solutions to the components.

The second part: to purify the solution by mechanical and ion-exchange filtration.

System consists of:

- Filtration line with 2 mechanical filters, a cationite filter, an anionite filter and a trap;
- 4 tank for boron injection with a capacity of 400 m3 each;
- 3 pumps for boron solution.

3.3.2.2 Reactor Heat Removal – Ultimate Heat Sink

Main ultimate heat removal from the reactor to units 5 and 6 is provided by water of the Danube River. At loss of the main ultimate heat sink systems are provided to remove heat to the alternative ultimate heat sink which is the atmosphere.

Heat removal to main ultimate heat sink is ensured from circulating water system.

Heat removal to alternative ultimate heat sink is ensured from group "A" service water system. The system circulates in closed circuit and transfers heat from the reactor to the atmosphere by spray ponds.

In hot condition the heat from the steam generators to the reactor is removed BRU-K for the turbine condensers that are cooled by system.

With the reactor cooled down (temperature lower than 90^{o}C) heat removal from the reactor takes place via the Low Pressure Emergency and Normal Core Cooling System by group "A" service water system to spray ponds. In this case, is the ultimate sink is the atmosphere.

In the process of the reactor cooldown within the temperature range from 90 to 150^{o}C, both ultimate heat sinks can be used.

In the modes of loss of the main ultimate heat sink, heat is removed from the reactor to the atmosphere through the steam generators and group "A" service water system.

Reactor heat removal to alternative ultimate heat sink by steam generators are ensured through the steam discharge into the atmosphere through BRU-A and SG makeup from the emergency feedwater system.

3.3.2.2.1 Existing Channels and Means for Reactor Heat Removal by the Ultimate Heat Sink

3.3.2.2.1.1 Low Pressure Emergency and Normal Core Cooling System

Low Pressure Emergency and Normal Core Cooling system combines the functions of protection system and safety system of normal operation. The system is designed for:

- emergency cooling of the reactor core and subsequent continuous removal of residual heat from the core under emergency conditions associated with leakage from the primary circuit, including MCL rupture (DN 850) in the full section with smooth two-side coolant leakage when providing injection of boric acid solution of concentration more than 16 g/kg to the primary circuit with flow-rate of $250 \div 300$ m3/h at pressure in the primary circuit ≤ 2.16 MPa and $700 \div 750$ m3/h at pressure in the primary circuit 0.098 MPa;
- primary circuit normal cooldown during the reactor outage and residual heat removal from the core for maintenance and refueling;

The system consists of three independent channels, each of the channels providing for 100% of the system functions. The main elements of each channel are:

- Emergency cooldown pump
- Heat-exchanger for normal and emergency cooldown

- Emergency makeup tank (EMT) with volume of 707m3, common to all three channels.

EMT is a compound part of the floor of the containment and is filled with boron solution with concentration of 16 g/kg. The upper part of the tank has a general ceiling, which has three openings with an area of 1 m^2. These openings are located in the lowest elevation of the containment and through them leakage merging of the primary circuit water in an emergency is ensured.

The volume of the EMT is set to compensate for the loss of water supplied from the emergency pumps to the containment and ensure sufficient cavitations inventory for the normal operation of the safety systems pumps.

3.3.2.2.1.2 Medium Pressure Emergency Core Cooling System

The system is a protective safety system designed for emergency injection to the primary circuit of H_3BO_3solution with concentration ≥ 40 g/kg in emergency conditions maintaining high pressure in the primary circuit and also to compensate for leaks in the modes of loss of primary circuit tightness.

The system consists of three independent channels: each channel provides for 100% of system functions.

Each channel of the system consists of:

- 15 m3 tank with H_3BO_3 solution with concentration ≥40 g/kg;
- Pump;
- pipelines and valves.

Each channel is connected to the emergency boron solution pond (EMT).

3.3.2.2.1.3 Passive Emergency Core Cooling System (ECCS)

The system is designed to provide flow to the core during large LOCA if primary pressure decreases below 5.9 MPa (60 kgf/cm^2).

Hydraulic accumulators boron concentration of 16 g/kg ensures reactor core criticality.

The system is designed in such a way that boric acid solution volume in the three hydraulic accumulators is sufficient at occurrence of two-side rupture of the MCL for the core cooldown before the active part of the ECCS is actuated.

Passive Emergency Core Cooling system includes four independent loops, each consisting of:

- One hydraulic accumulator with water volume of 50 m^3;
- Connecting lines to the valves (con. diameter 300 mm);

- Per two isolating valves and two check valves on each pipeline.

3.3.2.2.1.4 Steam Generators Emergency Feedwater Supply System

Steam generator emergency feedwater supply system is a protective safety system designed to supply feedwater to the steam generators during accident conditions related to the loss unit of off-site power supply and at failure to normal feedwater supply to the steam generator.

The system consists of the tree independent channels. Each channel includes:

- One emergency feedwater pump;
- One chemically demineralized water inventory tank with capacity of 500 m^3;
- pipelines and valves.

The first channel can supply water to all the steam generators.

The second channel can supply water to steam generators 1 and 4.

The third channel can supply water to steam generators 2 and 3.

3.3.2.2.1.5 Steam Generators Alternative Make-Up System

SG alternative make-up system is designed to ensure feedwater supply to the steam generators at condition of total loss of the alternate current sources with the purpose of ensuring reliable core decay heat removal during at least 24 h. [1][2]

Power supply to the pump and I&C is provided from the mobile diesel generator.

Alternative make-up of the SG consists of:

- Demineralized water source – Steam Generators Emergency Feedwater Supply system tanks;
- Pump;
- pipelines and valves.

Suction pipeline of the pump is connected through the connector to the suction pipeline of first channel of Steam Generators Emergency Feedwater Supply System pump. Connection of the pump is provided through one manual valve.

Pressure pipeline is connected to the pressure header of first channel Steam Generators Emergency Feedwater Supply System pump. Connection to the pump is provided through two sequentially located manual valves.

3.3.2.2.1.6 Secondary Circuit Overpressure Protection System

The system is designed to protect the secondary circuit against pressure increase above the permissible level using the following:

- Protection of the secondary circuit against overpressure (BRU-A and the safety valve located in part of the main steam lines);
- Isolating of the specific SG in case of pipeline rupture within the boundaries of the reactor building as well as the turbine hall (check valves, isolating valves located in those parts of the steam lines and feedwater pipelines).

There are two safety valves located on the pipelines of each steam generator – one control valve and operating that protect steam generator vessels and the steam lines against excess pressure increase during accident conditions. After the safety valves, BRUA device is located on the streamline, and then MSIV and the check valve are located sequentially.

Two isolating valves are mounted at the feedwater line to each SG as well as two sequentially located check valves.

At pressure increase over 74 kgf/cm^2 the valves are forced to open due to actuation of interlock and actuators controllers operation. After pressure decrease below 74 kgf/cm^2 the controllers start to maintain pressure of 68 kgf/cm^2 with uncertainty 0.5 kgf/cm2.

At pressure decrease to 64 kgf/cm^2 BRU-A completely closes.

To ensure controlled steam relief from the steam generator to atmosphere BRUA fast acting atmospheric reduction valves are mounted. BRU-A and safety valve are used to cooldown the primary circuit at turbine trip and absence of availability to supply steam to the condensers.

Modernization of the system has been completed at which the design safety valves of the SG of Kozloduy NPP Units 5 and 6 were replaced with the new-design safety valves qualified for operation with steam-water media and possibility of manual control within the entire range of SG pressure change.

3.3.2.2.1.7 Condenser Steam Removal Systems

Condenser steam removal system is a normal operation safety related system performing the following functions:

- Limiting main steam pressure in the event of turbine or generator tripping
- Main steam pressure control in unit startup mode
- Unit cooldown speed control in shutdown mode

Bypass drainage system of main steam from main steam header to the condenser includes:

- BRU-K control valves;
- Steam intake devices for condenser steam intake and temperature decrease;
- Valves on the main condensate injection line in the steam intake devices;
- Filters on the main condensate injection line in the steam intake devices

3.3.2.2.1.8 Group A Components Service Water Supply System

Nuclear components service water system is a supporting safety system designed to remove heat from the cooling equipment during all operational modes and emergency conditions. Each of the system trains is supported by two-section spray ponds which represent the ultimate heat sink of core heat. Each train of the system can independently perform core cooling function. Water in the system circulates via the self-contained circuit. Initial filling and replenishment for lost fluid of the system are provided through make-up from the two CPS 3 and 4 (circulation pumping stations of Units 5 and 6), that are used as water source available to the site (in the cold channel). For standby water supply to the spray ponds SPS (shaft pumping stations) are used through underground metal pressure pipeline.

The system consists of two parts with different designations:

- in the Reactor building;
- outside the Reactor building.

Part outside the Reactor building supplies cooling water to the following equipment and systems:

- Cooling of the diesel generators (DG);
- Ventilation systems of the diesel generators;
- Conditioners in the diesel generators rooms;
- Compressor plant systems.

Part in the Reactor building supplies cooling water to the following equipment and systems:

- Protective safety systems equipment;
- Supporting ventilation systems equipment;
- Spent fuel pool cooling.

3 independent channels are mounted on each unit for service water supply to the reactor building to ensure operation of the plant during normal operational mode, planned cooldown mode, as well as accident mode.

3.3.2.2.1.9 Circulating Water System

System for circulating (main cooling) water is a normal operation system providing heat removal from the condensation of steam from the turbine condensers, condensers of turbo drives of feedwater pumps and also providing distribution of water to steam ejectors to vacuum system of the turbine condensers and to the unit transformers and other in the turbine hall.

It consists of 6 circular lines, per one for each condenser section, each of which includes a circulating pump and circulation water ducts.

The work of the primary cooling system is a DC scheme with water intake from the River Danube through the cold channel and hot water discharge in the River Danube by the warm channel.

3.3.2.3 FSP Heat Removal by the Ultimate Heat Sink

3.3.2.3.1 Heat Removal Means

Fuel in the FSP is cooled by FSP Cooling System in all design modes except for the isolation of the containment system SHO (CS).

3.3.2.3.1.1 FSP Cooling System

System consists of three channels and includes three pumps for the pond cooling, three heat exchangers on the suction side of each of the pumps, pipelines and valves. The channels are interconnected with connections of suction and pressure pipelines, which enable switching from one channel to another in case of failure to one of the channels. The pressure and suction lines have per three localizing fast acting valves, one of which is located in a tight casing. System heat exchangers are cooled by the Group A Components Service Water Supply System and each channel is cooled by a separate channel of Group A Components Service Water Supply System.

Productivity of each of the three channels of the system is such that each channel can independently ensure removal of residual heat from the pond in all modes of the system operation.

Each channel of the FSP Cooling System provides recirculation of water in the pond with a flow-rate of 300 to 600 m^3 h and a temperature not exceeding 45°C which corresponds to the maximum temperature of cooling water from Group A Components Service Water Supply System - 33°C.

The fuel storage pond is always filled with boric acid solution with nominal level and temperature.

In the storage mode the fuel has sufficient reserves to allow in case of failure of one channel for ensuring a reliable removal of residual heat from the fuel. The backup channel is started remotely by the operator.

In accident mode with failure to all channels of the cooling system of the FSP, removal of residual heat of the fuel stored is ensured by the evaporation of water from the pool. To prevent unacceptable level decrease in the fuel compartments and uncovering of the fuel at isolation of the confinement system emergency makeup by Spray System from EMT is used. The emergency makeup line is connected remotely by the operator.

3.3.2.4 Containment Heat Removal to the Ultimate Heat Sink

The last barrier against fission products release to the environment containment system. It performs the following functions:

- Prevention or localization of the radioactive substances released during accident conditions within the boundaries of the confinement;
- Limitation of ionizing radiation exposure.

Total free volume of the CS is approximately 60 000 m^3. The system consists of the following elements:

- Reinforced concrete enclosure structures of the containment including containment prestressing;
- Hermetic metal lining;
- Locks for ensuring radiation safety of the transportation communications connecting the hermetic area of the reactor building with the environment.
- Localizing valves designed to ensure tightness of the containment system by isolating the pipelines connecting the systems and their components located in the containment with the systems and components located outside the containment.
- Hermetic penetration ensuring tight isolation of the containment from such communication lines as electric cables, pipelines, ventilation lines, ionizing chambers trains;
- Localizing valves;
- Sections of the process pipelines performing functions of the containment components.

The containment is made of the prestressed reinforced concrete in the form of the cylinder. The reinforced structure of the containment is designed to withstand accident loads using special rope system within which stressing is provided for this structure is designed to withstand 5 kgf/cm2 within the containment.

The containment structure is designed to withstand shock wave with direct pressure of up to 0.03 MPa and duration of compression phase 1 s.

Internal part of the containment is lined with the metal lining to meet design requirements to the leak-tightness.

Layout of the internal structures of the containment is provided in such a way that this protects the containment system against impacts of jet, shock waves, missiles and other accident conditions.

In the framework of implementation of units 5 and 6 modernization program control system for concentration and binding and recombination of hydrogen has been mounted.

The system for measuring hydrogen serves as a continuous and on-line measurement of local hydrogen concentrations at various locations in the CS during operation and after loss of coolant accident (LOCA).

Hydrogen binding and recombination system serves to reduce the concentration of hydrogen in the CS during and after loss of coolant accident (LOCA).

Hydrogen binding and recombination system is designed to limit the concentration of H_2 to 4% vol. To avoid an explosion of hydrogen in the containment during LOCA.

3.3.2.4.1 Heat Removal Means

3.3.2.4.1.1 Spray System

The system is designed to localize accidents through condensation of the evaporated part of the coolant released from the containment.

The system performs also the following functions:

- Cools and reduces pressure in the containment to atmospheric level through cold borated water injection to the CS;
- Emergency make-up of the FSP in case the FSP cooling system fails.

The system consists of three independent trains for supply of the spray solution to the containment system, and each train consists of:

- Concentrated boron solution tank;
- Pump;
- Water jet pump;
- pipelines and valves

3.3.2.4.1.2 Pressure Reduction Filter System

The system has been installed under the program for modernization of units 5 and 6. The reason for placing the system is to increase the safety level of units in beyond design basis accidents.

PRFS system mitigates consequences of a severe accident preventing from overpressure in CS.

The PRFS system consists of:

- Venturi type scrubber with integrated filter with metal threads and Venturi type nozzle;
- inlet pipeline of PRFS system;
- tear-off membrane, designed to break at an absolute pressure 0.5 MPa, i.e. near the design pressure in the CS;
- CS isolating valves, which are normally closed;
- PRFS system outlet pipeline.

3.3.2.5 Location, Time Constraints and Power Supply Sources for Heat Removal Channels

3.3.2.5.1 Heat Removal Channels Location

Equipment and pipelines of all active safety systems designed to cool the reactor, the fuel storage pond and CS are physically separated and protected against external influences in the reactor building premises.

Channels of nuclear component service water supply system are spatially separated.

The spray ponds are located to the west of the main buildings of units 5 and 6. Underground pipelines from the ponds to the pump are traced to the appropriate DGS.

The pumps are located in the respective DGS - two pumps (operating and backup) in one DGS cell.

Pressure pipelines are traced to and from the cooling equipment.

3.3.2.5.2 Time Constraints for Operation of the Various Heat Removal Channels

There are no restrictions on operation time for different channels for normal heat transfer by the main ultimate heat sink.

In accident modes at loss of off-site power heat removal by the alternative ultimate heat sink is via Group A Components Service Water Supply System whose spray ponds (SP) have maximum volume of 8250 m^3 each. The volume of water in each SP is sufficient to supply water cooling systems for 30 hours without replenishment. The minimum amount of water in each SP is 1650 m^3.

3.3.2.5.3 AC and DC Power Supply Sources for Each Channel

Power supply to the active components for each systems safety channel is ensured by the channel of I and II category uninterruptible power system.

The diesel generator is designed to operate at nominal load without maintenance for 240 hours if possible to fill the tank for fuel storage.

The capacity of the fuel storage tank is $100m^3$ and it allows for 72 hours' operation at full load.

3.3.2.5.4 Cooling of Equipment to the Heat Removal Channels

Long-term heat removal from the equipment of the channels is ensured through the operation of ventilation systems for the premises, the most important of these are powered from category 2 uninterruptible power supply system sections.

3.3.2.6 AC-DC Power Supply.

3.3.2.6.1 Off-Site Power

Off-site power for Kozloduy NPP Units 5 and 6 is supplied through OSY 400 and 220 kV.

3.3.2.6.1.1 Reliability of the Off-Site Power

The connection of the units with the OSY is on 400kV side of Unit transformers. To increase the reliability a generatior circuit breaker is installed (HEC 7), as well as auxiliary transformers are connected between the switch and the unit transformers. Thus, auxiliaries of the unit can be powered by house loads transformers with the generator disconnected. Connections between the elements of the main unit circuit and the variation of house loads transformers are made using busbars and 24kV. In the variation of house loads transformers there is no commutation unit to increase reliability.

To increase reliability and reduce the likelihood of a total deenergizing of 400 kV switchyard, the system is designed as a sectioning dual bus system.

The two back-up power transformers to each of these two groups should work together, i.e. both backup power supply transformers in the group are also considered an element in terms of reliability.

On outdoor switchyard between 400 kV and 220 kV systems, and between 220 kV and 110 kV systems connections are made through the automatic transformers.

At failure of the EPS, the main sources of the off-site power of Kozloduy NPP are the following:

- 1.Getting power from the neighboring EPS;
- 2.Getting power from HPP with capability of "black start";
- 3.Emergency corridors for nuclear and thermal power plants auxiliary power supply (to each facility there is at least two corridors from two independent sources).

3.3.2.6.1.2 Connection to the External Network

Kozloduy NPP is connected to EPS with transmission line of 110, 220 and 400 kV voltage systems.

The number of electricity lines of 400 kV outgoing from the NPP is 8, 2 of which are between the lines to the EPS system in Romania and 6 lines to substations in the country.

The number of outgoing 220 kV lines from the NPP is 5, from which 3 transit lines to substations in the country and 2 radial lines to shore pumping station (SPS).

The number of 110 kV lines outgoing from the NPP is 4, of which 3 are transit lines and 1 is radial line to regional substations in the country.

The NPP is connected to 17 electric wires from which 14 are transit and 3 are radial.

The connection of Units 5 and 6 and the OSY is made by 400kV through the unit transformers, through double sectioning bus system.

To 220 kV OSY two groups per two backup power supply transformers 220/6/6 kV, 63/31.5/31.5 MVA are connected.

Between 400 kV and 220 kV systems, and between 220 kV and 110 kV systems connections are made through the automatic transformers.

3.3.2.6.2 Distribution of the Plant Auxiliary Power

Auxiliaries of units 5 and 6 of the NPP during normal operations is provided from 24 kV busbar after the generator circuit breaker HEC-7 through the auxiliaries transformer with 24/6/6 kV- 5(6) uncoupled coils, 63/31.5/31.5 MVA power, to supply power to normal operation sections. For the Unit normal operation, simultaneous operation of both house loads transformers auxiliaries transformers is required, i.e. both auxiliaries transformers of each Unit are considered a single element from reliability point of view.

At loss of power to house loads transformers auxiliaries transformers, automatic switchover to back-up equipment is used to supply power from the group of start-up standby transformers from 220 kV Outdoor switchyard depending upon ASBE selected priority.

The backup unit busses are powered by a 220 kV switchyard though backup power supply transformers with power 2x63MVA, which is intended to transform the voltage from 220 kV to 6.3 kV and to provide the unit auxiliary backup power.

6 kV and 0.4 kV switchgears are designed for distribution of electricity from power transformers to the NPP auxiliary consumers.

Connections of 400 kV and 220 kV are implemented by overhead lines connecting the transformer sites of units 5 and 6 with OSY.

Transformers at the site are separated by protective walls and have an automatic spray system.

The system of 24 kV bus is designed in the form of busbars with gas-insulated installation.

Connections of 6 kV and 0.4 kV are made through cables located in protected corridors and cable channels with automatic fire pits.

NPP power supply auxiliary loads are divided into 3 categories according to types of electricity consumed by them and the reliability of their power:

Category 1 consumers are consumers of alternate and direct current, not allowing according to safety provisions for interrupting power supply for more than part of a second (one semiperiod of 20 ms) during all modes including complete loss of alternate current voltage from operational and standby auxiliaries transformers (blackout mode), requiring mandatory availability of power supply after the reactor emergency protection has actuated;

Category 2 consumers are consumers of alternate and direct current, having higher requirements to reliability of the power supply and allowing for interruption of power supply for the period specified according to the requirements of nuclear safety (15 s to 1 min – time required for actuating the DG and automatic connection of the TLS);

Category 3 consumers are consumers of the alternate current not having higher requirements to availability of the power supply as requirements to the power supply to the essential loads of conventional thermal power plant.

For category 3 consumers normal operation system represents the only electric power supply source. These consumers include the NPP auxiliaries mechanisms powered from 6 kV и 0.4 kV sections and their sets in the main building.

3.3.2.6.3 Main In-House sources of backup power supply

3.3.2.6.3.1 Uninterruptible Power Supply System to Category 2 Auxiliaries Consumers

Uninterruptible power supply system to Category 2 auxiliaries consumers.

The operating source of uninterruptible power supply is category 3 busbars 6 kV normally energized from the EPS.

Emergency power supply to category 2 safety systems at loss of voltage to the normal operation busbars is provided from the diesel generator stations.

At occurrence of accident, signal to start DGs is delivered independently on each diesel generator. Connection of the DG to category 2 busbar is provided during longer time than time required to actuation of the ASBE of the unit busbars 6 кV (1.2 s).

Load developing from the DG is provided Automatically step by step (TLS).

DG power selection is provided considering the worst combination of the accident situation. Power of the start-up step shall not exceed maximum permissible power of the DG.

DG is in continuous hot standby mode to be ready to automatic startup and accept the load.

The DG is designed to work at nominal load without maintenance during 240 hours.

Category 2 uninterruptible power supply system (SNEZ-II) is designed to supply power to the safety systems.

3.3.2.6.3.2 Uninterruptible Power Supply to Category 1 Auxiliaries Consumers

Category one consumers are divided into the consumers of direct and alternate current.

At normal operation of the unit these consumers are energized from the rectifier of the correspondent direct current board (DCB), and if this looses power - from the related battery at the DCB.

DCB energize the related devices for the relay protections, control and alarm, automatic devices for safety systems automatics, automatics and alarm on fire annunciation and fire extinguishing, emergency lighting, etc.

Uninterruptible power supply to category 1 consumers from the uninterruptible power supply system (SNEZ-I) is provided through uninterrupted power supply units (UPSU).

UPSU is designed for:

- Power supply with direct and alternative current to category 1 consumers during normal operation of the plant, as well as during accident modes including station blackout;
- Recharging of the accumulators during normal operation.

3.3.2.6.4 Permanently Installed Additional Backup Power Sources

At loss of off-site power supply units 5 and 6 have common unit diesel generator stations, which according to the procedure supply power to normal operation sections and part of normal operation equipment which is necessary to start the reactor cooldown mode.

Additional DG unit has rated power 5.2MW, its purpose is to ensure power supply to the unit consumers at failure to the network, and also to ASBE of the first and second backup power supply. The capacity of the fuel storage tank is 100 m^3 which allows for operating for 72 hours at full load.

3.3.2.6.5 Other Power Supply Sources

To ensure emergency power supply to safety systems loads at total and continuous loss of auxiliary voltage mobile DG (MDG) is provided. Due to its mobility the MDG performs the functions of emergency DG that, if needed can ensure electric power to a part of the safety systems loads.

Mobile DG is a last means to supply 6 kV voltage for category 2 UPSS power supply where all other sources of power cannot be used.

3.3.2.7 DC Power Batteries

The batteries are a compound part of the UPSS providing continuous power to category 1 consumers of safety systems.

In each channel of category 1 uninterruptible power supply systems one battery is mounted without the element switches.

The batteries operate under constant load, such as charging and recharging provided by rectifiers of UPSS. (See t.3.3.2.6.3.2).

The batteries are equipped with 106 HCP-23 type cells connected in series.

The capacity of one battery is 927A/h for 10 hours. Batteries are discharged during 3 hours with power 305a or a real load (inverter) to 1,8 V for the first two weakest elements (according to the requirements). When performing real tests on unit 5 of battery of the unit safety system channel 2 with all possible loads connected in station blackout mode, the battery supplied power to those loads for 10 hours and 18 minutes reaching voltage 191 V. The battery should start automatically on 180 V and the performed extrapolation estimated that the battery can operate for up to 11 hours and 32 minutes.

The recommended recharging voltage for battery HCP23 is 2,25 V/el. This voltage is applied to all temperatures in the range of 5°C to 35°C.

Battery is equipped with 106 pcs. Rechargeable Vb-2412 type elements connected in series mounted on seismic metal racks. The rack is equipped with 10 VB2412-type batteries.

Battery capacity is 1200 A/h with 10 hour discharge to 1,8 V of the element at temperature of 20°C.

The actual load of the system is approximately 250 A battery which can be discharged to a final element minimum voltage of 1,75 V.

The recommended charging/recharging voltage is 2.23V/cell or 2.25V at 20°C.

3.4 Main Data for SFSF and SFDSF

3.4.1 Main Characteristics of the Spent Nuclear Fuel Storage Facility

Spent nuclear fuel storage facility (SFSF) of Kozloduy NPP is designed for temporary storage under water of spent nuclear fuel (SF) of WWER-440 and WWER-1000 for 10- year period of operation of all units of Kozloduy NPP as well as for execution of transportation and technological operations related to its acceptance its loading in the storage compartments, storage and transportation from the SFSF in compliance with the safety requirements. Storage facility is an independent building on the NPP site where a pond with four storage compartments is located with the respective transportation and technological equipment needed for handling of SF.

Nuclear safety during SF storage is provided by:

- restriction of the step of arrangement of the assemblies in the containers ("baskets"), racks, packing;
- use of heterogeneous and homogeneous absorbers and control of their absorbent capacity or their concentration;
- control of the burning in depth of the NF when used as a nuclear safety parameter;
- control of the arrangement of the assemblies and absorbers;
- control of the technological parameters of the complex of systems for storage and handling of NF.

3.4.1.1 Location and Properties of the Site

SFSF building is located in the South-Western part of the site of Kozloduy NPP, Southern from the Auxiliary Building 2.

All site characteristics important for the buildings and facilities of Kozloduy NPP are valid also for the SFSF.

3.4.1.2 Main Design Solutions

3.4.1.2.1 Conceptual Solutions

Design concept foresees construction of a space filled in with water where the containers with spent fuel assemblies are stored. Water is used both as a biologic shielding and for cooling of SF. There is a structure constructed around the reactor pond allowing execution of the transportation and technological operations with SF and protection of both the environment and the people against impermissible radiation impact. The pond itself is split with internal compartments thus increasing the reliability of fuel storage.

In the rooms of the building including the reactor pond there are structures, facilities and systems ensuring safe performance of the activities related to:

- acceptance of SF from the Spent fuel Pond (SFP) of WWER -440 and WWER-1000;
- internal transportation and handling of SF;
- storage of SF under water in the compartments of the pond;
- transportation of the fuel for reprocessing;
- loading of containers for dry storage of SF.

3.4.1.2.2 Principal Technical Solutions

Civil structure is verified for re-assessed seismic characteristics. Proposed support is implemented and thus it is qualified as seismic category 1.

3.4.1.2.3 Systems and Equipment for Storage and Transportation of SF

System is a compilation of facilities and equipment designed for acceptance, continuous storage under water of assemblies and packing with SF.

A SF Pond (SFP) is located in the storage facility with four storage compartments (FSC) as well as with the respective transportation and technological equipment needed for SF handling.

System includes: a hall of the storage facility room 201 (the part located above the FSC) and SFP consisting of: 4 four compartments for storage of SF ; preliminary corridor, connecting all compartments of the storage facilities; frameworks for the hydraulic gates.

Spent fuel assemblies shall be disposed at least 3 years in the Spent Fuel Pond, in the reactor compartment and after that the spent fuel assemblies shall be placed in the transportation baskets and shall be transported by the transportation containers to the storage compartment in the SF.

Transportation and technological equipment and facilities shall consist of:

- Bridge crane 160/32/8 t;
- Bridge crane 16 t;
- Reloading machines МПХОГ-440/1000;
- Facilities and equipment for transportation and storage of SF of reactors WWER-440: containers ТК-6 and SF Transport container, basket T12/B-4, basket T13/B-4,pressurized cases T6/B-4,crossbeam125 t, rod with lighting type I, rod with lighting type II, capture for the work assemblies, capture for ARC assemblies, capture for

pressurized cases seat for container and cover, seat for baskets, site for the container, stand for the container cover;

- Facilities and equipment for transportation and storage of SF of reactors of WWER - 1000: container TK13/3, basket 37/3, crossbeam 130 t, road with lighting, capture of container cover TK13/3 and cover of basket 37/3, cover of assemblies WWER 1000, stand under container TK13/3, site for container TK13/3, stand for the container cover TK13/3, seat for baskets type 37/3, container cooling system TK–13;
- Facilities and equipment for transportation and technological operations with container "Constor 440/84" in SFSF: Reinforced concrete container "Constror 440/84", transportation overturn device for container "Constor 440/84" (movable downtilter), Hoist clamp of container "Constor 440/84" (crossbeam 135 t), auxiliary equipment for loading, draining, compacting, vacuum drying, Device for control of the pressurization of the assemblies, Welding platform, cutter for elimination of the welds on the container cover.

3.4.1.2.4 Spent Fuel Pond Water Cooling System

System is designed for removal of the residual heat discharge of the SF during its storage in the pond. System includes: pumps for water supply from the compartments of the SFP to the heat exchangers for its following cooling; laminar heat exchangers for water cooling from the SFP; pipelines and valves of stainless steel 12X1810T and 08X18H10T.

Cooling water through the heat exchangers shall be provided by the systems for reliable technical water supply group "A" of units 3 and 4, by the pipelines for cooling water to DGS-2. Connection for emergency supply of cooling water from the Fire Protection System-2.

3.4.1.2.5 Spent Fuel Pond Water Treatment System

Water treatment system of the pond is designed to maintain such physical and chemical characteristics of the water, which guarantee the corrosion resistance of the coatings of the fuel elements stored in it.

System includes: two pumps, 3 ion-exchange filters, resin catching filter, valves, pipelines Dn 50, Dn 100, bellows fittings.

3.4.1.2.6 System for Filling, Feeding, Transfusion and Discharge of the FSP

The system is designed for:

- initial filling with the chemical demineralized water in the compartments of the SFSF;

- acceptance of the overflowing waters in the compartments of the SFSF;
- feeding with chemical demineralized water in the compartments of the SFSF;
- acceptance and storage of the water in the ponds during the maintenance in the compartments.

System includes: buffer tanks each of 60 m^3 capacity, submerge pumps type ГНОМ 100-25; tank for treated condensate of 25 m^3 capacity; pump for supply of pure condensate; pipeline of Auxiliary building 2 for supply of Pure Condensate from the Tank for pure condensate or chemical demineralized water from the chemical workshop; pipelines from the Auxiliary Emergency Feedwater System 4 and БЗК-4; back up pump for chemical demineralized water; pipelines and valves.

3.4.1.2.7 System for Collection and Return of the Leakages of the Water from the spent Fuel Storage Pond

It is used for collection and control of the leakages from the inter-lining area of the SFP, overflows of the compartments for storage of the SF; acceptance compartment, compartment for reloading of the fuel assemblies, transfer corridor and from the drainage collector of the sensors of the I&C in the SFSF; return of the leakages to the compartment of the SF or supply to the active drainage pipework.

System includes: tank, receiver of controlled leakages; pumps for control leakages; pipelines and valves.

3.4.1.2.8 Ventilation System

The system is designed for maintenance of the normal climate in the rooms of the SFSF and for prevention of the propagation of the radioactive contaminations and non-exceeding of the dose rates of the staff by the means of:

- air exchange in the rooms assumed by the calculation for creation of discharge of 50 Pa and air movement velocity of 1 m/sec in the open door of the room of the primary area and creation of needed sanitary hygiene conditions for the staff in the rooms of second, third and fourth area;
- rooms of the different sanitary zones serviced by different ventilation systems;
- excluded passing of the air ducts of the suction ventilation from rooms of 1/2 area through the rooms of 3/4 area;
- excluded passing of the air ducts of the supply ventilation of the rooms of the third area into the rooms of the first and second areas;

- back up fans of the suction systems, auxiliary rooms of the first and the second zone started up in case of shut down of the main ones;
- discharge into the atmosphere of air released from the rooms of 1/2 area through a pipe of 45 m height;
- air ducts of the suction ventilation in the rooms of the first and second zones are made by welding without any splitting connections.

System includes: 4 suction ventilation systems with constant operation mode, 33 supply ventilation systems with constant operation mode; 10 suction ventilation systems with non-constant operation mode; 7 supply ventilation systems with non-constant operation mode.

3.4.1.2.9 Radiation Control System

Radiation Control System in the SFSF is designed for control of the main radiation parameters typical for the operation of the storage facility in all modes of operation including the design based accidents. System includes: 7 devices УИМ with two channels for measurement with sensors БДМГ 41-01 (14 pcs.); radiometer РКС 2-03 “Kalina” with one instrument channel, consisting of detector block БДАБ2-01, intermediate block УСМ2-03 and electronic instrument board УВК2-04; 2 gas blow downs 1A11-30-4A; 14 gamma radiation monitor GIM 204-7 with one channel for continuous instrumentation of the equivalent dose rate γ-; Aerosol monitor NGM204 with two instrument channels - instrument channel ABPM 201 – L, designed for continuous monitoring of the volume activity and alpha and beta active aerosol particles and instrument channel NGM 204-L designed for continuous monitoring of the volume activity of noble gases; 20 air sampling devices from rooms with analytical filters АФА-РМП-20; arcs for control of the contamination of the staff РЗБ 04-04М– 2 pcs and PBG-61 – 1 pc; 6 signaling devices for surface contamination with β-emitters СЗБ-04; movable radiation control devices; 40 pcs electronic signaling individual dose meters RAD100S.

3.4.1.2.10 Power Supply System

Electric Power Supply system of SFSF is designed for supply of electricity to its consumers both in normal and emergency operation modes. System includes: 2 transformers - 6/0,4 кV, 2 compartments of the houseload switchgear -0,4kV, back up diesel generator of SFSF 0,4 kV.

System provides power supply to the consumers with different requirements to the reliability of the power supply - allowing continuous disconnection, allowing short disconnection and not allowing any failure of the power supply - depending on their functional designation.

Main power supply shall be supplied from the uninterruptible power supply compartments of units 3 and 4, supplying voltage respectively to inter-redundant sections I and II of SFSF.

For the consumers of the safety systems an additional power supply from own DG in the SFSF is designed, which is connected to compartment I. In case of emergency it is possible direct power supply from the mobile DG to be provided.

3.4.1.2.11 Process Control System

The system provides metering and registration of parameters typical for the operation of the technological systems and equipment of SFSF for all possible metering ranges.

Process Control System performs the following functions:

- collection of information with the means of the local indicating devices;
- collection of information with the means of primary converters;
- conversion of the thermal technical parameters in electrical unified signals 0-5 mA;
- sending of signals of the automatic control and blocking;
- provision to the operator of information about the instant values of the controllable parameters and their registration.

Indications and signalization are displayed in the control room (CR) in the SFSF and in MCR-4 as well.

3.4.1.2.12 Heat removal from SFP to the Ultimate Heat Sink

Main ultimate heat sink is the Danube River by the system for service water group A of Unit 4 in mode of open cooling diagram. As an alternative ultimate heat sink the atmosphere with the closed cooling system is used by the means of sprinkler ponds by the operation of the system for service water group A of Unit 4.

3.4.2 Main Characteristics of the Spent Fuel Dry Storage Facility

One of the measures stipulated in the national strategy for control of spent nuclear fuel and radioactive waste adopted by the Bulgarian Government with Decision N: 693/09.11.1999 is the construction of spent nuclear fuel dry storage facility (SFDSF), which shall accept the processed nuclear fuel from all units on the Kozloduy NPP site. For the execution of these requirements 50-year minimal period of time is selected.

3.4.2.1 Location and Properties of the Site

The site is located North-West from the existing building of the SFSF of Kozloduy NPP and is situated within the framework of the fence of Kozloduy NPP.

All site characteristics important for the buildings and facilities of Kozloduy NPP are valid also for the SFDSF.

3.4.2.2 Main Design Solutions

3.4.2.2.1 Conceptual Solutions

The storage technology means a container system for storage with use of containers with air cooling by natural convection. Containers CONSTOR® 440/84 are used for storage of the assemblies with spent nuclear fuel of WWER-440. Containers are located in the building providing physical protection and protection against the natural impacts regarding the activities for storage and auxiliary equipment.

Containers are loaded with spent nuclear fuel and are prepared for storage in the existing SFSF. Priority principle of the container system for storage of the pressurized closure of the assemblies with spent nuclear fuel in the containers in case of normal and emergency conditions. All main safety related functions are achieved by the container itself. These safety related functions are:

- Radiation shielding;
- Removal of the heat from radiation decay;
- Tight closure;
- Maintenance of the subcriticality.

3.4.2.3 Systems and Equipment Description

Systems and equipment comply with the Bulgarian normative documents, with the international standards and meet the requirements of the technical design as well. Main systems related to the SFSF design are as follows:

- Ventilation and air conditioning of the storage room;
- Systems for handling operations in the SFSF;
- Transportation systems;
- Security systems;
- Equipment providing protection measures in compliance with the IAEA requirements;
- Monitoring and control system;

- Monitoring systems:
  - Radiation monitoring;
  - Container monitoring system.
- Electric power supply.

3.4.2.3.1 Ventilation and Air Conditioning in the Storage Room

Ventilation in the storage rooms provides removal of the heat generated in the containers thus ensuring safety storage temperature.

Natural ventilation of the system is designed to ensure sufficient air flow around the containers for removal of the heat generated in them thus maintaining safety temperature for storage of SF. System is passive and is based on the natural convection and will not need any mechanical equipment or power supply.

Walls of the storage facilities have several outlets for inflow of the outside air and its direction to the level of the floor of the storage facility. Outlets are secured against impermissible access and there is considerable radiation protection provided too. Air is heated and goes up to the roof where on the comb there are some outlets designed for release of the heated air.

Ability of the system to maintain the needed temperatures in the containers for the entire range of the design temperatures of the environment is substantiated in IPSA [15] of the SFDSF.

3.4.2.3.2 System for Handling Operations in the SFDSF

3.4.2.3.2.1 Crane

In compliance with the general practice in the spent nuclear fuel dry storage facilities and pursuant to Article 64(3) of the Ordinance for provision of the safety during the control of spent nuclear fuel [16] the containers are lifted and moved to the lowest possible height above the floor not exceeding 0,3 m. This maximal lifting height of the containers is restricted by an interlock.

3.4.2.3.2.2 Other Equipment

For the purposes of servicing of the SFDSF a crane, horizontal crossbeam, turnover device (stationary), transportation turnover device (440), work platforms and absorber (installed in the floor) are used;

3.4.2.3.3 Transportation System

According to the design of SFSF a transportation system is provided for transfer of the containers between SFDSF and SFSF. It consists of:

- Existing in Kozloduy NPP trailer Goldhofer, adapted for transportation of CONSTOR containers.
- Device for transportation and turnover with built in absorber and hoist for CONSTOR® 440 fixed to the Goldhofer trailer.
- Existing tractor for traction of the trailer.
- Needed roads with suitable location and lifting capacity for access of the trailers are also provided.

3.4.2.3.4 Security System

For protection purposes the SFDSF is located within the framework of the security area formed by extension of the protection fence of the SFSF around the building. Lighting and monitoring with cameras are provided. Access to the SFDSF building is controlled. Control of the access to the SFSF building is not changed.

3.4.2.3.5 Protection Equipment Pursuant to the Requirements of the IAEA and EUROATOM

Safety measures concept includes the following elements:

- Identification of the container;
- Determination of operations related to the service of the loaded containers;
- Determination of movements of the loaded containers.

3.4.2.3.6 Instrumentation and Control Equipment

DNFSF has instrumentation and control equipment (I&C) for safety execution of daily operations for acceptance and storage of the SF in compliance with the technological processes and procedures in SFDSF.

3.4.2.3.7 Monitoring Systems

3.4.2.3.7.1 Radiation Monitoring

Storage room has reliable radiation monitoring system connected with the station radiation monitoring system available in the power plant. It is designed to ensure reliable and safety control with normal conditions and emergency situations:

- Radiation monitoring in the storage room;
- Stationary systems used for control of the dose rate;
- Stationary systems used for control of the radioactivity of LLA and inert gases;

- Mobile systems used for control of aerosols radioactivity.

3.4.2.3.7.2 Container Monitoring System

CONSTOR® containers are specially designed for long term fuel storage. Foreseen control measures for indirect monitoring of the integrity of the barrier provide in time warning about any deviations. Neither direct intervention on the container nor systematic verification of the containers is required in the area of acceptance or elsewhere.

3.4.2.3.8 Electric Power Supply

SFDSF is equipped with the following electric power supply systems:

- Electric power supply from Kozloduy NPP in case of normal operation;
- Backup power supply by diesel generator;
- Backup power supply from separate uninterruptable power supply.

3.4.2.4 Heat removal from SFP to the Ultimate Heat Sink

Main ultimate heat sink is the atmosphere, which by natural ventilation system provides sufficient air flow around the containers for heat discharge generated in them thus maintaining the operation temperature that is safe for them. System is based on natural convection and during its functioning it does not rely on any mechanical equipment or electric power supply; this is a passive system.

3.5 Scope and Main Results of the PSA

3.5.1 Scope and Main Results of the PSA of Unit 3 and 4

For units 3 and 4 of Kozloduy NPP a full scope of PSA is made as well as a lot of updates of the probabilistic analyses. They have been extended many times regarding the scope, methodology and structure in order to reflect the current condition of the power plant following the multiple modernizations as well as the development of the analysis methods. PSA results level 1 in the table below present the frequency of core failure (events occurred during one reactor grid) for the on load operation of the power unit as well as the change of the results during the year:

TABLE 3.5.1-1 Core damage frequency

IE\year	1997	1999	2001	2002
Internal events	1,31E-04	9,51E-05	8,82E-05	1,26E-05
Seismic impact	3,38E-05	2,77E-06	1,98E-06	1,98E-06
Internal fires	6,77E-05	6,36E-05	6,36E-05	4,00E-06
Total Core damage frequency	2,32E-04	1,61E-04	1,54E-04	1,85E-05

Core damage frequency (events for reactor year) for operation of the power unit of low capacity is 3,26E-05, and the total fuel failure frequency in the SFP is 2,4E-05.

Total frequency of the big early releases into the environment, when it is needed to undertake urgent protection measures of the population is 9.59E-07 s/g.

Based on PSA a Risk Monitor is developed, which is used in the daily operation of Units 3 and 4 of Kozloduy NPP. Besides, there are also some other applications of the PSA elaborated, e.g. probabilistic assessment of the operation events.

For the current condition of the power unit the relevant are the values of the fuel failure frequency in the SFP as a part of the PSA of a shutdown reactor. However, this assessment is too conservative considering the constantly reduced heat discharge in the assemblies and additional technical and organizational measures for prevention and mitigation of the consequences of the possible emergency conditions as a result of the examined initial events of the PSA.

3.5.2 Scope and Main Results of the PSA of Units 5 and 6

Probabilistic safety analyses have a long-term tradition in Kozloduy NPP In principle, the first probabilistic safety analysis of WWER-1000 reactors has been elaborated namely for Units 5 and 6 of Kozloduy NPP. Since that time the scope, methodology and structure of the analyses have been extended many times. The objective was to reflect both the present condition of the power plant following the multiple modernizations made and the development of the analysis methods as well. Results of the PSA level 1 in the table below present the current condition at the moment:

TABLE 3.5.2-1: Level 1PSA results

	Internal initial events	Internal fires	Internal floods	Seismic impact	Total
Core damage frequency during the on load unit operation, g	9,32E-06	3,11E-06	1,98E-07	3,34E-06	1,6E-05
Core damage frequency during the low load operation and shut down reactor, g	5,22E-06	1,98E-07	2,94E-08	3,66E-09	5,45E-06
Fuel Failure Frequency in SFP/g	1,50E-06	1,21E-07	1,67E-09	1,66E-08	1,64E-06
Total events/reactor years.	**1,60E-05**	**3,43E-06**	**2,29E-07**	**3,36E-06**	**2,30E-05**

Within the framework of the modernization some additional facilities have been constructed, which shall impact both the prevention of failure of the pressurized containment and the discharges of radioactive decay products into the environment. Result obtained by 2001, which is being updated now for Units 5 and 6 of the Kozloduy NPP Units shows a value of low early releases' frequency (LERF) of

5.37E-06 1/year.

Based on PSA a Risk Monitor is developed, which is used in the daily operation of Kozloduy NPP. Besides, there are some other PSA appendices that are elaborated or are under elaboration - risk-aware tests, risk-aware technical service, risk-informed Technological Rules.

3.5.3 Scope and Main Results of the PSA of the SFSF

Full-scope probabilistic safety analysis of SFSF has not been made. There are safety analyses of the entire range of possible output events made under the method of analysis of the potential hazards and operability (HAZOP).

4 KOZLODUY NPP STRESS TESTS CURRENT STATUS

4.1 Chronology of Stress Tests Performance

Following the events occurred in Fukushima 1 NPP, revoking the international response and concern about the tragedy all around the world, on 06/01/2011 Kozloduy NPP JSC has received a letter from the NUCLEAR REGULATORY AGENCY (NRA), which was a response to the request of the Council of Europe on March 25, 2011, in accordance with decisions of the plenary meeting of 13 May 2011 between ENSREG and the European Commission for a in-depth risk and safety assessment of nuclear power plant in the European Union and in connection with the responsibilities of the licensees to ensure the safety of nuclear plants.

After analysis of the letter from the NRA, management of Kozloduy NPP set up a working group of experts to carry out necessary activities related to preparation and performance of "stress tests" of Kozloduy NPP, whose task was communication, coordination and monitoring of the work related to preparing a methodology for conducting "stress tests", analysis, approval of contractors reports, submission of documents to the NRA and others.

According to the guidance set out in the letter from the NRA, Risk Engineering on 15/6/2011 develop Methods for conducting stress test as targeted reassessment of safety margins of nuclear facilities in the NPP Kozloduy for natural disaster events leading to a severe accident. Basis for the development of the methodology is the document EU Stress Tests Specifications endorsed by ENSREG on 13/05/2011, and sent to the Kozloduy NPP with letter from the NRA № 47-00-77/31.05.2011 , and document ENEF Safety Task (STORE) focused on safety and risks reassessment of applicable to nuclear power plants in the European Union in the light of events in Fukushima 1 NPP issued on 04/05/2011 by Working Group "Risk" (“SAFETY TERMS OF REFERENCE (STORE) - Targeted Safety and Risk Reassessment applicable to Nuclear Power Plants in the EU in the light of the Fukushima events”).

As a basis for the development of a progress report document - Stress tests to European nuclear power plants following the accident in Fukushima - format and content of the additional safety assessment report" sent with NRA letter № 47-00-77/18.07 0.2011 to Kozloduy NPP has been used.

In order to create a better organization of work together with the selected performers an integrated schedule for implementing activities has been developed according to which from 12.07.2011 until 29.09.2011 the stress tests performance task will be accomplished with the following activities:

- presentation of the Quality Assurance programme (QAP);

- collection and transfer of input data;
- description of the design bases and assess their adequacy;
- analysis of protective measures and compliance with current plant licensing bases for initiating events, such as earthquake, flood, extreme external pressure and loss of off-site power supply;
- issuance of a preliminary progress report on various scenarios;
- assess existing management measures for different stages of accident scenarios with loss of function of core cooling, protection function of RAW retention in case of failure of fuel, limiting the effects of loss of integrity and loss of fuel cooling function in the fuel ponds.
- margins assessment for initiating events such as earthquakes, floods, a combination of earthquakes and floods, extreme external impacts and management of severe accidents;
- stock assessment for initiating events such as earthquakes, floods, extreme external pressure and loss of off-site power and issue a report on the assessment;
- issuing a progress report;
- issuing a final summary report with the NRA current reporting requirements.

In accordance with the tasks set out in the integrated schedule and to support the work of contractors in Kozloduy NPP working groups were formed which had the task to provide the necessary input data for contractors to carry out transfer of data in a consistent and timely request.

With well established organization, the process of transfer of the collected input data passed quickly enough, thus, allowing the performers to focus their efforts on a deeper analysis and specific tasks.

4.2 Individual Stress Tests Status

After starting the implementation of stress tests by 05/08/2011 and in accordance with the integrated schedule, the following activities in the implementation of stress tests are performed:

4.2.1 Earthquakes

Stress test with initiating event "earthquake" is performed by "Risk Engineering" JSC. The company was contracted and started work for the stress test implementation. Set of input data has been prepared and delivered by Kozloduy NPP, the experts of the contractor consult regularly with experts of the Employer and if necessary new input to the original package. Quality assurance program of conducting the stress test has been provided to the Employer. Status of implementation of stress tests on the main points is as follows:

4.2.1.1 Design Basis

Stress test progress report "Earthquakes – design basis of the nuclear facilities on Kozloduy NPP site" (revision 0) has been prepared and submitted for approval. Currently internal review of the report by Kozloduy NPP experts is ongoing with regard to completeness, accuracy and compliance with the NRA requirements in order to approve the report.

4.2.1.2 Safety Margins Evaluation

Ongoing works on stress test progress report "Earthquakes – evaluation of safety margins of the nuclear facilities on Kozloduy NPP site" (revision 0).

4.2.2 Flooding

Stress test with initiating event "flooding" is performed by "Risk Engineering" JSC. The company was contracted and started work for the stress test implementation. Set of input data has been prepared and delivered by Kozloduy NPP, the experts of the contractor consult regularly with experts of the Employer and if necessary new input to the original package. Quality assurance program of conducting the stress test has been provided to the Employer. Status of implementation of stress tests on the main points is as follows:

4.2.2.1 Design Basis

Stress test progress report "Flooding – design basis of the nuclear facilities on Kozloduy NPP site" (revision 0) has been prepared and submitted for approval. Currently internal review of the report by Kozloduy NPP experts is ongoing with regard to completeness, accuracy and compliance with the NRA requirements in order to approve the report.

4.2.2.2 Safety Margins Evaluation

Ongoing works on stress test progress report "Earthquakes – evaluation of safety margins of the nuclear facilities on Kozloduy NPP site" (revision 0).

4.2.3 Extreme Meteorological Conditions

Stress test with initiating event "extreme meteorological conditions" is performed by Nuclear Research and Nuclear Energy Institute (NRNEI) at Bulgarian Academy of Science. The performer was contracted and started work for the stress test implementation. Set of input data has been prepared and delivered by Kozloduy NPP, the experts of the contractor consult regularly with experts of the Employer and if necessary new input to the original package. Quality assurance program of conducting the stress test has been provided to the Employer. Status of implementation of stress tests on the main points is as follows:

4.2.3.1 Design Basis

Stress test progress report "Extreme meteorological conditions– design basis of the nuclear facilities on Kozloduy NPP site" (revision 0) has been prepared and submitted for approval. Currently internal review of the report by Kozloduy NPP experts is ongoing with regard to completeness, accuracy and compliance with the NRA requirements in order to approve the report.

4.2.3.2 Safety Margins Evaluation

Ongoing works on stress test progress report "Extreme meteorological conditions – evaluation of safety margins of the nuclear facilities on Kozloduy NPP site" (revision 0).

4.2.4 Loss of Safety Function due to Any Credible IE on the Site

Stress test for "Analysis of consequences of loss of safety function due to any credible IE on the site" is performed by "ENPRO-Consult" JSC. The performer was contracted and started work for the stress test implementation. Set of input data has been prepared and delivered by Kozloduy NPP, the experts of the contractor consult regularly with experts of the Employer and if necessary new input to the original package. Quality assurance program of conducting the stress test has been provided to the Employer. Status of implementation of stress tests on the main points is as follows:

4.2.4.1 Design Basis

Stress test progress report "Analysis of consequences of loss of safety function due to any credible IE on the site – design basis of the nuclear facilities on Kozloduy NPP site" (revision 0) has been prepared and submitted for approval. Currently internal review of the report by Kozloduy NPP experts is ongoing with regard to completeness, accuracy and compliance with the NRA requirements in order to approve the report.

4.2.4.2 Safety Margins Evaluation

Ongoing works on stress test progress report "Analysis of consequences of loss of safety function due to any credible IE on the site – evaluation of safety margins of the nuclear facilities on Kozloduy NPP site" (revision 0).

4.2.5 Severe Accident Management

Stress test for "Severe accident management" is performed by Westinghouse Energy Systems JSC. The performer was contracted and started work for the stress test implementation. Set of input data has been prepared and delivered by Kozloduy NPP, the experts of the contractor consult regularly with experts of the Employer and if necessary new input to the original package. Quality assurance program of conducting the stress test has been provided to the Employer.

5 REFERENCE DOCUMENTS

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[4] Updated safety analysis report for unit 4, 2004

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