



STATE NUCLEAR REGULATORY INSPECTORATE OF UKRAINE

# NATIONAL REPORT OF UKRAINE



2011

## ABBREVIATIONS

AC	Alternating Current
AEFS	Additional Emergency Feedwater System
BDBA	Beyond Design Basis Accident
BRU-A	Steam Dump Valve to Atmosphere
BRU-K	Steam Dump Valve to Condenser
C(I)SIP	Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs
ChNPP	Chornobyl Nuclear Power Plant
CP	Cooling Pool
CPS	Control and Protection System
DBA	Design Basis Accident
DC	Direct Current
DCH	Direct Containment Heating
DG	Diesel Generator
DSF	Dry Spent Fuel Storage Facility (Zaporizhzhya NPP)
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EDG	Emergency Diesel Generator
EFS	Emergency Feedwater System
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency Operating Procedure
FCP	Fuel Channel Cooling Pool
FSS	Full-Scope Simulator
HPIS	High-Pressure Injection System
ISF	Interim Spent Fuel Storage Facility (Chornobyl NPP)
ISF-1	Wet Interim Spent Nuclear Fuel Storage Facility
ISF-2	Dry Interim Spent Nuclear Fuel Storage Facility
KhNPP	Khmelnytsky Nuclear Power Plant
LOCA	Loss-of-Coolant Accident
LPIS	Low-Pressure Injection System
MCCI	Molten Corium/Concrete Interaction
MCP	Main Coolant Pump
MCR	Main Control Room
MDGPU	Mobile Diesel Generator and Pumping Unit
MDGS	Mobile Diesel Generator Station
MGV	Main Gate Valve
MSB	Multi-Assembly Sealed Basket
MSIV	Main Steam Isolation Valve
NF	Nuclear Facility
NNEGC	National Nuclear Energy Generating Company <i>Energoatom</i>
Energoatom	
NPP	Nuclear Power Plant
NPP	Nuclear Power Plant
PGA	Peak Ground Acceleration
PRZ PORV	Pressurizer Pilot-Operated Relief Valve
PSA	Probabilistic Safety Assessment
PSS	Plant Shift Supervisor
RCS	Reactor Coolant System
RF	Reactor Facility
RNPP	Rivne Nuclear Power Plant
RPV	Reactor Pressure Vessel
RV	Relief Valve
SAMG	Severe Accident Management Guideline

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SAR	Safety Analysis Report
SEWM	Site Emergency Work Manager
SFA	Spent Fuel Assembly
SFAP	Spent Fuel Assembly Cooling Pool
SFP	Spent Fuel Pool
SFPU	State Fire Protection Unit
SG	Steam Generator
SG SV	Steam Generator Safety Valve
SLP	Sequential Loading Program
SNRIU	State Nuclear Regulatory Inspectorate of Ukraine
SSC	Structures, Systems and Components
SUNPP	South Ukraine Nuclear Power Plant
SV	Safety Valve
SVO	Active Water Treatment System
TBSC	Transfer Basket Storage Compartment
WWER	Water-Cooled Water-Moderated Power Reactor
ZNPP	Zaporizhzhya Nuclear Power Plant

## INTRODUCTION

On 11 March 2011, the Fukushima-1 NPP (Japan) suffered a severe accident caused by an earthquake and subsequent tsunami. The Fukushima-1 event exposed the nuclear community to the following tasks: carry out a detailed analysis of the accident causes and lessons learned, develop and implement measures to prevent the severe accident and mitigate its adverse impact on the public and environment.

On 25 March 2011, the European Council declared that the safety of the European NPPs should be reviewed on the basis of a comprehensive and transparent risk assessment (stress tests). At the Fifth Review Meeting of the Convention on Nuclear Safety (4–14 April 2001), the Contracting Parties pointed out the need for NPP safety reassessment and immediate additional measures on NPP safety improvement, in their joint statement on the Fukushima Daiichi accident.

On 13 May 2011, the European Nuclear Safety Regulators Group (ENSREG) and the European Commission agreed technical requirements for stress tests for the European NPPs (Declaration by ENSREG, Annex 1: EU stress test specifications). Pursuant to the ENSREG technical requirements, stress test is a targeted reassessment of the safety margins of nuclear power plants in the light of the events which occurred at Fukushima-1. The stress tests shall include a detailed analysis of extreme natural events and their combinations that challenge the plant safety functions and may cause a severe accident.

The safety improvement of Ukrainian NPPs in the light of the Fukushima-1 accident was discussed at the meeting of the National Security and Defense Council of Ukraine (NSDC) on 8 April 2011. The NSDC decision, which emphasizes the need for in-depth safety reassessment of Ukrainian NPPs, including seismic analyses, was endorsed by Presidential Decree No. 585/2011 of 12 May 2011.

The SNRIU Board approved an action plan for a targeted safety reassessment and further safety improvement of Ukrainian NPPs in the light of the Fukushima-1 accident and an action plan for a targeted safety reassessment and further safety improvement of Chornobyl NPP units 1-3 and Interim Spent Fuel Storage Facility (ISF-1) in the light of the Fukushima-1 accident on 19 May and 5 July, respectively. One of the actions defined in the Action Plans was a targeted safety reassessment of operating nuclear facilities at NPP sites (stress tests). The Action Plans also provide for the following:

- targeted review of emergency preparedness;
- review and amendment (based on stress tests results) of the Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs;
- updating of the Plan for Safety Improvement of the Chornobyl NPP ISF-1;
- analysis and improvement of the regulatory framework for nuclear and radiation safety, strengthening of safety requirements for operating and new nuclear power units.

The SNRIU developed the Recommendations on the Structure and Contents of the Report on Targeted Safety Reassessment of On-site Nuclear Facilities to Learn Lessons from the Fukushima-1 Accident and introduced them by Order No. 91 of 23 June 2011. The Recommendations are based on the ENSREG EU stress test specifications.

In the framework of the stress tests, the operators (NNEGC “Energoatom” and Chornobyl NPP) analyzed in detail:

- external extreme natural events (earthquakes, flooding, fires, tornadoes, extremely high/low temperatures, extreme precipitations, strong winds, combinations of events);
- loss of electrical power and/or loss of ultimate heat sink;
- severe accident management.

At the operating nuclear power plants (Zaporizhzhya, Khmelnytsky, Rivne and South Ukraine), the stress tests focused on nuclear fuel in the reactor cores, spent fuel pools, fresh fuel rooms, dry spent nuclear fuel storage facility (Zaporizhzhya NPP).

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At the Chornobyl NPP, the stress tests focused on spent nuclear fuel stored in the spent fuel pools of units 1 and 2 and ISF-1. The spent fuel pool of unit 3 that may be used to store spent nuclear (as a standby facility) was analyzed as well.

Based on the completed analyses, conclusions were made on robustness of the nuclear facilities against external extreme events, and recommendations on their safety improvement were provided.

On 3 November 2011, the SNRIU Board discussed the stress tests results for Chornobyl NPP units 1-3 and ISF-1 and presented findings of the state nuclear and radiation safety review of Chornobyl NPP stress tests. The Board's meeting was attended by representatives of the State Exclusion Zone Management Agency of Ukraine, Ministry of Emergencies of Ukraine, Supreme Council Fuel and Energy Committee, European Bank for Reconstruction and Development, non-governmental organizations and mass media.

On 24-25 November 2011, the SNRIU Board discussed the results of stress tests for the operating NPPs of Ukraine and findings of the state nuclear and radiation safety review. The session was attended by representatives of non-governmental organizations, mass media, Secretariat of the Cabinet of Ministers of Ukraine, dedicated committees of the Supreme Council of Ukraine, Ministry of Energy and Coal Industry of Ukraine, regional state administrations and regional councils, and representatives of nuclear regulatory authorities of Bulgaria, Germany, the Russian Federation, Slovak Republic.

On 24 June 2011, Ukraine signed the Declaration for stress tests at Ukrainian NPPs (within the Ministerial Conference for preparation for and implementation of stress tests at NPPs in the EU and neighboring countries). The stress test results for Ukrainian NPPs will be assessed by independent European experts (peer reviews). Approaches to peer reviews of the stress test results for the European NPPs and requirements for national reports on stress tests were adopted at the ENSREG meeting of 11 October 2011.

This Report has been developed by the State Nuclear Regulatory Inspectorate of Ukraine based on the results of stress tests performed by the operating organizations for all operating NPPs and units 1-3 and ISF-1 of the Chornobyl NPP, with account for findings of the state nuclear and radiation safety review. The Report is in line with the ENSREG requirements for the contents and format of National Reports (Post-Fukushima stress tests of European Nuclear Power Plants – Contents and Format of National Reports). The Report was reviewed and approved at the session of the SNRIU Board on 20 December 2011.

Part 1 of the Report describes the stress test results for the operating NPPs (Zaporizhzhya NPP, Khmelnytsky NPP, Rivne NPP and South Ukraine NPP). The stress test results for the Chornobyl NPP are presented in Part 2.

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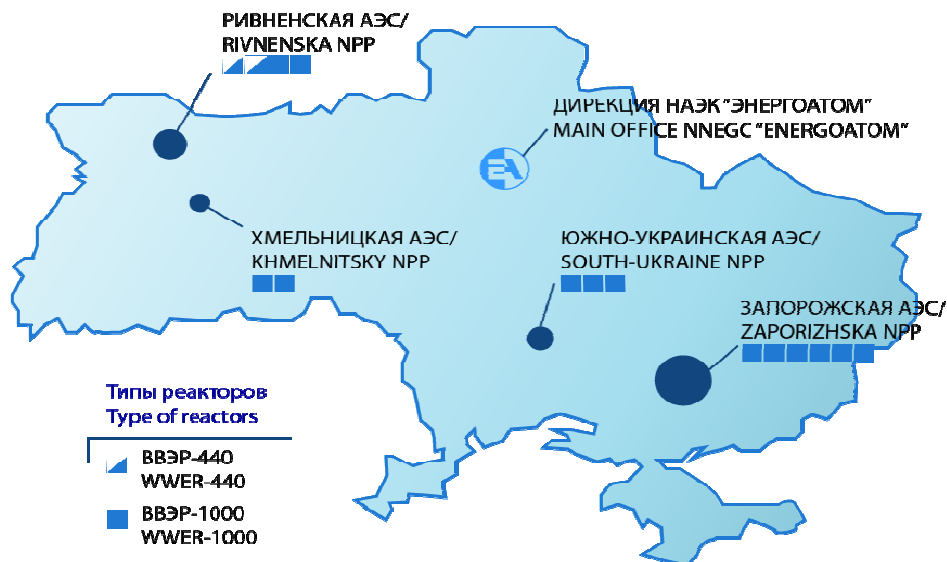
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## PART 1. OPERATING NPPs OF UKRAINE

### 1. GENERAL DATA ON THE SITES AND NUCLEAR POWER PLANTS OF UKRAINE

There are currently 15 power units operated at four NPP sites in Ukraine with a total installed electrical power of 13,835 MW, which constitutes 26.3% of the total installed electrical power of all power plants in the country.



Ukrainian Nuclear Power Plants

Summary Information on Operating Units of Ukrainian NPPs:

NPP, Unit No.	Electric power, MW	Reactor type	Connection to the grid	Design lifetime expiry date
<i>ZNPP</i>				
1	1000	V-320	10.12.1984	10.12.2014
2	1000	V-320	22.07.1985	22.07.2015
3	1000	V-320	10.12.1986	10.12.2016
4	1000	V-320	18.12.1987	18.12.2017
5	1000	V-320	14.08.1989	14.08.2019
6	1000	V-320	19.10.1995	19.10.2025
<i>SUNPP</i>				
1	1000	V-302	31.12.1982	31.12.2012
2	1000	V-338	09.01.1985	09.01.2015
3	1000	V-320	20.09.1989	20.09.2019
<i>RNPP</i>				
1	420	V-213	22.12.1980	22.12.2010*
2	415	V-213	22.12.1981	22.12.2011*
3	1000	V-320	21.12.1986	21.12.2016
4	1000	V-320	10.10.2004	10.10.2034
<i>KhNPP</i>				
1	1000	V-320	22.12.1987	22.12.2017
2	1000	V-320	08.08.2004	08.08.2034

\*The operational lifetime was extended (December 10, 2010) based on the results of periodic safety review for power units.

The National Nuclear Energy Generating Company "Energoatom" (NNEGC "Energoatom") is the licensee of all operating NPPs. All the operating units of Ukrainian NPPs have a license for nuclear installation operation:

- Zaporizhzhya NPP. The license was issued by the State Nuclear Regulatory Committee of Ukraine on 16 June 2004;
- Khmelnytsky NPP. The license was issued by the State Nuclear Regulatory Committee of Ukraine on 5 October 2004 (unit 1) and on 15 December 2005 (unit 2);
- Rivne NPP. The license was issued by the State Nuclear Regulatory Committee of Ukraine on 10 December 2010 (units 1, 2), on 10 December 2010 (unit 3) and on 6 April 2006 (unit 4);
- South Ukraine NPP. The license was issued by the State Nuclear Regulatory Committee of Ukraine on 19 July 2002.

Detailed information on the Zaporizhzhya NPP units and dry spent fuel storage facility (DSF) is provided in Section 1Z, on the Khmelnytsky NPP units in Section 1Kh, the Rivne NPP in Section 1R and the South Ukraine NPP in Section 1SU.

### **Use of PSA as Part of Safety Assessment**

According to regulatory requirements, the operating organization shall justify the safety of NPP units in a comprehensive manner and represent the results in the Safety Analysis Report (SAR). The safety analysis methodology combines complementary deterministic and probabilistic methods.

The SAR structure and contents are determined in the regulatory document "Requirements for the Contents of the Safety Analysis Report for Operating Units of Ukrainian NPPs with WWER Reactors". The SAR consists of the Integrated Safety Analysis Report and mandatory appendixes:

- Technical Safety Substantiation (TSS);
- Additional Materials on Safety Analysis (AMSA);
- Analysis of Design Basis Accidents (DBA);
- Analysis of Beyond Design Basis Accidents (BDBA);
- Probabilistic Safety Assessment (PSA).

The SAR structure and contents for the new power units commissioned in 2004, Khmelnytsky-2 and Rivne-4, are determined in document KND 306.302-96 "Requirements for the Contents of the Safety Analysis Report for NPPs with WWER Reactors for Issuing a Commissioning License". The SAR for Khmelnytsky-2 and Rivne-4 consists of 20 Chapters:

- Chapter 1 General Description of the Nuclear Power Plant;
- Chapter 2 Characteristics of the NPP Area and Site;
- Chapter 3 Design Substantiation for Buildings, Structures, Systems and Components;
- Chapter 4 Reactor;
- Chapter 5 Reactor Coolant System and Related Systems;
- Chapter 6 Safety Systems;
- Chapter 7 Process Control Systems;
- Chapter 8 Power Supply;
- Chapter 9 Power Unit Auxiliary Systems;
- Chapter 10 Steam Turbine Facility;
- Chapter 11 Radioactive Waste Management;
- Chapter 12 Radiation Protection;
- Chapter 13 Operation;
- Chapter 14 Pre-commissioning Program;
- Chapter 15 Accident Analysis;
- Chapter 16 Safe Operation Limits and Conditions;
- Chapter 17 Quality Assurance;
- Chapter 18 Decommissioning;



- Chapter 19 Probabilistic Safety Assessment;
- Chapter 20 Fire Protection System.

The Probabilistic Safety Assessment (PSA) is a mandatory part of the Safety Analysis Report (SAR). Both qualitative and quantitative safety assessments of NPPs are performed within the PSA. The PSA results are used to:

- demonstrate that the safety criteria are within the established limits for the core damage frequency and the large early release frequency;
- identify and analyze risk contributors and safety issues;
- identify and analyze the key phenomena emerging in severe accidents;
- improve emergency operating procedures and training programs;
- identify NPP safety issues, evaluate effectiveness and adequacy of compensatory measures.

In compliance with the regulatory requirements, a full-scope PSA shall be developed as part of the SAR to account for internal initiating events, internal and external hazards (fire, flooding, etc.) for all operational states (nominal and low power, shutdown). The scope of developed PSAs for NPP units that has been approved by the State Nuclear Regulatory Inspectorate of Ukraine is as follows:

*Nominal power:*

- Level 1 PSA for internal initiating events;
- Level 1 PSA for internal fires;
- Level 1 PSA for internal flooding;
- Level 1 PSA for external extreme events (hazards);
- Level 1 PSA for consequences of compromised residual heat removal from the spent fuel pool;
- Level 2 PSA for internal initiating events.

*Low power and shutdown:*

- Level 1 PSA for internal initiating events.

Besides, the level 2 PSA for low power and shutdown was developed for Zaporizhzhya-5 and got the approval of the State Nuclear Regulatory Inspectorate of Ukraine.

Upon results of deterministic and probabilistic safety assessments (within the SAR) the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs” was developed. On 30 November 2010, the State Nuclear Regulatory Inspectorate of Ukraine and the Ministry of Energy and Coal Industry of Ukraine approved this Program.

In addition to various hardware upgrades, the Program includes comprehensive activities on extending PSA studies for all NPPs to address all potential internal and external hazards for the core and spent fuel pool in all operating modes, as well as development of the seismic PSA studies.

## 1 Z GENERAL DATA ON THE ZAPORIZHZHYA NPP

### 1.1 Z Brief Description of the Site Characteristics

The Zaporizhzhya NPP site is situated in the Kamenka-Dniprovsk district of the Zaporizhzhya region on the left bank of the Kakhovka water reservoir (Dnipro river).



**Zaporizhzhya NPP**

The district center, the town of Kamenka-Dniprovsk, is situated at a distance of 12 km from the Zaporizhzhya NPP site, 52 km from the regional center, the city of Zaporizhzhya, and at a distance of 5 km from the satellite town of Energodar.

The local relief of the Zaporizhzhya NPP site is flat, with alternating sand hummocks and hollows. The site leveling elevation is taken as 22.0 m.

The vehicle access is provided via the 6 km motor road built during Zaporizhzhya NPP construction.

The railway siding adjoins the railway access of Zaporizhzhya thermal power plant with an exit to the external railway network.

There are following industrial facilities within a 10-km radius of ZNPP (the distance is measured from the Zaporizhzhya NPP site):

- Zaporizhzhya thermal power plant rated 3600 MW, 2.5 km off the site;
- industrial area (house-building factory, plant of non-standard metal and reinforced concrete panel and honeycomb structures, plant of auxiliary boiler equipment, plant of non-standard equipment and pipes), 3 km off the site;
- cargo terminal for the Zaporizhzhya NPP and thermal power plant at the Kakhovka water reservoir, 0.83 km off the site;
- storage facilities of the Zaporizhzhya NPP, 4 km off the site.

There are no chemical or oil-refining plants, mines, open-cast mines, oil pipes, well drills, boreholes of underground gas reservoirs, docks, ports, airports and other facilities in the area that could affect the Zaporizhzhya NPP.

There are six power units operated at the Zaporizhzhya NPP site with WWER-1000/320 reactors with the total electric power of 6000 MW.

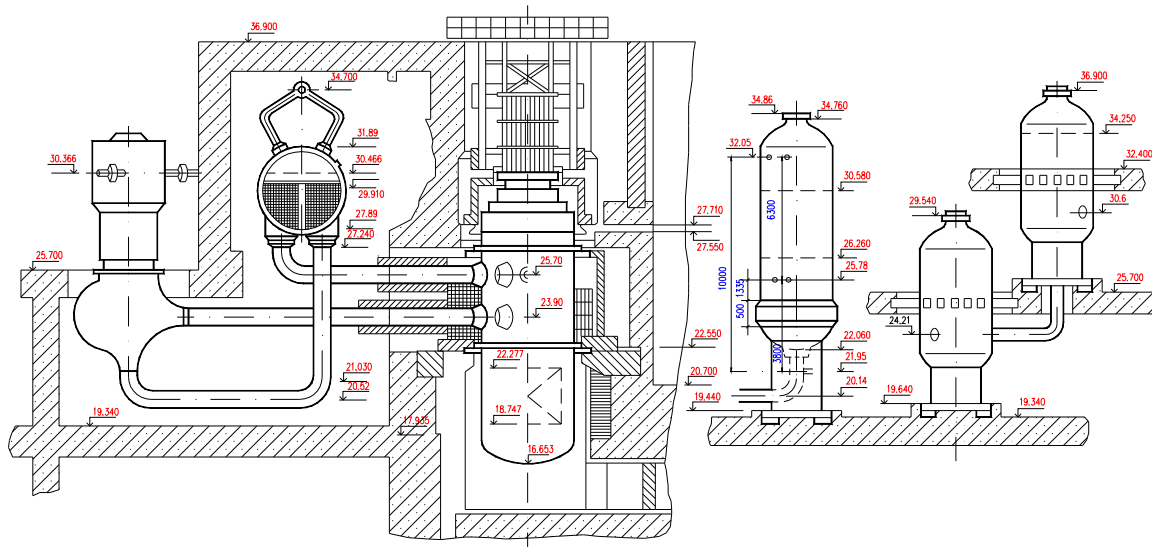
### 1.1.1 Z Main Characteristics of the Units

#### 1.1.1.1 Z Reactor Facilities

The main equipment of Zaporizhzhya NPP units 1-6:

- Water-cooled water-moderated pressurized power reactor WWER-1000 (V-320 series);
- Turbine unit K-1000-60/1500-2;
- Electric generator TVV-1000-4.

The reactor facilities of Zaporizhzhya NPP units 1-6 are designed to consist of two circuits. Each reactor facility (RF) is equipped with a water-cooled water-moderated pressurized power reactor WWER-1000 (V-320 series).



**Layout Diagram of Main Equipment of WWER-1000/320**

Systems and equipment of WWER-1000/V-320 assure:

- safe and reliable reactor operation under all normal operation modes envisaged by the design;
- operation control and monitoring for reactor major equipment;
- integrity of the reactor in all design-basis accidents;
- integrity and reliable cooling of the fuel assemblies in the reactor core under all operational modes envisaged by the design.

RF equipment is housed in the pre-stressed leak-tight reinforced-concrete containment having the shape of a hollow cylinder with a spherical dome and flat bottom. The reinforced concrete wall of the containment is 1.2 m thick in its cylinder-shaped section and 1.1 m thick in its dome section. There is leak-tight 8-mm metal lining on the internal side of the containment.

#### 1.1.1.2 Z Spent Fuel Storage Facilities

The spent fuel assemblies unloaded from the core are stored in racks in the spent fuel pools (SFP). Before placing for storage, the fuel assemblies are subjected to fuel cladding leak testing. Based on the testing results, a spent fuel assembly is placed either in the rack slots or in a sealed canister. The spent fuel storage system is housed in the reactor compartment provided with all necessary rooms and equipment to receive and store the spent fuel assemblies.

The SFP is housed inside the containment and consists of three compartments designed for storage of spent fuel assemblies, and a well which is an area for loading of transport

casks with spent fuel assemblies and unloading of fresh fuel casing. Dividing the SFP into three compartments allows for maintenance in one of them while spent fuel assemblies are placed into the remaining two. The well stands separately from the fuel storage area, which permits installation of a fresh fuel casing into the dry well.

The SFP is adjacent to the reactor and is connected with upper part of the reactor cavity by a refueling channel for transport of fuel assemblies. The pool is equipped with spent fuel storage racks.

The SFP is filled with a boric acid solution of at least 16 g/dm<sup>3</sup> concentration. Water in each SFP compartment circulates through the SFP cooldown system that keeps the SFP water temperature within a permissible limit of no more than 70°C (with the reactor core completely unloaded). This value is established to prevent boiling of the cooling water and fuel melting induced by decay heat. In addition, the cooldown system ensures a protective level of water in the SFP during spent fuel storage and provides radiation protection for maintenance personnel.

In an emergency situation there is a possibility for makeup and fuel cooling in SFP compartments with a 16 g/dm<sup>3</sup> boric acid solution from pumps of the containment spray system. In this case makeup and water cooling in SFP proceeds as follows: containment sump – emergency cooldown heat exchanger – containment spray pump – SFP compartment – SFP flow through the transport opening to the refueling pool – water flow through the lower elevation of the containment to the containment sump.

### **Dry Spent Fuel Storage Facility**

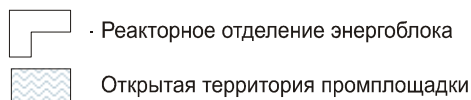
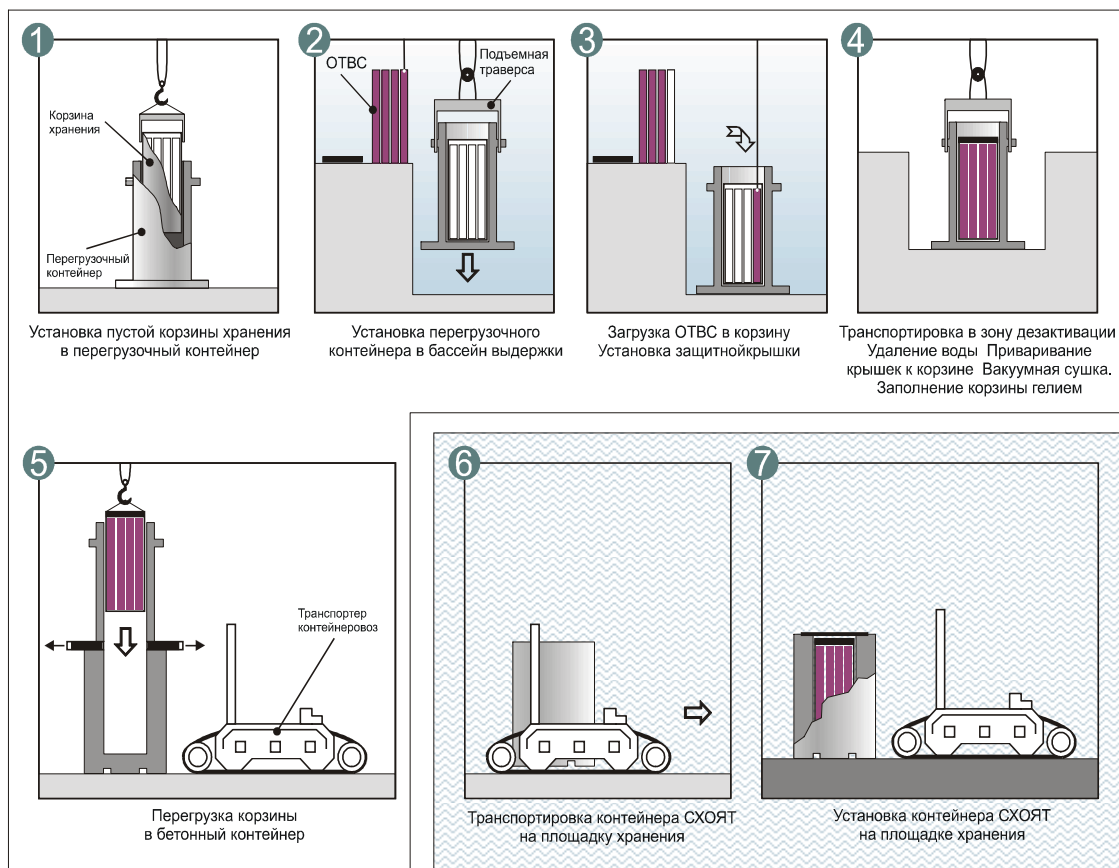
For further storage of spent fuel from all Zaporizhzhya power units, there is a dry spent fuel storage facility) on the Zaporizhzhya NPP site.

The DSF is designed for safe storage of 380 containers VSC-WWER (over 9000 spent fuel assemblies). The system is passive; thus, after concrete containers are placed on the storage site no major maintenance efforts will be required. As of 30 June 2011, there was DSF Stage 1 in operation designed to accommodate 100 containers.

The DSF is situated in the north-eastern part of the Zaporizhzhya NPP site at 21.80 m elevation. The DSF is protected along the perimeter with a radiation protection structure, which is a concrete wall 300 mm thick and 6 m high.

The DSF loading areas are situated in reactor buildings of all power units at 36.9 m elevation and in passageways at 0.0 elevation. These areas for spent fuel loading and transportation are equipped with transfer and handling equipment: overhead pivoting cranes and refueling machines.

The DSF transportation area is a road network used to deliver VSC-WWER containers from the loading area to the storage area. A dedicated container transporter envisaged by the DSF design is used for container transportation. The DSF storage site is located in the open area. To install the containers in their storage place, the above-mentioned container transporter is used.



Cycles of Operations for Transportation of Spent Fuel Assemblies from the SFPs of Zaporizhzhya NPP Units to the DSF Site

### 1.1.2 Z Description of the Systems for Conduction of Main Safety Functions

The reactor coolant system (RCS) consists of a reactor and four circulation loops, each consisting of a steam generator PGV-1000M, main coolant pump (MCP) GTsN-195M, and RCS piping. The RCS piping consists of pipes 850 mm in inner diameter and 70 mm in thickness. Each loop of the primary circuit is divided into “cold” and “hot” legs. The “hot” leg supplies the coolant from the reactor to the steam generator, while the “cold” leg returns the coolant from the steam generator to the reactor via the MCP. The design of all loops is similar, ensuring equal hydraulic resistance of the loops and, thus, equal coolant flow rate. The RCS piping is designed to meet the requirements for prevention of radioactivity release into the environment. The design ensures normal operation of the reactor facility subjected to earthquake-induced loads reaching the design-basis values, and ensures safe shutdown and cooling. Thermal expansions of the piping are compensated by moving the steam generators and MCP that rest on rolling supports.

The PGV-1000M steam generator is a single-shell recuperative horizontal heat-exchanger with a tubing bundle immersed into it and is designed for generation of dry saturated steam.

The main coolant pump GTsN-195M is designed for coolant circulation in the primary circuit and is a vertical centrifugal single-stage pump with hydrostatic seal, console work wheel, axial water supply and a self-contained three-phase asynchronous electric motor with a short-circuited rotor.

To achieve the main safety goal, which is to prevent radioactive releases beyond the physical barriers, the design provides for the following safety functions:

- reactivity control;
- reactor core cooling;
- confinement of radioactive substances within the established boundaries.

These safety functions in compliance with the defense-in-depth principle are inherent in the WWER design.

### **Reactivity Control**

Reactivity control and reactor transfer to sub-criticality as well as maintaining the reactor core sub-critical are ensured through reactivity control systems based on different principles:

- system of control rods mechanical movement in the reactor core (reactor power control and scram – reactor protection system);
- boric acid control system to regulate boric acid concentration in the coolant (the system is designed to control reactivity related to fuel burnup and transients as well as to keep the core subcritical in reactor shutdown and transfer to cold shutdown or hot standby mode).

The control rods are controlled through the reactor protection system (RPS). Control rods are placed in the uppermost position when the reactor is at nominal power, except for the one control rods group. Reactor scram is achieved by insertion of all rods into the core in the shortest time.

In normal operation, the boron control function rests with the primary makeup and letdown system.

In emergency modes, there are provisions for the regulation of boric acid concentration in the primary coolant using safety systems integrated in the ECCS (emergency core cooling system):

- high-pressure boron injection system (TQ14,24,34);
- high-pressure injection system (TQ13,23,33);
- passive ECCS (hydroaccumulators YT11,12,13,14B01);
- low-pressure injection system (TQ12,22,32).

Each of the ECCS systems has three trains. The passive ECCS consists of 4 hydroaccumulators: with the two supplying water to the reactor upper plenum, and the other two to the lower plenum.

Emergency systems are actuated automatically in response to emergency signals, the operating mode and sequence of actuation and operation depends on the initiating event.

In accidents, the ECCS systems supply the primary circuit with the boron-containing coolant (thus ensuring the reactor core transfer to subcriticality and maintaining the core subcritical for a long time after reactor scram) and perform the “core cooling” safety function.

To ensure nuclear safety in reactivity control, neutron flux parameters (density, neutron power, reactivity and period) are monitored on a permanent basis by appropriate instrumentation and control systems; concentrations of boric acid in the reactor and trains of safety systems are monitored as well.

### **Core Cooling**

The key objective of this function is to prevent fuel failure as a result of overheating. For this purpose, there are heat removal systems and equipment in all operational modes. Heat removal from the reactor core under WWER normal operation proceeds as follows: reactor core – primary coolant – steam generator – secondary coolant – ultimate heat sink – atmosphere. During accidents and after reactor shutdown, residual heat is removed in the same sequence. If steam generators are unavailable for heat removal function, the heat is removed through the emergency core cooling system.

*Fuel cooling in normal operation and anticipated operational occurrences under primary circuit integrity*

At the first cooldown stage, the heat from the reactor core is removed through the secondary circuit system. Steam is removed through the steam dump valve to condenser (BRU-K) or the steam dump valve to atmosphere (BRU-A) in case of loss of off-site power or any other initiating event when BRU-K operation is impossible, as well as through steam generator safety relief valves in case both BRU-K and BRU-A are unavailable. When water supply to steam generators through the main and auxiliary feedwater systems is impossible, feedwater is supplied by the emergency feedwater system (TX10-30).

Long-term cold shutdown is ensured through the low-pressure injection system (TQ12,22,32). ECCS heat exchangers are cooled through the essential service water system, which is also used to cool other components important to safety.

*Core cooling under loss-of-coolant accident*

In case of LOCA which cannot be compensated for by the makeup and letdown system, the emergency core cooling systems are actuated. ECCS supplies the core with concentrated boric acid solution to maintain sufficient coolant inventory in the primary circuit to ensure core cooling, residual heat removal and maintain the core subcritical due to primary circuit makeup with borated water.

*Core cooling during cold shutdown or refueling*

Low-pressure injection system is used. All three system trains provide borated water supply to the reactor lower or upper plenum, with the flow direction selected automatically depending on the mode; in case of cooldown for maintenance purposes, water is supplied only to the upper plenum. Two trains are connected to the main borated water pipes from hydroaccumulators, while the third one to cold and hot legs of the primary circuit.

**Limitation of Radioactivity Release**

This safety function is aimed at preventing radioactive releases beyond the nuclear power plant site.

In accidents, fuel can be damaged due to overheating and thus radioactive substances will release into the primary circuit. To limit radioactivity releases in accidents, the design provides for localization safety system (components), which are systems (components) designed to prevent or limit release of radioactive materials and ionizing radiation beyond the boundaries specified in the design.

The localization safety system consists of a containment system and various process systems ensuring confinement of accident products and pressure decrease in the reactor containment (RC): containment spray system (TQ11,21,31), air-operated isolation valves, penetrations, hydrogen concentration monitoring systems, passive autocatalytic hydrogen recombiners, etc.

**Safety Systems**

The safety systems are designed to perform safety functions in case of an initiating event and any independent single failure of one active or passive component with mechanical moving parts, or an additional human error.

All NPP safety systems are designed according to the redundancy principle. Each safety system consists of three independent trains. Each train in terms of its efficiency, response time and other factors is sufficient to perform the intended functions in the full scope defined for a given system under any operational mode, including maximum design-basis accident (double-ended break of the main coolant piping DN 850 mm under loss of off-site power).

Independence of each safety system trains is ensured by complete physical separation of the trains. Equipment of the respective safety system trains is actuated automatically in response to independent signals.

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**WWER-1000/V-320 Main Safety Systems and their Functions**

System	Intended Use under Normal Operation	Intended Use in Emergencies
<b>Protective safety systems</b>		
Reactor Protection System (mechanical side)	Compensate for reactivity changes and reactor power control	Reactor emergency power decrease or reactor scram by insertion control rods into the core
High-pressure Boron Injection System TQ14,24,34	Under normal operation the system is in standby; the system is periodically tested in compliance with the operating procedures	In emergencies and accidents, each train of the system ensures boron concentrate injection ( $40\text{g/dm}^3$ ) into the primary circuit at any primary pressure (up to the maximum permitted) at a flow rate of $6\text{m}^3/\text{h}$ , when the makeup and letdown system performance is not sufficient or its function of boron injection into the primary circuit is lost. The pump is actuated automatically upon the signal from the sequential loading program (SLP) or by the operator. To inject the boron concentrate into the primary circuit, the operator must close the valves on the recirculation line using control switches.
High-pressure Injection System TQ13,23,33	Under normal operation, the system is in standby; the system is periodically tested in compliance with the operating procedures	“Under primary pressure decrease below $100\text{ kgf/cm}^2$ , each safety system train ensures boron concentrate ( $40\text{g/dm}^3$ ) injection into the primary circuit at a flow rate of $\approx 130\text{t/h}$ . After the boric acid tank is emptied, the train is automatically connected to the RC sump, thus initiating the closed-cycle system operation. The pump is actuated automatically upon the ECCS signal or the SLP signal.
Passive ECCS YT11, 12, 13, 14	Under normal operation, the system is in standby; the system is periodically tested in compliance with the operating procedures	The systems ensure injection of boric acid of $16\text{g/dm}^3$ concentration and $55\text{-}60^\circ\text{C}$ temperature into reactor under primary pressure below $5.9\text{ MPa}$ ( $60\text{ kgf/cm}^2$ ) thus ensuring core cooling and filling under loss of coolant accidents.
Low-pressure Injection System TQ12,22,32	Residual heat removal from the reactor core and primary circuit cooldown.	Emergency core cooling and residual heat removal from the reactor core in accidents requiring the reactor transfer into “cold shutdown” mode. The pump is actuated automatically upon the ECCS signal or the SLP signal.
Primary Overpressure Protection System YP	PORV is in standby. There are three PORVs installed – one control and two operating.	To prevent primary equipment and piping damage in pressure increase beyond the safe operation limits. To decrease primary pressure to ensure the feed and bleed procedure by means of the high-pressure injection system.
Emergency Gas Evacuation System YR	During unit startup the system ensures gas evacuation when the primary circuit is filling When reactor is at nominal power, the system is not used and is maintained available. During unit shutdown, the system ensures that the primary equipment metal is cooled evenly (reactor head, SG headers pressurizer upper head)	The system ensures complete filling of the core with coolant for reliable heat removal due to gas bubbles in the upper dead-end sections of the primary piping that release from the coolant when its temperature decreases to saturation margin in case of emergency pressure decrease above the core. Primary pressure decrease to restore conditions permitting actuation of high-pressure injection system pumps.
Secondary Overpressure Protection System TX50,60,70,80	SG RV is in standby.	Steam generator vessel and piping overpressure protection in operational occurrences and emergencies.



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System	Intended Use under Normal Operation	Intended Use in Emergencies
Steam Dump Valve to Atmosphere TX50,60,70,80S05 (BRU-A, Russian designation)	The steam dump valves to atmosphere (BRU-A) is designed to ensure: <ul style="list-style-type: none"> <li>• reactor residual heat removal;</li> <li>• primary circuit cooldown in steam mode;</li> <li>• steam generator pressure control in case of failure of the steam dump valves to the condenser or prohibition of their opening.</li> </ul>	For secondary circuit overpressure protection under normal operation and emergency modes.
Emergency Feedwater System TX10,20,30.	In standby.	Emergency feedwater system (EFS) actuated in case of failure of the normal SG feedwater system. The EFS is actuated automatically for residual heat removal at the first stage of emergency cooldown.
Main Steam Isolation Valve System	In standby.	The main steam isolation valve system (MSIV) is designed to disconnect the broken steam line of the emergency steam generator in non-isolated section (before MSIV) from steam lines of serviceable steam generators to limit deep cooldown of the primary circuit through additional steam makeup of leak places: measure to prevent development of a reactivity accident.
<b>localization safety system</b>		
Containment System	Under normal operation, the containment system performs functions of a normal operation system, which include provision of a closed space to confine radioactive fission products released during NPP operation under various design modes.	Limitation of radioactive releases into the atmosphere in emergencies and design-basis accidents including maximum design-basis accident.
Containment spray system TQ11,21,31	In standby.	Accident confinement by condensation of steam generated as a result of primary coolant leak into the containment. Pressure decrease in the containment. Immobilization of iodine in the steam and air of the containment. Makeup and cooling of spent nuclear fuel in the spent fuel pool.
Containment Hydrogen Measurement System, passive catalytic hydrogen recombiners	In standby.	The hydrogen concentration reduction in the containment air to the non-explosive value in design-basis accidents.
<b>Control safety systems</b>		
Control and Protection System (CPS)	Reactor control during startup, operation at nominal power and in shutdown states	Reactor control during accidents. Emergency reactor unloading or reactor scram by the emergency protection system.
Control Safety System (CSS)	In standby until actuation signal is received, generates data on process parameters to be used by personnel and actuates alarms when major failures occur.	Actuation of safety systems, their control and monitoring during performance of safety functions.
<b>Supporting safety systems</b>		
Emergency diesel generators (DG)	In standby.	Independent emergency supply to safety systems. In emergency modes requiring the operation of safety systems, the diesel generators are actuated automatically and sequentially loaded.

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System	Intended Use under Normal Operation	Intended Use in Emergencies
Essential Power Supply System BV, BW, BX	Power supply from the working transformer bus through two bus switches on 6 kV bus and 6/0.4 transformers of the second reliability group. Power supply to equipment fed from the 0.4 kV essential power supply bus, first reliability group. Power supply to the uninterruptible power supply unit (UPS) of the direct-current board and charging of the battery.	Power supply to all safety system equipment.
Essential Service Water System (VF/QF)	Heat removal to the ultimate heat sink from normal operation system equipment (spent fuel pool, intermediate circuit, ventilation systems ...), as well as from safety system mechanisms.	Heat removal from the reactor core through the ECCS heat exchangers, cooling of safety systems' mechanisms.

The following upgrades were performed at Zaporizhzhya NPP units 1-6 to facilitate accident management:

- establishing long-term make-up of emergency feedwater tanks and steam generators (SG) under primary circuit emergency cooldown modes in loss of house loads power. In 2011 stationary piping was installed at unit 1 to supply cooling water from mobile sources (fire engines) under loss of power;
- introduction of long-term make-up of emergency feedwater tanks and SGs under primary circuit emergency cooldown modes in loss of house loads power using fire hoses and mobile pumping units (fire engines) with the following connection points: emergency feedwater pump discharge line TX10D01; emergency feedwater tank inlet pipe TX10D01 on the feedwater line; suction pipe from tanks TB40B01,02 of distillate pumps TN21(22,23) D01.

At Zaporizhzhya NPP units 1-6, the following is ensured to supply voltage to 6 kV auxiliary buses:

- power supply from the available units of the NPP site;
- power supply to 6 kV standby lines from diesel generators of any safety system of any power unit on the site or from the common-unit diesel generators;
- power supply from Dnipro Hydraulic Power Plant-1, Dnipro Hydraulic Power Plant-2, Zaporizhzhya Thermal Power Plant and Kakhovka Hydraulic Power Plant.

### **1.2 Z Significant Differences between Units**

Zaporizhzhya NPP units 1-6 are of typical WWER-1000/V320 design and have similar layout and arrangement. At the same time, there are the following significant differences between the units:

- the design provides for common-unit emergency diesel generators for units 5-6 that are housed in a separate building and are designed to provide power for the categories 1 and 2 loads along with normal operation loads which provide for good condition of basic technological equipment under loss of off-site power. The system consists of two trains with autonomous diesel generators;
- In 2011 stationary piping was installed at unit 1 to supply cooling water from mobile sources (fire engines) in case of loss of power.

There are no other significant differences between the Zaporizhzhya NPP units that would influence the assessment of units' vulnerability to external hazards or performance of safety functions and severe accident management.

### 1.3 Z Use of PSA as Part of Safety Assessment

Information on PSAs for Zaporizhzhya NPP units 1-6 is provided in Section 1 "General Data on the Sites and Nuclear Power Plants of Ukraine".

#### ***Safety Substantiation of Dry Spent Fuel Storage Facility (DSF)***

To substantiate safe operation of the DSF, a safety analysis report for the DSF was developed and approved by the State Nuclear Regulatory Inspectorate of Ukraine. The analysis included:

- strength analysis;
- analysis of thermal and pressure effects;
- biological shielding analysis;
- nuclear safety analysis;
- personnel dose analysis.

Based on the SAR results, conclusions were made on the DSF robustness against all postulated external hazards and its safe operation.

## 1 Kh GENERAL DATA ON THE KHMELNITSKY NPP

### 1.1 Kh Brief Description of the Site Characteristics

The Khmelnytsky NPP site is located in the Slavuta district of the Khmelnytsky region.



**Khmelnytsky NPP**

The distance from the Khmelnytsky NPP site to the district center (Slavuta town) is 15 km and to the regional center (Khmelnytsky town) is 100 km. The residential area closest to Khmelnytsky NPP (2.7 km of reactor buildings) is Neteshin town situated north-west of the NPP. The natural southern boundary of Neteshin is the Goryn river. An off-channel cooling reservoir is located in the south west of the NPP site in the floodplain of the Goryn river, which includes a retaining dam. In the south, the outdoor switchyard is sided by lands owned by the Slavuta State Forestry.

All industrial facilities within 30 km from the NPP are mainly located in the towns of Neteshin, Slavuta, Iziaslav and Ostrog. All NPP facilities are provided with an asphalt-

covered road network linked with the main road siding going from the NPP surrounding area to the regional roadway Stryi–Ternopil (P-05) on the Shepetovka-Ostrog-Berdichev highway.

The NPP site is connected with the South Western Railway (SWR) through the NPP railroad station adjacent to the Seltso station of SWR.

There are no gas or oil pipelines, factories or chemical plants within the 30-km area of Khmelnytsky NPP.

The Khmelnytsky NPP has been designed as a four-unit plant with WWER-1000/320 reactors 1000 MW each. To date, two power units have been constructed and commissioned, units 1 and 2, with a total electrical power of 2000 MW.

### **1.1.1 Kh Main Characteristics of the Units**

#### **1.1.1.1 Kh Reactor Facilities**

There are two WWER-1000/B-320 units with a total electrical power of 2000 MW at the Khmelnytsky NPP site.

The main equipment of Khmelnytsky NPP units 1 and 2 includes:

- water-cooled water-moderated pressurized power reactor WWER-1000 (V-320 series);
- turbine unit K-1000-60/3000;
- generator TVV-1000-2UZ.

The reactor facilities of Khmelnytsky NPP units 1 and 2 are of the WWER-1000/V-320 standard design and are similar to Zaporizhzhya NPP units 1-6.

#### **1.1.1.2 Kh Spent Fuel Storage Facilities**

The Khmelnytsky NPP does not have any on-site spent fuel storage facilities.

Spent fuel removed from the reactor core is stored in storage racks of spent fuel pools. The spent fuel storage system is arranged in the reactor building with all needed rooms and equipment to receipt and store spent fuel assemblies. The design of spent fuel pools of Khmelnytsky NPP units 1 and 2 is similar to that of Zaporizhzhya NPP units 1-6 (See Section 1.1.1.2 Z).

### **1.1.2 Kh Description of the Systems for Conduction of Main Safety Functions**

The configuration of the main safety systems and normal operation systems important to safety of Khmelnytsky NPP units 1 and 2 is similar to that of Zaporizhzhya NPP units 1-6. The systems that perform safety functions are described in Section 1.1.2 Z.

In addition, several upgrades were implemented at Khmelnytsky NPP units 1 and 2 for accident management:

- establishing of long-term make-up of emergency feedwater tanks and steam generators under primary circuit emergency cooldown modes in loss of house loads power ;
- introduction of long-term make-up of emergency feedwater tanks and SGs under primary circuit emergency cooldown modes in loss of power to in-house loads using fire hoses and mobile pumping units (fire engines) with the following connection points: emergency feedwater pump discharge line TX10D01; emergency feedwater tank inlet pipe TX10D01 on the feedwater line; suction pipe from tanks TB40B01,02 of distillate pumps TN21(22,23)D01.

The upgrades implemented at Khmelnytsky NPP units 1 and 2 also included the following features to supply voltage to 6kV auxiliary buses:

- power supply from the available units of the NPP site;
- power supply to 6kV standby line from diesel generators of safety systems of any unit located on site or from emergency diesel generators.

## 1.2 Kh Significant Differences between Units

Khmelnitsky NPP units 1 and 2 are of identical design and have similar reactor facilities, layout and arrangement. However, similarly to Zaporizhzhya NPP Units 5 and 6, the design of Khmelnitsky NPP unit 2 additionally provides for a common-unit essential power supply system including two trains with independent diesel generators. The power of each diesel generator (5600 MW) was selected to ensure its startup in case of blackout of both power units (in case one of the diesel generators fails).

It should be noted that an ALSTOM hydrogen monitoring system has been installed in the containment of the unit 2 to monitor hydrogen concentration in all operating modes, while this system has not been installed at unit 1, which is a significant difference between the units relative to accident management capabilities.

In all other respects, there are no significant differences between the units that may affect the assessment of their vulnerability to external hazards and performance of safety functions.

## 1.3 Kh Use of PSA as Part of Safety Assessment

Information on PSAs for Khmelnitsky NPP units 1 and 2 is provided in Section 1 "General Data on the Sites and Nuclear Power Plants of Ukraine".

## 1 R GENERAL DATA ON THE RIVNE NPP

### 1.1 R Brief Description of the Site Characteristics

The Rivne NPP site is located in the north west of Ukraine at the border of Rivne and Volyn' regions.



**Rivne NPP**

The distance from the Rivne NPP site to:

- Kuznetsovsk town westward – 3 km;
- district centre of Vladimirets to north-east – 20 km;
- regional centre of Rivne to south-east – 80 km;
- regional centre of Lutsk to south-west – 75 km.

All industrial plants within the 30-km area of the NPP are mainly located in Kuznetsovsk and Vladimirets of the Rivne region and Manevichi of the Volyn' region.

There are no gas or oil pipelines, factories or chemical plants in the Volyn' region within the 30-km area of the Rivne NPP.

The Kiev–Kovel highway extends 20 km to the south of the Rivne NPP site. To provide the road connection to the Rivne NPP and in-site connections between different plant facilities a system of roadways was constructed.

The main river crossing the whole territory of the 30-km area from south to north over 113 km is Styř' river that is used to supply service water to the Rivne NPP.

There are four units with a total electrical power of 2835 MW at the Rivne NPP site: two WWER-440/V-213 units (units 1 and 2) and two WWER-1000/V-320 units (units 3 and 4).

**1.1.1 R Main Characteristics of the Units**

**1.1.1.1 R Reactor Facilities**

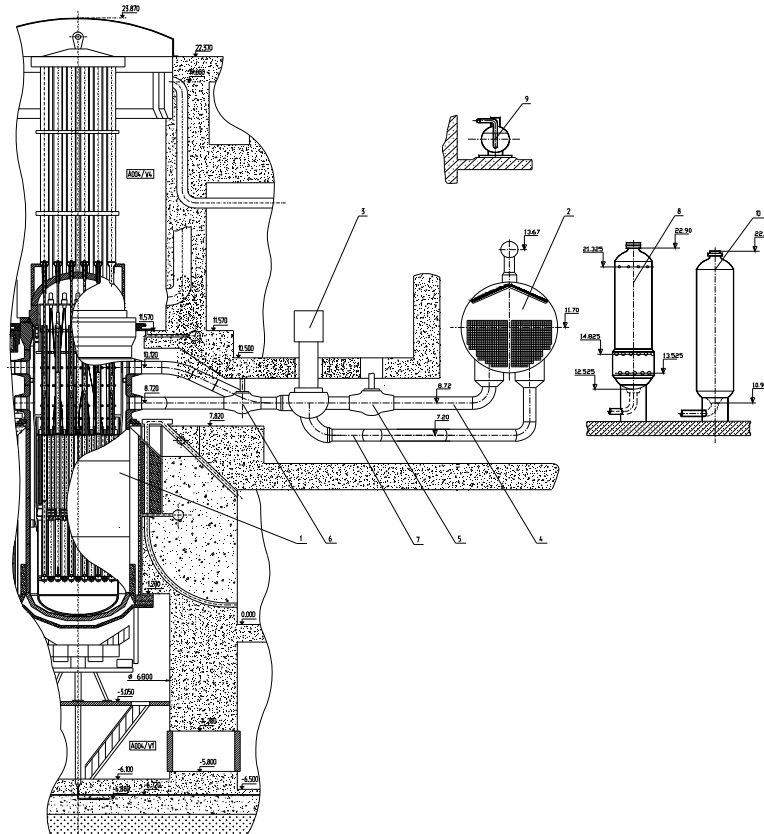
**Units 1 and 2**

There are two WWER-440/213 units (units 1 and 2) with a total electrical power of 835 MW at Rivne NPP.

Major equipment of RNPP units 1 and 2 includes:

- water-cooled water-moderated reactor WWER-440 (V-213);
- turbine unit K-220-44;
- generator TVV-220-2AUZ.

The reactor facilities of Rivne NPP units 1 and 2 consist of two circuits. The reactor facilities of each power unit are located in separate buildings, and turbine generators in a common turbine building.



**Layout of WWER-440/V-213 Major Equipment**

1 - reactor; 2 – steam generator; 3 – main coolant pump; 4 – “hot” leg; 5 - main gate valve on hot leg; 6 – main gate valve on cold leg; 7 - “cold” leg; 8 - pressurizer; 9 – pressure relief tank; 10 – ECCS hydroaccumulator

Systems and equipment of WWER-440/V-320 assure:

- safe and reliable operation of the reactor in all normal operation modes envisaged by the design;
- control of operations and monitoring of major reactor equipment;
- integrity of the reactor in all design-basis accidents;
- integrity and reliable cooling of fuel assemblies in the reactor core under all operation modes envisaged by the design.

The design of the reactor facility provides protection of NPP personnel and the environment against radiation according to health and safety standards in all operation modes provided by the design (including anticipated transients and design-basis accidents).

To remove heat from turbine condensers and heat exchangers of auxiliary equipment, cooling towers of 100,000 m<sup>3</sup>/h capacity (each tower) are used. To remove heat from essential loads, spray ponds are used.

The reactor facilities of units 1 and 2 are located inside containments of reactor buildings.

#### **Units 3 and 4**

There are also two WWER-1000/V-320 units (units 3 and 4) with a total electrical power 2000 MW at the Rivne NPP.

Major equipment of RNPP units 3 and 4 includes:

- water-cooled water-moderated reactor WWER-1000 (V-320);
- turbine K-1000-60/3000;
- generator TVV-1000-2UZ.

The reactor facilities of Rivne NPP units 3 and 4 are of WWER-1000/V320 standard design and are similar to Zaporizhzhya NPP units 1-6.

##### **1.1.1.2 R Spent Fuel Storage Facilities**

There are no on-site spent fuel storage facilities on the Rivne NPP site.

#### **Units 1 and 2**

The spent fuel assemblies unloaded from the core are stored in racks in the spent fuel pools (SFP). Before placing for storage, the fuel assemblies are subjected to fuel cladding leak testing. Based on the testing results, a spent fuel assembly is placed either in the rack slots or in a sealed canister. The spent fuel storage system is housed in the reactor compartment provided with all necessary rooms and equipment to receive and store the spent fuel assemblies.

The SFP is located within the steam generator box and is connected with the reactor through refueling channels for transportation of fuel assemblies. Spent fuel is stored and reloaded under a protective layer of water. The pool is equipped with a compacted fuel storage system and storage racks that accommodate spent fuel assemblies, shield assemblies, sealed canisters for damaged fuel and special tools.

The SFP is filled with a boric acid solution of at least 16 g/dm<sup>3</sup> concentration. Water in each SFP compartment circulates through the SFP cooling system that keeps the SFP water temperature within a permissible limit of no more than 70°C (with the reactor core completely unloaded). This value is established to prevent boiling of the cooling water and fuel melting induced by decay heat. In addition, the cooling system ensures a protective level of water in the SFP during spent fuel storage and provides radiation protection for maintenance personnel.

In case of emergency unloading of spent fuel from one power unit, the spent fuel pool of the other power unit may be used if necessary.

## Units 3 and 4

The spent fuel assemblies unloaded from the reactor core are stored in racks in the spent fuel pools (SFP). The spent fuel storage system is housed in the reactor compartment provided with all necessary equipment to receive and store the spent fuel assemblies. The design of SFP of Rivne NPP units 3 and 4 is similar to that of Zaporizhzhya NPP units 1-6.

### 1.1.2 R Description of the Systems for Conduction of Main Safety Functions

## Units 1 and 2

The reactor coolant system (RCS) of WWER-440/V-213 consists of six circulation loops. Each loop includes: RCS piping; main coolant pump (MCP); steam generator; and main gate valves to cut off any loop from the reactor. Each of the six circulation loops consists of "hot" and "cold" legs: "hot" leg is used to transfer the coolant from the reactor to the steam generator; "cold" leg is used to transfer the coolant cooled down in SG to MCP intake and back to the reactor. The cold and hot legs of each loop are provided with main gate valves (MGV) for isolation of the loop with a damaged component in case of accidents (SG or RCS piping leakage in the cutoff section etc.).

The RCS piping is designed to prevent radioactive releases to the environment. The design ensures normal operation of the reactor facility subjected to earthquake-induced loads reaching the design-basis values, and ensures safe shutdown and cooling.

The PGV-213 steam generator is a single-pass recuperative horizontal heat-exchanger with a tubing bundle immersed into it and is designed for generation of dry saturated steam.

The main coolant pump GTsN-317 is designed for coolant circulation in the primary circuit and is a vertical single-stage centrifugal pump with hydrostatic seal, console work wheel, axial water supply and a self-contained three-phase asynchronous electric motor with a short-circuited rotor.

The MGV with an in-built motor is used to cut off the loop during accidents caused by steam line break and primary-to-secondary SG leakage when the reactor is already scrammed. In normal operation, MGV is designed to transfer the loop to hot standby with relevant decrease in reactor power.

To achieve the main safety goal, which is to prevent radioactive release beyond the physical barriers, the design provides for the following safety functions:

- reactivity control;
- reactor core cooling;
- confinement of radioactive substances within the established boundaries.

These safety functions in compliance with the defense-in-depth principle are inherent in the WWER design.

### **Reactivity Control**

Reactivity control and reactor transfer to sub-criticality as well as maintaining the reactor core sub-critical are ensured through reactivity control systems based on different principles:

- system of control rod mechanical movement in the reactor core (reactor power adjustment and scram – reactor protection system);
- boric acid control system to regulate boric acid concentration in the coolant (the system is designed to control reactivity related to fuel burnup and transients as well as to keep the core subcritical in reactor shutdown and transfer to cold shutdown or hot standby mode).

The control rods are controlled through the control and protection system (CPS). Control rods are placed in the uppermost position when the reactor is at nominal power, except for the control rod CPS. Reactor scram is achieved by insertion of all absorber rods into the core in the shortest time.

In normal operation, the boron control function rests with the primary makeup and letdown system.



In emergency modes, there are provisions for the regulation of boric acid concentration in the primary coolant using safety systems integrated in the ECCS (emergency core cooling system):

- high pressure injection system;
- passive ECCS (hydroaccumulators);
- low-pressure injection system.

Each system, except for hydroaccumulators, is provided with three trains.

Emergency systems are actuated automatically in response to emergency signals, the operating mode and sequence of actuation and operation depends on the initiating event.

In emergencies, the ECCS systems supply the primary circuit with the boron-containing coolant (thus ensuring the reactor core transfer to subcriticality and maintaining the core subcritical for a long time after reactor scram) and perform the “core cooling” safety function.

To ensure nuclear safety in reactivity control, neutron flux parameters (density, neutron power, reactivity and period) are monitored on a permanent basis by appropriate instrumentation and control systems; concentrations of boric acid in the reactor and trains of safety systems are monitored as well.

### **Core Cooling**

The key objective of this function is to prevent fuel damage as a result of overheating. For this purpose, there are heat removal systems and equipment in all operational modes. Heat removal from the reactor core under WWER normal operation proceeds as follows: reactor core – primary coolant – steam generator – secondary coolant – ultimate heat sink – atmosphere. During accidents and after reactor shutdown, residual heat is removed in the same sequence. If steam generators are unavailable for heat removal function, the heat is removed through the emergency core cooling system (ECCS) and essential service water system.

#### *Core cooling under loss-of-coolant accident*

In case of LOCA which cannot be compensated for by the makeup and letdown system, the emergency core cooling systems are actuated. ECCS (HPIS, LPIS and hydroaccumulators) supplies the core with concentrated boric acid solution to maintain sufficient coolant inventory in the primary circuit to ensure core cooling, residual heat removal and maintain the core subcritical due to primary circuit makeup with borated water.

#### *Core cooling during cold shutdown or refueling*

Low-pressure injection system is used for this purpose. All three system trains provide borated water supply to the reactor lower or upper plenum, with the flow direction selected automatically depending on the mode; in case of cooldown for maintenance purposes, water is supplied only to the upper plenum. Two trains are connected to the main borated water pipes from hydroaccumulators, while the third one to cold and hot legs of the primary circulation loop.

### **Limitation of Radioactivity Release**

This safety function is aimed at preventing radioactive releases beyond the nuclear power plant site. In case of primary circuit integrity failure, the radioactive releases to the environment are prevented by the localization safety system maintaining negative pressure by means of ventilation systems.

To limit radioactivity releases in accidents, the design provides for localization safety systems (components), which are systems (components) designed to prevent or limit release of radioactive materials and ionizing radiation beyond the boundaries specified in the design.

The localization safety system includes structural components and devices to limit the accident confinement area represented by space where, according to the design, radioactive substances are confined during and after an accident.

All equipment operating on radioactive coolant is located in containment compartments. All containment compartments are made of reinforced concrete and designed to reduce radiation doses both in normal operation and during all accidents. The localization safety system is designed to sustain all design-basis loads resulting from coolant system failure, up to DN 500 rupture of the main circulation piping and to confine radioactive materials inside the containment, which ensure radiation parameters outside the containment within the permissible limits according to current standards.

### Safety Systems

The safety systems are designed to perform safety functions in case of an initiating event and any independent single failure of one active or passive component with mechanical moving parts, or an additional human error.

All NPP safety systems are designed according to the redundancy principle. Each safety system consists of three independent trains. Each train in terms of its efficiency, response time and other factors is sufficient to perform the intended functions in the full scope defined for a given system under any operational mode, including maximum design-basis accident (double-ended break of the main coolant piping DN 500 mm under loss of of-site power).

Independence of each safety system trains is ensured by complete physical separation of the trains. Equipment of the respective safety system trains is actuated automatically in response to independent signals.

### WWER-440/V-213 Main Safety Systems and Their Functions

System	Composition of System (Train)	Intended Use in Emergencies	Intended Use in Normal Operation
<b>Protective Safety Systems</b>			
Control & Protection System (CPS) (mechanical side)	The mechanical circuit of CPS consists of CPS control rods (37 groups of control rods). All control rods are divided in groups (1÷6)	Transfer to and keep the reactor core in subcritical state	Compensate for reactivity changes and control reactor power
High-pressure Injection System	The system consists of three independent trains. Each train includes: boric acid storage tank with 40g/kg boron solution; primary emergency makeup pump; valves; piping	Emergency injection of boron solution to the reactor core	Under normal operation, the system is in standby; the system is periodically tested in compliance with the operating procedures
Passive ECCS (ECCS hydroaccumulators)	Consists of two independent trains. Each train includes two boric acid hydroaccumulators with 16g/kg boron solution	Emergency supply of boric acid to the reactor for core cooling and filling in case of LOCA with primary pipe break	Under normal operation, the system is in standby
Low-pressure Injection System	Consists of three independent trains. Each train includes: boric acid storage tank with 16 g/kg boron solution; reactor coolant pump; ECCS heat exchanger; valves; piping	Emergency cooldown of the reactor core and further residual heat removal for a long period in LOCA	Under normal operation, the system is in standby; the system is periodically tested in compliance with the operating procedure; it is used during refueling operations to transfer heat from the reactor core to ultimate heat sink
Primary Overpressure Protection System	Consists of two self-contained pilot-operated safety relief valves of the pressurizer	Protect RCS against primary overpressure	Under normal operation, the system is in standby;

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System	Composition of System (Train)	Intended Use in Emergencies	Intended Use in Normal Operation
Secondary Overpressure Protection System	Each steam generator is equipped with two pilot-operated safety relief valves	Secondary system overpressure protection in operational occurrences and emergencies.	Under normal operation, the system is in standby;
Primary Emergency Gas Evacuation System	Consists of piping connecting SG headers and RPV head with the pressure relief tank	The system ensures complete filling of the core with coolant for reliable heat removal due to removal of gas bubbles in the upper dead-end sections of the primary piping that release from the coolant when its temperature decreases to saturation margin in case of emergency pressure decrease above the core.	Under normal operation, the system is in standby;
Emergency Reactor Cooldown System	Consists of two Steam Dump Valve to Atmosphere	Remove heat from the reactor in emergencies caused by unit blackout, design-basis seismic loads and other emergencies resulting from compromised heat removal from the reactor	Not used, maintained available (in automatic standby)
Main Steam Isolation Valves System	Each SG is equipped with one MSIV, and three valves are installed on the main steam header dividing it into two symmetric sections	The system is designed to disconnect the emergency steam generator with broken steam line in non-isolated section (before MSIV) from steam lines of serviceable steam generators to limit deep cooldown of the primary circuit due to additional steam makeup: measure to prevent development of a reactivity accident.	In standby.
Emergency Feedwater System	Consists of two feedwater tanks, three auxiliary motor-driven emergency feedwater pumps, piping, valves; control	Support the required feedwater level in SG to ensure core heat removal and prevent unacceptable overheating and damage of fuel in accidents. The system is used in unit emergency cooldown	In standby.
<b>Localization safety system</b>			
Containment System	Consists of: reinforced concrete enclosing structures with leak-tight steel lining (embedded parts, including anchors); penetrations (process, cable, ventilating etc.), airlocks, hatches, doors and their embedded parts; overflow and safety valves; distribution line sections intersecting the containment within confining structures; isolating structures	Confine radioactive substances releases within the accident confinement area, isolate from the environment the systems and components, protect personnel and the public against radiation and radioactivity effects	The system is used to create and maintain rarefaction inside compartments adjacent to the reactor, is maintained available and periodically inspected according to the established procedures

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System	Composition of System (Train)	Intended Use in Emergencies	Intended Use in Normal Operation
Passive Hydrogen catalytic Recombiners (PAR)	Consists of PAR modules to ensure sufficient hydrogen removal rate inside the containment,	The hydrogen concentration reduction in the containment air to the non-explosive value in design-basis accidents.	In standby.
Passive Steam Condenser	Consists of passive confining features – air trap rooms and suppression pools filled with boric acid solution	Protect containment compartments against overpressure in medium and large primary coolant leaks by condensation of steam generated during an accident that caused by primary piping break; remove radioactive products from the steam-gas mixture during operation of the passive steam condenser; prevent emergency radioactive releases into the environment	Not used
Spray System	Consists of three independent trains. Each train includes: containment spray pump; boric solution tank; piping; valves	Reduce pressure to rarefied atmosphere and maintain it inside the containment during accidents caused by primary and secondary piping rupture; Immobilize iodine contained in steam-gas medium; mix the steam-gas medium in compartments	In standby.
<b>Supporting Safety Systems</b>			
Essential Service Water System	Consists of three independent trains. Each train includes: on spray pond section; emergency storage tank; two essential service water pumps; piping; valves	Heat removal from the reactor core through the ECCS heat exchangers, cooling of safety systems' mechanisms.	Heat removal to the ultimate heat sink from normal operation system equipment (spent fuel pool, intermediate circuit, ventilation systems ...), as well as from safety system mechanisms.
Emergency Power Supply System	Consists of three independent trains. Each train includes self-contained power supply sources (batteries); transducers (rectifiers, inverters); switchyards and distribution system (cables in cable routings)	Provide uninterruptable power supply to safety system loads of reliability group 1 in normal operation, operational occurrences and emergencies (including unit blackout)	Used for equipment designed to perform safety functions; remains in continuous operation; the system is periodically tested in compliance with the operating procedures
System of Emergency Diesel Generators	Consists of three independent trains. Each train includes one diesel-generator with self-contained fuel, oil, cooling water, power, ventilation and compressed-air systems	Provide power supply to safety system loads of reliability group 2 during unit blackout	Not used, maintained available (in automatic standby); the system is periodically tested in compliance with the operating procedures
<b>Control Safety Systems</b>			

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System	Composition of System (Train)	Intended Use in Emergencies	Intended Use in Normal Operation
Control and Protection System (CPS)	Consists of: electric control equipment for control rods; power supply system and system for conversion of power from NPP in-house sources and distribution to other loads; two independent sets of process control equipment; two independent process monitoring equipment sets; two independent neutron flux monitoring sets (AKNP-7); power limiting device ROM-2S; automatic power controller ARM-5S	Initiate automatic reactor core transfer to subcritical state; protect the reactor core against unacceptable deviation of basic parameters; provide automatic insertion of all control rods to the lowermost position	Reactor power control, including automatic control; limitation of reactor thermal power; monitoring of neutron power, control of the neutron flux rate; the system is periodically tested in compliance with the operating procedures
ECCS Protection Alarm System	Consists of three independent trains. Each train includes: process instrumentation circuit that consists of three independent instrumentation channels; signal monitoring circuit and sensor mismatch alarm circuit; fuse check or cutoff circuit; power control circuit for output and testing relays; ECCS protection testing circuit; safety system actuation relay system	Activate safety systems in case of emergencies (including unit blackout)	Not used, maintained available (in automatic standby); the system is periodically tested in compliance with the operating procedures

In case of common-cause failure of all main and emergency feedwater supply systems, feedwater is supplied through the steam generator additional emergency steam generator feedwater system (AEFS). The system was installed and commissioned in 2010. The AEFS is located in a separate building that was specially constructed; feedwater tanks are located outside the AEFS building on individual foundations and are connected with the AEFS building by an underground channel. The system is designed to:

- keep the required coolant level in steam generators to ensure heat removal from the reactor core,
- provide emergency shutdown of units 1 and 2 and their safe transfer to cold state in case the main and emergency feedwater supply systems fail under different initiating events.

This system is common for Rivne NPP units 1 and 2 and consists of three independent subsystems, each designed for the relevant unit. The subsystems consist of:

- one diesel-operated pump RR91(92)D01 with auxiliary generators;
- one feedwater tank RR91(92)B01 (1000 m<sup>3</sup>);
- main pump pressure header.

This system is independent and survivable because its performance does not depend on other safety systems and normal operation systems. AEFS pumps are driven directly by their own air-cooled diesel engines. In case of station blackout, the AEFS inventory is sufficient to ensure long-term decay heat removal. Also, the AEFS tanks can be refilled.

The following power supply options are ensured at Rivne NPP units 1 and 2 in case of blackout:

- power supply from the available units of the NPP site power supply to 6 kV standby lines from diesel generators of any safety system of any power unit on the site or from the common-unit emergency diesel generators.

#### **Units 3 and 4**

The configuration of the main safety systems and normal operation systems important to safety of Rivne NPP units 3 and 4 is similar to that of Zaporizhzhya NPP units 1-6. The systems that perform safety functions for WWER-1000/320 are described in Section 1.1.2 Z.

The following upgrades were performed at Rivne NPP unit 3 and 4 for accident management:

- introduction of long-term make-up of emergency feedwater tanks and steam generators (SG) under primary circuit emergency cooldown modes in loss of house loads power ;
- introduction of long-term make-up of emergency feedwater tanks and SGs under primary circuit emergency cooldown modes in loss of power to in-house loads using fire hoses and mobile pumping units (fire engines) with the following connection points: emergency feedwater pump discharge line TX10D01; emergency feedwater tank inlet pipe TX10D01 on the feedwater line; suction pipe from tanks TB40B01,02 of distillate pumps TN21(22,23)D01.

The upgrades implemented at Rivne NPP units 3 and 4 also included the following options in case of blackout:

- power supply from the available units of the NPP site;
- power supply to 6 kV standby lines from diesel generators of any safety system of any power unit on the site or from the common-unit emergency diesel generators.

### **1.2 R Significant Differences between Units**

#### **Units 1 and 2**

Rivne NPP units 1 and 2 are of standard WWER-440/V213 design and have similar layout and arrangement and have no significant differences that may affect assessment of their vulnerability to external hazards, performance of safety functions or severe accident management.

#### **Units 3 and 4**

Rivne NPP units 3 and 4 are of standard WWER-1000/V320 design and have similar reactor facilities, layout and arrangement. However, the design of unit 4 additionally provides for a common-unit emergency diesel generator system including two trains with independent diesel generators. The power of each diesel generator (5600 MW) was selected to ensure its startup in case of blackout of both power units.

There are also differences between units in terms of implementation of safety improvement measures: modernization of CPS control system, including power supply system has been implemented at unit 3 and has not been started yet at unit 4.

In all other respects, there are no significant differences between units 3 and 4 that may affect the assessment of their vulnerability to external hazards and performance of safety functions.

### **1.3 R Use of PSA as Part of Safety Assessment**

Information on PSAs for Rivne NPP units 1-4 is provided in Section 1 "General Data on the Sites and Nuclear Power Plants of Ukraine".

## 1 SU GENERAL DATA ON THE SOUTH UKRAINE NPP

### 1.1 SU Brief Description of the Site Characteristics

The South Ukraine NPP site is located in the Arbuzinka district in the Mykolayiv region, on the left bank of the Yuzhny Bug river.



### South Ukraine NPP

The South Ukraine NPP with all main components of the water supply system is located on the left bank of the Yuzhny Bug river, at a distance of approximately 159 km from its estuary. From south to north, the territory of the South Ukraine NPP site is crossed by the Tashlyk water reservoir. From north-west to south-east the Yuzhny Bug river flows 60 km through the 30 km area. The nearest main building (unit 1) is 2.7 km from the river bank.

There is no woodland in the NPP area.

The Ulianovka-Mykolayiv primary motor road constructed around the Tashlyk water reservoir passes at a distance of 1050 m from the South Ukraine NPP site fencing. This motor road is intended for general use and is closest to the South Ukraine NPP.

The closest railroad station is located at a distance of 2.2 km from the NPP site.

There are no gas or oil pipelines, factories or chemical plants within the 30-km area of the South Ukraine NPP.

The town of Yuzhnoukrainsk is located at a distance of 2.5 km from the South Ukraine NPP. The town of Voznesensk (at a distance of 30 km) and several urban-type settlements and villages are located within the 30-km area. The nearest big city situated beyond the borders of the 30 km area is the regional center, Mykolayiv, located at a distance of 112 km from the SUNPP.

There are three WWER-1000 units in operation at the South Ukraine NPP. They are unit 1 with V-302 reactor, unit 2 with V-338 reactor and unit 3 with V-320 reactor, with a total electrical power of 3000 MW.

### 1.1.1 SU Main Characteristics of the Units

#### 1.1.1.1 SU Reactor Facility

Major equipment of South Ukraine NPP unit 1 includes:

- water-cooled water-moderated reactor WWER-1000 (V-302);
- turbine unit K-1000-60/1500 with side-inlet condensers;
- generator TVV-1000-4UZ.

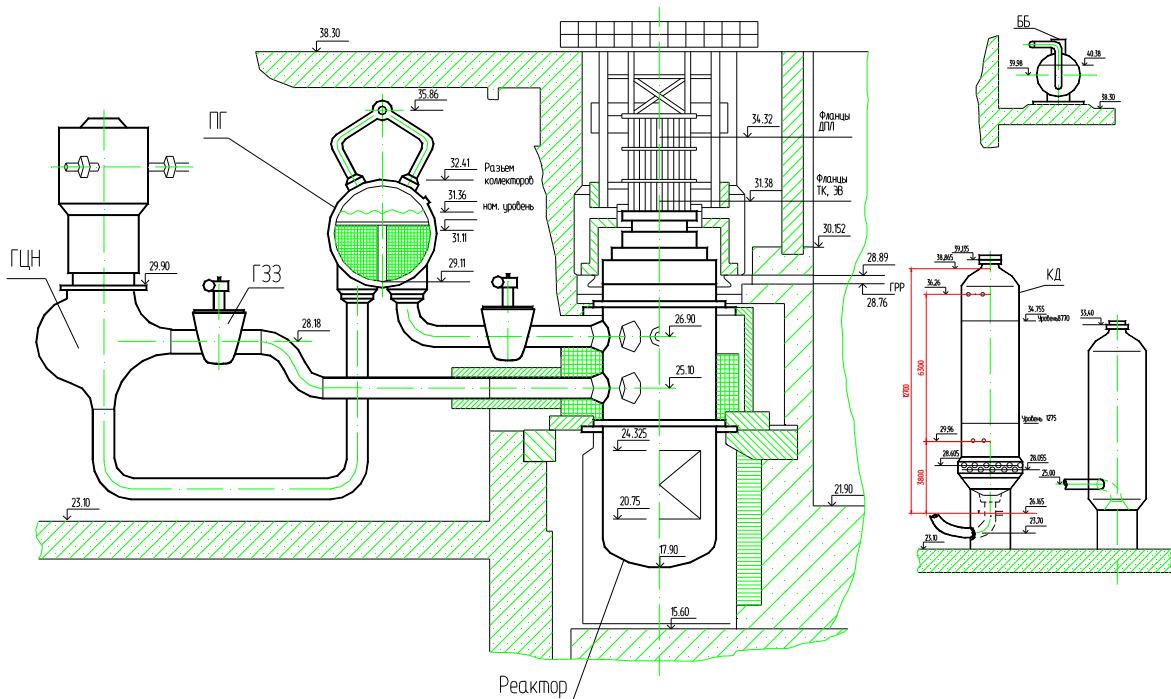
Major equipment of South Ukraine NPP unit 2 includes:

- water-cooled water-moderated reactor WWER-1000 (V-338);
- turbine unit K-1000-60/1500;
- generator TVV-1000-4UZ.

Major equipment of South Ukraine NPP unit 3 includes:

- water-cooled water-moderated reactor WWER-1000 (V-320);
- turbine unit K-1000-60/3000;
- generator TVV-1000-2UZ.

South Ukraine NPP units 1 and 2 are designed to consist of two circuits with water-cooled water-moderated reactors WWER-1000 (V-302/338 design) operating on thermal neutrons with pressurized water with an electrical power of 3000 MW. The designs of South Ukraine units 1 and 2 are similar.



Layout Diagram of WWER-1000/V-320 Main Equipment

Systems and equipment of WWER-1000/V-302/338 provide:

- safe and reliable operation of the reactor in all normal operation modes envisaged by the design;
- control of operations and monitoring of major reactor equipment;
- integrity of the reactor in all design-basis accidents;
- integrity and reliable cooling of fuel assemblies in the reactor core in all operation modes provided by the design.

The reactor facility of South Ukraine NPP unit 3 is developed according to WWER-1000/V-320 standard design and is similar to Zaporizhzhya NPP units 1-6.

Reactor equipment is housed in the pre-stressed leak-tight reinforced-concrete containment having the shape of a hollow cylinder with a spherical dome and flat bottom.



### **1.1.1.2 SU Spent Nuclear Fuel Storage Facilities**

The spent fuel assemblies unloaded from the core are stored in racks in the spent fuel pools (SFP). Before placing for storage, the fuel assemblies are subjected to fuel cladding leak testing. Based on the testing results, a spent fuel assembly is placed either in the rack slots or in a sealed canister. The spent fuel storage system is housed in the reactor compartment provided with all necessary rooms and equipment to receive and store the spent fuel assemblies.

The SFP of units 1 and 2 is housed in the reactor containment and consists of two compartments: assembly compartment designed for storage of spent fuel assemblies and the container compartment provided with a stationary rack and multi-purpose slot that is used as the area for loading of transport casks with spent fuel assemblies and unloading of a fresh fuel assembly casing. Dividing the SFP into compartments allows for maintenance in one of them while spent fuel assemblies are placed into the other. The SFP is adjacent to the reactor pressure vessel and is connected with it by a refueling channel for transport of fuel assemblies. The pool is equipped with spent fuel storage racks.

The SFP cooldown system is intended for removal of residual heat from spent nuclear fuel stored in the cooling pools.

The SFP is filled with a boric acid solution of at least 16 g/dm<sup>3</sup> concentration. Water in each SFP compartment circulates through the SFP cooldown system that keeps the SFP water temperature within a permissible limit of no more than 70°C (with the reactor core completely unloaded). This value is established to prevent boiling of the cooling water and fuel melting induced by decay heat. In addition, the cooldown system ensures a protective level of water in the SFP during spent fuel storage and provides radiation protection for maintenance personnel.

In an emergency situation there is a possibility for makeup and fuel cooling in SFP compartments with a 16 g/dm<sup>3</sup> boric acid solution from pumps of the containment spray system. In this case makeup and water cooling in SFP proceeds as follows: containment sump – emergency cooldown heat exchanger – containment spray pump – SFP compartment – SFP flow through the transport opening to the refueling pool – water flow through the lower elevation of the containment to the containment sump.

The design of the spent fuel pool of South Ukraine unit 3 is similar to that of Zaporizhzhya NPP units 1-6.

### **1.1.2 SU Description of the Systems for Conduction of Main Safety Functions**

#### **Units 1 and 2**

The reactor coolant system (RCS) consists of a reactor and four circulation loops, each consisting of a steam generator PGV-1000M, main coolant pump (MCP) GTsN-195M, and RCS piping. Each loop of the primary circuit is divided into “cold” and “hot” legs. The “hot” leg supplies the coolant from the reactor to the steam generator, while the “cold” leg returns the coolant from the steam generator to the reactor via the MCP. The design of all loops is similar, ensuring equal hydraulic resistance of the loops and, thus, equal coolant flow rate. The MCP system is designed to meet the requirements for prevention of radioactivity release into the environment. The RCS piping design ensures normal operation of the reactor facility subjected to earthquake-induced loads reaching the design-basis values, and ensures safe shutdown and cooling. Thermal expansions of the RCS piping are compensated by moving the steam generators and MCP that rest on rolling supports. To achieve the main safety goal, which is to prevent radioactive releases beyond the physical barriers, the design provides for the following safety functions:

- reactivity control;
- reactor core cooling;
- confinement of radioactive substances within the established boundaries.

These safety functions in compliance with the defense-in-depth principle are inherent in the WWER design.

### **Reactivity Control**

Reactivity control and reactor transfer to sub-criticality as well as maintaining the reactor core sub-critical are ensured through reactivity control systems based on different principles:

- system of control rod mechanical movement in the reactor core (reactor power adjustment and scram – reactor protection system);
- boric acid control system to regulate boric acid concentration in the coolant (the system is designed to control reactivity related to fuel burnup and transients as well as to keep the core subcritical in reactor shutdown and transfer to cold shutdown or hot standby mode).
- The control rods are controlled through the control and protection system (CPS). Control rods are placed in the uppermost position when the reactor is at nominal power, except for the control rod CPS. Reactor scram is achieved by insertion of all control rods into the core in the shortest time.

In normal operation, the boron control function rests with the primary makeup and letdown system.

In emergency modes, there are provisions for the regulation of boric acid concentration in the primary coolant using safety systems integrated in the ECCS (emergency core cooling system):

- high-pressure boron injection system (TJ13);
- high-pressure injection system (HPIS) (TJ10);
- passive ECCS (hydroaccumulators TH51-54);
- low-pressure injection system (TH10-30).

Each of the ECCS systems has three trains. The passive ECCS consists of 4 hydroaccumulators: with the two supplying water to the reactor upper plenum, and the other two to the lower plenum.

Emergency systems are actuated automatically in response to emergency signals, the operating mode and sequence of actuation and operation depends on the initiating event.

In accidents, the ECCS systems supply the primary circuit with the boron-containing coolant (thus ensuring the reactor core transfer to subcriticality and maintaining the core subcritical for a long time after reactor scram) and perform the “core cooling” safety function.

### **Core Cooling**

The key objective of this function is to prevent fuel damage as a result of overheating. For this purpose, there are heat removal systems and equipment in all operational modes. Heat removal from the reactor core under WWER normal operation proceeds as follows: reactor core – primary coolant – steam generator – secondary coolant – ultimate heat sink – atmosphere. During accidents and after reactor shutdown, residual heat is removed in the same sequence. If steam generators are unavailable for heat removal function, the heat is removed through the emergency core cooling system (ECCS) and essential service water system.

#### *Fuel cooling in normal operation and anticipated operational occurrences under primary circuit integrity*

At the first cooldown stage, the heat from the reactor core is removed through the secondary circuit system. Steam is removed through the steam dump valve to condenser (BRU-K) or the steam dump valve to atmosphere (BRU-A) in case of loss of off-site power or any other initiating event when BRU-K operation is impossible, as well as through steam generator safety relief valves in case both BRU-K and BRU-A are unavailable. When water supply to steam generators through the main and auxiliary feedwater systems is impossible, feedwater is supplied by the emergency feedwater system. In normal operation, small primary leaks are compensated by the makeup and letdown system. In case of LOCA which can not be compensated by the make-up system, ECCS systems (HPIS, LPIS and

hydroaccumulators) are used for supplying borated water to the reactor for residual heat removal under accident conditions.

**Limitation of Radioactivity Release**

This safety function is aimed at preventing radioactive releases beyond the nuclear power plant site.

In accidents, fuel can be damaged due to overheating and thus radioactive substances will release into the primary circuit. In case of loss-of-coolant accident, the containment prevents the radioactive release into the environment.

To limit radioactivity releases in accidents, the design provides for localization safety system, which are systems (components) designed to prevent or limit release of radioactive materials and ionizing radiation beyond the boundaries specified in the design.

**Safety Systems**

The safety systems are designed to perform safety functions in case of an initiating event and any independent single failure of one active or passive component with mechanical moving parts, or an additional human error.

All NPP safety systems are designed according to the redundancy principle. Each safety system consists of three independent trains. Each train in terms of its efficiency, response time and other factors is sufficient to perform the intended functions in the full scope defined for a given system under any operational mode, including maximum design-basis accident (double-ended break of the main coolant piping DN 850 mm under loss of of-site power).

Independence of each safety system trains is ensured by complete physical separation of the trains. Equipment of the respective safety system trains is actuated automatically in response to independent signals.

**WWER-1000/V-302 Major Safety Systems and Their Functions**

System	Intended Use under Normal Operation	Intended Use in Emergencies
<b>Protective Safety Systems</b>		
High-pressure Injection System	Standby mode	Supply boric acid solution to the primary circuit at a flow rate of 6 m <sup>3</sup> /h, concentration of 40 g/dm <sup>3</sup> , at pressure (160-0 kg/cm <sup>2</sup> ) ranging 16.0-0 MPa Supply boric acid solution to the primary circuit with a flow rate of not less than 130 m <sup>3</sup> /h and initial concentration of 40 g/dm <sup>3</sup> , within the range of primary pressures (110-15 kgf/cm <sup>2</sup> ) of 11-1.5 MPa from a 150 m <sup>3</sup> tank, and then, after its emptying, supply boric solution with a concentration of 16 g/dm <sup>3</sup> from tanks V=750 m <sup>3</sup> Maintain temperature of the supplied water not lower than 20°C in all modes In emergencies, supply boric solution from the high-pressure injection pump, at the latest in 35-40 sec once the primary pressure has reached 90 kgf/cm <sup>2</sup>
Low-pressure Injection System	Residual heat removal from the reactor core and primary circuit cooldown.	Emergency core cooling and residual heat removal from the reactor core in accidents requiring the reactor transfer into "cold shutdown" mode. Initiated when ECCS protection setpoints are generated or under loss of power to auxiliary buses on the automatic load sequencer signal.
Passive ECCS (ECCS hydroaccumulators)	Under normal operation, the system is in standby; the system is periodically tested in compliance with the operating procedures	The systems ensures injection of boric acid of 16g/dm <sup>3</sup> concentration and 55-60°C temperature into reactor under primary pressure below 5.9 MPa (60 kgf/cm <sup>2</sup> ) thus ensuring core cooling and filling under loss of coolant accidents.
Primary Overpressure Protection System	Standby mode: pilot-operated and main valves are closed and are ready to actuate from all control lines in case of need	To prevent RCS damage in pressure increase beyond the safe operation limits. To decrease primary pressure to ensure the feed and bleed procedure by means of the high-pressure injection system.

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System	Intended Use under Normal Operation	Intended Use in Emergencies
Secondary Overpressure Protection System: SG Safety Valves	Standby mode	Secondary system overpressure protection in operational occurrences and emergencies.
Steam Dump Valve to Atmosphere	Steam Dump Valve to Atmosphere RA13-43S02 (BRU-A) is designed for removal of residual heat from the reactor; for cooldown of the primary circuit in the steam mode; for control of SG pressure in case of BRU-K failure or prohibition for opening	Protection of the secondary circuit against overpressure in anticipated operational occurrences and emergencies
Emergency Gas Evacuation System	During unit startup the system ensures gas evacuation when the primary circuit is filling. When reactor is at nominal power, the system is not used and is maintained available. During unit shutdown, the system ensures that the primary equipment metal is cooled evenly (reactor head, SG headers, pressurizer upper head)	The system ensures complete filling of the core with coolant for reliable heat removal due to gas bubbles in the upper dead-end sections of the primary piping that release from the coolant when its temperature decreases to saturation margin in case of emergency pressure decrease above the core. Primary pressure decrease to restore conditions permitting actuation of high-pressure injection system pumps.
Control & Protection System (CPS) (mechanical side)	Compensate for the reactivity changes and reactor power control	Reactor emergency power decrease or reactor scram by the reactor control and protection system (CPS)
Main Steam Isolation Valve System	In standby.	The system is designed to disconnect the emergency steam generator with broken steam line in non-isolated section (before MSIV) from steam lines of serviceable steam generators to limit deep cooldown of the primary circuit due to additional steam makeup: measure to prevent development of a reactivity accident
Emergency Feedwater System	In standby.	Emergency feedwater system (EFS) actuated in case of failure of the normal SG feedwater system. The EFS is actuated automatically for residual heat removal at the first stage of emergency cooldown.
<b>Control Safety Systems</b>		
Control and Protection System (CPS)	Reactor control during startup, operation at nominal power and in shutdown states	Reactor control during accidents. Emergency reactor unloading or reactor scram by the emergency protection system.
Control Safety System (CSS)	In standby until actuation signal is received, generates data on process parameters to be used by personnel and actuates alarms when major failures occur.	Actuation of safety systems, their control and monitoring during performance of safety functions.

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<b>Localization safety system</b>		
Containment System	Under normal operation, the containment system performs functions of a normal operation system, which include provision of a closed space to confine radioactive fission products released during NPP operation under various design modes.	Limitation of radioactive releases into the atmosphere in emergencies and design-basis accidents including maximum design-basis accident
Containment spray system	Standby mode	Accident confinement by condensation of steam generated as a result of primary coolant leak into the containment. Pressure decrease in the containment. Immobilization of iodine in the steam and air of the containment. Makeup and cooling of spent nuclear fuel in the spent fuel pool.
Containment Hydrogen Measurement System, passive catalytic hydrogen recombiners	In standby.	The hydrogen concentration reduction in the containment air to the non-explosive value in design-basis accidents.
<b>Supporting Safety Systems</b>		
Emergency Power Supply System	In standby.	Independent emergency supply to safety systems. In emergency modes requiring the operation of safety systems, the diesel generators are actuated automatically and sequentially loaded.
Essential Power Supply System	Power supply from the working transformer bus through two bus switches on 6 kV bus and 6/0.4 transformers of the second reliability group. Power supply to equipment fed from the 0.4 kV essential power supply bus, first reliability group. Power supply to the uninterruptible power supply unit (UPS) of the direct-current board and charging of the battery.	Power supply to all safety system equipment.
Essential Service Water System	Heat removal to the ultimate heat sink from normal operation system equipment (spent fuel pool, intermediate circuit, ventilation systems ...), as well as from safety system mechanisms.	Heat removal from the reactor core through the ECCS heat exchangers, cooling of safety systems' mechanisms.

### Unit 3

The configuration of the main safety systems and normal operation systems important to safety of South Ukraine unit 3 is similar to that of Zaporizhzhya NPP units 1-6. The systems that perform safety functions for WWER-1000/320 are described in Section 1.1.2 Z.

However, the design of unit 3 additionally provides for a common-unit reliable power supply system including two trains with independent diesel generators and batteries. The power of each diesel generator (5600 MW) was selected to ensure its startup in case of blackout of both power units (in case one of the diesel generators fails).

As additional capability that can be used in case of emergencies, additional makeup of emergency feedwater tanks was introduced at South Ukraine units 1, 2 and 3 from a fire water collector through the feedwater piping to maintain SG emergency make-up in case of transfer of design-basis accidents into beyond-design basis scenarios leading to full loss of

water in emergency feedwater tanks and condensate storage tanks, prior to operation of low-pressure injection system (LPIS).

The following power supply options are also ensured at South Ukraine NPP units 1, 2 and 3 in case of blackout:

- power supply from the available units of the NPP site;
- power supply to 6 kV standby lines from diesel generators of any safety system of any power unit on the site or from the common-unit emergency diesel generators (for unit 3);
- power supply from the Kremenchuk Hydroelectric Plant and Tashlyk Hydroelectric Pumped Storage Power Plant.

### 1.2 SU Significant Difference between Units

South Ukraine NPP units 1 and 2 are of identical design and have similar reactor facilities, layout and arrangement and have no significant differences that may affect assessment of their vulnerability to extreme external natural events, performance of safety functions or severe accident management.

There are some differences between the units in terms of safety improvement measures: some measures implemented at one unit are either in progress or planned for future at the other unit.

Taking into account objectives of stress tests, a list of differences between all South Ukraine NPP units was made. These differences were taken into account during stress tests at the South Ukraine NPP.

In addition to the differences between the designs of V-302/338 and V-320 containments, the absence of the main gate valve in the V-320 reactor and the presence of the common-unit emergency diesel generators (unit 3), the following differences were taken into account during stress tests of the South Ukraine NPP:

Difference	Power Unit 1	Power Unit 2	Power Unit 3
Type of PRZ PORV	Sempell, VS99/66 type	"Bopp&Reuther", SiH 3111/3115 type	Sempell, VS99/66 type
Type of SG PORV	"Bopp&Reuther", SiZ 2507 type	Chekhov Power Engineering Machine-Building Plant (Russia), 696-250/300-0-01 type	"Bopp&Reuther", SiZ 2507 type

### 1.3 SU Use of PSA as Part of Safety Assessment

Information on PSAs for South Ukraine NPP units 1-3 is provided in Section 1 "General Data on the Sites and Nuclear Power Plants of Ukraine".

## 2. EARTHQUAKES

### 2.1 Design Basis

#### 2.1.1 Earthquake against Which the Plants Are Designed

The siting and design of all operating nuclear power plants in Ukraine were performed in the mid-1970s to meet the regulatory requirements in force at that time. Later, taking into account national and international experience and IAEA recommendations, the regulatory framework was improved and a number of new regulations were implemented. In particular, two levels of earthquakes were introduced (design earthquake and maximum calculated earthquake) and also seismic categorization of structures, systems and components (SSCs), specific requirements for SSCs (depending on their seismic categories), requirements for seismic investigations of sites etc.

According to the design documentation, the seismicity of all operating Ukrainian NPP sites was specified as follows:

Design Earthquake<sup>1</sup> (DE) – intensity of 5 according to the MSK-64 seismic scale with probability of being exceeded of  $1 \times 10^{-2}$  and peak ground acceleration (PGA) of 0.025 g;

Maximum Calculated Earthquake<sup>2</sup> (MCE) – intensity of 6 according to MSK-64 with probability of being exceeded of  $1 \times 10^{-4}$  and PGA=0.05 g.

In seismic assessment of SSCs at the design stage, the calculated accelerograms and response spectra were used. Depending on site-specific soil conditions, the accelerograms were revised in terms of site peak ground acceleration (PGA).

Seismic analysis of SSCs was carried in accordance with their categorization. SSCs of seismic category I are designed to withstand MCE and must perform their functions to ensure plant safety during and after MCE. SSCs assigned to seismic category II are designed to withstand DE impact and must be operable after DE. SSC seismic categorization is presented in the corresponding sections of the safety analysis reports.

According to regulations, civil NPP structures and buildings are classified into three categories (I, II, III) depending on their responsibility for nuclear and radiation safety. Category I<sup>3</sup> structures are required to withstand a series of extreme hazards such as snow, wind, temperature, tornado, maximum design-basis accident and MCE.

In assessment of SSCs, the simultaneous seismic load of three components was considered (two horizontal components and a vertical one), the vertical component was taken with a factor of 0.5 relative to a horizontal one.

In seismic assessment of equipment and piping, a combination of normal operation and operating design earthquake (normal operation + DE) and a combination of normal operation and safe shutdown earthquake (normal operation + MCE) were considered, and a joint impact of MCE and the most unfavorable accident involving piping rupture was taken into account.

Seismic loads on equipment and its components were calculated for MCE using accelerograms determined in calculations for equipment mounting elevations. Seismic loads were calculated using two methods: spectral and dynamic. Floor response spectra, in particular for the unified design, were calculated for a 1% damping. The considered equipment was represented as a three-dimensional stick system with discrete mass distribution. The rigidity of the stick system simulates stiffness properties of equipment and its supports.

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<sup>1</sup> MCE corresponds to Seismic Level 2 (SL-2) in IAEA practice and Safe Shutdown Earthquake (SSE) in USA practice

<sup>2</sup> DE corresponds to Seismic Level 1 (SL-1) in IAEA practice and Operating Basis Earthquake (OBE) in USA practice

<sup>3</sup> Category I includes buildings and structures whose destruction or damage can affect safety-related operation systems and thus cause radioactive releases in amounts that may lead to exposure of personnel and the public to radiation exceeding the established doses for the maximum design-basis accident or cause failure of safety systems that maintain the reactor core subcriticality, ensure reactor heat removal and confine radioactive materials

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In accordance with IAEA recommendations and state-of-the art international practice, to specify the seismic hazards of Ukrainian NPPs sites, additional instrumental seismic investigations were conducted:

- 1999-2001 - Khmelnytsky NPP and Rivne NPP sites;
- 2009-2010 - South Ukraine NPP site;
- starting from 2011 – seismic investigations at the Zaporizhzhya NPP are ongoing.

The additional investigations showed that PGA and calculated accelerograms established in the design were specified. The summary of the investigations is presented below.

Seismicity of the sites of operating power units

Power unit	Type of reactor	Design seismicity of the site				Seismicity based on additional investigations			
		MCE		DE		MCE		DE	
		Intensity	PGA	Intensity	PGA	Intensity	PGA	Intensity	PGA
ZNPP 1-6	V-320	6	0,05	5	0,025	-	-	-	-
KhNPP 1-2	V-320	6	0,05÷0,06	5	0,5*SSE	6	≈0,06	5	0,5 * SSE
RNPP 1-2	V-213								
RNPP 3-4	V-320								
SUNPP 1-2	V-302/ 338								
SUNPP-3	V-320								
							≈0,093/ 0,12 <sup>4</sup>		

Seismic PSAs for operating Ukrainian power units are currently developed under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”.

## 2.1.2 Provisions to Protect the Plants against the Design Basis Earthquake

### 2.1.2.1 Identification of Structures, Systems and Components

In accordance with Ukrainian regulations, the seismic resistance of SSCs is assessed considering their seismic categorization.

The list of systems performing safety functions is defined in the design documents and safety analysis reports. These systems include:

- reactor scram system;
- high-pressure injection system;
- low-pressure injection system;
- ECCS hydroaccumulators;
- primary makeup and letdown system;
- primary overpressure protection system;
- primary steam-gas mixture removal system;
- SG emergency feedwater system;
- auxiliary feedwater system;
- secondary overpressure protection system;
- containment spray system;
- isolation valve system;
- essential service water supply system;
- auxiliary power supply system;
- common-unit reliable power supply system;
- air-conditioning and ventilation systems.

<sup>4</sup> The additional seismic investigations demonstrated that PGA for SUNPP site is 0.093g, but SNRIU Board made a decision on 8 September 2011 to establish an engineering margin of 30% of PGA=0.093g for seismic assessments of SSCs, i.e. PGA= 0.12g is accepted.



Seismic resistance of buildings and structures which contain equipment and piping of the above-mentioned systems has been analyzed as well.

The list of the mentioned buildings and structures subjected to analysis within the framework of stress tests includes:

**For WWER-1000/V-320:**

- Main building. Reactor building including the auxiliary building;
- Main building. Turbine hall;
- Building of emergency diesel generators;
- Building of common-unit emergency diesel generators;
- Auxiliary building ventilation stack (considered in terms of consequences of its damage and drop on the neighboring building, blocking of access roads etc.);
- Auxiliary building. Fresh fuel storage;
- Boron solution storage building;
- Spray ponds (essential service water system).

**For WWER-1000/V-302, 338r:**

- Main building. Reactor building;
- Main building. Turbine hall;
- Boron solution storage building;
- Building of emergency diesel generators;
- Special building. Fresh fuel storage;
- Special building ventilation stack;
- Mechanical-draft tower (essential service water system).

**For WWER-440/V-213:**

- Main building. Reactor building;
- Main building. Turbine hall;
- Main building. Fresh fuel storage;
- Special building. Ventilation stack;
- Building of emergency diesel generators;
- Cooling towers (essential service water system).

**2.1.2.2 Evaluation of Robustness of Structures, Systems and Components to Assess Potential Margins**

To assess the impact of earthquakes on the main safety functions, seismic loads with peak ground accelerations of 0.05g, 0.1g, 0.15 g, and 0.2 g were analyzed in the framework of stress tests.

To assess the threshold effects of earthquakes on safety-related equipment, the seismic qualification results for Ukrainian NPP equipment were used. The seismic qualification was intended to confirm seismic resistance of equipment and its ability to perform the main safety functions.

The stress test results for the Ukrainian NPPs confirmed that the safety-related equipment is capable of performing its main safety functions under design-basis seismic impacts.

The piping required for emergency shutdown, heat removal and confining of radioactive materials within the specified boundaries for all types of reactors retains its safety functions during design-basis earthquake.

The structures of category I (containment, reactor building, building of emergency diesel generators, essential service water spray ponds) and category 2 (including turbine hall) are resistant to the design-basis seismic impacts.

Safety margins with respect to seismic impacts determined during the stress tests are provided in Section 2.2.1 of this Report.

### **2.1.2.3 Main operating contingencies in case of damage that could be caused by an earthquake**

In the framework of the stress tests, potential secondary effects of earthquakes were analyzed. The consequences of earthquakes that are common for all NPP sites (fire, explosion, loss of off-site power supply) and specific for each site were identified.

Secondary earthquake effects that may be observed at each site are as follows:

- fires and/or explosions at fire hazardous and highly explosive facilities located at the NPP site (considered in Section 4 of this Report);
- loss of off-site power supply (considered in Section 5.1 of this Report);
- destructions at the NPP site and infrastructure around the plant (it should be noted that inspections of each NPP have demonstrated that there is no potential for significant damage that would block access to the site and buildings of categories I and II under external hazards);
- damage of hydraulic structures at or near the NPP site;
- damage of tanks containing various chemicals.

The consequences of secondary effects may be site-specific. In particular, specific consequences of secondary effects for the Zaporizhzhya NPP site may be as follows:

- loss of the ultimate heat sink due to damage of the dam of the Kakhovka Hydroelectric Plant and, as a consequence, loss of Zaporizhzhya NPP cooling pond (discussed in detail in Section 5.2 of this Report);
- potential flooding of the Zaporizhzhya NPP site due to accidents at Dnieper dams (considered in Section 3 Z of this Report);
- destruction of the ammonia tank in the joint auxiliary building, it is not dangerous since the isolation area exceeds the distance to the ventilation systems of the reactor building.

For the Rivne NPP and Khmelnytsky NPP sites, specific consequences of secondary effects may be as follows:

- leak of chlorine gas resulting from damage of chlorine piping inside the warehouse and chlorinators during an earthquake with an intensity higher than 6. Spread of chlorine is dangerous for personnel of the main control room and emergency control room and thus may affect accident management. Measures are envisaged to ensure survivability of the main control room and emergency control room (see Section 6.1.3 of this Report).
- flooding of the site resulting from the break of water storage dams is not dangerous. A considerable distance from the water reservoirs to the Rivne NPP water intake, small amount of water in them, and sufficiently wide floodplain of the Styr river were considered for the Rivne NPP. For the Khmelnytsky NPP, on-site structures are located substantially above the maximum flooding wave horizon (considered in detail in Sections 3 R and 3 Kh of this Report).

For the South Ukraine NPP, specific consequences of secondary effects may be leaks and destruction of tanks with chemicals (nitric and sulfuric acid, etc.). These events are not dangerous since the isolation area exceeds the distance to the air intake of ventilation systems of the reactor building.

External floods resulting from accidents at hydraulic structures caused by an earthquake are not hazardous for buildings and structures (considered in detail in Section 3 SU of this Report).

### **2.1.3 Compliance of the Plant with Its Current Licensing Basis**

The NPP compliance with licensing requirements means mandatory compliance with standards, rules and regulations on nuclear energy and radiation safety, as well as operating regulations and procedures.

To confirm that safety-related SSCs meet the requirements for resistance to external hazards, maintenance and inspections are conducted on a regular basis in accordance with operating procedures. Inspections are conducted on a regular basis both by proper supervisory plant services and by SNRIU inspectors.

NPP maintenance and repair are conducted in accordance with regulatory requirements and operating procedures to restore operability and keep equipment and systems operable within the lifetime established by the design. For this purpose, regular maintenance and repair of systems and equipment are conducted after NPP commissioning to prevent decrease in their reliability resulting from aging, wear and other operational factors.

For repairs, each NPP provides for relevant divisions (departments, services, laboratories, workshops and sections, including specialized ones), which constitute the NPP maintenance and repair service.

The maintenance and repair operations include:

- well-arranged and documented maintenance of equipment;
- lifetime analysis of components and equipment to determine technically and economically sound standards;
- analysis of operating experience and determination of the optimal periodicity for major, intermediate and routine repairs;
- implementation of advanced methods for arrangement and management of repairs;
- quality assurance of repairs using appropriate instrumentation;
- logistics planning for timely provision of materials, spare parts and componentry including their quality control.

The requirements for arrangement and documentation of maintenance and repair of NPP equipment are established in appropriate industrial standards. The procedure for equipment outage, repair and acceptance and quality inspection after repair is determined in appropriate procedures.

In addition to maintenance and repair of equipment and piping, regular monitoring of civil structures is carried out by each NPP. The monitoring of civil structures covers routine observations, management and control, technical assessments, analysis of processes, timely detection of changes in load-bearing capability and serviceability, as well as lifetime reassignment. NPP civil structures and buildings forming a comprehensive production system are subjected to monitoring.

The monitoring is arranged as follows:

- monitoring is carried out permanently in accordance with schedules for technical, visual and instrumental inspections;
- databases (electronic and paper) and technical certificates of buildings and structures are kept.

Monitoring is based on the technical supervision system including different types of inspections (with various periodicities) depending on the necessity. To arrange and conduct monitoring, operating services for buildings and structures are established at each NPP.

Technical inspections conducted to check the state and repair of NPP industrial buildings and structures are scheduled on a planned basis (according to annual plans for technical inspections) and include:

- visual technical examinations;
- instrumental inspections and examinations.

Final and intermediate reports are developed to record monitoring results. These reports provide the basis for making conclusions on the technical state and lifetime of buildings and structures.

The results of inspections and examinations confirmed that no buildings or structures were found in emergency state at all NPPs. Maintenance and repairs, testing and inspections

conducted on a regular basis in accordance with regulatory requirements and operational documents allow timely detection and eliminations of drawbacks. This, eventually, confirms the capability of safety-related SSCs to fulfill their functions.

## **2.2 Evaluation of Safety Margins**

### **2.2.1 Range of Earthquake Leading to Severe Nuclear Fuel Damage**

Ukrainian NPPs operate nuclear fuel manufactured by TVEL Ltd (Russian Federation) and by Westinghouse (USA).

According to the design data, TVEL fuel retains its functions under the following seismic impacts: 1.6 g in horizontal direction; 2.6 g in vertical upward direction and 1.13 g in vertical downward direction. According to the design data, the Westinghouse fuel retains its functions under an axial impact up to 4g and a transverse impact up to 6g. Therefore, the nuclear fuel is designed to withstand loads greatly exceeding the seismic loads for fuel assemblies in the core.

Nuclear fuel damage may be caused by loss of coolant and failure of heat removal resulting from damage of primary equipment and piping. In particular, according to the design data, the WWER-1000/320 was designed to withstand seismic loads up to MCE of 8 intensity. A combination of the maximum design-basis accident and MCE is selected as the boundary case in the design, and the capability of the reactor pressure vessel to retain its function in these conditions has been verified.

Additionally, it should be noted that analysis of design documentation has demonstrated safety margins for a great number of pipes and steam lines under impacts of at least 0.1g. For instance, the main coolant piping and piping of the passive part of ECCS of V-320 are capable of resisting earthquakes with 0.2g ground acceleration, which substantially exceeds the design values. Performance of the safety functions under impacts with 0.1g ground acceleration is ensured by the pressure control system piping, SG main steam system piping and SG feedwater piping.

According to the design documentation for V-302/338 and V-213 reactors, their process equipment and piping (reactor, steam generators, main coolant pump scroll, pressurizer, relief tank, main coolant piping, ECCS piping, etc.), as well as equipment and piping of the reactor protection and control system, ECCS, primary pressure control system, SG emergency feedwater system, and secondary overpressure protection system, are capable of performing their safety functions under seismic impacts up to 0.1g; i.e. there is a margin relative to the design-basis impact (0.05g).

In the framework of technical assessment and long-term operation measures for South Ukraine NPP unit 1 (V-302), seismic resistance of the primary coolant piping and pressurizer surge line was calculated. The results of calculations confirmed the seismic resistance of these components for earthquakes with 0.1g ground acceleration and, correspondingly, their capability to perform basic safety functions (calculation results for South Ukraine NPP unit 1 can be applied to South Ukraine NPP unit 2 as well in the frames of stress tests). The resistance to seismic impacts of 0.15g has been confirmed with appropriate strength calculations for the steam and feedwater lines of South Ukraine NPP unit 1 (V-302), provided that compensatory measures are taken.

Based on assumptions made in the stress tests, quite a large amount of WWER-1000 safety-related equipment is resistant to seismic impacts of 0.1-0.2g. It should be noted that the operating organization (NNEGC Energoatom) currently continues seismic qualification (0.1g PGA) of safety-related equipment at the Zaporizhzhya, Khmelnytsky, Rivne and South Ukraine NPPs (these efforts include identification of the current state of equipment qualification, replacement of non-qualified equipment by seismically-resistant one or implementation of compensatory measures to increase seismic resistance).

Buildings and structures of all NPP units of category I (containment, reactor building, building of emergency diesel generators, and spray ponds of essential service water system) are resistant to seismic impacts of 0.1g.

To assess the threshold values of seismic impacts on the spent fuel pools and dry storage facility (Zaporizhzhya NPP), 0.1g ground acceleration was accepted during the stress tests. The results of analysis demonstrate the following:

- structures and components of the fuel storage pools of Ukrainian NPPs retain their functions under seismic impacts at the level of 0.1g ground acceleration;
- structures and components of the dry spent fuel storage facility are designed to withstand 0.2g ground acceleration and perform their functions under this seismic impact.

### 2.2.2 Range of Earthquake Leading to Loss of Containment Integrity

In the frameworks of stress tests, V-320, 302 and 338 reactor containments as well as V-213 reactor containment were analyzed to assess the seismic impact that could lead to loss of their integrity.

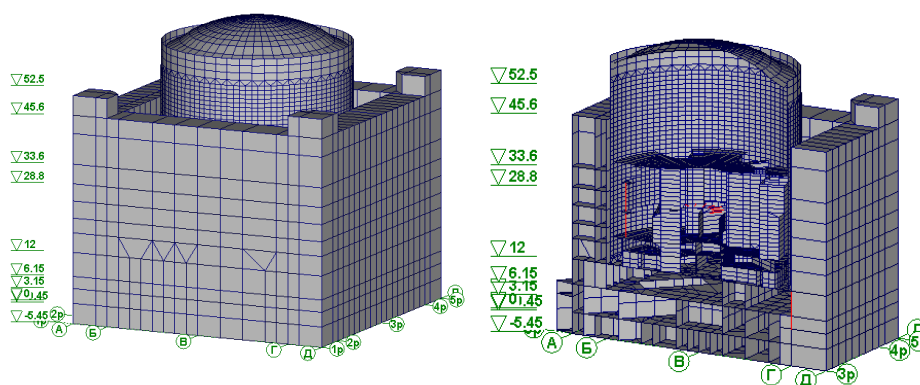
According to design documentation, the seismic resistance of WWER-1000/V-320 containment is maintained at least for 0.1g PGA.

Under stress tests, additional calculations were performed to evaluate seismic resistance of the V-320 and V-302 containments. The seismic resistance of the V-213 containment was assessed in 2010 in the framework of long-term operation measures at Rivne NPP units 1 and 2.

The results of preliminary calculations performed under stress tests demonstrate the following.

#### WWER-1000/V-320

To determine the threshold value at which the V-320 containment remains integral, preliminary containment calculations were performed.



Calculation Model for the V-320 Reactor Containment

The following loads were addressed:

- deadweight of structures;
- weight of equipment;
- polar crane;
- preliminary containment compression with tendons;
- 7, 8, 9-point seismic intensities;
- combination of loads.

The containment foundation is considered to be elastic to characterize its performance under seismic impacts.

The following results have been obtained in calculations:

- mode and frequency of natural oscillations (50 modes considered);
- stresses in containment components using a linear-spectral theory of seismic resistance.

Containment robustness is assessed for the averaged cross-section without regard to strengthening at penetrations, etc.

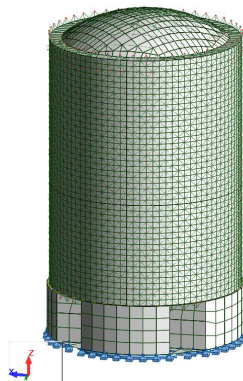
The calculations showed that the threshold value of seismic impact at which the V-320 containment remains integral is at least 0.17g by conservative assessment.

#### **WWER-1000/V-302 (V-338)**

Strength calculation of the V-302 reactor containment was performed as part of the seismic assessment. This effort included:

- development of a containment finite-element model;
- identification of all loads and their combinations in the calculation model, including seismic loads with intensity of 6 (PGA=0.05g) and PGA of 0.15g (for reinforced-concrete structures, containment pre-stressing system, auxiliary reactor building, foundation base);
- linear calculation for static impacts;
- calculation of mode and natural frequency;
- calculation of dynamic impacts;
- analysis of stresses in the most loaded structural components to assess compliance of the main supporting structures with operability criteria;
- assessment of the containment integrity.

The containment calculation model demonstrates the actual operational conditions and corresponds to the calculation cases considered. The calculation took into account interaction between structural components, spatial performance of structures, mechanical properties of materials and grounds and actual tension of containment tendons.

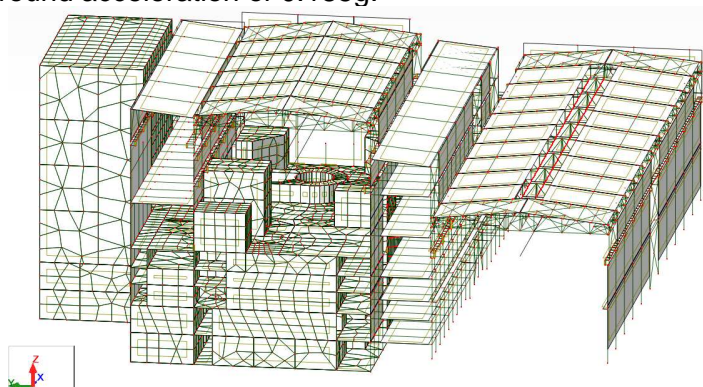


Finite-Element Model for the Containment of South Ukraine NPP Unit 1

Therefore, the V-302 containment is seismically resistant to an earthquake of PGA=0.15g.

#### **WWER-1000/V-213**

According to calculations performed in the framework of long-term operation measures, the threshold value of seismic impact for civil structures of units 1 and 2 is not lower than the impact with peak ground acceleration of 0.185g.



Finite-Element Model for the Reactor Building of Rivne NP Unit 1

### **2.2.3 Earthquake Exceeding the Design Basis Earthquake for the Plants and Consequent Flooding Exceeding Design Basis Flood**

Zaporizhzhya NPP. An event associated with the breach/destruction of Dnieper dams located upstream the Zaporizhzhya NPP may represent a potential hazard for this plant. An assessment has been carried out for the most conservative event, i.e. the avalanche-like break of all the said dams, with the Kakhovka Hydroelectric Plant remaining operable (located downstream the Zaporizhzhya NPP). The maximum possible flooding level for such a scenario will be 19.36 m, which is lower than the level of Zaporizhzhya NPP site (22.0 m). Hence, there is no direct risk of Zaporizhzhya NPP site flooding in the event of dam breach/destruction resulting from an earthquake. However, there is a potential for the failure of equipment in the buildings of unit pump stations and spray ponds, which, nonetheless, will not lead to loss of safety functions.

Khmelnitsky NPP. There are small water reservoirs on the Goryn river at a substantial distance from the Khmelnitsky NPP site (more than 50 km). Their possible breach/destruction under seismic impacts cannot pose a hazard to the Khmelnitsky NPP buildings and structures. The breach of the Khmelnitsky NPP cooling pond does not pose a hazard either, since a breaking wave caused by a seismic impact beyond the design basis will spread along the Goryn river floodplain in the western direction, considering lowering ground surface levels. The leveling elevations of the Khmelnitsky NPP site and top of the dam are the same and constitute 206.0 m, while the maximum initial level of the flooding wave will be 203.0 m. Thus, Khmelnitsky NPP structures are located higher than the maximum level of the flooding wave.

Rivne NPP. The breach of the water reservoir dam located upstream the river Styr may be a potential cause of an earthquake-driven flooding of the Rivne NPP. Rivne NPP water intake structures are the only large hydraulic constructions on the Styr river within the 30 km area. There are three Styr basin water reservoirs upstream the river at a great distance from the Rivne NPP. Considering the substantial distance of these water reservoirs from the Rivne NPP water intake structures, small amount of water in them, and sufficient width of the Styr riverbed in the area of Rivne NPP water intake (2-3 km), the water level near the Rivne NPP water intake area will hardly increase (by no more than 0.1 m) in the event of breach of the water reservoir dams, considering water flow spreading along the riverbed. The analyses and assessments demonstrate that the Rivne NPP site is resistant to external flooding.

South Ukraine NPP. Rise of the water level in the Yuzhny Bug river does not pose a hazard for buildings and structures located on the South Ukraine NPP site since its level (104 m) is more than 70 m higher than the water level in the river. Moreover, the South Ukraine NPP site is 3 km away from the river shoreline. To ensure water cycle, a flood spillway was arranged in the Tashlyk water reservoir, whereby the water level may reach up to 101.5 m in the reservoir, which is 2.5 m lower than the leveling elevation of the South Ukraine NPP. This means that there is no risk of flooding for the site.

However, in the framework of stress tests, resistance of the South Ukraine NPP against external flooding was addressed also in terms of potential flooding of the Tashlyk water reservoir makeup pump station, which supplies service water from the Yuzhny Bug to the South Ukraine NPP. The maximum water levels of the Yuzhny Bug river in the area of the cooling pond makeup pump station constitute 33.0 m. The makeup pump station's entry level is 32.407. The Alexandrovka Hydroelectric Plant located within the NPP area allows control of the river water level and flow, thus excluding the potential for flooding of the makeup pump station. Potential failure of the South Ukraine NPP cooling pond makeup pump station will not result in loss of safety functions.

#### **2.2.4 Measures Which Can Be Envisaged to Increase Robustness of the Plants against Earthquakes**

The following measures are considered to increase seismic resistance of all operating NPPs:

- complete equipment seismic qualification for 0.1g PGA for the Rivne, Khmelnytsky, and Zaporizhzhya NPPs and for 0.12g for the South Ukraine NPP, including replacement of non-qualified equipment by seismic-resistant one or implementation of compensatory measures to increase the seismic resistance of equipment;
- complete activities to confirm resistance of buildings, structures and piping against seismic impacts of 0.1g PGA (0.12g for the South Ukraine NPP);
- complete additional instrumental seismic investigations at the Zaporizhzhya NPP site;
- implement permanent seismic monitoring systems at all Ukrainian NPPs;
- develop a seismic PSA for all Ukrainian NPPs.

The above-mentioned measures are envisaged by the current “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”.



### 3. FLOODING

#### 3 Z FLOODING (ZAPORIZHZHYA NPP SITE)

##### 3.1 Z Design Basis

The Zaporizhzhya NPP water supply system includes the following main hydraulic structures: a cooling pond, spray ponds No. 1 and 2, discharge channel of the cooling pond makeup, blowdown channel and unit pump stations. The Kakhovka Water Reservoir on the Dnipro River is a source of service water supply for the Zaporizhzhya NPP. In the area of Zaporizhzhya NPP, the reservoir is 8-13 km wide, has an average depth of 8 km and the maximum depth of 14-16 km (in the bed of the Konka river and Dnipro river).

The cooling pond was made by separating the riverside part of the Kakhovka Reservoir by means of a hydraulic-fill sand dam. Non-recoverable losses in the cooling system during operation are compensated by makeup from the discharge channel of the Zaporizhzhya Thermal Power Plant (ZTPP). One of the main tasks of the service water pump station is to make up water losses of the essential service water in case of ZTPP shutdown.

##### 3.1.1 Z Flooding against which the Units Are Designed

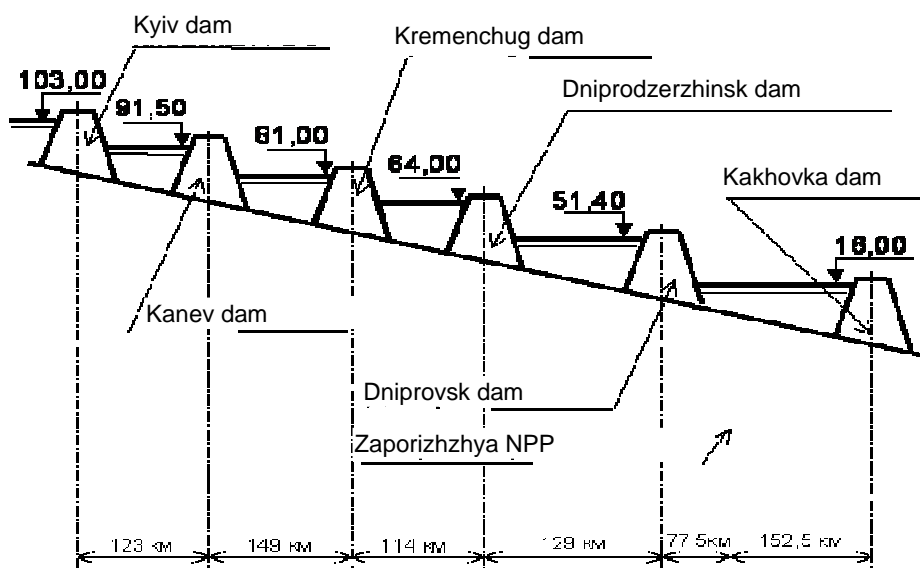
The Zaporizhzhya NPP site layout elevation is 22 m. The normal headwater level (NHL) of the Kakhovka Reservoir is 16.0 m.

From 1958 to 1998, specific watermarks of the reservoir were as follows: an average annual water mark was 15.6 m, the highest water mark was 16.47 m (in 1969), the lowest water mark was 12.71 m (in 1960).

#### **Assessment of Zaporizhzhya NPP vulnerability to flooding caused by break of the Dnipro Cascade dams**

There are large reservoirs of the Kyiv, Kanev, Kremenchug, Dniprodzerzhinsk, Dniprovsk and Kakhovka Hydroelectric Plants in the Dnipro River basin.

Zaporizhzhya NPP vulnerability to flooding induced by a possible break of the Dnipro cascade dams was assessed in the framework of the safety analysis reports. The events related to the break of the dams were assessed for the most conservative assumption providing for failure of all Dnipro cascade dams upstream the Zaporizhzhya NPP with the Kakhovka Hydroelectric Plant dam (located downstream of Zaporizhzhya NPP) remaining operable. Using a detailed analysis, the maximum possible level in the Kakhovka Reservoir is determined. It equals 19.36 m, that is below the Zaporizhzhya NPP site elevation of 22.0m.



**Longitudinal Section of the Dnipro Cascade Reservoirs**

To analyze operation of Zaporizhzhya NPP when a breaking wave passes through the Kakhovka Reservoir, the wave impact can be conditionally divided into several stages specified by different conditions of the cooling pond filling and emptying and by respective water level impact on Zaporizhzhya NPP operation.

At the initial stage when NHL in the cooling pond is stable, the NPP structures are in normal operating mode. Afterwards, due to overtopping of both the damp and blowdown structure, cooling pond water level will rise and exceed the design NHL by 0.20 m. Later, the water level in the pond will be lowering through the blowdown structure to the design NHL. While the cooling pond level rises or drops, all NPP structures including the cooling system for turbine equipment condensers will operate normally. In case of short-term flooding of the discharge channel of cooling towers (operating in summer) located on the dam, all coolers of the auxiliary turbine equipment located on the dam must be de-energized.

When the Kakhovka Reservoir water level rises, the ground water level will also increase within its adjacent areas, specifically, on the dam of the cooling pond up to values approximating the water level in the reservoir. Therefore, rising ground water level will not be so significant for the structures located on the dam. There will be no additional deformations of the structure foundations because of a minor change in the ground water level and spray ponds and unit pump station are protected by drains; moreover, the cooling towers were built on the pile foundation. At the subsequent stages, the ground water level will decrease to normal values.

When the cooling pond level rises, the ground water level will also rise in the adjacent areas including the site along the service water supply channel. The ground water level rising by 0.10...0.15 m will cover the 50-80 m area extending from the unit pump stations to the main buildings of all power units. There will be no additional deformations of the structure foundations and changes in soil filtration parameters of foundation and earthwork structures because the ground water level changes within the normal deviation range and during a short period of time.

When the breaking wave passes through the Kakhovka Reservoir, water flow velocities reach their maximum, the outer side slope of the cooling pond levee is undercut and its design profile is disrupted.

The analysis of calculations made for Zaporizhzhya NPP cooling pond filling and emptying when the breaking wave travels through the upstream Dnipro Cascade reservoirs shows the following:

- dams of all upstream reservoirs are referred to the first reliability class and are designed for snowmelt-induced flood of 0.01% occurrence;
- level in the cooling pond can rise up to 16.70 m and exceed NHL value. Such a high level can be reached within 6 days;
- additional water volume coming to the cooling pond is about 2.0 million m<sup>3</sup>;
- water level in the cooling pond comes down to the design-basis NHL value of 16.50 within 29 days (43 days after the break);
- filling and emptying of the cooling pond does not affect the power unit safety (site design elevation is 22.0 m, i.e. 2.64 m above the maximum level of the Kakhovka Reservoir);
- in case of overflow to the cooling pond, the turbine service water coolers may fail (cooling towers and spray ponds on the dam are inoperable);
- water level will be rising and lowering slowly (with maximum daily variations of less than 1.0 m/day), that will not affect the structural stability;
- wave height equal to  $h=0.90\text{m}$  in the Kakhovka reservoir was obtained as a result of calculations made for the north-west direction of wind with a speed of  $W=17\text{ m/sec}$  (of 50% occurrence) (for the period when the breaking wave passes through the Kakhovka Reservoir). Wind-induced surge height is expected to be about 0.02 m, with slope swash of 0.24 m. Wave impacts will cause negligible

change to the dam slope 5-10 m wide, that actually will not affect the design basis profile of the dam (crest width of 250-400 m);

- when the peak breaking wave passes, the water flow velocity in the Kakhovka Reservoir will reach 0.82 m/sec. This entails scouring of fine-grained sands of the unfortified outer slide slope of the cooling pond dam. Along the banks, when the water flow velocity exceeds 0.5 m/sec, scouring of the cooling pond dam by the water flow will occur, i.e. within 5 days. Washed-down earth will be deposited on the underwater part of the dam slope. Width of a faulted area of the dam will be less than 10 m;
- short-term wave impacts and increased flow velocity in the Kakhovka Reservoir during snowmelt flood will not result in significant negative influence on Zaporizhzhya NPP structures and will not affect safety of the Zaporizhzhya NPP.

### ***Possibility of ultimate heat sink loss as a result of hydraulic structure damage after an earthquake***

In the framework of stress tests, a possibility of water loss in the Zaporizhzhya NPP cooling pond induced by the Kakhovka Hydroelectric Plant dam failure after an earthquake was reviewed.

The lowest water level in the Kakhovka Reservoir on the Zaporizhzhya NPP site will be 10.0 m in case of the break of the Kakhovka Hydroelectric Plant dam. Owing to the significant width of the cooling pond levee, the dam will not be broken; and water losses because of filtering will remain actually unchanged in comparison with the design-basis conditions (NHL of the pond is 16.5 m, minimum volume level in the Kakhovka reservoir is 12.7 m).

Also, a possibility of water outflow from the cooling pond through the blowdown canal is reviewed. This canal is designed for water discharge from the cooling pond to the Kakhovka Reservoir to improve chemical composition of water in the circulation system. The dam structure includes a blowdown structure to provide the cooling pond blowdown. The structure consists of a blowdown canal and water discharge gate. Width of discharge openings is 3.0 m. Discharge channel bottom width is 8.0m. Elevation of the sluiceway structure bottom is 13.0 m. Maximum water flow rate passing through the blowdown structure according to the design is 40.0 m<sup>3</sup>/sec.

In place of blowdown structure, backfill of the dam is made using loose earth with layer-to-layer compaction. In case the blowdown structure's discharge openings are destroyed as a result of an earthquake, the water discharge gate may be covered with surrounding soils.

A failure of the Kakhovka Hydroelectric Plant dam after an earthquake may lead to loss of water in the Zaporizhzhya NPP cooling pond and, consequently, loss of make-up of the essential service water spray ponds.

### ***Assessment of impacts with regard to the Dry Spent Fuel Storage Facility (DSF)***

The system of residual heat removal from the spent fuel assemblies in the DSF is completely passive (natural convection of atmospheric air). During operation, the personnel cannot control the heat removal process. The only external factor that can influence the process of heat removal from the spent fuel assemblies is the change in atmospheric air temperature and humidity and deterioration of the air possibility through the ventilation ducts of the VSC-WWER containers (anticipated operational occurrences).

The most vulnerable point of the structure seems to be ventilation apertures of the VSC-WWER containers. Blockage of these apertures may occur during flooding since water flow may carry with it a sufficient amount of aquatic vegetation, silt and waste. Although it is not expected that DSF can be flooded due to hydrogeological conditions of the Zaporizhzhya NPP location, hypothetical full blockage of VSC-WWER containers apertures will not be followed by disastrous consequences and critical increase in fuel temperature.

### **3.1.2 Z Provisions to Protect the Units against the Design Basis Flood**

According to the design, the territory around the buildings is planned with slope from the buildings to drain snowmelt and rain waters to the storm-water inlets of storm sewers. The site territory around the main buildings of reactor compartment has the overall slope to the direction of the cooling water pond. Therefore, layers of the water which are formed on roads will flow down a side slope into the cooling water pond. The network operating staff controls network performance, as well as a storm sewer provides remedial maintenance and cleaning-up in case of clogging or degradation of the discharge capacity.

All the Zaporizhzhya NPP waterside pumps are equipped with drainage pumps to avoid flooding of the turbine hall. For pumping water from the drainage sumps, all Zaporizhzhya NPP emergency diesel generators (and common-unit emergency diesel generators) are provided with water drainage pump systems which operate in automatic mode. Process alarm is envisaged in the main control room in case of water level rising in drainage sumps of the emergency diesel generator building.

The turbine halls of the main buildings, as well as the reactor auxiliary building of “non-contaminated area” on below-zero elevations of staircases, are provided with drainage sumps of the industrial sewage system, with waste water pumped out by means of drainage pumps in the automatic mode. The main control room is equipped with a sump overflow process alarm. If water ingresses into the main building, it will flow down the staircase into the drainage sump and will be pumped out by drainage pumps.

The Zaporizhzhya hydraulic engineering service monitors the cooling pond water level on a regular basis. In case of emergency on water bodies, the plant management will be informed, therefore, adequate measures will be taken.

### **3.1.3 Z Compliance of the Units with Their Current Licensing Basis**

Compliance with unit license conditions is discussed in Section 2.1.3.

## **3.2 Z Evaluation of Safety Margins**

### **3.2.1 Z Estimation of Safety Margin against Flooding**

In case of a break of the Dnipro cascade dams, the highest possible level in the Kakhovka Water Reservoir is 19.36 m, that is below the elevation of Zaporizhzhya NPP site 22.0m. The rise of Kakhovka Water Reservoir level will not have a direct impact on Zaporizhzhya NPP units, but some faults in operation of equipment located in the unit pump station may occur, as well as failure in operation of the spray pond located in cooling pond channels.

### **3.2.2 Z Measures That Can Be Envisaged to Increase Robustness of the Units against Flooding**

To prevent impact of accident consequences on Zaporizhzhya NPP hydraulic engineering structures, the following actions are planned:

- carry out a detailed analysis of the potential water loss of Zaporizhzhya NPP cooling pond due to the break of the Kakhovka Hydroelectric Plant dam resulting from an earthquake;
- develop additional measures to provide for makeup of the spray pond of the essential service water system.

Despite the absence of direct influence on Zaporizhzhya NPP safety from flooding that may be caused by a possible break of Dnipro cascade dams, the following activities have been implemented at the Zaporizhzhya NPP:

- water pumping circuits in case of flooding of the reactor building or turbine were been developed;
- a purchase order for additional underwater pumps of high-level performance was issued;
- a list of buildings to be sealed to prevent flooding of the reactor building was developed;

- activities were completed to seal the rooms at below-zero elevation of the reactor compartment of units 1 and 2 and emergency diesel generator building;
- equipment was manufactured to seal the relief valves in rooms at below-zero elevations of the reactor compartment; procedure for their use was determined;
- procedures were developed to de-energize the equipment of the emergency control room in case of flooding;
- procedures for personnel actions in case of flooding of the reactor building, emergency diesel generator building, turbine hall, unit pump station were developed.

### **3 KH FLOODING (KHMELNITSKY NPP SITE)**

#### **3.1 Kh Design Basis**

The source of Khmelnitsky NPP service water supply is Goryn river and Gnyloy Rog river. The major hydraulic engineering structures consist of:

- off-channel cooling reservoir;
- earthen water-retaining dam;
- flood spillway;
- drainage canal of seepage water draw-off;
- pump station of seepage water return;
- cooling water intake channel;
- unit pump stations;
- pre-filter cameras;
- pressure and return circulating water lines;
- closed outlet channels with siphon structure to maintain water levels in closed outlet channel;
- open outlet channel;
- conjugating structures;
- pump station of additional water.

Each unit pump station provides water to the appropriate unit. The following basic technological equipment is located in the unit pump station:

- circulation pumps of the main cooling system, supplying water for cooling the turbine equipment in the main building;
- non-essential pumps supplying water to the cooling equipment located in the reactor and turbine halls, as well as in the special building;
- fire water pumps.

The essential service water system is designed for heat removal from the equipment important to safety, located in the reactor compartments and emergency diesel generator building, and for heat transfer to the ultimate heat sink. The essential service water system performs the following functions:

- remove heat from the coolant through the ECCS heat exchangers;
- remove heat from the equipment of safety systems;
- remove heat from the equipment of normal operation systems important to safety.

The essential service water system combines the functions of the normal operation system important to safety and the supporting safety system. The system consists of three independent cooling channels and it is isolated from other cooling systems. Spray ponds are made for each channel for cooling. The main source of the make-up water system is an open-pit type pond, where the pump station is located. To maintain the water quality in the coolant system, additional water is treated at the water purification plant. The reserve makeup of the system is provided from the nonessential service water system.

#### **3.1.1 Kh Flooding against which the Units Are Designed**

A threshold value of external flooding impact (seasonal floods and flash floods on the Goryn river and Gnyloy Rog river) is flooding of compartments in the Khmelnitsky NPP

buildings where safety systems and safety important systems are located. There are compartments located below a relative building elevation of 0.0. The relative elevation of 0.0 of the main buildings and emergency diesel generator buildings corresponds to an absolute elevation of 206.15 m. Thus, the threshold value of the external flooding impact is water level at Khmelniysky NPP reaching 206.15 m abs.

The relative elevation of 0.0 of unit pump station-1 and unit pump station-2 corresponds to an absolute elevation of 204.5. However, the unit pump stations are isolated from the Goryn river by the Khmelniysky NPP territory. Thus, for the unit pump station's equipment, the threshold value of the external flooding impact as a result of seasonal floods and flash floods on the Goryn river is the water level at Khmelniysky NPP reaching elevation 206.0 abs., which corresponds to an elevation of road planning in the vicinity of the main buildings.

The highest level of a spring tide in the Goryn basin over 1946-1999 was observed in 1956: on the Khmelniysky NPP water intake area near the town of Neteshin, this level reached an absolute elevation of 195.96. The absolute elevation of the maximum design-basis level in the vicinity of NPP water intake is 197.84 m.

Taking into account floor elevations of the site (206.0 m) and the outdoor switchyard area (222.0–24.0), the highest water levels induced by snowmelt and rain waters on the Goryn river do not threaten the NPP structures.

The makeup water pump station of the water reservoir is located at the downstream slope of the dam. Water is fed to the pump station under gravity through a bypass channel from the Goryn river. The relative elevation of the pump station floor of 0.0 corresponds to the absolute elevation of 198.0. In the event of seasonal floods, the pump station may be flooded, however the pump station flooding does not jeopardize functioning of the cooling water reservoir of the Khmelniysky NPP and the systems that use this water (the NPP comprised of two power units can function without charging from the cooling water reservoir under emergency conditions over a year).

The essential service water makeup pump station is isolated from the cooling water reservoir by the Khmelniysky NPP site with a grade level of 206.0. Thus, the water level rise in the water reservoir will not have an impact on functioning of the essential service makeup water pump station.

On the Goryn river, there are some small water reservoirs far away from the Khmelniysky NPP (more than 50 km) and their break cannot threaten the Khmelniysky NPP buildings and structures.

The break of the dam of the cooling water reservoir do not pose any threat as a breaking wave in the event of an accident (i.e., as a result of beyond-design-basis seismic effects) will spread over the Goryn floodplain westward downstream the river considering the lowering of ground levels.

### **3.1.2 Kh Provisions to Protect the Units against the Design Basis Flood**

Equipment of the systems involved into safe shutdown and maintaining of the reactor facility under "cold" shutdown conditions is located in the reactor and turbine compartments, emergency diesel generator building, electrical stack structure, and also includes cooling ponds and underground piping.

According to the design, the territory around Khmelniysky NPP buildings is planned with a slope from the buildings to transfer rain and snowmelt waters into storm sewage system. The territory of the area next to the main buildings of the reactor compartment is sloped towards an inlet channel. Therefore, if layers of water form on road passages, water will run down into the intake channel over the ground slope. The network operating staff controls operability of networks including the storm water system, conducts maintenance and repairs if necessary, cleans them in case of blockage and deterioration of their discharge capacity.

All the pump stations at the Khmelniysky NPP are equipped with drain pumps to prevent turbine hall flooding. For water pumping from drainage sumps, all emergency diesel generator buildings at the Khmelniysky NPP are provided with automated drainage systems.

In case of water level rise in the drainage sumps of the emergency diesel generator building, process alarm at the main control room is envisaged.

In the turbine compartments of the main buildings at Khmel'nitsky NPP unit 1 and Khmel'nitsky NPP unit 2, as well as in the reactor auxiliary building of the so-called "clean" area, drainage sumps of the industrial sewage system are provided at below-zero elevations of stair cages from which sewage water is drained automatically with drain pumps. The main control room is equipped with process alarms in case of sump overflowing. In case of water ingress into the main building rooms, water will run down into the drainage sump and will be drained with the help of drainage pumps.

### **3.1.3 Kh Compliance of the Units with Their Current Licensing Basis**

Plant compliance with licensing conditions is considered in Section 2.1.3.

## **3.2 Kh Evaluation of Safety Margins**

### **3.2.1 Kh Estimation of Safety Margins against Flooding**

Floor elevations of the dam top and ground level at the Khmel'nitsky NPP are 206.0 m abs., and the maximum initial elevation of a flooding wave in case of breaking of the cooling water reservoir dam is 203.0 m abs. Thus, structures at the Khmel'nitsky NPP site are located above the maximum horizon of the flooding wave.

### **3.2.2 Kh Measures That Can Be Envisaged to Increase Robustness of the Units against Flooding**

There is no need for developing and implementing additional actions to increase the robustness of the Khmel'nitsky NPP against potential external floods.

## **3 R FLOODING (RIVNE NPP)**

### **3.1 R Design Basis**

Rivne NPP service water supply is provided from the Styr river. Water intake structures are located on the right bank of the river and consist of a silt-settling tank, intake channel, additional water pump station and discharge water conduits. The water intake structures were built as a part of the first construction stage (power units 1-3).

For heat removal from turbine condensers and auxiliary equipment heat-exchangers, the Rivne NPP uses cooling towers of 100 000 m<sup>3</sup>/h capacity each. Spray ponds are used for heat removal from essential loads.

To provide for a scale-free mode, the main cooling system includes additional water treatment structures. To maintain specific water chemistry, the main cooling system includes a blowdown system. The essential and non-essential cooling systems are also fed from the Styr river by means of additional water piping. It is envisaged that the blowdown of essential and non-essential cooling systems is performed by the wind through droplet entrainment of water from the spray pond.

The essential service water system of units 1 and 2 is designed to cool down equipment that assures nuclear and radiation safety of the units (equipment of the reactor compartment, emergency diesel generators, turbine compartment equipment involved in unit cooldown). The essential service water system together with spray ponds is a recycling water system that is not connected with any other service water systems. This system is shared between units 1 and 2 and is composed of three independent trains. Each train is designed to supply the maximum design flow in case of main coolant piping break. Each safety system train is provided with one spray pond. Unit pump station-1 is common for both power units. The essential service water system must operate in all design-basis modes including emergency ones.

The essential service water system of power units 3 and 4 is intended for removing heat from equipment of the reactor compartment and emergency diesel generator building in

normal operation and in case of emergencies. The essential service water system is a recycling system provided with spray ponds as coolers. The system includes three trains, with one spray pond for each train. The system, which is completely isolated from other cooling systems, operates continuously. The design flow rate of cooling water is 9000 m<sup>3</sup>/h per unit and 3000 m<sup>3</sup>/h per train. Essential service water pumps are located in the emergency diesel generator building.

### **3.1.1 R Flooding against Which the Units Are Designed**

The Styr river is the closest surface water body that is also used as Rivne NPP service water source. The river water level rise primarily affects water intake structures, nearby structures and, to a lesser extent, structures located far from it.

In designing the Rivne NPP first stage, maximum design water levels were defined for water intake points of the Styr river. They were also taken into consideration in developing the design of the second stage of the Rivne NPP. The maximum level of the Styr river in the Rivne NPP intake area is 164.4 m. From 1947 to 2010, the highest level of a spring tide in the Styr basin was observed on 9 April 1956: in the Rivne NPP water intake area, this level reached 163.2 m. Approximately the same high flood level (163.0 m) was observed in 1979.

Since the floor elevation of the NPP site is 188.5 m, it cannot be flooded as a result of catastrophic river high floods.

The relative floor elevation of 0.0 m of the additional water pump station located on the Styr bank corresponds to an absolute elevation of 165.0 m. The absolute design ground elevation near the pump station is 164.85 m. In designing the pump station, the maximum design water level of the intake channel was taken as 163.8 m. The threshold value which, if reached, may result in flooding and resulting inoperability of the additional water pump station is the Styr water level of 165.0 m.

If the additional water pump station fails and, correspondingly, the plant cooling makeup systems fail, power units must be shut down. Makeup of the essential service water system after reactor shutdown can be provided using water inventory of the spray ponds, intake and discharge channels of the main circulating water cooling system, and cooling tower ponds. The total water inventory of the main cooling system structures is 170 000 m<sup>3</sup>, this volume will be sufficient for 27 days of plant operation in "cold shutdown" condition.

It should be stressed that emergency procedures and water drain pumps are provided in the event of flooding of the make up service pumps.

The potential flooding of Rivne NPP may be caused by an accident on hydraulic engineering structures, such as breaking of dams of water reservoirs located upstream the Styr river. Rivne NPP water intake structures are the largest hydraulic engineering structures constructed on the Styr river. There are no other large hydraulic engineering structures within the 30-km area of the Rivne NPP.

There are three upstream water reservoirs located in the Styr river basin at a significant distance from the Rivne NPP. The largest one is the Khrennikovske water reservoir located 212 km upstream of the RNPP water intake. The second water reservoir is Mlynovske water reservoir located approximately at the same distance from the Rivne NPP, on the Ikva river (the right tributary of the Styr river). The Kholonovsk water reservoir is located on the Bezmyanka river that flows into the Khrennikovske water reservoir.

Considering the long distance of the water reservoirs from the Rivne NPP water intake structures, their small water inventory and sufficient width of the Styr river floodplain (the Khrennikovske water reservoir that is 0.5 - 0.6 km wide in the downstream wall extends up to 2-3 km towards the Rivne NPP water intake structures), the water levels in the vicinity of water intake will not practically rise in case of hypothetical dam break, with the account of flow spreading over the river floodplain.



### **3.1.2 R Provisions to Protect the Units against the Design Basis Flood**

At present, the state of geological, hydrological and hydrogeological conditions is monitored at the Rivne NPP site and its 30-km area. These monitoring activities are conducted using different methods and monitoring facilities. Monitoring activities are performed by means of the following systems:

- surface water quality monitoring and control system;
- ground water quality monitoring and control system;
- system for monitoring of geological processes and soil conditions.

Hydrological situation on the Styr river is monitored by hydrometric stations of the Ukrainian Hydrometeorological Committee's network including those located upstream and downstream of Rivne NPP water intake (villages of Kolki and Mlynok). Hydrological observations of the Styr river are also performed by the Rivne NPP on the hydrological station located in the town of Kuznetsovsk. In case of dangerous water level rise of the Styr river that may result in flooding of the additional water pump station, the plant management will be informed and power units will be shut down if necessary.

The entire territory of the Khmel'nitsky NPP site has a common slope spreading from the north to south-west, towards the Styr river. Therefore, water layer (if formed) will run downhill to lower places. In accordance with the design, the territory surrounding the buildings was planned with regard for slope to drain rain and snowmelt waters to storm sewage. The Network Operations Department controls their operability, conducts maintenance and repairs required, as well as cleanup in case of blockage and deterioration of their discharge capacity (these activities are also performed for the storm sewage system).

Each Rivne NPP emergency diesel generator building is provided with a drain system to pump-out water from sumps. Rivne NPP main buildings at below-zero elevations are provided with drainage sumps of the industrial sewage system from which discharges are pumped out by drain pumps. In case of water ingress into the main building rooms, it will run down through stair cages to the drainage sump at elevation minus 3.6 m and will be pumped out using drain pumps.

### **3.1.3 R Compliance of the Units with Their Current Licensing Basis**

Plant compliance with its licensing basis is considered in Section 2.1.3.

## **3.2 R Evaluation of Safety Margins**

### **3.2.1 R Estimation of Safety Margins against Flooding**

The design ground elevation of the Rivne NPP site is 188.5 m and for additional water treatment structures is 188.9 m. Outdoor switchyards of 110 kV and 330 kV are located on the terrace arranged on the elevation of 192.5 m. The design ground elevation of 750 kV outdoor switchyards varies in the range of 180.0 m to 179.5 m. The maximum design water level of the Styr river is 164.4 m.

The NPP site is located well above the design water level of the Styr river with no risks of Rivne NPP flooding induced by an accident at hydraulic engineering structures.

### **3.2.2 R Measures That Can Be Envisaged to Increase Robustness of the Units against Flooding**

There is no need for developing and implementing additional actions to increase robustness of the Rivne NPP against potential external floods.

## **3 SU FLOODING (SOUTH-UKRAINE NPP)**

### **3.1 SU Design Basis**

The South Ukraine NPP with all main components of water supply system is located on the left bank of the Yuzhny Bug river about 159 km from its estuary. The water supply system includes the following main hydraulic engineering structures: cooling pond, intake channel,

water makeup and supply pump station, and spillway structures. South Ukraine NPP service water supply is based on pumped storage constructed on the Tashlyk gorge and is fed from the Yuzhny Bug river during operation.

The Yuzhny Bug is situated 4 km to the west of the site. The total length of the river is 806 km, water intake area is 63700m<sup>3</sup>. The river valley is a canyon 170-350 m wide, with slopes of approximate height of 50 m. There is no floodplain, the right bank is steep and is composed of granite, and the left bank is moderately steep, steadily washed out.

There are several hydroelectric plants on the Yuzhny Bug River: Pervomaisk Hydroelectric Plant with water reservoir capacity of 3.3 mln. m<sup>3</sup> and the Aleksandrovka Hydroelectric Plant with the water reservoir capacity of 134 mln. m<sup>3</sup>.

Construction of the Tashlyk Hydroelectric Pumped Storage Plant began in 1981. This plant is an integral part of the South Ukraine power system that also includes the South Ukraine Nuclear Power Plant and Aleksandrovka Hydroelectric Plant (small-sized).

According to the design, the Tashlyk Pumped Storage Plant had to consist of six recycling (generator/engine) units 150 MW each in turbine mode (225 in pump mode) and four ordinary hydraulic units 250 MW each. The total installed capacity would be:  $6 \times 150 + 4 \times 250 = 1600$  MW (turbine mode)/  $6 \times 225 = 1350$  MW (pump mode).

### **3.1.1 SU Flooding against Which the Units Are Designed**

Effect of floods on the Tashlyk stream is considered in terms of potential flooding of the South Ukraine NPP site. To characterize water rise levels, long-term observation data from hydrometric stations located within the South Ukraine NPP site are used.

To arrange for water exchange in the Tashlyk reservoir, flood spillway is provided, through which water is discharged during the flood. Water level in the reservoir may reach a 101.5 m watermark, that is 2.5 m lower than the design site elevation and, therefore, there is no risk of site flooding.

In case an accident occurs on flood spillway (gate closed) or transmission of discharged water through spillway is impossible, extreme rain flood volume will be transferred by overtopping (overflow of the crest) of the dam (102.3 m). Short-term wave actions and increased flow speed in the reservoir during the flood will not affect the NPP structures and safety of the South Ukraine NPP.

Therefore, a conclusion can be made that Tashlyk stream flood and further water rise in the Tashlyk reservoir do not jeopardize the safety of South Ukraine NPP buildings and structures.

The rise of the Yuzhny Bug river level does not pose a direct threat to facilities and structures located on the NPP site because the design site elevation is 104 m and is more than 70 m above the river level. Meanwhile, the distance from the Yuzhny Bug bank line to the South Ukraine NPP site is 3 km. Taking this into consideration, the flooding effects on facilities and structures located on the NPP site may not be taken into account due to the considerable site elevation above the river level.

It should also be noted that, in case of flooding on the makeup pump station and termination of water blowdown in the Tashlyk water reservoir, the water will be subjected to mineralization. This process is not critical for NPP operation as water reservoir has not been blown down for several years.

Based on the operating experience of the Tashlyk reservoir makeup pump station, the system is to be put in operation no more than once a year. It is obvious that a failure of the makeup pump station equipment will not lead to undesired consequences for the power unit.

### **3.1.2 SU Provisions to Protect the Units against the Design Basis Flood**

To prevent turbine hall flooding, all pump stations at the South Ukraine NPP are provided with drain pumps. For pumping water from drainage sumps, the automated drain water pumping systems are used at all emergency diesel generator systems. If water level rises in

drainage sumps of the emergency diesel generator building, the relevant alarm signal is generated at the main control room.

The turbine halls of the main buildings and auxiliary building of the reactor compartment in “non-contaminated area” at below-zero elevations of staircases are provided with industrial sewage system drain sumps from which wastewater is automatically pumped out by drain pumps. The main control room provides process alarms in case of sump overflowing. If water ingresses into the main building it will flow down the staircase into the drain sump and will be pumped out by drainage pumps.

South Ukraine NPP information services collect, process and store all corresponding meteorological and hydrological data in order to assure safety and to avoid any potential negative effects.

### **3.1.3 SU Compliance of the Units with Their Current Licensing Basis**

Unit compliance with the licensing conditions is considered in Section 2.1.3.

## **3.2 SU Evaluation of Safety Margin**

### **3.2.1 SU Estimation of Safety Margin against Flooding**

The level of the Tashlyk reservoir can reach a 101.5 m watermark, that is 2.5 m lower than the design site elevation and, therefore, does not pose a risk of site flooding.

The Yuzhny Bug water level rising does not pose a direct threat to facilities and structures located on the South Ukraine NPP site since the design site elevation is considerably higher than that of river water.

### **3.2.2 SU Measures That Can Be Envisaged to Increase Robustness of the Units against Flooding**

There is no need for developing and implementing additional measures to improve the South Ukraine NPP robustness against potential external flooding.

## 4. EXTREME WEATHER CONDITIONS

In the framework of stress tests, the following external hazards (except for earthquakes and floods) were analyzed for all operating NPPs:

- tornados;
- extreme wind;
- external fires;
- heavy precipitations, extreme snowfall;
- extreme temperatures;
- different combinations of external hazards.

The stress tests revealed no additional external hazards or their combinations that were not taken into account in the safety analysis reports.

### 4.1 Design Basis

#### *Extremely high/low temperatures*

The following temperature parameters were used in the design of operating NPPs:

Outside temperature, °C	ZNPP	KhNPP	RNPP	SU NPP
Absolute maximum temperature	+41	+41	+38	+41
Absolute minimum temperature	-34	-36	-38	-36

Long-term data were used to specify the characteristics of climatic conditions:

- for the Zaporizhzhya NPP site - data taken from the weather stations of Zaporizhzhya, Nikopol and Prishyb. The Nikopol weather station was selected as a reference one;
- for the Khmelnytsky NPP site – data taken from the Shepetovka weather station;
- for the Rivne NPP site – data taken from the Sarny weather station;
- for the South Ukraine NPP site – data taken from the Pervomaisk and Konstantinovka weather stations.

#### *Heavy precipitations*

In compliance with standards for design of rainwater drain system and industrial storm sewer systems, the design flow rate of rainwater is determined on the basis of 20-min rainfall intensity (q<sub>20</sub>).

For the Zaporizhzhya NPP site, this design criterion constitutes 85 L/secxha.

For the Khmelnytsky NPP site, this criterion constitutes 100 L/secxha.

For the Rivne NPP site, this criterion constitutes 100 L/secxha.

For the South Ukraine NPP site, this criterion constitutes 90 L/secxha.

#### *Tornadoes*

The standards that were in force at the time the Ukrainian NPPs were designed did not require to consider tornado impacts. In accordance with new standards and regulations and taking into account statistics of observations, calculation parameters and characteristics for a potential tornado have been established for Ukrainian NPP sites.

As statistical data are accumulated, tornado characteristics may be specified. Therefore, the assessment of buildings and structures for tornado impact in safety analysis reports used more conservative values of tornado loads as compared with calculation ones.

Tornado characteristics represented in the SAR for each site include, in particular, the following parameters:

- tornado class;
- annual tornado probability;
- maximal tornado rotational speed;

- maximal tornado forward speed;
- pressure drop between the center and periphery of tornado.

Impact of tornado on buildings and structures is determined by three factors:

- wind frontal pressure;
- pressure drop (between the tornado center and periphery);
- flying missiles.

According to regulatory requirements, tornado impacts must be taken into account for buildings and structures of category I responsibility for nuclear and radiation safety. They include: reactor building, building of emergency diesel generators and spray ponds of essential service water system of group A.

### **External fires**

When mixed (coniferous and hardwood) forests growing on dry soil are on fire, a safe distance to the facility shall be at least 100 m on the leeward side (according to the wind rose at an average annual wind speed up to 10 m/sec).

## **4.2 Evaluation of Safety Margins**

### **4.2.1 Estimation of Safety Margin against Extreme Weather Conditions**

#### **Extremely high/low temperatures**

Long-term observations at the operating NPP sites were used to obtain the following temperature parameters:

<b>Outside temperature, °C</b>	<b>ZNPP</b>	<b>KhNPP</b>	<b>RNPP</b>	<b>SU NPP</b>
Maximum temperature	+39*	+36,6	+37,8	+39
Low temperature	-34	-33,6	-36,9	-34

\*The maximum temperature measured by the Nikopol weather station was up to +39°C and by the Zaporizhzhya weather station was +41°C.

The comparison of the parameters provided in the table and the data presented in Section 4.1 shows that absolute temperatures observed at all operating NPPs are not exceeded.

#### **For WWER-1000 units**

To analyze the impact of extreme temperatures on operation of NPP systems, all systems are divided into two categories depending on the cooling method:

- systems that use different types of water-cooled heat exchangers. Water with a temperature from +5 to +33 °C shall be supplied to these systems to ensure their normal operation;
- ventilation systems that use atmospheric air for cooling. The indoor temperature shall promote absorption of excessive heat and the equipment temperature shall be kept in compliance with technical specifications for normal operation of such systems.

Different ventilation systems are used for cooling of supporting safety systems. As a rule, these systems use water from the essential service water system as a coolant. These systems include air conditioning and recirculation ventilation. The required air temperature in ventilation and air conditioning systems is maintained within operational limits by auxiliary equipment (heat exchangers, cooling equipment, etc.) and the air is supplied from rooms which are not directly connected with the atmosphere.

The operability of auxiliary ventilation systems is determined by the capability of spray ponds to maintain temperature of the cooling water in the range +5÷+33°C. The system performance criterion is supply of cooling water through the emergency and regular cooldown heat exchanger at a flow rate of 3000 m<sup>3</sup>/h for the respective safety system train.

Service water is cooled in spray ponds during hot summer months by spraying it through nozzles. During the cold period when temperature is below 15°C, idle water discharge without spraying is provided.

The capacity of spray ponds was selected to ensure performance without supply of make-up water during the specified time (24 hours) required for repair and recovery of any make-up water system of the ponds. The volume of water in one pond is 21250 m<sup>3</sup>. Water from make-up stations of spray ponds is supplied to compensate for evaporation and wind-induced losses of spray water ponds.

It should be considered that water is renewed every 7 hours in the pond during the cold period, and its temperature is mainly determined by the temperature of water supplied from the reactor compartment cooling system. Taking into consideration that the temperature of water supplied from the reactor compartment cooling system is maintained in the range 12-45°C, it can be stated that extremely low air temperature will not decrease water temperature in the spray pond below the design value (5°C). Hence, low temperatures do not pose hazards for cooling systems that use water from spray ponds. Control of water temperature and the redundant make-up system allow maintaining water temperature in spray ponds within design limits as required.

#### **For WWER-440 units**

The analysis demonstrated that heat removal to the ultimate heat sink was characterized by the following conditions:

- when one heat removal channel is in use, the required temperature (33°C) for ECCS heat exchangers is maintained at the mean daily temperature for the five hottest days per year, with 10% probability, at  $T_{air}=25.6^{\circ}\text{C}$ , 70% humidity, 2.7 m/sec wind speed, and flow rate of circulating water in the channel of 3888 m<sup>3</sup>/h at thermal load (26.4 + 24.2) = 50.6 Gcal/h).
- when two heat removal channels are in use (which meets the single-failure criterion), the required temperature of 33°C for ECCS heat exchangers is maintained at  $T_{air}=30.0^{\circ}\text{C}$  (minimum threshold of exceeding the maximum temperature per year with  $k=0.964$ , 65% humidity, 2.7 m/sec wind speed, flow rate of circulating water in two channels of 7776 m<sup>3</sup>/h at thermal load (26.4 + 24.2) = 50.6 Gcal/h).

In determining a threshold value for extreme temperatures, it was established that the essential service water system is capable of ensuring the required temperature of cooling water at air temperature of 36.4°C in operation at rated power. The highest temperature above 25°C can last no more than 6-7 hours per day and above 30°C no more than 1.5-2 hours.

Extremely low temperatures do not affect cooling safety systems due to the following reasons:

- safety system piping is laid at a depth of more than 5 m, which is much lower than frost penetration;
- a heat conservation mode is envisaged for spray ponds to prevent water overcooling in winter. For this purpose, a greater amount of warm water from heat exchangers of safety systems is supplied under the spray pond. A smaller amount of warm water is sprayed through nozzles to prevent their freezing and keep them operable to accept full thermal loading from the emergency cooldown heat exchanger;
- a combined cooling system is envisaged for emergency heat removal: part of the spray pond operates without spraying and the other part (36 nozzles) with spraying.

### ***Heavy Precipitations***

Rain water drains and industrial and storm water sewage systems are envisaged to protect buildings and facilities at the Zaporizhzhya, Khmeltsky, Rivne and South Ukraine NPP sites from precipitations. Surface water is drained from the sites and 750 kV open switchyards via bladed surfaces to road gutters or trenches, as well as via storm-water collectors around sites with subsequent discharge to storm sewage. Excessive rain water is absorbed into the soil.

#### ***Zaporizhzhya NPP***

Water accumulation at the Zaporizhzhya NPP site in case of heavy rains can be potentially hazardous if its level is higher than certain site structures and thus may lead to an emergency situation. Rain water drains and industrial and storm sewage systems are designed to protect buildings and facilities at the Zaporizhzhya NPP site against precipitations. The designed capacity of the industrial and storm sewage system is sufficient to drain rainwater from the on-site road surfaces with precipitation intensity up to 90 L/secxha.

The maximum observed precipitation levels at the Zaporizhzhya NPP site are lower than the design criteria established for Zaporizhzhya site engineering structures. Site walkdowns and analysis of site structures did not reveal any components whose flooding can cause a site emergency in case of heavy rains.

Entrances into site buildings and structures are sufficiently elevated above the ground level. Additionally, the Zaporizhzhya site surface is sloped down to the Kakhovka reservoir shoreline.

#### ***Khmeltsky NPP***

Water accumulation at the Khmeltsky NPP site in case of heavy rains can be potentially hazardous if its level is higher than certain site structures and thus may lead to an emergency situation. Rain water drains and industrial and storm sewage systems are designed to protect buildings and facilities at the Khmeltsky NPP site from precipitations. The designed capacity of the industrial and storm sewage system is sufficient to drain rainwater from the on-site road surfaces with the precipitation intensity up to 100 L/secxha.

The maximum observed precipitation levels at the Khmeltsky NPP site are lower than the design criteria established for site engineering structures. Site walkdowns and analysis of on-site structures did not reveal any components whose flooding can cause a site emergency in case of heavy rains.

The zero level of all Khmeltsky NPP site buildings is located at least 150 mm above the actual ground level close to the building. The site is leveled in a way that the slopes are oriented from the buildings to prevent flooding of the main facilities and buildings. Land and roads around the reactor buildings are sloped down to the intake channel to drain rainwater if the storm sewage is failed.

#### ***Rivne NPP***

Water accumulation at the Rivne NPP site due to heavy rains can be potentially hazardous if its level is higher than certain site structures and thus may lead to an emergency situation. Rain water drains and industrial and storm sewage systems are designed to protect buildings and facilities at the Rivne NPP site against precipitations. The designed capacity of the industrial and storm sewage system is sufficient to discharge rainwater from the on-site road surfaces with the precipitation intensity up to 100 L/secxha.

The maximum observed precipitation levels at the Rivne NPP site are lower than the design criteria established for site engineering structures. Site walkdowns and analysis of structures did not reveal any components whose flooding can cause a site emergency in case of heavy rains.

The zero level of all RNPP site buildings is located at least 150 mm above the real ground level close to the building. The site is sloped from North to South-West down to the Styr river. Thus all accumulated water will run down to lower areas.

*South Ukraine NPP*

Water accumulation at the South Ukraine NPP site due to heavy rains can be potentially hazardous if its level is higher than certain site structures and thus may lead to an emergency situation. Rain water drains and industrial and storm sewage systems are designed to protect buildings and facilities at the South Ukraine NPP site against precipitations. The designed capacity of the industrial and storm sewage system is sufficient to drain rainwater from the on-site road surfaces with the precipitation intensity up to 90 L/secxha.

The maximum observed precipitation levels at the South Ukraine NPP site are lower than the design criteria established for site engineering structures. Site walkdowns and analysis of structures did not reveal any components whose flooding can cause a site emergency in case of heavy rains.

Entrances into site buildings are sufficiently elevated above the ground level. Surface water is drained via the bladed surfaces to road gutters or trenches, as well as via storm-water collectors around the site with subsequent discharge to storm sewage. Excessive rain water is absorbed into the soil.

**Tornados**

In the framework of stress tests, safety margins of buildings and structures have been assessed for all three factors of tornado impact, mentioned in Section 4.1.

**Frontal Wind Pressure.** The wind frontal pressure for all NPP sites calculated using statistics data is in the range from 6.0 to 12.0 kPa. Category I buildings and structures for all operating units are designed to withstand an air shock wave with frontal pressure compression pressure  $\Delta P = 30$  kPa. Therefore, all category I buildings and structures have safety margins in terms of tornado frontal wind pressure.

**Pressure Drop.** To assess safety margins of buildings and structures in terms of pressure drop caused by tornado, calculation load (according to statistics data) is compared with a more conservative value from SAR.

Tornado impact on buildings and structures resulting from pressure drop (drop of atmospheric pressure) has the following features: in case of completely open structures, including skeleton-type buildings with openings, cooling towers, ventilations stacks, etc., internal and external pressure become equal almost immediately and additional loads on structures induced by pressure drop are negligibly small. In closed buildings and structures, pressure does not change in case of tornado, and internal explosion effect may occur. Closed buildings and structures (without openings) include reactor containment and building of emergency diesel generators. Assessed safety margins for closed buildings and structures in terms of pressure drop between the tornado center and periphery are provided in Table. It is shown that there are safety margins for this impact.

Safety Margins for Pressure Drop in Tornado

Site	ZNPP	KhNPP	RNPP	SUNPP
Load				
Calculation value (according to statistics) (kPa)	10.3	7.0	7.0	6.4
Conservative value (accepted in SAR) (kPa)	10.9	10.0	10.0	8.1



**Flying Missiles.** Analysis shows that tornadoes of classes 0 – 3 cannot cause flying missiles hazardous for NPP buildings and structures.

Hence, it can be stated that there is no significant negative effect from a tornado on buildings and structures and safety systems and safety-related systems located in them. There is potential for some damage to window and door openings of skeleton-type buildings and structures, which will not lead to degradation of safety functions.

In terms of heat removal to the ultimate heat sink, there is a potential risk of failure of open components of the essential service water system due to loss of water in case of tornado. This event is analyzed in SAR. The stress tests confirmed the need for compensatory measures to ensure heat removal to the ultimate heat sink in case of loss of water in spray ponds following a tornado.

Analysis of tornado impact on the dry spent fuel storage facility at the Zaporizhzhya NPP site also revealed no negative impact. Robustness against tornadoes is ensured by the design and structural materials of the storage facility. The ventilated storage cask (VSC) is the first barrier for protection against tornado. The strength calculations performed in SAR demonstrate that the design features of the storage facility ensure complete VSC robustness against tornado impact. The most vulnerable place in terms of tornado can be clogging of air vents of VSC. Partial and complete clogging of air vents, which is a realistic situation, has been analyzed in detail in the SAR for the dry spent fuel storage facility. The results show that safety is not affected in this case.

### **External Fires**

The 10-km areas of the operating NPPs are free from fire hazardous sources like oil and gas mains, delivery ducts, underground and surface storage tanks with flammable substances for more than 1000 m<sup>3</sup>, as well as airports, air communication and flight lines (limited to 5 km in radius from NPP and 6 km in height), coal and peat mining facilities.

Forests adjacent to NPPs are considered the main potential external fire source for the sites.

The actual distance from Zaporizhzhya NPP units to the forest is not less than 400 m, which exceeds the design safety limit of 100 m from a leeward side.

The actual distance from Khmelnytsky NPP units to the forest is not less than 550 m, which exceeds the design safety limit.

The actual distance from Rivne NPP units to the forest is not less than 500 m, which exceeds the design safety limit.

There are no woodlands around the South Ukraine NPP site.

### **4.2.2 Measures Which Can Be Envisaged to Increase Robustness of the Units against Extreme Weather Conditions**

Tornado strike on the Zaporizhzhya, Khmelnytsky, Rivne and South Ukraine NPP sites can potentially result in a failure of spray ponds of the essential service water systems (ESWS) due to its impact on the open water surface (water ejection; water funnels resulting in air plugs inside suction lines; drift of trash resulting in clogging of suction line baskets). Loss of ESWS can cause failure of emergency power supply from emergency diesel generators (EDG).

The following cooling water supply procedures were developed at the Zaporizhzhya NPP to compensate for ESWS pump failure:

- procedure for cooling water supply to EDGs from fire water posts;
- procedure for cooling water supply to safety system pumps from fire water posts;
- procedure for cooling water supply to EDG compressors US11(21,31)D01 from fire water posts (additional insets into ESWS are required).

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A concept for water supply to fire piping from a suction dredge ship was developed. In this case, the fire pipeline will be used to supply cooling water to safety system equipment and also for spray pond makeup.

Measures on tornado resistance enhancement for the Khmel'nitsky, Rivne and South Ukraine NPPs shall be developed and implemented.

No additional measures are required to enhance the resistance of the Zaporizhzhya, Khmel'nitsky, Rivne and South Ukraine NPPs to extreme winds, precipitations, temperatures, external fires and their combinations.

## 5. LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

### 5.1 Loss of Electrical Power

#### 5.1.1 Loss of Off-site Power

Loss of off-site power to WWER-440/1000 leads to loss of on-site power that causes disconnection of the main plant equipment. To ensure power supply to systems that are required to maintain the units in a safe state, WWER-1000 and WWER-440 reactors are provided with essential power supply systems (EPSS).

EPSS is designed using the same principles as safety systems: redundancy, physical separation, and independence of system trains. It has three independent trains with the identical components sets.

In case of loss of off-site power, power is supplied from emergency diesel generators and batteries. In terms of power and configuration of connected loads, each train of the emergency power supply system is capable of ensuring safe shutdown state in all design-basis accidents. In case of loss of off-site power, signals for startup of the diesel generators are actuated independently for each DG. Therefore, the high reliability of emergency power supply function is provided.

Each unit of Ukrainian NPPs has three DGs. In addition, all NPPs are provided with common-unit DGs (for detailed information see below):

- Zaporizhzhya NPP – 20 DGs. There are three DGs per unit and two additional common-unit DGs for units 5 and 6.
- Rivne NPP – 14 DGs. There are three DGs per unit and two additional common-unit DGs for unit 4.
- South Ukraine NPP – 11 DGs. There are three DGs per unit and two additional common-unit DGs for unit 3.
- Khmelnytsky NPP – 8 DGs. There are three DGs per unit and two additional common-unit DGs for unit 2.

The power of each common-unit DG (5600 kW) was selected to ensure power supply to all main equipment of two units, taking into account failure of one of the DGs.

All DGs are actuated automatically. When necessary, the DGs could be started by the MCR operator. Time delay to full power operation is less than 15 seconds. Loads are connected to DGs by the sequential loading program.

All DGs at NPPs are provided with 7-day reserve of diesel fuel, and DG design-basis non-serviceable operation in emergencies is 250 hours. Consequently, it may be concluded that the emergency power supply system was designed using the redundancy principle, has a sufficient number of DGs (3 DGs per unit as a minimum), and the design features permit high reliability of the system.

It should be pointed out that a series of measures to increase reliability of power supply to safety systems are ongoing under the Comprehensive (Integrated) Safety Improvement Program (C(I)SIP). For instance, modifications to allow supply of power to safety systems of an affected unit from the neighboring units of the same NPP have been implemented.

In case of loss of off-site power, the reactor protection system actuates signals (by MCP trip, or frequency decrease, by turbine-driven feedwater pumps trip) and the reactor is transferred into subcritical state by insertion of control rods. Heat is removed from the reactor via the secondary circuit through makeup of steam generators using emergency feedwater systems and steam dump devices (BRU-A, SG SV). Natural circulation is established in the primary circuit, the reactor is in stable state with high coolant pressure and temperature. Under normal operation of systems and interlocks, loss of off-site power does not lead to further increase of primary parameters and necessity of dumping primary coolant via the pressurizer pilot-operated relief valve (PRZ PORV).

Hence, in case of loss of off-site power, the WWER-1000 and WWER-440 reactors can be transferred into a safe state and ensure continuous decay heat removal for more than 72 hours.

#### **5.1.2 Loss of Off-site Power and Loss of the Ordinary Backup AC Power Source**

In case of an accident caused by loss of off-site power and failure of all DGs, decay heat removal function is not performed. Without operator actions, this will lead to loss of the primary coolant and uncovering and damage of fuel. However, the operator can restore power supply from one of the off-site sources.

There are additional sources and options for supply of power to 6 kV buses of Zaporizhzhya in-house systems:

- power supply from the Zaporizhzhya Thermal Power Plant if it remains operational;
- power supply from Dnipro Hydroelectric Plant through two independent supply lines;
- power supply from the Kakhovka Hydroelectric Plant;
- power supply from one (any) DG of any other Zaporizhzhya NPP unit that is automatically transferred to the in-house power supply mode;
- power supply from common-unit DGs;
- power supply to Zaporizhzhya NPP 750 kV outdoor switchyards from the Yuzhnodonbass 750 kV high-voltage line (if the Donbass electrical grid is successfully separated from the Ukrainian unified electrical grid) or from one of the high-voltage lines (Dnipro or Zaporizhzhya).

Corresponding procedures for power supply restoration were developed and emergency operating procedures (EOP) were updated for ZNPP units 1-6 (2010). Plant staff has passed corresponding training and their skills are maintained by scheduled trainings.

There are additional sources and options for supply of power to 6 kV buses of Khmelnytsky in-house systems:

- power supply to internal grid from the one (any) DG of any Khmelnytsky NPP unit that automatically transferred to the in-house power supply mode;
- power supply to internal grid from any common-unit DGs.

Plant staff is also recommended to try to power safety systems from alternative sources such as the power unit remaining operational or from the grid through 330/750 kV outdoor switchyard via the 330 kV high-voltage line (Shepetovka, Rivne or Khmelnytsky substation) or via 750 kV high-voltage line.

The Rivne NPP has the following additional sources and options for power supply to 6 kV buses of plant in-house systems:

- power supply to plant internal grid from one (any) DG of unit 1 or 2 as well as from one (any) DG of unit 3 or 4, which automatically transferred to the in-house power supply mode;
- power supply to plant internal grid from any common-unit DG.

The operator is recommended to try to power safety systems from alternative sources: from the units that remain operational or from the grid through outdoor switchgear buses via one of four 330 kV high-voltage lines (Rovno, Grabov, Kovel or Lutsk Sev substations) or via 750 kV high-voltage line (Zapadnoukrainskaya substation).

The South Ukraine NPP has the following additional sources and options for power supply to 6 kV buses of plant in-house systems:

- Kremenchug Hydroelectric Plant, which uses a renewable energy source and does not require special conditions for electricity generation;
- Tashlyk Hydroelectric Pumped Storage Plant, which is operated in generator mode. The startup of the Tashlyk plant in the generator mode is possible only if its internal grid is powered from the Alexandrovka Hydroelectric Plant;

- DGs of unit 2 connected via the transmission lines may be a source of power for one of the safety systems of the unit 1. It is necessary to monitor the DG power and load to prevent DG overloading;
- SUNPP Unit 3, which automatically is transferred to power supply mode to NPP internal grid. The long-term operation of Unit 3 in power supply mode to internal grid ensures:
  - in-house systems of units 1, 2;
  - Tashlyk Pumped Storage Plant operating in the engine mode;
  - loadings in the region adjacent to the South Ukraine NPP.

In case of an accident, the design requirements related to duration of battery operation for all the NPP units in full load is 1 hour. It should be noted that the actual time of operation from batteries will exceed the design-basis one. In addition, lists of loads to be disconnected to extend the battery operation time have been developed for all Ukrainian NPPs.

According to analysis of beyond design basis accidents involving long-term station blackout and failure of all DGs, it is required to restore power to one pump of the low-pressure injection system or one SG emergency feedwater pump to prevent core damage. In this regard, a safety upgrade has been implemented at Rivne NPP units 1 and 2 (WWER-440). A steam generator additional emergency feedwater system (AEFS) has been introduced at these units. This system is independent and survivable because its performance does not depend on other safety systems and normal operation systems. AEFS pumps are driven directly by their own air-cooled diesel engines. The water inventory of AEFS is  $2 \times 1000 \text{ m}^3$ . In case of station blackout, the AEFS inventory is sufficient to ensure long-term decay heat removal. The AEFS tanks can be refilled (detailed information on the AEFS is presented in Section 1.1.2 R).

Taking into account the above-mentioned information, it can be concluded that under loss of off-site power and loss of all DGs for WWER-1000 and WWER-440 type reactors the adequate operator actions aimed at transferring the reactor unit to safe state are envisaged.

### **5.1.3 Loss of Off-site Power and Loss of the Ordinary Backup AC Power Sources, and Loss of Permanently Installed Diverse Backup AC Power Sources**

In case of loss of off-site power, loss of the ordinary backup AC power sources and loss of permanently installed diverse backup AC power sources (station blackout), the power from the batteries remains available.

Current Ukrainian EOPs for WWER-1000 NPPs include procedures developed to increase time before core damage. These procedures include operator actions on SG makeup from turbine hall deaerators and primary system makeup from ECCS hydroaccumulators. Implementation of such activities allows significantly extending time before the core damage (more than for 12 hours). However, these actions after discharge of the batteries require mobile energy sources to monitor necessary parameters and control the respective valves. Therefore, EOPs recommend alternative actions based on connection of a mobile diesel generator and pumping unit (MDGPU). General requirements for MDGPUs, including their configuration and location on the site, have been developed in the framework of stress tests.

At WWER-440 units the same functions can be performed by AEFS.

Regarding SFP, the time available to fuel heatup above the design limits for WWER-1000/V-320 is 7.5 hours, for WWER-1000/V-302, V-338 is 6.5 hours, and for WWER-440/V-213 is 16 hours. These values were calculated in the framework of stress tests.

### **5.1.4 Conclusion on the Adequacy of Protection against Loss of Electrical Power**

Based on analysis of emergencies caused by loss of off-site power including total station blackout for Ukrainian WWER-1000 and WWER-440 units, the following can be concluded:

- in case of loss of off-site power, WWER-1000 and WWER-440 can be transferred into the safe state and decay heat removal can be provided within more than 72 hours in case of site isolation;

- in case of loss of off-site power and failure of all DGs, there are additional possibilities and adequate operator actions intended to transfer the reactor into the safe state (Section 5.1.2);
- in case of loss of off-site power and loss of the ordinary backup ac power sources, and loss of permanently installed diverse backup ac power sources, the main specific features of accident progression are:
  - loss of monitoring of the main reactor parameters due to discharged batteries that complicates transfer NPP units into the safe state and maintaining it;
  - potential deficiencies of SG and/or primary circuit makeup and cooldown, as well as SFP makeup and cooling for long time (up to 3 days) in case of NPP isolation.

In this case, EOPs recommend using mobile devices for SG emergency makeup and SFP makeup and cooling.

The AEFS is capable of ensuring long-term heat removal from the core in case of total station blackout at Rivne NPP unit 1 and 2 with WWER-440 reactors.

#### **5.1.5 Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Electrical Power**

Measures on possible accident management in case of total station blackout to provide emergency power supply and primary and secondary makeup within a long-term period were developed in the framework of stress tests. Corresponding calculations were performed to determine necessary water inventory and equipment configuration, and proposals on equipment layout were developed. A detailed description of the proposed measures is presented in Section 5.3.3.

### **5.2 Loss of the Decay Heat Removal Capability/Ultimate Heat Sink**

#### **5.2.1 Design Provisions to Prevent the Loss of the Primary Ultimate Heat Sink, Such as Alternative Inlets for Sea Water or Systems to Protect Main Water Inlet from Blocking**

Emergency cooling at WWER-1000 and WWER-440 units as well as heat removal to ultimate heat sink are provided by the following systems:

- emergency core cooling system (ECCS) with subsequent heat removal to the ultimate heat sink via the essential service water system;
- secondary circuit with subsequent heat removal to the ultimate heat sink via the secondary steam dump devices.

Design features related to prevention of loss of heat removal are presented in Sections 1.1.2 Z, 1.1.2 Kh, 1.1.2 R and 1.1.2 SU.

#### **5.2.2 Loss of the Primary Ultimate Heat Sink (e.g., Loss of Access to Cooling Water from the River, Lake or Sea, or Loss of the Main Cooling Tower)**

Loss of access to cooling water from the river or pond results in failure of the cooling water system. In such emergency conditions, operator actions are intended to decrease primary pressure, provide alternative heat removal from safety systems (using emergency service water tanks, alternative water sources), transfer the reactor into a safe shutdown state and perform recovery actions.

The reactor cooling is performed through secondary system by means of SG emergency feedwater supply and steam discharge via BRU-A or SG SV by operator.

The strategy of SG makeup from deaerators and opening of PRZ PORV to decrease primary pressure to allow discharge of hydroaccumulators can be considered as an alternative. This strategy for WWER-1000 reactors will allow decrease in primary pressure and ensure operation of the LPIS. Thereafter, alternative heat removal to the ultimate heat sink must be provided by one LPIS train or one SG emergency feedwater train. For WWER-

440 reactors, this strategy can be implemented even without possibility of SG makeup from deaerators (taking into account AEFS operation).

Therefore, in case of loss of access to cooling water from the river or pond or loss of the main cooling tower, adequate operator actions intended to decrease primary pressure, transfer the reactor to a safe shutdown state are envisaged. Consequently, Ukrainian NPPs with WWER-1000 and WWER-440 reactors are capable of ensuring long-term decay heat removal during more than 72 hours in case of site isolation.

### **5.2.3 Loss of the Primary Ultimate Heat Sink and the Alternate Heat Sink**

Progression of this accident is similar to progression of the total station blackout. The main difference is later failure of active systems since pumps will remain operational for some time without cooling and, thus, a longer time margin before core damage will be ensured. In this case, the operator has more time available for emergency actions. Secondary steam dump devices and PRZ PORVs will remain in operation.

EOPs (for WWER-1000 units) govern operator actions that are aimed at decreasing primary pressure to allow LPIS operation and are based on the use of SG makeup from TG deaerators and discharge of hydroaccumulators (if there is no possibility to use primary emergency makeup and cooldown system). This will allow significant increase in time to reactor core damage.

AEFS has been implemented at Rivne NPP units 1 and 2 with WWER-440 reactors. AEFS pumps are driven directly by their own air-cooled diesel engines. The water inventory of AEFS is  $2 \times 1000 \text{ m}^3$ . Taking into account the possibility to refill system tanks, AEFS pumps can ensure long-term heat removal from the core in case of total loss of primary heat removal (detailed information on the AEFS is presented in Section 1.1.2 R).

In the framework of stress tests, the calculation analysis for the most unfavorable accident conditions, i.e. combination of station blackout and loss of decay heat removal to the ultimate heat sink has been performed, and time available to reactor core damage as well as cliff-edge effects have been identified (the results are provided in Section 5.3).

### **5.2.4 Conclusion on the Adequacy of Protection against Loss of Ultimate Heat Sink**

The following conclusions can be made upon results of accident analysis related to loss of ultimate heat sink:

- in case of loss of primary heat removal (emergency trip of cooling tower fans and pumps, drop of pressure in the discharge line etc.) and NPP isolation, adequate operator actions are envisaged. They are aimed at decreasing primary pressure, use of alternative heat removal sources and reactor transfer into a safe state;
- in case of loss of primary heat sink and impossibility to use an alternative heat sink, the main features of accident progression are as follows:
  - possible deficiency of SG and/or primary circuit makeup and cooldown as well as SFP makeup and cooling for long time (up to 3 days) in case of NPP isolation;
  - inability to provide normal heat removal through the secondary and/or primary circuit for a long-term period (more than 72 hours) under loss of ultimate heat sink;
  - inability to provide normal heat removal from SFP for a long-term period (more than 72 hours).

In this case, EOPs recommend applying procedures based on connection of alternative power supply sources and MDGPU aimed at providing SG emergency makeup, as well as SFP makeup and cooling.

### **5.2.5 Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Ultimate Heat Sink**

Measures on possible accident management in case of loss of ultimate heat sink to provide primary and secondary makeup within a long-term period were developed in the framework of stress tests. Corresponding calculations were performed to determine necessary water inventory and equipment configuration, and proposals on equipment layout were developed. A detailed description of the proposed measures is presented in Section 5.3.3.

#### **5.3 Loss of the Primary Ultimate Heat Sink, Combined with Station Blackout**

Safety systems and power supply from unit-specific DGs, common-unit emergency DG and other on-site AC sources are not considered for initiating event (IE) “loss of primary heat sink combined with station blackout”. Battery-powered equipment is the only equipment that remains in operation.

Natural circulation of primary coolant is established upon the MCP coastdown. When secondary pressure exceeds specific setpoints, BRU-As are opened and remain in this position after battery discharge (estimated discharge time of batteries is 1 hour). The rise in primary coolant temperature results in increase of pressure due to thermal expansion of the coolant. Periodical spring-driven actuation of the PRZ RV leads to decrease in the primary coolant inventory. Decrease of the coolant level in the reactor core and loss of fuel heat removal capability leads to fuel overheating with further potential melting. Prevention of severe fuel damage in the reactor core requires restoration of the reactor coolant level and heat removal from fuel rods. Results from evaluation of time available for operator actions and emergency strategy efficiency are provided below.

##### **5.3.1. Time of Autonomy of the Site before Loss of Normal Cooling Condition of the Reactor Core and Spent Fuel Pool (e.g., Start of Water Loss from the Primary Circuit)**

Numerical analyses of the IE “total station blackout and loss of heat removal to ultimate heat sink” have been carried out for all types of Ukrainian NPPs in the framework of stress tests. This accident was selected because, in terms of degradation of heat removal conditions, a combination of station blackout and loss of heat removal to the ultimate heat sink is the most conservative and limiting case. The calculation analyses were performed based on conservative assumptions. The analysis was intended to:

- identify the minimum time span available until the maximum design limit of fuel rod damage is exceeded;
- identify cliff-edge effects;
- analyze efficiency of operator actions according to EOPs;
- identify the minimum SG makeup flow rate and the required water inventory to prevent severe core damage during 3 days;
- identify the minimum primary makeup flow rate and water inventory necessary to prevent severe core damage during 3 days.

The analyses were carried out for:

##### *Reactor operation at rated power:*

- basic scenario “total station blackout and loss of heat removal to the ultimate heat sink” (taking into account operation of first category non-interruptible power supply equipment and capacity of batteries, operator actions according to EOPs);
- total station blackout and loss of heat removal to the ultimate heat sink with secondary “feed and bleed” procedure using a low-head MDGPU;
- total station blackout and loss of heat removal to the ultimate heat sink with primary “feed and bleed” procedure using MDGPU;
- total station blackout and loss of heat removal to the ultimate heat sink with pressurizer safety relief valve stuck open (medium LOCA);



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- total station blackout and loss of heat removal to the ultimate heat sink with pressurizer safety relief valve stuck open and primary “feed and bleed” procedure using MDGPU (pressurizer SRV stuck open was assumed conservatively to cover the possible primary leaks, including leaks through MCP seals).

To determine the minimum time available to reactor core damage, additional analysis of total station blackout and loss of heat removal to the ultimate heat sink without operator actions according to EOPs has been carried out.

*Reactor in shutdown states:*

- total station blackout and loss of heat removal to the ultimate heat sink, reactor is open;
- total station blackout and loss of heat removal to the ultimate heat sink, taking into account measures aimed at preventing reactor core damage.

*Spent fuel pool:*

- station blackout and loss of decay heat removal from SFP to the ultimate heat sink;
- station blackout and loss of decay heat removal from SFP to the ultimate heat sink, taking into account measures aimed at preventing fuel damage.

The calculation analyses have been performed using validated and verified models for the RELAP/Mod 3.2 computer code developed previously in the framework of safety analysis report.

The time margin to core melting in case of station blackout and loss of heat removal to the ultimate heat sink is determined by the ratio of decay heat power to primary and secondary water inventory. In the first stage of the accident, heat is removed from the primary circuit through evaporation of SG water and stable natural circulation in the primary circuit. Operator actions according to EOPs, aimed at SG makeup using pressure drop between SG and deaerator induced by elevation difference, provide additional time and substantially extend the duration of the first stage.

The second stage of the accident starts after exhaustion of SGs and deaerators and termination of primary-to-secondary heat transfer due to decrease of the SG water level below the tube bundle. Power generated in the reactor core is removed through evaporation and discharge of the primary coolant through pressurizer safety relief valves. The analysis of time available to core melting is presented below for WWER-1000/V-320 (Zaporizhzhya-5).

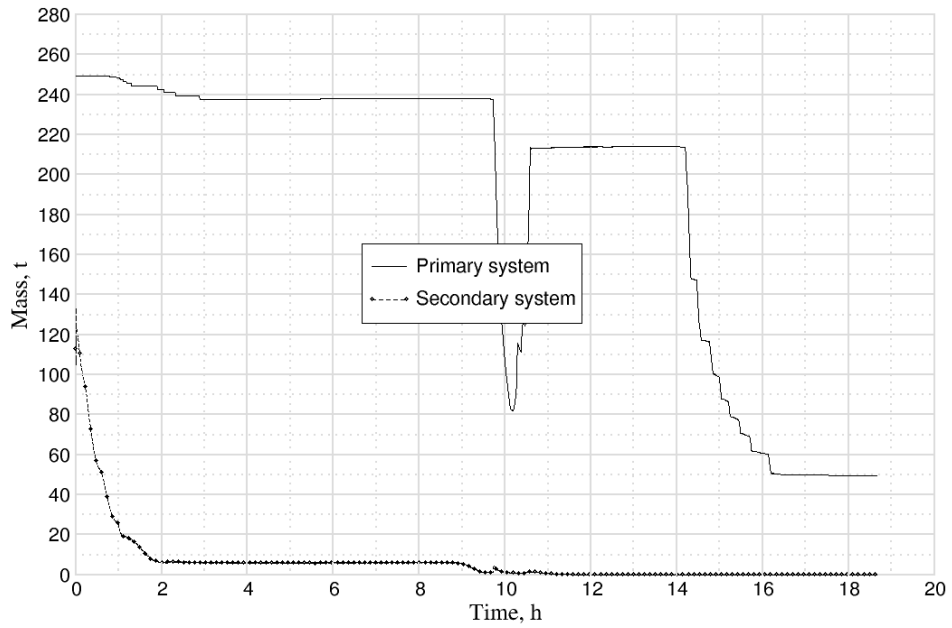
Stage 1				Stage 2			
Decay heat power %/MW	SG water inventory / deaerator water inventory, t	Time, h	Flow rate, t/h	Decay heat power %/MW	Primary coolant inventory /tank water inventory, t	Time, h	Flow rate, t/h
1/30.0	120/ 334	8.3	54.5	0.75/ 22.5	238/ 167	10.3	34.5

This table demonstrates the following: if it is not possible to restore operation of the safety/safety related systems for a long-term period, the emergency strategies are limited to using independent mobile sources for:

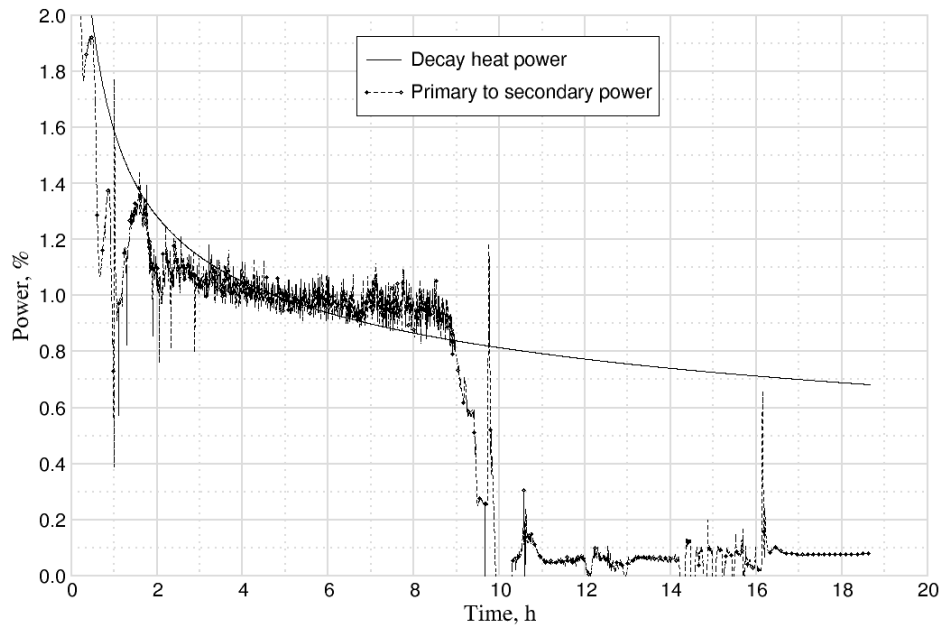
- makeup of the secondary circuit during the first stage (with minimum flow rate 54.5t/h)
- makeup of the primary circuit with boric acid solution during the second stage (with minimum flow rate 34.5t/h).

More detailed analysis of emergency strategies data is provided in Section 5.3.3.

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Primary Coolant and SG Mass Inventory in Case of Total Station Blackout and Loss of Heat Removal to Ultimate Heat Sink



Decay Heat Power and Power Transferred from Primary to Secondary Circuit in Case of Total Station Blackout and Loss of Heat Removal to Ultimate Heat Sink

Based on the calculation analyses, the time margin to severe core damage was determined as follows:

- for WWER-1000/V-320 – more than 18 hours;
- for WWER-1000/V-302, WWER-1000/V-338 – 12 hours;
- for Rivne NPP units 1 and 2 (WWER-440/V-213) – about 10 hours.

It should be noted that the available time margin for WWER-440/V-213 was calculated very conservatively, since operation of AEFS was not taken into account. Operation of AEFS

can provide long-term decay heat removal from the reactor core (more than two days without tank refilling).

Results of the calculation analysis for station blackout and loss of the ultimate heat sink for shutdown reactor states demonstrated that in case of the opened reactor, the operator should start actions aimed at discharge of hydroaccumulators. In this case, the time available to severe reactor core damage will be 8-10 hours. Further actions of the operator imply alternative organizational and technical measures relating to the use of mobile water and power supply sources.

The following cliff edge effects were identified upon the calculation analysis:

*For WWER-1000/V-320 reactors:*

- 1 hour – conservatively estimated time for discharge of batteries (the operator has no possibility to monitor reactor parameters and use PRZ PORVs and secondary steam dump devices to reduce pressure);
- 9 hours – time to decrease in the coolant level in the reactor below the hot legs (secondary makeup becomes inefficient due to loss of natural circulation in the reactor coolant system (RCS) loops, end of the first stage);
- 16.5 hours – the latest possibility for the operator to intervene (irreversible heatup of the reactor core starts and leads to severe core damage).

*For WWER-440/V-213 reactors:*

- 1 hour – conservatively estimated time for discharge of batteries (the operator has no possibility to monitor reactor parameters and use PRZ PORVs and secondary steam dump devices to reduce pressure);
- 7.5 hours (analysis with a conservative approach with no credit taken for operator actions and operation of AEFS) – time to decrease in the coolant level in the reactor below the hot legs (secondary makeup becomes inefficient due to loss of natural circulation in the RCS loops, end of the first stage);
- 9 hours (analysis with a conservative approach with no credit taken for operator actions and operation of AEFS) – the latest possibility for the operator actions (irreversible heatup of the reactor core starts and leads to severe core damage).

*For SFP of Ukrainian NPPs:*

- the time margin to fuel heatup above the design limits established for the most unfavorable conditions, with the reactor core unloaded to SFP, constitutes about 6.5-7 hours.

### **5.3.2. External Actions Foreseen to Prevent Fuel Degradation**

Calculation analyses of station blackout and loss of the ultimate heat sink with operator actions for Ukrainian WWER-1000 demonstrated that core damage can be avoided in case of makeup of one SG using a MDGPU having the flow rate of at least 13 kg/sec with pump head of 7 kgf/cm<sup>2</sup>, which is sufficient for continuous decay heat removal. This strategy can be implemented even at the late stage of accident management.

In case of total station blackout, SG makeup can be (heat removal by secondary circuit) performed using a ATs-40/4 fire truck having the following characteristics:

- nominal flow rate at suction height of 3.5 m - 15 L/sec;
- pump head – 0.9 MPa (9 kgf/cm<sup>2</sup>);
- water inventory in the tank – at least 4500 L (or 5800 L);
- load-bearing capacity – 7000 kg (or 10000 kg);
- base unit chassis is 6×6 (or 6×4).

Similar makeup requirements are determined for SFP at the required flow of 6 kg/sec with pressure of 4 kgf/cm<sup>2</sup>.

The primary “feed and bleed” procedure and primary system makeup with boric acid solution using another MDGPU having the flow rate more than 13 kg/sec at pressure of 20 kgf/cm<sup>2</sup> can be a possible way of emergency makeup and cooling. Boric acid solution with

the minimum concentration of 16g/kg must be injected into the primary circuit to prevent recriticality. This can be possible even for water with 0°C temperature taking into account the solubility of boric acid.

Borated water can be injected into the primary circuit using inventory from the primary coolant purification system.

To prevent severe core damage in WWER-1000 units, the required water inventory for MDGPU for 3 days is 3370 t per unit; this inventory must be available on-site or delivered to the MDGPU during an accident with required frequency.

The calculation analysis of the total station blackout and loss of ultimate heat sink with operator actions for Rivne NPP-1, 2 (WWER-440) demonstrated that core damage can be avoided in case of makeup of one SG using MDGPU with the flow rate of at least 4 kg/sec at pressure of 4 kgf/cm<sup>2</sup>. This flow rate is sufficient for continuous decay heat removal.

The primary “feed and bleed” procedure and supply of boric acid solution using another MDGPU with the flow rate of at least 4 kg/sec with pump head of 25 kgf/cm<sup>2</sup> can be a possible way of emergency makeup and cooling for WWER-440 primary circuit.

Similar requirements are determined for the SFP, the required flow rate is 3 kg/sec with pressure of 3 kgf/cm<sup>2</sup>.

To prevent severe core damage in WWER-440 units, the required water inventory for MDGPU for 3 days is 1037 t per unit; this inventory must be available on-site or delivered to the MDGPU during an accident with a required frequency.

At reactor cooldown using MDGPUs, it is very important to monitor temperature in the cold legs and primary pressure to avoid reactor vessel cold overpressure.

The following parameters should be monitored in accident conditions with assumed power supply from the MDGPU:

- reactor power;
- reactor period;
- coolant temperature at fuel assembly outlet or under the reactor cover;
- coolant temperature in hot legs;
- coolant temperature in cold legs;
- primary coolant pressure;
- SG secondary pressure;
- reactor coolant level;
- pressurizer coolant level;
- SG water level;
- pressure in containment compartments;
- hydrogen concentration in containment compartments;
- radioactivity in steam lines.

There is an alternative possibility to monitor pressure and temperature using manometer readings and infrared scanners.

The expediency of borated water storage for primary system makeup or its delivery and treatment on the NPP site with a required frequency shall be analyzed additionally to select the best option.

### **5.3.3. Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Primary Ultimate Heat Sink, Combined with Station Blackout**

Proposals on possible accident management in case of total station blackout and/or loss of the ultimate heat sink to provide primary and secondary makeup within a long-term period have been developed in the framework of stress tests. Specific calculation analyses were performed to determine necessary water inventory and equipment configuration, and proposals on equipment layout were developed. To ensure decay heat removal it is recommended to implement the following measures:

- increase capacity of batteries;
- ensure SG emergency makeup:
  - water injection into SGs from fire trucks located at the NPP site;
  - restoration of power supply to stationary makeup pumps from a mobile diesel generator;
  - water injection into SGs from MDGPU;
  - water injection into SGs from available stationary pumps of different systems, which can potentially be used;
- ensure emergency makeup of the primary circuit:
  - borated water injection into the primary circuit from MDGPU;
  - restoration of power supply to stationary makeup pumps from a mobile diesel generator;
- ensure SFP makeup and cooldown:
  - restoration of power supply to regular SFP makeup and cooling pumps;
  - water injection into the SFP from independent MDGPU or from the fire extinguishing system;
  - possibility of SFP passive heat removal.

Depending on the accident progression, the following facilities can be used to feed the primary and secondary sides and spent fuel pool:

- primary purification system and water treatment facility water storage tanks;
- distillate, boron concentrate and borated water tanks;
- water inventory in cooling towers of the service water supply system (V-302/338);
- water of the fire extinguishing system;
- spray ponds of group A (reactor hall) of the service water supply system;
- artesian wells.

## **5.4 Loss of Electrical Power and Loss of Ultimate Heat Sink for Dry Spent Fuel Storage Facility (DSF)**

### **5.4.1 Loss of Electrical Power**

The DSF is a passive system. Electrically-driven mechanisms are used in the DSF system only during loading of WWER ventilated storage casks at the Zaporizhzhya NPP. All other operations, including transportation of loaded WWER storage casks to the DSF site, as well as their storage on the DSF site, do not require electricity supply.

Consequently, loss of off-site electrical power will not result in loss of heat removal on the DSF site. In the framework of stress tests, special calculation analysis was carried out for the loss of off-site power supply during in-unit transportation and handling operations.

The blackout may lead to termination of handling operations at any time with spent fuel being inside the unit. Preliminary analysis revealed that the worst condition, in terms of heat exchange in multi-assembly sealed basket (MSB), is its location in an uncovered transfer cask (i.e. filled with air). In that case, additional thermal resistance decreases heat exchange rate at the boundaries of MSB. Under maintenance in normal conditions, the time of MSB remaining in this position is restricted. However, in case of blackout, this time may extend up to several days.

Conservative initial and boundary conditions have been selected for analysis. The maximum residual power and minimum conductivity of cooling air in the model are selected as initial conditions. The conservative heat transfer coefficient corresponding to conditions on the outer surface of a concrete cask is assumed. The radiation heat transfer inside the fuel assemblies in the MSB and close to the MSB cover is not taken into account, which allows increasing the energy stored in the MSB. The cooling agent is air under atmospheric pressure. Air temperature is assumed to be 40°C for normal conditions and 150°C for accident conditions in the containment (maximum design limit). The flow of the cooling agent is absent. The calculation was performed up to the moment of reaching the steady temperature condition inside MSB.

The analysis confirmed that the steady temperature condition inside the basket is reached in 2 days; it means that that maximum design limit of fuel assembly temperature is not reached.

Calculation results confirm safety of DSF in case of blackout.

#### 5.4.2 Loss of Ultimate Heat Sink

Heat is removed from spent fuel assemblies through natural circulation of environmental air via air ducts of WWER storage casks.

Loss of heat sink may occur if all inlet holes of air ducts are clogged. Since a ventilated storage cask has four independent inlets of air ducts located on two opposite sites, clogging of all inlets with debris, etc., is considered an unlikely case.

Despite this fact, in the framework of stress tests, hypothetical scenarios involving clogging of all inlets of air ducts of WWER storage casks were analyzed with various initial assumptions.

The results of the calculations are summarized in the table below.

Conditions	Environment temperature	Temperatures (sustained), °C					
		Sun heating consideration	Air outlet	Outer concrete surface	Inner concrete surface	Storage basket	Fuel cladding, max
Allowable values	-	-	-	66	107	-	350
Normal storage conditions, steady state	24	No	73	30	84	131	346
Extremely low temperature, steady state	-40	No	-1	-35	7	69	297
Limiting values	-	-	-	93	177	-	450
Maximum thermal load of transients	52	Yes	106	73	120	158	369
Extremely high temperature, steady state	38	Yes	89	59	103	145	358
Half of inlets closed	24	No	78	31	88	133	348
All inlets closed	24	No	104	32	109	149	361
Basket in a transfer cask, helium	24	-	-	-	-	203	407

Based on the calculations for all selected extreme and emergency values of environmental temperatures, the temperature of system components remains within the range permitted for fuel storage.

Calculation results confirm DSF safety in case of loss of the ultimate heat sink.

#### 5.4.3 Combination of Blackout and Loss of Ultimate Heat Sink

A combination of blackout and loss of the ultimate heat sink due to design features and passive principle of DSF performance represents no additional negative consequences as compared with individual losses of safety functions discussed in Sections 5.4.1 and 5.4.2.

## 6. SEVERE ACCIDENT MANAGEMENT

### 6.1 Organization and Arrangements of the Licensee to Manage Accidents

#### 6.1.1 Organization of the Licensee to Manage the Accident

##### *Staffing and shift management in normal operation*

The organizational structure of the Ukrainian nuclear operator (NNEGC Energoatom) and its separated entities – Rivne, Khmel'nitsky, Zaporizhzhya and South Ukraine NPP (henceforth - NPPs) - is determined in accordance with the "Provisions on Organizational Structure of NNEGC Energoatom".

Shift operating personnel is headed by the Plant Shift Supervisor; his main task is to ensure safe operation of NPP equipment and satisfy requirements for its safety. The Plant Shift Supervisor is obliged to:

- evaluate the nature and scope of any unforeseen deviations of NPP unit parameters and operating modes including those which can lead to radioactivity releases;
- take initial measures to minimize consequences of radioactive releases;
- ensure personnel safety and take measures to implement NPP emergency plans.

Operating personnel at each power unit is headed by the Unit Shift Supervisor; his responsibilities include management of the MCR personnel and management of the NPP power unit in accordance with the Technical Specifications and procedures.

Operating personnel of the NPP departments manage technological process in accordance with the limits and conditions set by the operating procedures; they conduct scheduled activities on maintaining reliability of the systems and components, perform inspections, tests, outages for maintenance, monitor process parameters of the operating equipment, detect deviations and malfunctions and respond to them in accordance with operating procedures.

Personnel with necessary qualification and skills are available for all operational modes, including normal operation, anticipated operational occurrences and accidents at nuclear power plants.

##### *Measures taken to enable optimum intervention by personnel*

Optimum personnel intervention during accident management is achieved by the following:

- taking into account the human factor in the NPP design;
- provision of operating and emergency procedures;
- professional staffing with highly qualified personnel;
- conducting emergency exercises.

##### *Use of off-site technical support for accident management*

Off-site technical support during accident management is ensured under the Emergency Preparedness and Response System of the NNEGC Energoatom for accidents and emergencies at Ukrainian nuclear power plants. The Emergency Preparedness and Response System includes a series of technical means and resources as well as organizational, technical and health and safety measures implemented by the operating organization to prevent or decrease radiation impact on personnel, the public and the environment in case of a nuclear or radiation accident at the NPP.

The Emergency Preparedness and Response System is included into the National State System for civil protection of the public and territories and has two interrelated levels: level of the NNEGC Energoatom management and NPP level. The procedures for actions in case of an emergency are specified in the Emergency Response Plan of the NNEGC Energoatom management and NPP Emergency Response Plans.

The Emergency Response Plan of the NNEGC Energoatom management envisages that NNEGC Energoatom, if necessary, provides support to the NPP at which the accident has occurred, including mobilization of means and resources of other nuclear power plants, Emergency Technical Center (ETC) and AtomRemontService (ARS). The total number of the emergency organizational structures of the NNEGC Energoatom is:

- KhNPP - 607 persons;
- ZNPP - 1195 persons;
- SUNPP - 790 persons;
- RNPP - 729 persons;
- ETC - 220 persons;
- ARS -195 persons.

If necessary, nuclear power plants organize additional shifts for emergency groups and brigades of the emergency NPP. The other nuclear power plants provide the emergency NPP with all the necessary technical means from their emergency sets and stocks, including spares, tools, materials and specialized vehicles.

If resources of the operating organization are not sufficient, the NNEGC Energoatom management has opportunities to involve the following additional resources:

- Ministry of Energy and Coal Industry: mobilization of resources to support the NPP, preparation of a request to provide resources from the state reserve;
- Ministry of Emergencies: use of specialized emergency and rescue teams to respond to an emergency; information exchange with interdepartmental headquarters and the Government commission for emergency mitigation; preparation of a request for support in accordance with the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency;
- Ministry of Internal Affairs: protection of material values, organization of checkpoints to limit the access of vehicles and public to the controlled area and contaminated territories; organization of free traffic of transport with evacuees and transport of forces and means for accident mitigation to the NPP; protection of public order;
- Ministry of Defense: support in decontamination of the contaminated territory and transport, deliver of forces, means and material resources to the accident region by means of military aviation;
- Ministry of Public Health: special and qualified medical assistance to injured emergency personnel in medical institutions;
- Security Service: mitigation of the accident consequences in conditions of a threat or actual terrorist acts.

Each NPP has dedicated services of the Ministry of Emergencies, Ministry of Internal Affairs, and Ministry of Public Health which arrive at the request of the NPP management for support and act at the plant site and controlled area.

#### *Dependence on the functions of other reactors on the same site*

Emergency procedures for NPP power units envisage the possibility to supply power to in-house loads of the affected power unit (including the battery recharge system) from one of the plant units in operation, automatically switched to in-house power supply mode or powered from diesel generators of any safety train of any plant power unit (connection through backup power lines).

#### *Procedures*

To mitigate accidents, operating personnel uses emergency operating procedures. Emergency operating procedures specify personnel actions for safety assurance during normal operation, design basis and beyond design basis accidents to prevent core damage.

Emergency operating procedures and operating instructions envisage personnel activities during accidents in the spent fuel pool and Zaporizhzhya NPP dry spent fuel storage facility (DSF). Additional measures on further enhancement of the emergency operating procedures



for the spent fuel pool and DSF are planned to extend covering the beyond design basis accidents.

Symptom-based emergency procedures have been recently developed for all Ukrainian nuclear power units.

At present, the emergency procedures at all Ukrainian nuclear power units include:

1. Procedures for elimination of anticipated operational occurrences. These procedures are intended for safe elimination of anticipated operational occurrences which have not led to the conditions of scram and/or setpoints of safety system actuation.

Elimination of operational occurrences involves implementation of activities aimed at the following:

- transfer of the reactor into safe state;
- exclusion of loss of control over the reactor neutron power and reactor period;
- termination of occurrences;
- exclusion of impact on personnel and the public;
- exclusion of damage to equipment not affected by the operational occurrence;
- confinement of possible radioactive releases;
- reliable decay heat removal.

Use of the procedures for elimination of anticipated operational occurrences results in one of the following:

- reactor unloading;
- stabilization of the reactor parameters and transition to the normal operating procedures;
- transition to emergency operating procedures.

2. Emergency operating procedures (EOPs). These procedures are intended for efficient mitigation of emergencies and accidents and prevention of severe accidents.

EOPs are divided into diagnostic actions and actions aimed at the optimum restoration to allow MCR operators to take efficient actions required in each particular situation after the malfunction is diagnosed.

The accident management process begins with diagnostics of the power unit. The diagnostics process includes the following main stages:

- check actuation of automated systems in case of emergencies;
- take actions to stabilize the primary and secondary parameters and transfer the reactor into a safe state;
- identify procedures for optimal restoration or procedures for restoration of critical safety functions.

Critical safety functions are determined for each of the safety barriers to be constantly maintained to preserve the integrity of the corresponding barrier. Conditions which ensure integrity of the safety barriers are controlled on the basis of critical safety functions.

The following priorities (hierarchy) are set among the critical safety functions in EOPs:

- subcriticality;
- reactor core cooling;
- heat removal;
- integrity;
- containment.

When the critical safety functions are maintained, there is no threat to safety barriers.

EOPs envisage that mitigation of emergencies at one power unit is managed by the Unit Shift Supervisor; if an emergency occur on two or more power units at one plant site, the management is the responsibility of the Plant Shift Supervisor.

The EOPs specify personnel actions during the operational modes: hot shutdown, hot zero power, operation at rated power. At present, under the “Comprehensive (Integrated)

Safety Improvement Program for Ukrainian NPPs” (C(I)SIP), EOPs are developed for shutdown states (measures No. 19203, 29203, 39203 “Improvement of the emergency operating procedures for reduced power and shutdown states”).

### 3. Severe accident management guidelines (SAMG)

Under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”, severe accident management guidelines are being developed. The development of SAMG follows the branch “Program for Analysis of Severe Accidents and Development of SAMG” PM-D.0.41.491-09.

The main objectives of severe accident management are the following:

- termination of core damage at early stages;
- maintenance of containment functions for the longest possible period;
- minimization of consequences of radioactive releases both on-site and off-site.

#### *Training and exercises*

All the operating personnel of the nuclear power plants are obligatorily trained in accordance with regulations and standards. The training course comprises the following stages:

- professional staffing, psycho-physiological and medical examination, obtaining permit to work at nuclear facilities;
- entry exam to check knowledge level;
- professional training in accordance with individual programs;
- check of knowledge and obtaining of all the required licenses;
- permit for work without supervision and further scheduled retraining including mandatory emergency and fire-prevention exercises.

All Ukrainian nuclear power plants have training centers equipped with training means including full-scope simulators (FSS) required for training and skill improvement of the personnel.

The MCR personnel are trained on the full-scope simulator to acquire the following capabilities and skills:

- equipment control in normal operation, anticipation operational occurrences and accidents;
- organization of work of subordinated personnel;
- interaction with personnel of the MCR and other plant departments.

The list of subjects for FSS exercises also covers accidents that can be caused by external hazards.

Emergency training tasks are as follows:

- systematic check of personnel’s ability and preparedness for efficient actions in emergencies;
- training of skills and abilities to act in accordance with procedures;
- training of personnel to prevent emergencies and mitigate accident consequences;
- training of personnel to interact with shift staff, firemen, medical staff, etc.

Every three years nuclear power plants conduct a common-plant emergency exercise involving departments of the emergency response system at the NNEGC Energoatom management level. The SNRIU also participates in such exercises.

In the frames of short-term activities under the “Action plan for targeted safety assessment and further safety improvement of Ukrainian NPPs in the light of the events at Fukushima-1”, the following emergency exercises were conducted additionally:

- 25-26 May 2011, Zaporizhzhya NPP: «General emergency at Zaporizhzhya NPP caused by extraordinary geological and hydrological conditions in the region of Zaporizhzhya NPP location, which led to failure of reactor heat removal»;
- 29 September 2011, Khmelnytsky NPP: «General emergency at Khmelnytsky NPP caused by extraordinary geological and hydrological conditions in the region of Khmelnytsky

NPP location, which led to failure of the NPP main and auxiliary power supply and to failure of reactor heat removal»;

– 10-11 November 2011, Rivne NPP: «General emergency at Rivne NPP caused by extraordinary geological and hydrological conditions in the region of Rivne NPP location, which led to failure of the NPP main and auxiliary power supply and, consequently, to failure of reactor heat removal».

– 24 November 2011, South Ukraine NPP: “Station blackout”.

Emergency training at NPPs was organized at high level. Personnel of the NPPs and involved organizations showed their preparedness to act in beyond design basis accidents and to transfer and maintain the reactor in the safe and controlled condition. Based on the exercises, some measures were developed for further enhancement of the emergency preparedness and response system.

Plans for improvement of plant site accident management are provided in Section 6.1.4.

### **6.1.2 Possibility to Use Existing Equipment**

#### *Provisions to use mobile devices*

Under the safety improvement programs, measures were implemented to ensure the emergency inventory of feedwater and emergency SG makeup in accident conditions, using fire hoses and mobile pumps (fire trucks), at Zaporizhzhya NPP units 1-6, South Ukraine NPP units 1-3, Khmelnytsky NPP units 1 and 2 and Rivne NPP unit 3.

The time period necessary to start water supply to the SG from a mobile source is about 1 hour; corresponding exercises of the operating staff of the reactor departments and fire protection shifts are conducted.

To ensure access of the mobile facilities to the connection places in case of possible destructions at the NPP site due to external hazards, appropriate mechanisms are foreseen for clearing access roads.

#### *Provisions for and management of supplies*

The reliable power supply systems at all Ukrainian NPP sites envisage reserves of fuel for seven days of work for each emergency diesel generator; the time of design unattended work for each diesel generator is 250 hours.

#### *Management and limitation of radioactive releases*

Accident progression with possible radioactive releases can be divided into the following stages.

The first stage begins with PORV actuation in response to accidents conditions, resulting in rupture of the relief tank diaphragm, or LOCA accident. The primary boundaries are affected and radioactive coolant gets into the containment. Due to pressure increase in the containment, radioactive coolant vapors may release into the environment in minor amounts limited by the design containment leakiness.

The main strategies of radioactive release management at this stage are primary circuit cooldown via the secondary circuit to decrease the primary parameters lower than setpoints for PORV actuation and measures aimed at pressure decrease in the containment for condensation of radioactive vapors. To implement these strategies, it is necessary to restore functions of the secondary makeup systems and the containment spray system.

The second stage begins after the level in the RPV has decreased and uncovered the core, leading to increase of fuel cladding temperature. Gas leak defects may emerge and fission products may release from fuel cladding into the primary circuit and containment. The maximum activity is limited by the coolant activity and the fuel assembly gas gap; release is limited by the design containment leakiness. The main strategies of radioactive release management at this stage are organization of the primary makeup and measures to decrease the pressure and temperature in the containment.

The third stage begins after the fuel melting temperature is reached. The maximum radioactive amount includes the activity of the primary coolant, gas gap and fuel. Release is limited by the design containment leakiness. The main strategies of radioactive release management are makeup and cooldown of the primary circuit and pressure and temperature decrease in the containment. During the accident, the greatest importance is placed to the recovery of cooling functions of the containment using the containment spray system. The spray system allows pressure decrease in the containment and condensation of a significant part of radioactive aerosols. Pressure decrease in the containment leads to decrease in the release of non-condensed gases.

The next stage is reaching of the design pressure limits and consequent failure of the containment. The strategy of radioactive release management is cooling of the containment using spray system. The objective is to decrease pressure and condense radioactive vapors inside the containment. Additionally, there are plans to implement measures for filtered pressure relief from the containment to avoid containment failure due to overpressure.

It can be concluded that in case of an accident, it is possible to use the primary and secondary makeup systems and the containment spray system. In addition, measures identified upon the stress tests for accident management are to be implemented.

### **6.1.3 Evaluation of Factors That May Impede Accident Management and Respective Contingencies**

*Extensive destruction of infrastructure or flooding around the installation that hinders access to the site*

There is an extensive infrastructure of access roads for different transport – automobile, railway, water transport (for Zaporizhzhya NPP) – for the NPP sites.

Major damage of these infrastructures can potentially occur as a result of an earthquake, tornado or flooding:

- earthquake (to 0.1g) poses no actual danger to roads to the extent that will make impossible transport operations;
- tornado may impede transport communication only locally due to blockage of routes that can be eliminated within short periods of time;
- flooding will not lead to serious impact on transport communication.

Possible damage to separate buildings and structures designed in accordance with general construction standards is conservatively assumed in stress tests. This assumption does not influence the unit functioning during an emergency but can lead to some difficulties in the organization of emergency response actions, including those connected with access to the site for personnel, mobile pumps and electrical generators (in case they are located off-site).

In the framework of the stress tests, plant walkdowns were performed at all the operating nuclear power plants. Based on their results, no potential for blockage of access roads to the sites due to destruction of neighboring buildings and structures was identified. It should be noted that to eliminate blockage, appropriate vehicles and mechanisms are envisaged.

*Communication systems (internal and external), loss of communication facilities/ systems*

Communication systems ensure redundant communication among the emergency centers of the nuclear power plants, workplaces of the Plant Shift Supervisor, MCR, ECR, radiation protection control boards. The format of information exchange among the emergency response participants, the procedure for preparation, registration, transmission and receiving of information, and the procedure for information exchange between the NPP emergency centers and the operating organization's emergency centers and other interested parties participating in emergency response are specified in plant procedures for information exchange among the emergency centers.

Connections of the dedicated telephone channels used for communication are duplicated by means of commuted telephone channels or other means of communication. To ensure

reliable connection among the emergency centers and emergency groups and brigades of the nuclear power plants, mobile radio communication on the basis of VHF connection is envisaged.

Equipment of the connection systems installed in the reactor and turbine halls has a seismic resistance category corresponding to regulations; it will be operable during design basis earthquakes.

In terms of possible flooding, the connection wells are protected from water ingress through roofs and have two leads each. Lower lids of the wells fit tight to the hatchways. The drainage system is in operable state.

*Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site*

In planning of emergency measures, potential high dose and radioactive contamination at the site are taken into account.

To ensure limitation of doses for personnel, a series of technical and organizational activities is implemented. The main activities on personnel protection are as follows:

- administrative management and supervision over radiation health and safety regulations specified in radiation safety standards to prevent their exceeding and to limit personnel exposure;
- determination of personnel protection modes; timely introduction of protective countermeasures;
- radiological surveys in NPP compartments and on-site;
- dose monitoring;
- provision of emergency personnel with prophylactic radio protective means (radio protectors and radioresistance means);
- personnel sheltering and evacuation;
- use of individual protection means.

Modes of radioactive protection of personnel are included in the emergency plan of each nuclear power plant. The emergency response system has a radiation monitoring group. Its main tasks are the following:

- evaluation of radiation conditions in the compartments, at the plant site, controlled area and observation area using of types of radiation monitoring data (monitoring of safety barriers, radioactive process monitoring, dose monitoring, radiation monitoring of the environment, radiation monitoring to prevent spread of contamination);
- evaluation of postulated or actual releases or discharges;
- evaluation of pathways of radioactive releases into the environment;
- prediction of radiation conditions for the compartments, plant site, controlled area and observation area taking into account forecasted meteorological conditions;
- evaluation and estimation of doses for personnel and the public; preparation of recommendations for protection of personnel and the public;
- determination and specification of locations of dose monitoring points;
- evaluation of optimum routes for personnel movements and evacuation.

In case of an emergency, additional dose monitoring of personnel is implemented and includes:

- monitoring of the working environment – radiation health and safety parameters – at workplaces, in compartments, on the plant site, in controlled area and observation area;
- emergency individual dose monitoring of personnel;
- system of timely planning and accounting of individual doses of personnel.

Monitoring of the working environment includes the following:

- monitoring of the gamma dose and beta particle flux density;
- monitoring of the neutron flux density and/or equivalent dose rate;

- monitoring of the activity concentration and nuclide composition of inert radioactive gases and aerosols in the air of compartments and in releases from the ventilation systems;
- monitoring of the radionuclide contamination levels of compartment surfaces, equipment, personnel bodies, personal and working clothes of personnel;
- monitoring of the contamination levels of transport, equipment, devices and materials transported beyond the restricted access area, buildings and site;
- monitoring of collection, removal and decontamination of solid and liquid radwaste;
- monitoring of the radionuclide contamination of working clothes and individual protection means delivered for sanitary treatment;
- monitoring of the contamination levels of facilities on the plant site.

For radiation monitoring, the nuclear power plants envisage stationary and mobile inspection devices. The composition of radiation monitoring instrumentation is provided in the “Radioactive Monitoring Procedures during Operation of NPP Facilities».

*Impact on the accessibility and habitability of the main and secondary control rooms, measures to be taken to avoid or manage this situation*

In accordance with regulatory requirements, the design envisages means to ensure accessibility and habitability of the MCR and ECR during emergencies and accidents.

The following requirements are set for the MCR:

- provide means ensuring MCR habitability and accessibility during normal operation and accidents;
- exclude MCR common-cause failures;
- envisage an independent conditioning system for the MCR;
- envisage an intake ventilation system the MCR, with aerosol and iodine filters;
- in case of atmospheric air contamination, ensure operation of the conditioning system in a closed mode;
- design supporting ventilation systems (conditioning) for external temperatures +42°C;
- maintain the following air parameters at workplaces during warm seasons (ambient air temperature +31,2 °C): optimum air temperature +23±2 °C, relative humidity 40-60%; tolerable upper level of ambient air temperature in the compartment +28°C ...+31°C, relative humidity of 55% at the temperature of +28°C;
- maintain the following parameters in the compartments with I&C instrumentation and equipment:
  - ambient air temperature +23°C...+25°C and relative humidity ≤80% at temperature of +20°C
  - minimum air temperature at the entrance to the compartment not lower than +14°C;
  - limiting air temperature in accordance with the technical specification for I&C instrumentation and equipment not lower than +5°C and not higher than +50°C;
- maintain the specified temperature at the MCR in normal operation and during blackouts;
- create increased pressure in MCR for smoke protection, not lower than 20 Pa.

Regulatory requirements set for the ventilation and air conditioning system of the ECR are similar to those of the MCR specified above.

The capacity of the MCR and ECR ventilation and conditioning systems is accepted based on heat coming into the compartments from process equipment, artificial lighting, people and through protecting structures.

The air-conditioning system at the MCR is designed taking into account reliable maintaining of internal air parameters in the control room compartments in emergency modes of blackout, under seismic impacts, and hazardous parameters of external air and taking into account installed equipment. Autonomous conditioners powered from the reliable power grid are accepted as cooling sources.

Under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”, measures are being implemented/planned for further enhancement of the systems ensuring MCR and ECR habitability and accessibility.

*Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident*

The means of emergency response at the NPP level include:

- MCR and ECR;
- NPP technical support centers;
- on-site and off-site crisis centers;
- systems and means of announcement and communication;
- systems of NPP crisis center data collection, development, documentation, storage, display and transfer;
- NPP emergency set of measuring devices and equipment, individual protective means, decontamination and sanitary treatment means, tools and devices, special equipment, transport and other emergency and technical means (it is planned to extend the scope of material and engineering means with consideration of the events occurred at Fukushima-1).

When protective means and support systems of the on-site crisis center cannot provide conditions for people vital functions (habitability), the off-site crisis center shall be used.

Continuous emergency response management is ensured by means of providing the acceptability and habitability of at least one of the crisis centers, since the beginning of an emergency till mitigation of its consequences. The habitability of crisis centers is assurance of acceptable safe conditions for personnel, including the following: protection against radiation and chemical factors, fires and smoke formation, ventilation, air conditioning, heating and other required conditions.

*Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)*

In planning of activities for management of station blackout and loss of heat removal, possible destruction on site caused by external hazards is taken into account.

During NPP walkdowns, possible destructions have been identified on the NPP sites. These destructions may be along the routes of mobile equipment transportation and could impede potentially the mobile equipment delivery. For the clearance of possible NPP site blockage that impedes the mobile equipment delivery to the places of equipment connection, the location of necessary mechanisms within the area for mobile installations is envisaged.

The following specific requirements are set for location of mobile equipment:

1. Mobile diesel generators shall be installed in a seismic-resistant separate building off-site. This approach is based on the following:

- Common-cause (flooding, tornado damage, etc.) failure of mobile equipment is excluded.
- Location of mobile equipment off-site creates an additional way for its delivery (to a certain power unit).

2. Motor pumps shall be installed in a separate seismic-resistant building or directly near places of their connection, in order to provide their quick connection. The conditions for keeping the mobile equipment:

- preservation;
- minimum time for its delivery to the place of use;

- readiness for immediate use for the direct purpose:
  - the storage room shall be heated;
  - the mechanisms shall be filled up with combustible and oil materials;
  - for towed mechanisms – a special carrier or organizational measures shall be envisaged to ensure its continuous readiness for immediate arrival;
  - the mechanisms without chassis shall be installed on chassis or trailers for transportation and use of these mechanisms without any loading and unloading;
  - the mechanisms shall be provided with the required number of spare parts and intake and pressure piping fixed on chassis.

#### *Unavailability of power supply*

To eliminate consequences of accidents related to station blackout, it is required to restore/provide power supply to equipment of safety-related systems needed to perform safety functions. Each NPP has several possibilities for establishing power supply to the affected unit from different alternative sources as discussed in detail in Section 5. When restoration of power supply from regular or alternative sources is not possible, the use of mobile diesel generators is envisaged. The minimum required power of mobile diesel generators is determined based on minimum configuration of safe shutdown equipment.

For restoration of equipment of one of the safety system trains (ECCS pumps, pump of the essential service water supply system, ventilators of the cooling systems of safety system rooms, etc.), a diesel-generator up to 6kV/2.5 MW connected to group 2 reliable power supply buses is required (connectors are located in compartments of emergency diesel generators).

#### *Potential failures of instrumentation*

In the framework of equipment qualification, instrumentation and control systems are qualified for seismic impacts and harsh environments that can emerge during accidents. It allows ensuring the operability of measuring equipment in emergency conditions.

In the conditions of station blackout and complete discharge of batteries, the following is envisaged:

- primary pressure control with the use of a mobile manometer;
- containment pressure control with the use of a mobile manometer;
- temperature control in the reactor, main coolant piping with the use of a mobile millivoltmeter or resistance box on the racks of in-core monitoring system;
- level and temperature in spent fuel pools.

Currently, within the framework of the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”, measures on emergency and post-emergency monitoring of reactor parameters are carried out.

In addition, upon results of stress tests, measures to ensure the operability of instrumentation and control systems in the conditions of station blackout are carried out: each train of the emergency power supply system for I&C will be provided with a 300 V/200 kW diesel generator and a 250 kVA isolating transformer, 380/220 V.

#### *Potential effects from the other neighboring installations at site, including considerations of restricted availability of trained staff to deal with multi-unit, extended accidents*

There are the following significant potential effects from adjacent units:

- deterioration of radiation conditions on site in case of an emergency at several power units. The deterioration of on-site radiation safety shall be considered during planning of emergency activities (Section 6.1.3.);
- increase of the need for technical resources and manpower for accident management at several power units. The system of emergency preparedness envisaged the mobilization of additional resources of the operating organization (Section 6.1.1).



#### **6.1.4 Conclusions on the Adequacy of Organizational Issues for Accident Management**

Symptom-based emergency procedures were implemented at NPP units. Actions of personnel are aimed at prevention of severe core damage. Emergency plans have been introduced at NPPs and at the level of the operating organization.

An efficient accident management system was created, which includes the required administrative and technical measures, means and resources to prevent and limit accident consequences. External technical support is ensured in accident management at the branch and national levels to mobilize additional human and technical resources.

Measures for improvement of NPP accident management are provided in Section 6.1.5.

#### **6.1.5 Measures That Can Be Envisaged to Enhance Accident Management Capabilities**

The stress tests confirmed the importance of short-term completion of measures envisaged in Section 6 "Accident Management" of the current "Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs":

- improvement of emergency operating procedures for shutdown states;
- implementation of severe accident analysis and development of SAMG.

In addition, the stress tests determined the need to implement the following:

- develop measures for spent fuel pool severe accident management, together with respective analytical substantiations;
- implement measures based on results of emergency training including measures concerning the accessibility and habitability of MCR (ECR) and crisis center in severe accidents).

### **6.2 Accident Management Measures in Place at the Various Stages of a Scenario of Loss of the Core Cooling Function**

#### **6.2.1 Before Occurrence of Fuel Damage in the Reactor Pressure Vessel (Including Last Resorts to Prevent Fuel Damage)**

The measures on accident management to prevent severe core damage are determined in EOPs (Section 6.1.2).

The EOPs include the optimal restoration procedure ARZ-0.0 "Loss of AC power". This procedure determines actions in a case of loss of AC power for all 6 kV buses of safety systems (BV, BW, BX) as well as actions on reactor stabilization in blackout conditions. Personnel are required to make attempts to supply power to safety system buses from alternative sources.

The procedures VFT-1.1 "Loss of secondary heat removal" and VFZ-1.1 "Reactor core overheating" determine actions on restoration of reactor conditions in case of loss of heat removal in all SGs and actions on core cooling restoration.

If pressure in the containment increases to the values above the established limits, the actions of personnel are determined by the procedure VFG-1.1 "High pressure in containment".

The actions according to EOPs allow personnel to prevent or substantially extend the time available to severe core damage (if the severe core damage cannot be prevented).

If the actions according to the EOPs are not successful, accident management proceeds in compliance with emergency plans and severe accident management guidelines (SAMG).

#### **6.2.2 After Severe Core Damage before Failure of the Reactor Pressure Vessel**

Core degradation (melting) is one of the most significant effects of severe accidents. In case of loss of heat removal accidents, melting occurs as a consequence of the lack of coolant (heat sink is not available) in the reactor core. Loss of coolant can be a consequence

of an initiating event (primary leak) as well as the result of its evaporation (accident with integral primary circuit, including station blackout and/ or loss of ultimate heat sink).

There are two main phases of severe accident progression till failure of reactor vessel bottom: early phase (initial heating and melting of the reactor core structures) and late phase (formation of extensive blockage, origination and relocation of melts and reactor bottom heating).

At early accident stage, irrecoverable primary coolant loss and consequent core drying occur. Core uncovering leads to increase in fuel temperature (because fuel assembly cooling by steam is not sufficient to avoid core melting). As a result, the initial core deformation (swelling and failure of fuel assembly cladding) occurs. Besides, temperature increase in the reactor core leads to initiation of the steam-zirconium reaction. The heat generated in this reaction is comparable to the decay heat, and intensity of temperature increase for fuel assembly claddings after the beginning of zirconium oxidation rises substantially.

If temperature further increases, local melting of fuel cladding can occur in the reactor core. As a result, "drops" of molten material can form. When some of them flow into the lowest part of the reactor core, molten materials may harden and lead to blockage of open flow areas and clearances between fuel assembly claddings.

In order to prevent reactor vessel failure and melt flow into the containment as well as for mitigation of consequences of the severe accident at in-vessel phase, it is necessary to perform actions on accident management.

The SAMG that meet up-to-date knowledge of severe accident progression are currently under development within the framework of the "Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs" and in accordance with the industrial "Program on severe accident analysis and SAMG development".

Currently, major efforts on severe accident analysis and SAMG development have been completed:

Vulnerability analysis for all types of reactor plants that are in operation in Ukraine (WWER-1000/V320, WWER -1000/V302 и WWER -440/V-213), including:

- development of guidelines on vulnerability analysis and analytical substantiation for severe accident management strategies;
- determination and substantiation of a list of accidents with severe core damage, requiring a detailed analysis;
- analysis of the selected list of severe accidents;
- analysis of radiation consequences for the selected list of severe accidents;
- determination and preliminary substantiation of severe accident management strategies.

Analytical substantiation and SAMG development are carried out. These efforts include:

- development of a set of guidelines on the format and structure of SAMGs, verification and validation and development of general provisions on severe accident management;
- analytical substantiation of SAMG;
- technical substantiation of SAMG;
- analysis of radiation consequences of severe accidents with consideration of severe accident management strategy implementation;
- SAMG compiling and validation.

In the framework of the performed vulnerability analysis, severe accident progression has been analyzed and severe accident management strategies have been identified. The following strategies have been specified for accident management at the stage till reactor pressure vessel failure:

- primary coolant injection;
- primary pressure decrease;
- SG makeup;

- SG pressure decrease;
- external cooling of RPV.

### **Primary coolant injection**

The main objectives of this strategy are:

- heat removal from the reactor core (from debris) by means of long-term water supply into the primary circuit;
- prevention of reactor pressure vessel failure (or increase of time margin till reactor pressure vessel failure);
- reactor core flood with the purpose of fission product deposition/confining.

The configuration of equipment that can be used for implementation of the mentioned strategy depends on power supply availability. In case of power supply availability, the following ways of water supply into primary circuit are possible: HPIS pumps, LPIS pumps, makeup system pumps.

In case of station blackout, addition measures are needed to implement this strategy. Use of mobile equipment for primary coolant injection and melt cooling has to be justified.

### **Primary pressure decrease**

The main objectives of this strategy are:

- prevention of damage to SG tubes and reactor coolant system;
- prevention of corium flow from the reactor core under high pressure;
- flow rate reduction through the leak in case of containment bypass;
- conditions for LPIS operation.

The configuration of equipment that can be used for implementation of the mentioned strategy depends on power supply availability. In case of power supply availability (restoration), the primary pressure can be decreased by opening of PORV and/or valves of the emergency gas evacuation system. In case of blackout, the time for this strategy implementation will be limited by battery discharge.

### **SG makeup**

The main objectives of this strategy are:

- heat removal from the primary circuit;
- prevention of temperature-induced failures of SG tubes;
- retention of fission products that appear in steam generators via tube leakages.

The configuration of equipment that can be used for implementation of the mentioned strategy depends on power supply availability. If power supply is available (or restored), water supply to SG is possible from regular systems:

- auxiliary feedwater pump and/or emergency feedwater pump (ZNPP, KhNPP, SUNPP units and RNPP units 3-4);
- auxiliary feedwater pump, feedwater pump or auxiliary emergency power supply system (RNPP units 1-2).

In case of blackout or failure of regular systems of SG makeup, coolant supply to SG secondary circuit shall be accomplished from the following systems:

- passive feedwater supply from turbine hall deaerators (WWER-1000);
- pump of the condensate storage tank;
- fire pumps;
- fire engines.

Passive steam generator makeup means SG pressure decrease to the value below the pressure in feedwater line to discharge water from deaerator to SG. SG pressure can be reduced through BRU-A or SG safety valve, i.e. time margin for this strategy is limited by the capacity of batteries. Water supply to SG from the pump of the condensate storage tank or fire pumps is possible if SG pressure is close to the atmospheric pressure.

### ***SG pressure decrease***

The main objective of this strategy is to promote conditions for water supply to the SG. (This strategy is especially relevant if water supply to SG secondary circuit is envisaged from alternative sources, for example, fire pumps, deaerator). The configuration of equipment needed for implementation of this strategy depends on power supply availability. In case of blackout, secondary pressure can be decreased through BRU-A or SG safety valves.

In case of power supply availability, BRU-K can be used (if vacuum is in turbine condenser).

### ***External cooling of RPV***

The objective of this strategy is the organization of heat sink from molten fuel due to the cooling of the reactor bottom outer surface. There are plans to perform a detailed analysis of the possibility of external cooling of RPV and implementation of respective technical decisions. It will be necessary then to modify the reactor cavity and investigate the possibilities for steam removal from the reactor cavity with use of available equipment.

#### **6.2.3 After Failure of the Reactor Pressure Vessel**

After reactor vessel failure, the melt would relocate to the reactor cavity where the molten corium–concrete interaction (MCCI) begins. As a consequence of this interaction, the following failures may occur:

- loss of containment integrity as a result of static pressure increase in containment (containment overpressure);
- loss of containment integrity as a result of hydrogen detonation;
- containment bypass as a result of melting of ionizing chamber channels (WWER-1000/V-320);
- containment bypass as a result of the “failure” of penetrations located on the floor of room A201 (WWER-1000/V-302/338);
- melt-through of the sealed door and melt release outside the containment (WWER-440);
- containment failure as a result of melt-through of the containment lining and/or transport hatches at elevation 13.7 m after melt release outside the room GA301 (WWER-1000/V-320).

The main tasks of severe accident management after reactor pressure vessel failure are:

- decrease of containment parameters to limit radioactive release to atmosphere;
- radionuclide deposition from containment atmosphere;
- corium heat removal outside the reactor pressure vessel to terminate corium/concrete interaction;
- prevention of hydrogen detonation in the containment.

A detailed description of the strategies intended to maintain the containment integrity is provided below (Section 6.3).

### **6.3 Maintaining the Containment Integrity after Occurrence of Significant Fuel Damage**

#### **6.3.1 Elimination of Fuel Damage/Meltdown in High Pressure**

In the course of loss of heat removal, irrecoverable loss of primary coolant occurs (as a result of leak through the primary piping rupture or due to pressurizer PORV operation), and the core is consequently dried out. Reactor core uncovering leads to fuel temperature increase (because fuel cooling by steam is not sufficient to prevent the core from melting). This leads to initial core deformation (swelling and failure of fuel claddings) and severe damage (melting) of fuel assemblies. In case of loss-of-coolant accidents with integral primary circuit, core melting can occur under high pressure.

Further progression of a severe accident with core melting under high pressure can finally lead to corium release to the containment under high primary pressure, and so-called effect of direct containment heating (DCH) may occur. Very fast heating of the steam and gas mixture in the containment is governed by a complex effect of several processes (primary coolant release, intensive heat transfer from melt fragments, exothermic reactions of metal oxidation and hydrogen burning) can result in temperature and pressure jump in the containment. If the pressure is above the containment design limit for strength, containment failure can occur. Such a failure would occur immediately after reactor pressure vessel damage and core melting fragment spreading within the containment. This failure would lead to a significant radioactive release beyond the containment.

Although PSA-2 for Ukrainian NPPs showed that the contribution of the above-mentioned phenomenon to failure probability is minor, the reactor core damage under high pressure must be prevented because of a probable failure of the primary equipment as a result of high-temperature creep (so-called temperature-induced failure). In case of temperature-induced failures of SG tubes, the containment bypass and radioactive release to atmosphere can hypothetically occur.

Primary pressure can be decreased using pressurizer PORVs, emergency gas evacuation system and by means of secondary cooldown.

In the framework of stress tests, effectiveness of accident management actions according to EOPs before severe core damage was analyzed. The results of calculations have shown a significant increase in time margin before severe core damage and revealed no conditions for temperature-induced failure of SG tubes and/or corium release to the containment under high pressure.

### **6.3.2 Management of Hydrogen Risks inside the Containment**

Accumulation of inflammable gases in high concentrations during a severe accident with subsequent potential deflagration and detonation is the most specific contributor to containment damage for different plant types (PSA-2 for Ukrainian NPPs also confirmed the importance of this issue).

Significant amount of hydrogen and other inflammable non-condensable gases is generated and accumulated inside containment compartments during severe accidents. The amount of hydrogen accumulated during an accident is one of the main parameters that characterize the accident progression. Accumulation of significant amounts of inflammable gases in the containment is associated with danger of their combustion and explosion that may result in static and dynamic stresses that are higher than the containment design limit for strength or even the containment ultimate capacity and may lead to containment failure.

In the framework of the stress tests, extensive calculations were performed to investigate hydrogen danger, including:

- analyses to investigate issues of hydrogen generation and distribution inside the containment;
- analyses to investigate possibilities to reduce hydrogen concentration inside the containment.

Calculations for station blackout combined with loss of ultimate heat sink were carried out for the three types of reactors (WWER-1000/V320, WWER-1000/V302 and WWER-440/V-213).

In investigating the possibilities to reduce hydrogen concentration inside the containment, several calculation analyses were carried out and involved modeling of operation of passive autocatalytic hydrogen recombiners (PAR). The simulation of PARs as means of hydrogen concentration reduction in the containment is based on the fact that to date a considerable scope of studies in this area has been performed for Ukrainian NPPs. Based on these studies, a decision was made that PAR is the most appropriate applicable mean of hydrogen concentration reduction. Therefore, a pilot conceptual technical decision was developed for

implementation of the hydrogen removal system to ensure hydrogen explosion safety during beyond design basis accidents for WWER-1000/V-302, V-338.

The calculated amounts of hydrogen generated during the accident “blackout combined with loss of ultimate heat sink” for WWER-1000 vary from 2.5 to 3.0 t significantly depending on the composition of RPV concrete (assessed as 3.8 t for SUNPP unit 3) and for WWER-440 is about 2 t.

Generation of this considerable hydrogen amount results in high hydrogen concentrations accumulating in the containment (over 15%); however, in accordance with the Shapiro diagram, hydrogen detonation conditions are not reached because of high concentrations of steam (over 60%) and low oxygen concentration.

Considerable hydrogen amounts accumulate in the containment while the existing tools are not intended to cope with such amounts of hydrogen generated in severe accidents (passive autocatalytic hydrogen recombiners are installed at RNPP units 1,2, 3 and KhNPP unit 2 to reduce hydrogen concentration during design-basis accidents). Therefore, a potential hazard of detonation exists (for example, in case of steam condensation resulting from the recovery of spray system operability).

Under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs”, the following measures are ongoing to enhance hydrogen safety:

- implementation of measures on hydrogen concentration reduction in the containment during beyond design basis accidents;
- implementation of the hydrogen concentration monitoring system in the containment for beyond design basis accidents;
- implementation of the hydrogen concentration monitoring system in SG and MCP compartments (A201) and in pressurizer compartments (A527/1).

Conceptual technical decision was also developed for implementation of the hydrogen removal system ensuring hydrogen explosion safety during beyond design basis accidents for WWER-1000/V-302, V-338.

According to the technical decision, in order to reduce the hydrogen concentration in containment during beyond design basis accidents (including accidents with severe core damage), passive autocatalytic hydrogen recombiners are expected to be installed in the containment.

Therefore, taking into account the information stated above, it can be concluded that Ukrainian NPPs pay proper attention to hydrogen safety issues. These efforts are highly important and should be accelerated.

### **6.3.3 Prevention of Containment Overpressure**

The containment spray system is intended for prevention of containment failure due to overpressure at power units of Ukrainian NPPs. The spray system is used for:

- reduction of pressure inside the containment in case of primary and/or secondary piping break;
- binding of iodine isotopes in the containment in case of loss-of-coolant accident;
- emergency feeding of spent fuel pools (for WWER-1000/320).

The spray system consists of three functional groups. Each group has independent process communications, I&C equipment and interlocks and normal and emergency power sources. Each functional group of the spray system includes the following main equipment:

- spray pumps intended for boron solution supply;
- water-jet pump for potassium borate supply to spray pumps;
- emergency cooling heat-exchanger, that is shared between the spray system train and the low-pressure injection system train;
- emergency boron solution storage tank (for WWER-1000/V320 – compartment GA-201 inside the containment);
- spray nozzles.

In addition to the containment spray system, the containment of RNPP power units 1-2 (WWER-440/V213) is provided with passive steam condenser ("passive" spray) to limit peak pressure in compartments through condensation of steam generated during an accident and to bind airborne iodine and confine radioactive products.

The passive steam condenser consists of a number of trays and casings with tunneling caps that are installed as cascade along the building height. Trays are filled with boric acid solution (16 g/kg) up to the nominal level. The steam condenser tower is joined with SG box via two corridors.

Results of severe accident analyses performed previously in the framework of vulnerability assessments and stress tests revealed a potential for considerable pressure increase in the containment.

The acceptance criterion for pressure in the WWER-1000 containment is conservatively assumed as 5 kgf/cm<sup>2</sup> (abs) (0.49 MPa), that complies with design containment pressure limit. According to strength calculations performed in PSA-2, pressure value for containment failure is ~0.83 MPa.

According to calculations performed in framework of stress tests for WWER-1000/V302-338 units, the containment pressure reaches 5 kgf/cm<sup>2</sup> in ~41 hours after accident beginning. For WWER-1000/V-320 units, it occurs in ~27 hours. For WWER-440/V-213, the acceptance criterion is not exceeded. Since this acceptance criterion is conservative, the actual available time to containment failure is much longer and sufficient to implement the strategy for preventing containment failure.

Prior to completion of the stress tests, the SNRIU established requirements for implementation of measures on forced filtered pressure relief from the containment for WWER-1000 units (containment venting). The "Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs" was supplemented with additional measures, including:

- development and implementation of a technical decision on controlled filtered containment venting;
- development of a procedure for emergency containment venting in case of a severe accident, and its incorporation into SAMG.

The technical decision being developed should provide for:

- necessary calculations that prove effectiveness of the containment pressure decrease, effectiveness and feasibility of discharged steam-gas treatment, considering minimization of radioactive contamination of the environment;
- possibility to operate the system under station blackout.

A conceptual technical decision "Implementation of emergency containment venting at South Ukraine NPP units 1,2" has been developed.

Results of the calculation analyses performed in stress tests showed the possibility and effectiveness of this pressure discharge strategy.

#### **6.3.4 Prevention of Re-criticality**

The re-criticality in normal operation and accidents is prevented by means of:

- control rod insertion into the core;
- primary shutdown boron concentration.

Re-criticality may appear due to injection of non-borated water in a partially degraded core (reflooding). Accident management strategies of melt flooding (in- and ex-vessel) suggested for SAMG, and measures proposed based on stress tests results envisage supply to reactor of borated water only. The results of stress tests showed that the plants have inventories of borated water that may be used for implementation of these strategies.

Any new strategies or modifications associated with injection of non-borated water, for instance, as a latest measure to terminate the core degradation, should be subject to detailed analysis of the re-criticality issue. Such analyses should take into account

uncertainties in the knowledge about the core state at the beginning of reflooding as well as in the behavior of the fuel rods during a severe accident.

### 6.3.5 Prevention of Basemat Melt-through

According to calculations performed in the framework of the stress tests, the time to reactor pressure vessel failure for WWER-1000/V302-338 and WWER-1000/V-320 is ~23-24 hours. For WWER-440/V-213, additional efficient measures (additional emergency feedwater system, see Section 5) have been implemented to prevent (substantially extend time to) severe core damage and consequent reactor pressure vessel failure.

After reactor vessel failure, corium penetrates into the reactor cavity where it starts interacting with concrete (MCCI). This may lead to the following containment failures and pathways of fission product releases to the environment:

- containment bypass due to melt-through of ionizing chambers (for WWER-1000/320);
- containment bypass due to failure of penetrations located on the floor of A201 compartment (for SUNPP units 1-2);
- sealed door melt-through and melt penetration to compartments outside the containment (for RNPP units 1-2);
- containment failure due to melt-through of containment cladding and/or transport hatches on elevation 13.7 m after melt spread outside the GA301 compartment (for WWER-1000/320).

Results of PSA-2 for Ukrainian NPPs showed that the dominant types of containment failures at late stage are caused by melt-through of containment structures.

Based on the vulnerability analyses for severe accident conditions, a strategy of corium cooling in the reactor cavity was developed. The purposes of the strategy are:

- corium cooling in the reactor cavity;
- prevention/minimization of hydrogen generation and generation of other combustible gases due to MCCI;
- prevention of containment failure due to melt-through of the floor in the A301 compartment and corium ingress to the A201 compartment with subsequent melt-through of the floor in this compartment and containment integrity degradation (for WWER-1000/302(338));
- prevention of potential containment bypassing through ionization chamber trains (for WWER-1000/V320);
- prevention of sealed door failure and corium penetration to non-hermetic compartments (for WWER-440/B213);
- prevention of basemat melt-through.

Corium cooling is possible in case if coolant is supplied into the reactor cavity by ECCS pumps. Measures on corium ex-vessel cooling and retaining inside the containment are currently underway. One of the measures foreseen in the C(I)SIP is “Containment integrity preservation in case of its interaction with corium (core melt) during external stage of severe accident”.

### 6.3.6 Power Sources Necessary for Protecting Containment Integrity

Operability of the following equipment is to be ensured to preserve the containment integrity in accordance with developed severe accident management strategies:

- containment isolating valves;
- spray system;
- hydrogen concentration reduction system (if PARs are used, there is no need for electric power and compressed air supply);
- containment venting;
- reactor cavity melt cooling system.



In case of a station blackout accident, operability of the systems will be provided by:

- compressed air system and air-operated valves;
- emergency diesel generators;
- common-unit emergency diesel generators;
- other electric power sources in case of blackout, including mobile installations (described in Section 5).

### **6.3.7 Measuring and Control Instrumentation Needed for Protecting Containment Integrity**

To implement strategies on containment integrity preservation, the following parameters shall be measured:

- containment water level,
- pressure in the containment,
- hydrogen concentration in the containment,
- temperature in the containment,
- fission product release to the containment (containment radiation level).

Besides, additional indicators (in terms of their importance for severe accident management strategies) to be monitored for the strategies are:

- activity in steam line (to assess possibility of environmental release of fission products in case of containment bypass);
- position of valves on containment penetrations;
- containment isolation condition;
- water level in low-pressure injection system tanks;
- release of fission products to the environment (NPP radiation level).

Analysis of measuring instrumentation at Ukrainian NPPs shows that it is sufficient for the majority of these strategies.

In addition, several measures are being implemented under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs” for accident and post-accident monitoring of reactor parameters (Section 6.3.10).

### **6.3.8 Capability for Severe Accident Management in Case of Simultaneous Core Melt/Fuel Damage Accidents at Different Units on the Same Site**

In case of a severe accident at several power units on-site, the NNEGC Energoatom emergency response system envisages mobilization of operating organization resources, including mobilization of personnel and means from other NPPs and subordinated enterprises. If necessary, NPPs form additional shifts for emergency response teams and brigades to arrive at NPP in emergency, engaging staff of emergency response teams and brigades from other NPPs. NPPs provide the affected NPP with necessary material and technical means from their own emergency sets and spare part stocks, instruments, materials, special equipment.

External support in case of emergency situations which cannot be managed by forces of the affected NPP is addressed in detail in Section 6.1.1.

### **6.3.9 Conclusion on the Adequacy of Severe Accident Management Systems for Protection of Containment Integrity**

Information concerning the strategies for severe accident management is provided in Sections 6.2.2 and 6.2.3. The following strategies were identified for severe accident management:

1. Coolant injection into the primary circuit;
2. Primary pressure decrease;
3. SG makeup;
4. SG pressure decrease;
5. Injection into the containment;

6. Hydrogen concentration control in the containment;
7. Containment venting;
8. External reactor cooling;
9. Corium cooling in the reactor cavity.

Strategies No. 5-7 and 9 directly refer to the containment integrity preservation.

Analysis of the developed strategies permits a conclusion that there are necessary technical means needed for implementing most strategies. Additional measures are needed for the following strategies:

- hydrogen concentration control in the containment;
- containment venting;
- corium cooling in the reactor cavity.

For implementation of these strategies, measures are envisaged under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs” (see Section 6.3.10).

#### **6.3.10 Measures Which Can Be Envisaged to Enhance Capability to Maintain Containment Integrity after Occurrence of Severe Fuel Damage**

The following measures are ongoing under the “Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs” for severe accident management (including measures for containment integrity preservation after severe core damage):

- SAMG development and substantiation;
- implementation of filtered containment venting in case of severe accidents;
- implementation of measures for hydrogen concentration reduction in the containment in case of beyond design basis accidents;
- implementation of hydrogen control system in SG and MCP compartments (A201) and in pressurizer compartment (A527/1);
- implementation of hydrogen concentration monitoring system in the containment for case of beyond design basis accidents;
- preservation of the containment integrity in case of interaction with corium (active core melt) at the ex-vessel phase of severe accident;
- development and implementation of measures for diagnostics in case of severe accident.

The above-mentioned measures will allow prevention of the main types of containment failures:

- containment failure due to overpressure;
- containment failure due to hydrogen explosion;
- containment failure due to MCCI.

### **6.4 Accident Management Measures to Restrict the Radioactive Releases**

#### **6.4.1 Radioactive Releases after Loss of Containment Integrity**

Section 6.2 of the report describes strategies on severe accident management to prevent the main severe accident phenomena or to mitigate their consequences to:

- terminate core damage at early stage;
- maintain the containment function as long as possible.

Containment integrity failure can occur in case of unsuccessful implementation of the strategies described above. Three main types of containment integrity failures are possible:

- containment bypass (as a result of primary-to-secondary leak or as a result of temperature-induced failure of SG tubes);
- failure of containment isolation;
- containment failure.

In case of containment bypassing, the following measures must be implemented:

- reduction of primary pressure;

- injection of coolant into the secondary circuit at affected SG.

In case of an accident with containment failure, measures must be focused on:

- fission product deposition from containment atmosphere;
- containment pressure decrease to reduce pressure drop between the containment and environment.

An efficient way to reduce radioactive releases to the environment is long-term operation of the containment spray system. As a result of the containment spray system operation, pressure and temperature will decrease owing to steam condensation, that will lead to reduction of leak from the containment and, consequently, will lead to reduction of fission product releases to the environment. Besides, fission products from the containment atmosphere will be deposited due to the spray system operation.

#### **6.4.2 Accident Management after Uncovering of the Top of Fuel in the Cooling Pool**

In case of blackout, spent fuel pool cooldown pumps will trip, leading to termination of cooling water circulation through the spent fuel pool. Water inventory above fuel assemblies allows establishing natural circulation in the spent fuel pool. Decay energy heats spent fuel pool water, resulting in water boiling and gradual water level decrease.

Decrease of the water level below the fuel part of assemblies leads to gradual heating of fuel claddings and zirconium-steam reaction with hydrogen generation and release of additional energy that is commensurable with fuel decay heat. Further heating of fuel elements leads to their severe damage. After complete dry-out of the spent fuel pool, its lining is damaged and corium starts interacting with concrete floor structures of the spent fuel pool, leading to generation of hydrogen and other noncondensable gases and resulting in spent fuel pool floor melt-through.

Conservative calculation analyses were performed under the stress tests for the most severe accident scenarios. According to the calculations, time was assessed for melt-through of the spent fuel pool concrete floor structure, which is more than 1 day for WWER-1000 pool and more than 2 days for WWER-440 pool. The calculated time available before spent fuel pool damage is sufficient for emergency measures.

Also, the results of calculation analyses showed that during a severe accident in the spent fuel pool, the amount of hydrogen generated is comparable with the hydrogen amount generated in case of a severe accident (see Section 6.3.2). The methods of reducing the hydrogen concentration resulting from spent fuel pool severe accident are the same as described in Section 6.3.2. the calculated hydrogen amount generated in the spent fuel pool in case of a severe accident is about 2200-3800 kg.

It should be noted that effective measures were developed to prevent spent fuel pool severe accidents, including water supply from mobile facilities. In addition to the planned measures, guidelines for personnel actions on spent fuel pool severe accident management are under development.

#### **6.4.3 Conclusions on the Adequacy of Measures to Restrict the Radioactive Releases**

The strategies on severe accident management include personnel actions to limit radioactive releases. In case of severe accident progression with containment integrity preservation, the containment performs functions of radioactive release limitation.

In case of containment failure, an efficient strategy for limitation of radioactive releases to the environment envisages containment spray system operation.

## 7. GENERAL CONCLUSIONS

### 7.1 Safety Margins

The stress test results were represented at the open session of the SNRIU Board on 24-25 November 2011. The session was attended by representatives of non-governmental organizations, mass media, Secretariat of the Cabinet of Ministers of Ukraine, dedicated committees of the Supreme Council of Ukraine, Ministry of Energy and Coal Industry of Ukraine, regional state administrations and regional councils, as well as representatives of nuclear regulatory authorities of Bulgaria, Germany, Russian Federation and Slovak Republic.

Upon presentation of stress test results, the following was pointed out:

- the sequence of events that occurred at Fukushima-1 is impossible for Ukrainian NPPs. Safety improvement measures implemented at Ukrainian NPPs for the last 10-15 years have greatly reduced the probability of core damage and radioactive releases;
- analysis has revealed no new crucial external natural hazards or combinations of hazards additional to those considered in NPP design and analyzed in detail in NPP safety analysis reports.

### **Earthquake**

The operating NPPs were designed using the conservative principle, they are robust against design-basis hazards and have appropriate safety margins.

The seismic impact that established in design of the operating Ukrainian NPPs is accepted as follows: design earthquake with intensity of 5 and maximum calculated earthquake with intensity of 6 on the MSK-64 scale (peak ground accelerations of 0.025g and 0.05g, respectively).

Additional seismic investigations completed on sites (ongoing at the Zaporizhzhya NPP site) confirmed the design-basis seismicity of the Rivne and Khmelnytsky NPPs (2000-2002) and determined the peak ground acceleration (0.093g) for the South Ukraine NPP (2009-2010).

The structures of category I<sup>5</sup> of all NPP units (containment, reactor building, emergency diesel generators, essential service water spray ponds) and category 2 (including turbine hall) are resistant to design-basis seismic impacts. Category I structures have safety margins (at least 0.1g).

The following safety margins were determined for the containment:

- 0.185g for WWER-440 (technical state assessment in the framework of Rivne-1,2 long-term operation activities);
- 0.17g for WWER-1000/V-320 (assessment in the framework of stress tests);
- 0.15g for WWER-1000/V-302, 338 (technical state assessment in the framework of South Ukraine-1 long-term operation activities).

The equipment and piping required to perform the main safety functions are robust against the design-basis seismic impacts; the main reactor equipment and piping and safety systems have safety margins.

Seismic qualification of equipment is ongoing. The value of 0.1g was conservatively accepted in qualification according to IAEA recommendations (0.12g for the South Ukraine NPP with account of engineering margin).

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<sup>5</sup> Category I includes buildings and structures whose failure or damage can affect safety-related normal operation systems and thus cause radioactive releases in amounts that may lead to exposure of personnel and the public to radiation exceeding the established doses for the maximum design-basis accident or cause failure of safety systems that maintain the reactor core subcriticality, ensure reactor heat removal and confine radioactive materials

The structures and components of the spent fuel pools at Ukrainian NPPs are robust against design-basis seismic impacts and have safety margins (0.1g).

The structures and components of the dry spent fuel storage facility (DSF, Zaporizhzhya NPP) are designed to withstand seismic impact with 0.2g ground acceleration.

Detailed information on safety margins for seismic impacts is provided in Section 2.2 of this Report.

### ***Flooding***

There are no direct risks of flooding, including those resulting from failure of hydraulic structures caused by an earthquake. The safety margins are as follows:

- Zaporizhzhya NPP. The maximum possible level in the Kakhovka reservoir in case of the worst scenario of failure of the Dnipro river dams will be 19.4 m, which is lower than the leveling elevation of the Zaporizhzhya NPP site (22.0 m).
- Khmelnytsky NPP. The leveling elevation of the Khmelnytsky NPP site and the cooling water reservoir dam top constitute 206.0 m, while the maximum level of a flooding wave in case of failure of the dam is 203.0 m.
- Rivne NPP. The leveling elevation of the Rivne NPP site is 188.5 m and the maximum water level of the Styr river is 164.4 m.
- South Ukraine NPP. The leveling elevation of the South Ukraine NPP site is 104.0 m. Rise of the water level in the Yuzhny Bug river does not pose a hazard since it is 70 m lower than the site location. The water level in the Tashlyk reservoir may reach 101.5 m.

### ***Tornado***

The tornado impact on category I buildings and structures of all Zaporizhzhya, Khmelnytsky, Rivne and South Ukraine NPP units was analyzed taking into account wind frontal pressure, pressure drop and flying missiles.

The tornado frontal pressure values calculated on the basis of statistical data are in the range from 6.0 to 12.0 kPa for all NPP sites and are substantially lower than shock wave impact with frontal compression pressure  $\Delta P = 30$  kPa (design value).

The pressure drop of 6.4-10.3 kPa under tornado impacts calculated on the basis of statistical data is lower than 8.1-10.9 kPa conservatively accepted in the safety analysis reports for such loads.

Tornadoes that may occur in Ukraine do not pose hazards for NPP buildings and structures in terms of flying missiles.

Analysis of tornado impact on the dry spent fuel storage facility at the Zaporizhzhya NPP revealed no adverse effects.

### ***Other Extreme Weather Conditions (external fires; extreme high/low temperatures; extreme wind; extreme snow; combination of external natural hazards).***

New crucial external hazards or combinations of hazards additional to those incorporated in NPP designs and analyzed in detail in NPP safety analysis reports have not been revealed.

The robustness of the Zaporizhzhya NPP (and DSF), Khmelnytsky NPP, Rivne NPP, and South Ukraine NPP against the above-mentioned hazards was confirmed. No safety issues were revealed.

### ***Power Supply***

Robustness of power supply is inherent in the plant designs. Each WWER-1000 and WWER-440 power unit is designed to have three independent essential power supply trains, each train containing a diesel generator and batteries. Each train of the emergency power supply system in terms of its capacity and configuration of the connected loads can ensure emergency reactor cooldown and long-term decay heat removal. The independency of three

trains is ensured by their complete separation in terms of structures, processes, power supply and control. In addition, each site has a common-unit reliable power supply system containing two trains with independent diesel generators (DG) and batteries serving important equipment of one or several units.

Hence, the total number of DGs installed at the NPP sites constitutes:

Zaporizhzhya NPP- 20 DGs, Khmelnytsky NPP– 8 DGs, Rivne NPP – 14 DGs, South Ukraine NPP– 11 DGs.

All DGs at all NPPs are provided with a 7-day reserve of diesel fuel, while the time of designed unattended operation of DGs in emergency mode is 250 hours.

In accidents involving off-site power supply, the design provides for transfer of WWER-1000 and WWER-440 reactors into a safe state and long-term decay heat removal. Additionally, all NPPs have several alternative possibilities for powering plant house loads using the following schemes:

- Zaporizhzhya NPP: from Zaporizhzhya Thermal Power Plant, from hydrogenerators of the Dnipro Hydroelectric Plants–1, 2 and Kakhovka Hydroelectric Plant; from a diesel generator of any Zaporizhzhya NPP unit; from any common-unit emergency diesel generator; from 75 kV high-voltage lines. The construction of the 750 kV Zaporizhzhya NPP-Kakhovka substation high-voltage line, which can also be used as an additional power supply source, has been started.
- Khmelnytsky NPP: from a diesel generator of any Khmelnytsky NPP unit; from any common-unit diesel generator; from 330 kV and 750 kV high-voltage lines.
- Rivne NPP: from a diesel generator of any Rivne NPP units (for units 1, 2 and 3, 4); from any common-unit diesel generator; from 330 kV and 750 kV high-voltage lines.
- South Ukraine NPP: from hydrogenerators of the Kremenchug Hydroelectric Plant and Tashlyk Hydroelectric Pumped Storage Plant; from a diesel generator of unit 2 for unit 1 (and vice versa); from any common-unit diesel generator.

Appropriate connection programs were developed; shift personnel were trained and are re-trained on a routine basis.

*In accidents involving loss of off-site power supply and failure of DGs, Ukrainian WWER-1000 and WWER-440 reactors are provided with additional possibilities of power supply, and appropriate actions of the operating personnel are envisaged. Besides, additional measures have been proposed as a result of stress tests, as discussed in Section 7.3.*

### **Heat Removal**

To ensure long-term decay heat removal, the following upgrades were implemented at Ukrainian NPPs:

- WWER-440 (Rivne-1,2). The new additional emergency SG feedwater system was commissioned (2010). The system is independent, has air-cooled diesel-pumping facilities and water inventory of  $2 \times 1000 \text{ m}^3$ . Refilling of its tanks is possible. This system is described in detail in Section 1.1.2 R of the Report.
- WWER-1000. Long-term makeup of emergency feedwater storage tanks and steam generators in primary emergency cooldown in loss of house loads power using fire hoses and mobile pumping units (fire engines) was ensured. This upgrade is described in detail in Sections 1.1.2 Z, 1.1.2 Kh, 1.1.2 R, and 1.1.2 SU. Appropriate connection programs were developed, changes were introduced into the operational and emergency procedures, personnel were trained and are re-trained on a routine basis.

*Implementation of the above-mentioned measures for WWER-440 and WWER-1000 allows successful long-term decay heat removal in accidents, including loss of power supply. Besides, additional measures have been proposed as a result of stress tests, as discussed in Section 7.3.*

### ***Management of Beyond Design Basis Accidents***

Symptom-based emergency procedures were implemented at NPP units. Actions of personnel are aimed at prevention of accidents involving severe core damage.

Emergency plans have been introduced at NPPs and at the level of the operating organization.

An efficient accident management system was created, which includes the required administrative and technical measures, means and resources to prevent and limit accident consequences. External technical support is ensured in accident management at the operator and national levels to mobilize additional human and technical resources.

#### **7.2 Safety Issues**

The safety assessment of WWER (V-213, V-320, V-302/338) designs to determine their main safety issues and establish a unique international approach to prioritize the safety improvement measures was initiated by the IAEA in 1990. IAEA WWER safety "Issues Books" were developed to establish and rank WWER NPP design and operational safety issues.

The "Issues Books" provided the initial basis for developing the safety improvement program for the Ukrainian NPPs. In addition, the comprehensive safety reassessment studies that involved in-depth probabilistic and deterministic safety analyses for all operating units suggested further technical basis for planning safety upgrades and including additional measures into the safety improvement program.

An independent safety assessment of the Ukrainian NPPs was carried out in the framework of the common EC-IAEA-Ukraine project (Memorandum of Understanding on Co-operation in the Field of Energy between the European Union and Ukraine) in terms of the following aspects:

- Design safety assessment (ESRS);
- Operational safety assessment (OSART);
- Radioactive waste management assessment (RWRS);
- Integrated regulatory review (IRRS).

In the framework of this project, 14 inspection missions at the four NPP units and the IRRS mission were carried out. 62 experts from 23 countries and 32 IAEA representatives participated in the missions.

Resulting from the project, the experts made a conclusion that the Ukrainian NPPs meet most IAEA safety requirements and recommendations. For areas in which IAEA safety requirements and recommendations were not completely met, it was confirmed that Ukraine undertakes measures to eliminate these inconsistencies. No complete inconsistencies were revealed.

In the framework of the design safety assessment, Ukrainian NPPs were examined for compliance with the requirements of NS-R-1 "Safety of Nuclear Power Plants: Design". Following the assessment, it was confirmed that Ukrainian NPPs met completely 172 of 194 NS-R-1 requirements. Incomplete compliance was revealed for: equipment qualification, consideration of severe accidents, NPP seismic resistance, completeness of probabilistic and deterministic safety analysis and post-accident monitoring.

The operational safety review mission confirmed that operational safety of the Ukrainian NPPs was managed in a skilled and efficient manner.

Compliance with recommendations of "Issues Books", comprehensive safety analysis and safety reassessment of Ukrainian NPPs, as well as peer review results allowed determining areas that require safety improvement and developing necessary safety improvement measures.

The stress tests revealed no safety issues that were not identified previously. The stress tests demonstrated relevance of the safety improvement measures under the Comprehensive (Integrated) safety Improvement Program.

According to stress test results, the main areas that require further improvement are as follows:

- increase NPP robustness against tornado (in terms of potential loss of essential service water);
- provide additional means for long-term decay heat removal from the core and spent fuel pools under station blackout and/or loss of ultimate heat sink (including mobile equipment);
- improve procedures for management of beyond-design-basis accidents in terms of mitigation of severe accident consequences;
- conduct post-accident monitoring.

Stress tests have addressed in detail several possible safety improvement measures and confirmed their efficiency:

- main parameters of mobile facilities were determined (flow rate, pump head);
- requirements for location of mobile equipment were established;
- severe accident management strategies were analyzed: filtered pressure relief, hydrogen concentration control.

The following cliff edge effects were identified:

- discharge time of batteries – loss of monitoring of reactor parameters and control of steam dump devices for secondary systems;
- water level reduction in the reactor pressure vessel below the level of main coolant piping hot legs – makeup of secondary systems becomes ineffective due to absence of natural circulation in main coolant piping loops;
- start of irreversible heating of fuel claddings - the last possibility for the operator to intervene, after which severe damage of reactor core begins;
- reactor pressure vessel failure – strategies aimed at maintaining of the containment integrity have to be implemented;
- containment failure - strategies aimed at decrease of fission product releases have to be implemented.

The revealed cliff edge effects were considered in developing safety improvement measures.

### **7.3 Potential Safety Improvements**

In April 2011, based on analysis of the events that occurred at Fukushima-1, the SNRIU obliged the operating organization to revise the scope and priorities of measures under the Comprehensive Safety Improvement Program. The following measures have to be considered:

- implementation of the hydrogen concentration monitoring system in containment for beyond design basis accidents;
- reliability increase of emergency power supply system under station blackout (mobile diesel generators);
- additional seismic investigations of NPP sites and assurance of seismic resistance of equipment, piping, buildings and structures important to safety;
- detailed analysis of severe accidents and development of severe accident management guides;
- measures on severe accident management during ex-vessel stage;
- provision of instrumentation before and after accidents;
- modernization of NPP radiation monitoring systems;
- implementation of a closed-circle television system for fire/explosion hazardous and non-serviced rooms.

Additionally, prior to stress tests, measures on forced emergency pressure relief from the containment at WWER-1000 units (containment venting) were already identified and put into implementation.



Also a scope and deadlines for all measures had been revised and amended with due consideration of their relevance in the light of the Fukushima-1 accident.

In addition to the above safety improvement measures, the following improvements have been identified from the stress test results:

- assurance of operability of essential service water loads under loss of water in the spray ponds as a result of tornado;
- a series of measures aimed at assurance of long-term (up to 72 hours) residual heat removal from the core and spent fuel pool under station blackout and/or loss of ultimate heat sink:
  - makeup of secondary systems using mobile facilities;
  - makeup of spent fuel pools;
- emergency makeup of primary systems with boric water are under consideration (for compensation of potential loss of primary coolant);
- development of severe accident management guides for spent fuel pools;
- I&C upgrade to ensure the proper work under severe accident conditions.

The above measures have been additionally included into the Comprehensive Safety Improvement Program. The Program has acquired the status of a national-level program, its scope and funding sources have been determined on 7 December 2011.

In accordance with SNRIU Board Resolution No. 13 of 24-25 November 2011, the mandatory condition for the operating organization to extend the lifetime of NPP units beyond 30 years will be the full implementation of the following measures:

- 1) ensure robustness of equipment, piping, buildings and structures required for the main safety functions to seismic impacts not less than 0.1 g (0.12g for South Ukraine NPP);
- 2) ensure performance of the main safety functions by NPP equipment in “harsh” environments;
- 3) implement containment venting systems at WWER-1000 plants;
- 4) implement measures to ensure SG and spent fuel pool makeup (cooldown) under long-term station blackout and/or loss of ultimate heat sink.
- 5) introduce at NPP units:
  - severe accident management guides (relating to core and spent fuel pools);
  - symptom-based emergency operating procedures for shutdown states.

In addition to the stress tests and according to the “Action plan for targeted safety assessment and further safety improvement of Ukrainian NPPs in the light of the events at Fukushima-1” (approved by the SNRIU), targeted inspection of emergency preparedness was conducted. Emergency exercises were performed at operating NPPs (25-26 May 2011 at Zaporizhzhya NPP; 29 September at Khmelnytsky NPP; 10-11 November at Rivne NPP; 24 November at South Ukraine NPP). The training scenarios were developed taking into account the events at Fukushima-1 and included total station blackout with loss of ultimate heat sink.

## PART 2. CHORNOBYL NPP



### 1 Ch GENERAL DATA ON THE CHORNOBYL NPP

#### 1.1 Ch. Brief Description of the Site Characteristics

The Chernobyl NPP site is situated in the north of the Kyiv region on the right bank of the Pripiat river. The ChNPP is located in the 30 km exclusion zone that was established after the accident at power unit 4 in 1986.

The town of Chernobyl is located 15 km southeastward from the ChNPP, the town of Slavutych is 60 km eastward and Kyiv is 110 km southward. The territory of the ChNPP is bordered by the Pripiat river from north and east, the lands and solid woods of the State Forest Management Fund of Ukraine adjoin it from the south and west.

The following nuclear facilities are located on the ChNPP territory:

- units 1, 2, and 3 (under decommissioning);
- Interim Spent Nuclear Fuel Storage Facility (ISF-1).

The License holder is the State Specialized Enterprise Chernobyl NPP.

The decommissioning activities on units 1, 2, and 3 are based on the License issued by the State Nuclear Regulatory Committee of Ukraine on 22 March 2002.

The ISF-1 is operated according to the License issued by the State Nuclear Regulatory Committee of Ukraine on 25 June 2008.

Four nuclear facilities are located at the ChNPP site: three units with RBMK-1000 reactors and ISF-1.

#### 1.1.1 Ch Main Characteristics of the Units

##### 1.1.1.1.Ch Reactor Facilities

Units 1, 2, and 3 include RBMK-1000 reactor facilities.

Unit 1 was commissioned on 26 September 1977, shut down on 30 November 1996.

Unit 2 was commissioned on 21 December 1978, shut down on 11 October 1991.

Unit 3 was commissioned on 8 December 1981, shut down on 15 December 2000.

ChNPP units 1, 2 and 3 are under decommissioning.

Each power unit includes RBMK-1000 reactor (high-energy, channel-type, boiling-water thermal reactor) with a thermal output of 3200 MW. Pressurized water was used as the coolant and graphite as the moderator.

The power units contained one circuit, and the turbines received separated steam generated in the reactor.

Each power unit includes two turbine generators of K-500-65/3000 type equipped with TVV-500-2 electrical generators, 500 MW each.

The electricity generated by ChNPP Stage 1 was delivered through 110 kV and 330 kV transmission lines. The electricity of generated by ChNPP Stage 2 was delivered through 750 kV transmission line.

Spent fuel assemblies (SFA) have been unloaded from the reactor cores of units 1, 2 and 3.

There is no fresh nuclear fuel on the ChNPP site. There are 1365 SFAs in the unit 1 cooling pool and 1355 SFAs in the unit 2 cooling pool.

#### **1.1.1.2.Ch Spent Fuel Storage Facilities**

The ISF-1 is a wet storage facility intended for the receipt and storage of SFAs after preliminary storage in spent fuel pools for at least 1.5 years or in reactors and for SFA unloading from ISF-1 and subsequent transfer to the dry storage facility under construction (ISF-2).

The ISF-1 was commissioned in September 1986. The transfer of spent nuclear fuel from the spent fuel pools of units 1 and 2 and its placement into the spent fuel pool of ISF-1 began in the same year.

ISF-1 is situated on the ChNPP north-westward to the main building of ChNPP Stage II. The area occupied by the ISF-1 building, constructed from 1983 to 1986, is approximately 2500 m<sup>2</sup>.

ISF-1 is to be operated until the dry spent fuel storage facility (ISF-2) is commissioned, taking into account time required for the transfer of SFAs from ISF-1 to ISF-2. It is planned to commission ISF-2 in 2015. SFAs will be unloaded from ISF-1 during 9-9.5 years (~2025).

The spent fuel pools of ISF-1 contain 18564 SFAs. SFAs shall be cooled for more than 10 years.

### **1.1.2 Ch Description of the Systems for Conduction of Main Safety Function**

#### **Power supply systems**

The ChNPP site has power supply from the unified electricity grid of Ukraine through ORU-110/330/750 kV outdoor switchgears connected to the grid through two 750kV lines, four 330kV lines and two 110kV lines. The 110/330kV outdoor switchgears are connected through two 330/110kV autotransformers AT-1 and AT-2 and the 330/750kV outdoor switchgears are connected through two 750/330kV autotransformers AT-3 and AT-4. Power to the units, ISF-1 and other ChNPP loads is supplied by the 110/330 outdoor switchgear using three lines: 110 kV line through 1TR, using two independent 330kV lines through 2 and 3 TR. There is possibility to supply power to units 1, 2, 3 and ISF-1 by any transformer 1,2,3 TR through the 6kV backup busbar.

ISF-1 off-site power is supplied from two power sources: 6kV bus of 6RNA KRU-6kV metal-clad switchgear of unit 3 (using IIDG-3 as automatic backup in case of loss power from backup transformer 3TR) and 6kV but of 6RB KRU-6kV metal-clad switchgear of unit 3. ISF-1 loads are powered from metal-clad transformer substation (KTPSN-0.5-6/0.4kV) with two transformers 237T and 238T (TSZS-1000/6) through power distribution unit (RUSN-0.4kV) that includes two 0.4 kV buses 237N and 238N with automatic backup.

### ***Spent fuel pools (SFP) of units and ISF-1. SFP makeup and level control***

#### *Design of the ChNPP units 1-3 spent fuel pools*

There are SFA cooling pools (SFAP) and fuel channel cooling pools (FCP) to store fuel channels and special items.

The SFAP is a reinforced concrete reservoir of 4.200x10.300m in the cross-section, whose walls and bottom are lined with 12X21H5T steel of 2 mm and 4 mm in thickness, respectively. The Walls and bottom of cooling pools provide shielding for NPP personnel.

The capacity of two SFAPs is defined by complete unloading of the reactor core. SFAP water volume at a rated level is 720m<sup>3</sup>.

Each SFAP has:

- a slotted slab consisting of 34 semi-rows of beams from the right side and 37 semi-rows of beams from the left side designed to suspend high-active items in canisters, 852 in each pool,
- a transfer basket cell (characteristic load is 5,000 kg),
- a mechanism for loading of fuel assemblies into canisters from the refueling machine (characteristic load is 900 kg),
- emergency staircases.

SFAPs are also provided with piping to ensure:

- heated water intake,
- cooled water return,
- water intake for treatment,
- return of treated water,
- water overflow if permissible levels are exceeded;
- makeup and loading;
- blow-off to the process venting system.

The SFAP slotted slab is formed by:

- H-beams (VSt.3Sp 5 steel) embedded into the SFAP concrete wall. The beam effective length is 2,000 mm, the upper flange width is 120 mm, the bottom flange width is 70 mm, the beam height is 200 mm, the distance between beams' axes constitutes 250x1 mm, and the central slot width is 200 mm,
- rakes – 76x6 mm pipes (08X18N10T steel),
- cover plates designed to prevent ingress of foreign items to the cooling pool and improve steam removal to the process venting system.

The effective load on the slotted slab constitutes 400 kg/m<sup>2</sup>.

The slotted slab design provides for:

- vertical installation of canisters,
- fixation of canisters and their arrangement in the horizontal plane (canisters are installed tightly on the console, the flange of the neighboring canister serves as support),
- fixation of the end canisters installed 12th in the row to prevent their displacement towards a longitudinal slot and falling to SFAP with plates (h=10 mm) welded in the upper part of the console end,
- installation of canisters into the lower support ring for cover plate removal.

All spent fuel assemblies loaded into the SFAP are stored in canisters according to the design layout at a pitch of 250x160 mm provided by the 250 mm distance between the slotted slab beams' axes and 160 mm diameter of the canister flange.

The cooling circuit and the cooling pool water treatment circuit provide decay heat removal from SFAs and water treatment. The following conditions are ensured:

- water level in SFAP maintained within the operational limits (operational limit for units 1, 2 is not lower than +19.2m and for unit 3 is not lower than +34.58m),

- water temperature in SFAP is maintained within operational limits of 20-55°C.

There are systems for air venting above the cooling pools.

SFAP water quality is maintained by periodic actuation of the active water treatment system SVO-2 or partial replacement of SFAP water with feedwater.

The unit 1 SFAP contains 1365 SFAs and unit 2 SFAP contains 1355 SFAs.

The ISF-1 contains 18564 SFAs. Spent nuclear fuel in ISF-1 is stored in the cooling pool consisting of five compartments. ISF-1 cooling pool compartments are reinforced concrete reservoirs, whose walls and bottom are lined with corrosion-resistant steel. The bottom lining is hard-faced to reinforced concrete embedded items. Walls lining is welded to reinforced concrete plates installed earlier. The walls and bottom of the cooling pool compartments provide shielding for ISF-1 personnel. Cooling pool compartments have a slotted slab, which is a console-type support without rakes used for suspension of the fuel assemblies in canisters. Fuel assemblies are stored in the cooling pool compartments in canisters filled with water at a pitch of 230x110 mm. The slotted slab is closed from above with hinged metal cover plates designed to prevent ingress of foreign items to the cooling pool and improve steam removal to the process venting system.

The cooling pool compartment up to elevation 11.35 m is made of M300 and B-8 hydraulic concrete. Walls along the numerical axes are 125 cm thick (internal) and 150 cm thick (external). Standard M300 concrete is used above elevation 11.35 m. The walls and bottom of ISF-1 are made of 12X21N5T corrosion-resistant steel to ensure safety of the storage facility operation: walls are 4 mm thick and bottom 8 mm thick. Grooves ensuring moisture collection from under the lining are made on a concrete bed to prevent potential leakages. The grooves end with pipes 76 mm in diameter passing to the room for cooling pool leakage collection and monitoring.

The transfer basket storage compartment (TBSC) of ISF-1 is located in the section for receipt and reloading of a cask car. It is used for the receipt and storage of transfer baskets, reloading of spent fuel assemblies from a transfer basket to the canisters and from the canisters to a transfer basket, as well as for temporary storage (placement) of spent fuel assemblies at places of canister accumulation.

A trench is located in the section for cask car receipt and reloading and is used for storage (placement) of spent fuel assemblies and to ensure transfer of canisters with SFAs to/from the compartments of the ISF-1 cooling pool.

The TBSC and trench are reinforced concrete reservoirs whose walls and bottom are lined with corrosion-resistant steel. The wall lining is 4mm thick and bottom lining is 8 mm thick. The walls and bottom of the TBSC and trench provide shielding for ISF-1 personnel. The TBSC bottom is covered with special plate-steel covering to serve as a damper to reduce the probability of severe damage to TBSC structures that can be caused by falling of separate canisters or a basket with a fuel assembly during handling.

The cooling pool compartments, trench and TBSC are filled with feedwater. The rated level is +10.50 m. The rated level in the compartments of the cooling pool, trench and TBSC is ensured by operation of the automatic level control circuit. Any compartment can be isolated using hydraulic locks (sluice).

Cooling pool makeup for ISF-1 is ensured through two lines: operating makeup of at least 48 m<sup>3</sup>/h and additional makeup of at least 67 m<sup>3</sup>/h. The emergency makeup of at least 115m<sup>3</sup>/h is ensured through equipment redundancy.

## 1.2 Ch. Significant Differences between Units

Units 1 and 2 are located in one main building and have common auxiliary systems.

Before the 1986 accident, units 3 and 4 were a single structure; after the accident, unit 3 was separated from the destroyed unit 4 with a newly constructed reinforced-concrete wall.

The Chernobyl NPP was constructed in stages using a “two-boiler single-turbine” layout for two power units. ChNPP Stage I includes units 1 and 2 representing the design of first generation. Unit 3 represents the design of second generation and is a part of Stage II. A separate main building is envisaged for each Stage, where the main and auxiliary equipment is located. Besides the main buildings, auxiliary buildings and structures are located on the site to ensure NPP operation.

The main differences between the designs of first and second generations basically included distinctions between the accident confinement systems, configuration of emergency systems, reactor core and reactivity control systems, and layout of power units.

Fresh nuclear fuel has been fully removed from the Chernobyl NPP territory.

Spent nuclear fuel has been fully unloaded from unit 3. Spent nuclear fuel safe storage, preparation for spent nuclear fuel unloading from units 1 and 2 and decommissioning of systems and components that are not used are the main activities ongoing at units 1, 2, 3.

### **1.3 Ch. Use of PSA as Part of the Safety Assessment**

The analysis was carried out by postulation and analytical modeling of emergencies under extreme hazards. The worst scenarios and their combinations were analyzed in the report. Documents on safety assessment of the nuclear facilities and other on-site facilities were collected and analyzed; the worst probable threats to safety and major risks were identified; and safety measures supported organizationally or technically were determined. The report analyzes the ChNPP procedures and ChNPP personnel capabilities to mitigate emergencies including, in particular, a scenario of complete loss of power and heat removal from spent nuclear fuel and also beyond design basis accidents (bdba) involving damage of nuclear facility barriers.

## **2 Ch EARTHQUAKES**

### **2.1 Ch Design Basis**

#### **2.1.1 Ch Earthquake against Which the Plants are Designed**

The maximum possible earthquake intensity is estimated as  $4.5 \pm 0.5$ , with the depth of possible earthquake sources of 5÷25 km. The minimum distance from the ChNPP site to the eastern part of the Teterev fault is 16 km. In designing the plant, the design earthquake<sup>6</sup> (DE) was considered to be 5 on the Richter scale, and the maximum calculated earthquake<sup>7</sup> (MCE) to be 6 on the Richter scale.

#### **2.1.2 Ch Provisions to Protect the Plants against the Design Basis Earthquake**

According to the design, old dusty sands and loamy sands were removed from under the foundation of the main buildings and replaced with fine and medium-size alluvial sands, rolled to the condition close to natural state ( $\gamma = 1.7 \text{ kgf/cm}$ ).

Long-term observations of turbogenerators 1-6 and 7-8 have shown that serious vibration loads on the foundation sands have not led to manifestation of thixotropic and running properties in the sand mass and have not caused changes in stability of foundations.

Carpathian earthquakes in 1977-1986, that caused shocks with intensity of up to 5 points in the area of ChNPP location, had no influence on soils of foundations of main buildings and structures of units 1, 2 and 3.

Laboratory studies of dynamic stability of water-saturated sandy soils of foundations of the objects, carried out in 1993, indicate impossibility of their liquefaction in case of seismic actions of 7-point earthquake on the MSK-64 scale.

General technical state of buildings and structures of units 1, 2, 3 and ISF-1 is satisfactory. In course of their operation, the following is performed:

<sup>6</sup> DE corresponds to Seismic Level 1 (SL-1) in IAEA practice and Operating Basis Earthquake (OBE) in USA practice

<sup>7</sup> MCE corresponds to Seismic Level 2 (SL-2) in IAEA practice and Safe Shutdown Earthquake (SSE) in USA practice

1. Year-round operation of plenum and exhaust ventilation, as well as operation of heating during winter season, is ensured.
2. Regular control of state of water-supply communications and, if necessary, their prompt repair are carried out.
3. Maintenance, minor and major repairs, and engineering inspections are performed according to regulatory requirements.
4. Building operation is carried out in accordance with the following documents:
  - "Technical Operation of Power Plants and Networks. Rules GKD 34.20.507-2003".
  - "Provisions on Engineering Supervision of ChNPP Industrial Buildings and Structures, 81P-S".

### **2.1.3 Ch Compliance of the Plant with Its Current Licensing Basis**

The documents "Basic Regulatory Requirements and Estimated Characteristics of Earthquakes for the Chernobyl NPP Site" and "Basic Regulatory Requirements and Estimated Characteristics of Tornados for the Chernobyl NPP Site" were developed and approved in 2005. Studies conducted for units 1 and 2 and ISF-1 in 2005-2008 confirm the structural stability of the power units and cooling pools under earthquakes up to MCE.

## **2.2 Ch Evaluation of Safety Margins**

### **2.2.1 Ch Range of Earthquake Leading to Severe Fuel Damage**

It was postulated during stress tests that the earthquake with intensity of more than 6 on the Richter scale (exceeding MCE) will cause a beyond design basis accident involving nuclear fuel damage, whose radiological consequences will require personnel evacuation. Nuclear fuel damage means loss of fuel cladding integrity and release of fission products. In terms of nuclear fuel damage, the worst scenario involving loss of heat removal from spent nuclear fuel (loss of ultimate heat sink and loss of water in the cooling pools) and loss of power was considered. This situation is postulated as deterministic regardless of its probability.

### **2.2.2 Ch Range of Earthquake Leading to Loss of Containment Integrity**

During the period from September 2004 to May 2005, the Geophysics Institute of the National Academy of Sciences of Ukraine recorded earthquakes at the Chernobyl NPP site. The estimated earthquake characteristics for the Chernobyl NPP site were based on instrumental surveys and analytical studies.

The estimated characteristics of seismic impacts at the Chernobyl NPP site are presented in the document "Basic Regulatory Requirements and Estimated Characteristics of Earthquakes for the Chernobyl NPP site", approved by letter No. 3/19-19 dated 8 July 2005 by the State Construction Committee of Ukraine.

Characteristics (accelerograms, ground response spectra, etc.) are established in the range of seismic vibrations 0.033÷100 Hz. The MCE intensity is 6 according to the MSK-64 scale and the DE intensity is 5. The peak ground acceleration for MCE (DE) is 0.1 (0.05) g.

The list of extreme hazards considered during stress tests of the ChNPP nuclear facilities is sufficient, their characteristics at the Chernobyl NPP site are established in compliance with Ukrainian regulatory requirements and studies were completed correctly and in adequate scope. The scope of characteristics is sufficient. Upon the results of ChNPP stress tests, additional studies were proposed to evaluate safety margins during extreme events.

### **2.2.3 Ch Earthquake Exceeding the Design Basis Earthquake for the Plants and Consequent Flooding Exceeding Design Basis Flood**

The cooling pond is located near the ChNPP site. A break of weakest dam area cannot be completely excluded in case of an earthquake with intensity of 6.

ChNPP site levels are not less than 113.7m. The normal design level of the cooling pond is 111.3m. The difference between levels permits a conclusion that there is no risk of ChNPP site flooding, but dam failure will affect radiological situation.

The water reservoir discharge will minimize the risk of dam failure due to caving and water head. At present, feasibility study for cooling pond decommissioning is under development.

#### **2.2.4 Ch Measures Which Can Be Envisaged to Increase Robustness of the Plants against Earthquakes**

According to the License issued by State Nuclear Regulatory Committee of Ukraine on 25 June 2008, the ChNPP implements measures envisaged in the "ISF-1 Safety Improvement Plan". A number of measures to improve ISF-1 safety were implemented in 2008÷2011, namely:

- structures were reinforced,
- additional line for cooling pool makeup was arranged,
- system for collection, treatment and return of water to the cooling pools in case of loss of their integrity loss was implemented.

Upon the results of stress tests, the Chornobyl NPP planned the following for category 1 buildings and structures at the ChNPP:

- studies of safety margins for structures in case of earthquakes exceeding MCE,
- studies of safety margins and failure probabilities for cooling pool lining (first of all ISF-1) in case of earthquakes exceeding MCE.

In addition, the ChNPP has plans to evaluate seismic resistance of equipment and service lines (e.g., cooling and water level control systems for cooling pools) according to regulatory requirements and using earthquake characteristics obtained in 2005 and then, based on the results, to implement measures to improve the seismic resistance of equipment and service lines.

### **3 Ch FLOODING**

#### **3.1 Ch Design Basis**

The extreme water level was determined as 111.3 m based on analyses using results of long-term hydrological studies.

#### **3.2 Ch Evaluation of Safety Margins**

External flooding does not create a direct danger for the ChNPP nuclear facilities because the levels of the ChNPP site (113.7÷114.0 m) are significantly higher than the extreme high water level (111.3 m). Therefore, this extreme impact has not been considered in the stress tests.

### **4 Ch EXTREME WEATHER CONDITIONS**

#### **4.1 Ch Design basis**

##### ***Tornado***

The characteristics are established upon results of analytical studies performed in 2002 using meteorological data for the ChNPP location region collected during many years. The maximum tornado is F 3.0 class on the Fujita-Pearson scale, design tornado is F 1.5. The pressure drop between the tornado vortex center and periphery for F 3.0 (F 1.5) is 8.1 (3.1) kPa, the maximum vortex rotation speed is 81 (50) m/sec, and the tornado width is 290 (50) m.

##### ***Extreme Wind***

The characteristics are established by extrapolation (according to Eurocode Standards) of the standard wind characteristics defined in State Construction Standards DBN B. 1.2-2:2006 for the ChNPP location region. The load for the height from the ground surface varying from 5 to 110 m is 0.656 to 1.92 kPa.



### ***Extreme Air Temperature***

The characteristics are established on the basis of State Construction Standards DBN B. 1.2-2:2006 and PiN AE-5.6 for the ChNPP location region and are - 43°C and +45°C.

### ***Extreme Snow***

The characteristics are established upon results of analytical studies using meteorological data for the ChNPP location region collected during many years. The load constitutes 2.1 kPa.

### ***Extreme Rainfalls***

The characteristics are established on the basis of regulatory requirements for the ChNPP location region. The maximum value is 72 mm for 20 minutes and 190 mm per day.

Extreme natural events are ranged in the following way:

1) earthquake and tornado are the most hazardous for the ChNPP nuclear facilities,  
2) extreme wind, snow, rainfall and air temperature are significantly less hazardous because:

- impacts of extreme wind, snow and rainfalls are much less intensive than tornado impact;
- structures of nuclear facilities have slow thermal response,
- temperature regime in buildings is maintained, snow is removed, rainfall drainage system is functioning, etc.

The list of extreme hazards considered during stress tests of the ChNPP nuclear facilities is sufficient, their characteristics at the Chernobyl NPP site are established in compliance with Ukrainian regulatory requirements and studies were completed correctly and in adequate scope. The scope of characteristics is sufficient.

Conclusions made on ranking of external hazards by their danger for nuclear facilities are justified.

## **4.2 Ch Evaluation of Safety Margins**

### **4.2.1 Ch Estimation of Safety Margin against Extreme Weather Conditions**

The extreme hazards for the ChNPP nuclear facilities such as extreme wind, snow, temperature, rainfall, flooding and external fires are less hazardous than an earthquake or tornado. Therefore, the operating organization considers earthquakes and tornadoes as initial events causing loss of safety functions (loss of power and heat removal) in this targeted safety assessment.

### **4.2.2 Ch Measures Which Can Be Envisaged to Increase Robustness of the Units against Extreme Weather Conditions**

According to regulatory requirements, category 1 structures of the ChNPP nuclear facilities shall be resistant to MCE, F 1.5 tornado, extreme wind, snow and temperature. The surveys and strength calculations of structures were performed, including those for extreme impacts, and measures for reinforcement/restoration of some structures of the nuclear facilities were implemented. As regards resistance of the walls and bottom of cooling pools (ISF-1 and units 1, 2, 3), this conclusion concerns reinforced-concrete structures only, but not the stainless-steel lining of cooling pool walls and bottom. The resistance of lining (in particular, welds) was not subjected to calculations. Some measures were implemented at ISF-1 in case of lining failure. They included, in particular, introduction of an additional system for cooling water collection, treatment and return.

The ChNPP has planned to additionally analyze the resistance of ChNPP category 1 structures and potential structural failures under F 3.0 tornado.

The ChNPP has planned to evaluate the resistance of outdoor service lines to extreme impacts (e.g., lines of systems for water supply to the cooling pools, storm sewage) and, if necessary, implement appropriate measures.

## **5 Ch LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK**

### **5.1 Ch Loss of Electrical Power**

#### **5.1.1 Ch Loss of Off-site Power**

The ChNPP site electrical power supply system is sufficiently reliable. Station blackout may occur in case of full failure of the electrical grid of Ukraine and loss of ChNPP 110/330/750 outdoor switchyard connections with the Central and Southwestern power systems and Kiev power center.

The operating organization has reliable operating procedures for restoration of off-site electrical power. The operating personnel are regularly trained according to these procedures. The emergency diesel generators (EDG) of ChNPP Stage II can supply power to all loads needed to maintain levels in the cooling pools of power units and ISF-1, temperature of SFA storage, operation of ventilation systems and radiation monitoring of the ChNPP. DGs can be started up within 15 seconds.

As of 30 June 2011, there are 75.6 tons of diesel fuel ensuring autonomous operation of two DGs at full load during 3.5 days, or operation of one DG during 7 days ensuring power supply to makeup systems of SFA storage locations. This time is sufficient to arrange uninterrupted supply of diesel fuel.

Based on results of emergency training conducted with emergency personnel (Emergency Training-50E "Full Loss of Power Supply to ChNPP Auxiliary Systems"), time required to restore power supply of units 1 and 2 from EDG-2 is equal to 42 minutes. This time is similar to the design time for unloading of one unit cooling pool in DN Ø100 pipeline rupture, which, taking into account the accepted conservative approach, confirms the sufficient time margin restoration for safety functions.

#### **5.1.2 Ch Loss of Off-site Power and Loss of the Ordinary Backup AC Power Source**

Unit 3 batteries ensure operation of the radiological monitoring control board in terms of DC loads, emergency lighting, 0.4 and 6 kV control bus for circuit breakers. Therefore, based on the load, operation of the batteries is ensured for 22 hours (group 1) and 33 hours (group 2).

#### **5.1.3 Ch Loss of Off-site Power and Loss of the Ordinary Backup AC Power Sources, and Loss of Permanently Installed Diverse Backup AC Power Sources**

Power loss scenarios, including loss of off-site power, were considered in the stress tests. Loss of pool water cooling and hydrogen removal functions are the most critical scenarios under loss of power. The following has been defined based on ChNPP stress tests results:

- in case of loss of pool water cooling function, SFA temperature will increase but will not exceed 70°C (under water presence in the cooling pool), and thus it will not cause an accident,
- if hydrogen removal function is lost, hydrogen concentration in the cooling pool will increase and achieve the lower explosive limit (4%) during 10 days according to the conservative calculations. Experimental studies at ISF-1 demonstrated that hydrogen concentration stabilizes at 0.2%.

If neither off-site power nor power from EDGs can be ensured, the operating organization has a mobile DG station (MDGS) and corresponding tested procedure for its connection and power supply to ISF-1 loads. Considering all organizational and technical measures, time required to connect MDGS does not exceed 8 hours. Based on the results of tests conducted on 7 November 2011, it has been established that in case of ISF-1 blackout the existing MDGS can ensure:

- lighting of all ISF-1 rooms,

- operability of ISF-1 radiation monitoring system,
- operability of ISF-1 alarm and process monitoring systems,
- operability of ISF-1 SFP automatic makeup system,
- collection and release of leakages from under the lining of ISF-1 cooling pools, TBSC and trench by pumps of ISF-1 active drain treatment system.

#### **5.1.4 Ch Conclusion on the Adequacy of Protection against Loss of Electrical Power**

Analysis of power loss performed in the framework ChNPP stress tests, conclusions and measures implemented based on the stress tests results (MDGS arrangement) are sufficient and reasonable.

Loss of power to units 1, 2, 3 and ISF-1 does not lead to accidents with nuclear fuel damage. The nuclear facility design envisages electrical power redundancy; layout and inter-unit connections ensure high reliability of nuclear facility auxiliary systems. At the same time, spent fuel design ensures hydrogen safety and there is a sufficient time margin for restoration of safety functions lost during an accident.

The actual time for restoration of power on-site is as follows:

- unit 3 and ISF-1  $\leq 15$  sec (time of DG-2 startup),
- units 1 and 2 ~40 min (based results of emergency training performed with personnel (Emergency Training-50E “Full Loss of Power Supply to ChNPP Auxiliary Systems”), time required to restore power supply of units 1 and 2 from DG-2 is 42 min),
- time of power supply restoration from the electrical grid is the time of restoration of normal electrical power supply of auxiliary systems with diesel generator shutdown and will constitute from 1.5 to 2 hours in case electrical grid integrity is not lost.

#### **5.1.5 Ch Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Electrical Power**

At present, technical measures to connect MDGPU to the ISF-1 power supply system without changing of actual power supply diagram are implemented. Time required for connection and startup of MDGPU has been established as  $\leq 8$  hours considering implementation of organizational and technical measures based on the results of practical tests of power supply and time needed for that.

### **5.2 Ch Loss of the Decay Heat Removal Capability/Ultimate Heat Sink**

#### **5.2.1 Ch Design Provisions to Prevent the Loss of the Primary Ultimate Heat Sink, Such as Alternative Inlets for Sea Water or Systems to Protect Main Water Inlet from Blocking**

Spent nuclear fuel temperature storage mode.

The typical time values to achieve the following parameters were obtained in the result of calculations performed:

- fuel cladding temperature will reach 48°C during 2 months in case of pump failure, and it will be limiting value (stationary state at cooling pool walls and environment temperature of 20°C),
- fuel cladding temperature can reach maximum 64°C in the cooling pool central part under the worst environmental conditions when cooling pool walls heat up to 45°C, and environmental temperature is equal to 45°C,
- conditions of fuel cladding heating to higher temperatures during pump failure and water presence in the cooling pool were not identified (100°C is not reached);
- if there is no water in the cooling pool, fuel cladding temperature (natural air cooling) will exceed 300°C in 57 days in the cooling pool central part;

- fuel cladding temperature will exceed 300°C in the entire cooling pool in approximately 1-2 weeks depending on the environment temperature and cooling pool wall temperature.

Possibility to restore operability of the cooling pool water supply systems for the period significantly less than 50 days was analyzed and justified during stress tests.

The conclusions are correct for the conditions of MCE impacts when stability of the buildings with cooling pools containing spent fuel is ensured.

#### **5.2.2 Ch Loss of the Primary Ultimate Heat Sink (e.g., Loss of Access to Cooling Water from the River, Lake or Sea, or Loss of the Main Cooling Tower)**

Time required for the restoration of the heat removal from spent fuel is conservative, not less than time required for the restoration of power supply to systems ensuring spent fuel makeup and constitutes ≤8 hours considering the time required for implementation of organizational measures and operating procedures.

#### **5.2.3 Ch Loss of the Primary Ultimate Heat Sink and the Alternate Heat Sink**

Maintenance of the level in cooling pools with spent fuel is sufficient to ensure heat removal. Therefore, the operating organization has paid special attention to issues of level stabilization during stress tests. Technical measures were arranged and personnel were trained on cooling pool level stabilization by returning of pool leakages to the supply line without their purification for ISF-1. Actual time to ensure level stabilization is 30 minutes.

#### **5.2.4 Ch Conclusion on the Adequacy of Protection against Loss of Ultimate Heat Sink**

Nuclear safety of spent fuel storage in ISF-1 cooling pool compartments under normal operation conditions, operational events and DBA is ensured. ISF-1 nuclear safety is substantiated taking into account the burnup. Loss of ultimate heat sink does not lead to fuel damage taking into account level stabilization measures, which were proven at ISF-1.

#### **5.2.5 Ch Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Ultimate Heat Sink**

The measures to ensure ChNPP equipment with cooling pool makeup mobile means have been proposed, and level stabilization measures have been proven in practice.

### **5.3 Ch Loss of the Primary Ultimate Heat Sink, Combined with Station Blackout**

The scenario of loss of heat removal combined with loss of power and adequate emergency measures do not differ significantly from specific scenarios of loss of heat removal from SFAs or electrical power loss. The operating organization sets electrical power supply to the systems supporting makeup as a priority considering fuel storage thermal conditions. This will prevent fuel damage considering the pool level stabilization measures.

#### **5.3.1 Ch Time of Autonomy of the Site before Loss of Normal Cooling Condition of the Reactor Core and Spent Fuel Pool (e.g., Start of Water Loss from the Primary Circuit)**

Time of the site autonomy in terms of the characteristics of ChNPP spent fuel in case of cooling pool lining integrity is more than 2 months.

#### **5.3.2 Ch External Actions Foreseen to Prevent Fuel Degradation**

Water supply from mobile sources (fire trucks) is envisaged to prevent fuel damage under long-term loss of water cooling. This measure is sufficient for the actual spent fuel characteristics.

#### **5.3.3 Ch Measures Which Can Be Envisaged to Increase Robustness of the Plants in Case of Loss of Primary Ultimate Heat Sink, Combined with Station Blackout**

Units 1, 2, 3 and ISF-1 blackout does not lead to accidents with fuel damage. Nuclear facility design envisages redundancy of electrical power supply; layout and inter-unit

connections ensure high reliability of nuclear facility auxiliary systems. At the same time, characteristics of stored spent fuel are such that hydrogen safety is ensured and there is a sufficient time margin for restoration of safety functions lost during an accident. According to the conservative approach, the time of ChNPP nuclear facility power supply restoration (in case power grid is not lost) does not exceed 2 hours.

## **6 Ch SEVERE ACCIDENT MANAGEMENT**

### **6.1 Ch Organization and Arrangements of the Licensee to Manage Accidents**

#### **6.1.1 Ch Organization of the Licensee to Manage the Accident**

Attribution of emergency states at the ChNPP facilities to one of the established categories shall be performed on the basis of quantitative and qualitative qualification features.

The plant shift supervisor (PSS) must immediately notify the ChNPP personnel, responsible persons, regulatory authorities, external entities about accident origination/hazard.

The ChNPP emergency structures include the following:

- Site Emergency Work Manager (SEWM);
- coordination and management body – SEWM Headquarters;
- permanent management body;
- ChNPP emergency teams and groups.

The ChNPP emergency documents, emergency measures and actions ensure comprehensive approach accounting all the facilities located on the site. The possibility to ensure heat and power supply of the emergency facility from systems and equipment of operating facilities is envisaged.

Common systems for the nuclear facilities (NF) are the following:

- systems ensuring water inventory;
- integrated service water supply system;
- DG-2.

All the specified systems (except for DG-2) are redundant as regards performance of main functions at the NF upon availability of power supply.

Availability of backup busbars (RShA-1,2,3, RShB-1,2,3), sectioned with switches and reserve transformers (1,2,3TR), at each unit allows power supply all the NF from one of the sources: diesel power station, 1,2,3TR and site substation. In case of loss of external power supply and damage of RShA-3, RShB-3 or RShA-2, RShB-2, there is no possibility to supply units 1, 2 from the diesel power station and site substation – only unit 3 and ISF-1 will stay under voltage. In case of damage of RShA-3, RShB-3, there is no possibility to power unit 3 and ISF-1 from 1,2,3TR and site substation.

#### **6.1.2 Ch Possibility to Use Existing Equipment**

On 7 November 2011, the ChNPP conducted tests for connection of the mobile diesel generator station (MDGS) to ISF-1 responsible consumers. In accordance with the obtained results:

- MDGS delivery time – depending on access ways condition, 0.5÷7 hours;
- MDGS connection time ~ 40÷60 min.

Procedures for connection of alternative sources for cooling pool makeup are under development. The results and experience of full-scale exercises and training according to the stress test scenario conducted on 15 September 2011 (under the program “Multiple failures of regular systems and equipment in conditions of extreme natural phenomena”) during which recovery of ISF-1 makeup was ensured during 30 minutes by forwarding ISF-1 cooling

pool leakages to SFP sections without treatment (“small circle”) have been taken as the basis of these procedures.

At the ChNPP 75.6 tons of diesel fuel is available to ensure autonomous operation of two DGs during 3.5 days at full load, or operation of one DG during not less than 7 days. This time is sufficient to arrange uninterrupted diesel fuel supply.

Water inventory for ISF-1 cooling pool makeup is ensured by:

- pure condensate inventory in amount not less than 1500 m<sup>3</sup>;
- water inventory in amount not less than 500 m<sup>3</sup>.

The total reserve of pure condensate is 13500 m<sup>3</sup>.

Resupply of water inventory is performed from different design systems in required amount. Use of mobile water supply equipment is envisaged in case of failures of design systems.

### **6.1.3 Ch Evaluation of Factors That May Impede Accident Management and Respective Contingencies**

Damages of the infrastructure around the ChNPP (roads, bridges over the Pripjat river) complicating access to the site are possible during a high intensity earthquake. Restoration of the infrastructure and access ways will be performed in the framework of the measures of the accident liquidation, for implementation of which at the ChNPP there is sufficient time margin (several weeks) in accordance with spent fuel characteristics and state. The ChNPP NF envisages development of measures for arrangement of prompt delivery of emergency teams from Slavutych by alternative routes in case of failure of Slavutych – ChNPP railway because of an earthquake.

In case of strong flood the access to the site will be limited; however, the flood does not result in an accident on the site.

### **6.1.4 Ch Conclusions on the Adequacy of Organizational Issues for Accident Management**

One of the main factors impacting emergency management is loss of power supply and damage of pipelines ensuring the cooling pool level and spent fuel cooling. There is sufficient time margin (weeks) for taking compensating measures to ensure heat removal from spent nuclear fuel.

### **6.1.5 Ch Measures That Can Be Envisaged to Enhance Accident Management Capabilities**

The ChNPP main organizational structures shall perform all the activities on emergency planning, assurance of emergency preparedness and response in case of accidents and emergencies at the ChNPP. In its actions the operator shall follow “The ChNPP Plan for Response to Accidents and Emergencies” intended to decrease the level of radiation impact on personnel, population and environment in case of accidents or emergencies at the ChNPP facilities.

The ChNPP emergency organizational structures include the following:

- ChNPP Site Emergency Work Manager (SEWM);
- Coordination and management body – SEWM Headquarters or the ChNPP Facility Commission for the issues of emergencies;
- Permanent management body – Emergency Preparedness and Response Department;
- ChNPP emergency teams and groups.

For accident liquidation the operator has emergency units amounting to 293 persons. The ChNPP emergency units include general and special purpose emergency groups and teams.

General purpose emergency groups and teams:

- Engineering support group;
- Group for laboratory analysis of radiation situation;

- Individual exposure dose account group;
- Dosimetry survey group;
- Shelter dosimetry survey group;
- Chemical survey group;
- Logistics group;
- Group of information support and communication with external entities;
- Communication team;
- Transport and work mechanization team;
- Personnel sanitary treatment team;
- Protective structures maintenance team;
- Physical protection assurance group.

Special purpose emergency groups and teams:

- Team for localization and liquidation of accidents related with spent nuclear fuel;
- Team for localization and liquidation of accidents on electric equipment;
- Team for instrumentation damage liquidation;
- Team for liquidation of accidents on heat underground communications and water supply networks;
- Team for localization of accidents on heat mechanical equipment;
- Team for localization of accidents at the Shelter heat mechanical equipment;
- Team for localization and liquidation of accidents on hydrotechnical structures;
- Team for emergencies during transportation of radioactive materials.

The following formations, sub-divisions and military units can be involved into emergency response actions to support the ChNPP emergency personnel as well as to perform special tasks at the ChNPP facility sites, in sanitary protective zone and observation zone in accordance with the established procedure:

- Military Unit 3041 shall perform tasks of physical protection of the ChNPP, guard the area exposed to contamination, control access to the site within radiation contamination zones, ensure protection of material assets at the NPP territory and at external facilities;
- 1-SFPU (State Fire Protection Unit at the ChNPP) shall extinguish fires arising at the ChNPP facilities in accordance with the “Operating plan of fire extinguishing at the Chornobyl NPP Main Building”, “Operating plan of fire extinguishing at the ChNPP Shelter”, “Operating plan of fire extinguishing at the ChNPP”, and provide emergency rescue services by means of emergency teams and groups;
- Specialized medical sanitary units No. 5 (Slavutych) and No. 16 (Chornobyl) shall provide first medical aid, qualified and specialized medical aid to victims, arrange evacuation of victims for hospitalization in medical institutions;
- Personnel and resources of the Ministry for Emergencies, Administration of Exclusion Zone and Zone of Obligatory (Mandatory) Resettlement, Ministry of Fuel and Energy, Ministry of Defense, Ministry of Internal Affairs, Security Service of Ukraine, Ministry of Health.

Units of external entities arriving to the Chornobyl NPP to help in localization of the accidents and localization of its consequences shall be subordinated to SEWM and follow its directions.

Under an accident at the ChNPP facilities and during its liquidation the management of emergency activities on site shall interface with the State nuclear regulatory authority at two levels:

- at facility level – with the State Nuclear Safety Inspectorate at the ChNPP;
- at state level – with the Information and Crisis Center of the State Nuclear Regulatory Inspectorate of Ukraine.

***Emergency response means***

*MCR (main control room), CCR (central control room)*

- Monitoring of the cooling pool and control of safety-related systems during routine operation and accidents shall be performed from the MCRs of Chornobyl NPP units. Means to ensure accessibility and habitability of the MCR for operation under accident conditions are envisaged in the design.
- Monitoring and control of the ChNPP power grid and reserve bus ducts RShA, RShB for the possibility to back-up power supply of consumers of units in case of station blackout shall be performed from the CCR. Control of fire extinguishing and fire alarm systems of the units. Coordination of the personnel actions during accident management until activation of the Crisis Center is ensured.
- The Chornobyl NPP Internal Crisis Center.

The Internal Crisis Center is located within the protective structure No. 1 and is a center for management of emergency actions at industrial site of the Chornobyl NPP facilities and in the controlled area. From the Crisis Center, the SEWM and its Headquarters shall perform management of emergency teams and groups on monitoring and forecasting radiation situation, actions for personnel protection, communication with crisis centers of ministries and departments, emergency structures of local authorities and other entities performing emergency activities in the area of the Chornobyl NPP facilities.

#### *Activation of the Crisis Center*

The Internal Crisis Center will be activated upon announcement at ChNPP on a site accident, industrial accident or communal accident.

The Crisis Center is considered as activated after arrival of assigned emergency personnel and under condition of preparedness of the systems and equipment of crisis centers to ensure support of functions assigned to the Crisis Center personnel.

The SEWM is a person making decisions on activation of crisis centers.

Identification of an accident shall be performed on the basis of the following:

- Readings of measurement devices, alarm devices, technological process control systems of units and ChNPP;
- Results of radiation technological monitoring of protective barriers state and radionuclide content in technological media;
- Results of radiation dosimetry monitoring;
- Results of individual dosimetry monitoring;
- Results of environment radiation monitoring;
- Information about damage of fuel assemblies during performance of transportation and handling operations with fuel;
- Information from the personnel on explosions in premises and at industrial sites, leakages, ruptures and deformations of pipelines and on other visible signs of damage of equipment, buildings and structures of the ChNPP;
- Information on erroneous actions of personnel, on failures of equipment;
- Signals of automated fire alarm system, information of the personnel on smoke and fires in premises and at the ChNPP facilities;
- Information on hazard of extreme nature phenomena (tornado, flooding, earthquake, etc.);
- Information on theft of ionizing radiation sources;
- Information on terrorist acts on the ChNPP facilities;
- Information on other unforeseen events, during which non-clarified or significant decrease of the ChNPP facilities safety level occurs.

Survivability of the MCR, central control room, Crisis Centers and other premises of obligatory personnel presence shall be ensured by redundancy of power supply control systems of pumps, communication, as well as by availability of emergency lighting, ventilation systems.

Two protective structures are located at the ChNPP site area:

- Protective Structure No 1 under administrative building-1;



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- Protective Structure No 2 under administrative building-2.

Protective Structure No. 1 (on-site crisis center) is designed for 1500 persons and is provided with the following:

- double-system ventilation;
- power supply (power grid and DG);
- drinking and service water inventory;
- autonomous fire extinguishing system;
- individual protection equipment, devices for radiation survey and dosimetry monitoring, medicines.

Protective Structure No 2. is designed for 1500 persons and is provided with the following:

- double-system ventilation;
- power supply (power grid and DG);
- drinking and service water inventory;
- autonomous fire extinguishing system;
- individual protection equipment, devices of radiation survey and dosimetry monitoring, medicines.

The ChNPP personnel's duties include immediate notification of the Plant Shift Supervisor (PSS) on arising of conditions, which may lead to an accident. Visual observation of visible signs of an accident (water leakage, open fire, deformations and damage of equipment, building structures, contortion or rupture of pipelines, strange noises, etc.) is one of the main ways to reveal an accident along with monitoring using instrumentation and monitoring systems.

The ChNPP personnel are forbidden to take any independent actions for liquidation of an accident conditions not foreseen by the ChNPP instructions in force without notification of the ChNPP PSS and without knowledge of radiation situation.

Assembly of the ChNPP emergency groups and teams shall be announced immediately after implementation of the Plan.

In case of impossibility to transport the emergency groups' personnel from Slavutych to the ChNPP by electric trains, the Transportation Shop shall ensure transportation by motor vehicles.

The Emergency Work Manager at the ChNPP industrial sites gathers emergency teams:

- during working time – in 20 minutes after notification signal "Gathering of management";
- during off-time – in 30 minutes after notification (at the boarding place of the duty bus of the Transportation Shop in front of the Bus Station of Slavutych).

A representative of Slavutich Medical Sanitary Station-5 shall inform the SEWM Headquarters about the time of his/her arrival.

Assembly of emergency groups and teams shall be performed after the signal "Assembly announced for emergency teams, groups and management":

- during work time – after notification in the places of assembly of emergency groups and teams, near the places of storage of the emergency set, Attachment 16, 17 of the Plan Part III;
- during off-time – the bus stop in front of the Bus Station of Slavutych.

Time of getting ready for actions of emergency groups and teams:

- during work time – not more than in 0.5 hour after the notification;
- during off-time – not more than in 5 hours after the notification.

In case of decrease in the cooling pool level (Stage I) below the level controlled using regular level indicators and impossibility of visual monitoring (because of radiation factors or other hazardous factors), it is possible to control safe operation limit by the cooling pool level by absence of disruption of 1,2NV-1(2). For this purpose, it is necessary to assemble the circuit for cooling of the affected cooling pool and to control stability of its operation

periodically by short-term (not less than 1 min.) turning-on of the pump and monitor stability of its operation.

For Stage I, operation stability of 1,NV-1(2) indicates that the level in the cooling pool is not lower than the safe operation limit and, consequently, that there is no emergency.

For Stage II, operation stability of 3NB-1(2) indicates that the level in the cooling pool 3CP-1(2) is not lower than the cooling water intake piping, elevation 33.6 m.

The level in SFAP and FCP (Stage I) may be identified using regular manometers of SVO-2 (for this purpose, it is necessary to connect SVO-2 to the affected pool), or by readings of manometers on pumps 1,2NO-1(2), taking into account levels of their location.

For more accurate identification, it is necessary to replace the regular manometer with the one graduated in meters with measurement range of 0÷4 kgf/cm<sup>2</sup>.

The level in FCP of Stage II using the manometer on the treatment line may be determined only for 3FCP-1, because 3FCP-2 has a hydraulic lock on the water intake piping for treatment.

If decrease of level in Stage II cooling pools is caused by failure of all makeup systems, it is possible to perform makeup of 1,2 CP-1,2 and 1FCP using water inventory in 2FCP (~160 m<sup>3</sup>). For this purpose, it is necessary to assemble the circuit for treatment of 2FCP, turn on 2NO-1(2) and open valves for return from treatment into the required pool.

Filling of the cooling pool with nuclear fuel after its full emptying should be performed in accordance with a separate program with indication of measures for nuclear and radiation safety.

Fission products released from the damaged fuel assemblies into CP water or into unit air present the main radiation hazard under absence of CP level monitoring. If there are indications of damage of SFA boxes, as a preventive protective measure, it is expedient to start cooling pool treatment with connection of the ion-exchange part of SVO-2 (SVO-6 for Stage II) until pump failure, even in case lining leakage is detected.

The operator has a well-developed emergency response system to prevent stored fuel damage. However, if under subsequent loss of safety functions development of an accident reaches fuel damage phase, the operator implements the following measures:

- Continuation of spent fuel cooling and creation of heat removal conditions using regular, restored or survived systems of units and storage facilities, as well as “mobile” pipes (fire hoses, hoses, polyvinyl chloride pipes), pumps, MDGPUs.
- Existing Stage II DG have fuel inventory in amount of 75.6 t, which ensures autonomy during 3.5 days of two DG operation. Operation of one DG to power pumps will allow ensuring independent power supply (as well as heat removal from spent fuel the NF site) during more than 7 days.
- Permanent monitoring of neutron rate and gas-aerosol activity in places of spent fuel accumulation is performed for monitoring of nuclear safety. Ensure gadolinium makeup into the damaged CP in case of an emergency caused by SFA accumulation due to collapse of FCP consoles.

On 15 September 2011, the operator along with the Ministry of Emergencies conducted a full-scale exercise in accordance with the approved program. The ChNPP emergency preparedness was checked during the exercise. The exercise was performed on the basis of the approved action plan for targeted safety assessment and further safety improvement of ChNPP units 1, 2, 3 and ISF-1 to learn lessons from the Fukushima-1 NPP. The Chernobyl NPP personnel achieved the objectives and tasks of the exercise. Evacuation of 2000 persons to the Semikhody checkpoint and their boarding into railway transport took 1 hour 15 minutes. Measures for improvement of the Chernobyl NPP emergency preparedness system will be developed on the basis of the exercise results.

There is no risk of steam-zirconium reaction for the ChNPP (for other NPPs, the risk exists). A steam-zirconium reaction is not achievable, because the temperature of zirconium

should be not less than 1200°C (all water will evaporate by that moment) to start it. Radiolytic hydrogen is generated during “wet” storage of spent nuclear fuel. The main factors determining conditions for formation of explosive gas mixture on hydrogen basis are amount of stored spent fuel, water volume and air volume within in spent fuel storage place. The calculation of rate and volume of generated radiolytic hydrogen was performed for the most hazardous facility, which is the ISF-1. The amount of stored spent fuel at the unit is 12 times less and the volume of air space over spent fuel is 6 times higher than in the ISF-1. Duration of idle state of the ventilation system, at which achievement of the lower limit of hydrogen explosive concentration in the ISF-1 is possible constitutes ~10.3 days. Thus, acceptable reasonable ventilation system downtime is not more than 10 days.

#### **6.2 Ch Accident Management Measures to Restrict the Radioactive Releases Accident Management Measures in Place at the Various Stages of a Scenario of Loss of the Core Cooling Function**

For Units 1, 2, 3, ISF-1 the main measure to restrict releases is restoration of the rated level in the FCPs, assurance of special ventilation operation at units and possibility to connect additional filtering cells the ISF-1.

Measures on maintenance of water level in the cooling pool and prevention of uncovering of spent fuel upper part are considered in Section 5.2.3 and 5.3.3.

## **7 Ch GENERAL CONCLUSIONS**

### **7.1 Ch Key Provisions Enhancing Robustness (already implemented)**

#### **7.1.1 Ch Robustness of ChNPP Units 1, 2, 3 and ISF-1 against External Hazards Such as Earthquakes, Hurricanes, Floods, External Fires, Extremely Low and High Temperatures**

Category 1 structures ensuring nuclear and radiation safety of ChNPP units 1-3 and ISF-1 comply with the design documents and are capable of withstanding external hazards including MCE with intensity of 6 and F 1.5 tornado.

Mutual redundancy (interfaces) of the ChNPP for power supply, fire and service water supply and makeup are ensured to maintain (or restore the possibility to maintain) safety functions in case of design-basis accidents caused by extreme natural impacts at units 1-3 and ISF-1.

The system for collection and return of leakages from under the ISF-1 cooling pool lining and maintenance of the acceptable water level in the CP under any design-basis accidents was upgraded at ISF-1. The upgraded system will allow collection, treatment and return of ISF-1 CP leakages during design-basis accidents with loss of integrity of CP lining. Implementation of this measure will sufficiently increase the time for removal of CP compartments for repair due to loss of integrity.

#### **7.1.2 Ch Assurance of Nuclear Fuel Safety under Loss of Power and Ultimate Heat Sink**

Nuclear safety of spent fuel storage within ISF-1 CP compartments in normal operation, operational occurrences and design-basis accidents is ensured.

All spent nuclear fuel on-site is cooled for more than 10 years. The nuclear physical properties of spent fuel storage places are such that boundary effects (spent fuel heating to 300°C with possible loss of spent fuel integrity) in case of loss of safety functions resulting in absence of water in CP do not emerge for a long time (about 2 months).

After loss of the ultimate heat sink (absence of water cooling in CP), it is evaluated that water in CP is heated to temperature not higher than 70°C with the largest amount of stored spent fuel (ISF-1).

The following estimations of hydrogen concentration increase within ISF-1 were obtained for failure of the ventilation system:

- design one with conservative suggestion – the lower limit of explosive concentration of 4% will be achieved in 10 days;
- experimental one – 0.2% equilibrium concentration of hydrogen in 20 hours, which does not increase.

The time of restoration of ChNPP power supply, upon integrity of the infrastructure, is not more than 2 hours. Upon damage to the infrastructure, the time of delivery of power supply to the systems ensuring maintenance of the level in the ISF-1 CP and operation of the ISF-1 ventilation systems is not more than 8 hours.

In terms of conservatism, the time for restoration of heat removal from spent nuclear fuel is not less than the time of restoration of power supply to the systems ensuring CP feeding and is not more than 8 hours. The systems of water supply to CP can be restored in 30 minutes by directing ISF-1 CP leakages to CP compartments without purification (“small circuit”).

Autonomous operation of DGs for ensuring heat removal from spent fuel during more than 7 days is ensured.

### **7.1.3 Ch Measures on Management of Accidents with Severe Consequences**

The Chernobyl NPP possesses the system of emergency preparedness and response ensuring mobilization of the main and additional forces and equipment.

It is proposed to use symptom-based instructions focused on implementation of preventive measures along with event-based emergency instructions. Such an approach will allow defining actions of personnel at the early stage of accident development, when data allowing to evaluate the accident development and to reveal its cause are absent, but there is information necessary to make decisions allowing confining or reducing the accident consequences.

The Chernobyl NPP takes measures to connect a mobile DG to ISF-1. This technical measure significantly enhances the operator capabilities to manage accidents with severe consequences.

Organizational technical measures ensuring the return of CP leakages without purification (“small circuit”) in case of destruction of external service lines are performed for ISF-1.

Analysis of management of accidents with severe consequences shows that reduction of the number of nuclear facilities will allow more efficient use of available resources, both human and technical ones.

## **7.2 Ch Safety Issues**

### ***Earthquake and Tornado***

For the ChNPP site, extreme natural phenomena such as extreme wind, snow, temperature, precipitation, floods and external fire are much less dangerous than earthquake or tornado. For this reason, the operator within this targeted safety assessment considers an earthquake and tornado as an initiating event resulting in loss of safety functions (power supply and heat removal).

As of today, the operator has not performed:

- analysis of safety margins of nuclear facility structures under earthquakes higher than MCE;
- analysis of safety margins and possibility of CP lining failures (primarily ISF-1) under MCE and higher impacts;
- seismic assessment of equipment and service lines (e.g., CP cooling and level control);
- resistance analysis of ChNPP category I structures, possible failures of structures upon F 3.0 tornado.

However, it should be mentioned that the above measures are envisaged within the safety improvement plan of the ChNPP.

***External Floods (as a result of accidents at hydro and technical structures)***

ChNPP resistance has been confirmed. Safety deficiencies have not been revealed.

***Other Natural Effects (external fires; extremely high/low temperatures; extreme wind; extreme snow; combination of external extreme effects)***

ChNPP resistance has been confirmed. Safety deficiencies have not been revealed.

***Loss of power supply and/or heat removal for ultimate sink***

Loss of power supply to units 1, 2, 3 and ISF-1 does cause accidents with damage of nuclear fuel. Designs of nuclear facilities envisage power supply backup, and layout and inter-unit communications ensure high reliability of the systems powering the nuclear facilities. Along with this, the characteristics of stored spent fuel are such that hydrogen safety is ensured and there is a sufficient time reserve to restore the safety functions lost during an accident.

***Management of severe accidents***

The Chornobyl NPP possesses the emergency preparedness and response system, which ensures mobilization of the main and additional forces and means. The ChNPP safety improvement plan includes measures to improve emergency preparedness and response.

**7.3 Ch Potential Safety Improvements and Further Work Forecasted**

The measures to improve resistance of nuclear facilities to external hazards, ensure safety of nuclear facilities in loss of power and/or ultimate heat removal, as well as additional measures to manage accidents and mitigate their consequences are specified in the "Safety Improvement Plan for ChNPP Nuclear Installations" agreed with the State Nuclear Regulatory Inspectorate of Ukraine.

In particular, the ChNPP Safety Improvement Plan includes the following measures:

- research for structures of category I responsibility for nuclear and radiation safety to identify a strength margin and probability of a failure under a tornado of class 3.0 and under seismic impact;
- research of seismic resistance of the ISF-1 cooling pool lining;
- analysis of durability and probable failure of ventilation stack-1 under seismic impacts and tornadoes;
- justifying calculations for the maximum temperature of SFAs taking into account probable radiation consequences at SFA "wet" storage;
- development of an Action Plan for Improvement of the Emergency Preparedness System for the cases of beyond design accidents caused by extreme hazards, including actions for emergency response at the scenario with a building collapse and CP depressurization;
- development of actions for fast transportation of emergency brigades from the Slavutich town by alternative ways in case of Slavutich-ChNPP railway disability due to an earthquake;
- reconstruction of radiation monitoring system;
- development of actions to restore operability of equipment ensuring the spent fuel on-site transportation damaged in case of an earthquake and tornado;
- procurement of a new car-cask for spent fuel transportation;
- improvement of the personnel training program under emergencies.

The ChNPP nuclear facilities were provided with a mobile diesel generator in November 2011.