NATIONAL REPORT

Stress Test for Armenian Nuclear Power Plant

Armenian Nuclear Regulatory Authority

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Armenian Nuclear Regulatory Authority
#4 Tigran Mets
Yerevan 0010, Republic of Armenia
Tel/fax: +374 10 543991
E-mail: info@anra.am
Web-site: www.anra.am
Prepared by Nuclear and Radiation Safety Center (www.nrsc.am)

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EXECUTIVE SUMMARY

Following the nuclear accident at the Fukushima nuclear power plant on 11 March 2011, the Armenian Government emphasized the need for urgent actions to reassess the preparedness of Armenian Nuclear Power Plant to respond to emergencies. In June 2011, ANRA required the conduct of an in-depth reassessment of the ANPP safety in the light of Fukushima accident (stress-tests), which should be in conformity with the methodology adopted by the ENSREG and the EC.

According to the ENSREG methodology the “stress tests” should summarize plant response and effectiveness of preventive measures in extreme situations by identifying any potential weaknesses and cliff-edge effects. This is necessary to assess the robustness of the defense-in-depth approach applied, the adequacy of existing accident management measures, as well as to identify potential technical and organizational (procedures, human resources, organization of emergency response or use of external resources) improvements to enhance the Armenian NPP safety. In this respect, stress tests objectives could be summarized as:

- Defining measures provided by the plant design and plant compliance with the design requirements with regard to external hazards;
- Defining plant capabilities to respond to beyond design basis events, namely assessing plant robustness and identifying potential weaknesses;
- Defining measures to improve the existing level of robustness of SSCs in order to improve the overall plant robustness against extreme natural phenomena.

The present report has been prepared on the basis of the Final ANPP self-assessment report. ANPP self-assessment covered Unit №2 and the wet spent fuel pools of Unit № 1 and Unit №2 in full compliance with the ENSREG specifications. ANPP was supported by EC support project to implement the self-assessment and prepare the report.

Based on ANPP stress test report ANRA with support of its TSO and EC technical support project elaborated national Stress Test report. ANRA has proposed safety improvement measures additional to those proposed by the licensee.

The first part of the report provides the main data for the site and ANPP with special emphasis being made on its characteristics. The design bases of ANPP, reassessment of safety margins and cliff edge effects are provided in the chapters 2-6; each chapter summarizes the proposed measures to improve plant robustness to extreme natural phenomena. In the chapter 7 conclusions and recommendations are summarized. Plant data and reassessment provided in report are as of July 2012.

ACRONYMS

ANPP Armenian Nuclear Power Plant

ANRA Armenian Nuclear Regulatory Authority

BCC Backup Crisis Centre

BDBE Beyond Design Basis Earthquake

BWST Borated Water Storage Tank

CD&ES Civil Defense and Emergency Situations

CDFM Conservative Deterministic Failure Margin

CNIISK Central Building Research Institute of former USSR

CP Cooling Pump

CSBO Complete Station Blackout, i.e. LOOP and loss of all back-up AC power-sources

CSG Complete Switchgear

CVS CKTI Vibroseism

CWPH Cooling Water Pump House

DAP Emergency Diesel (feedwater) Pump

DBE Design Basis Earthquake

DC Direct Current

DFP (NZS) “Dirty” Flows Pump

DG Diesel Generator

DGLS Diesel Generator Load Sequencer

DMWT Demi-water Storage Tank

DOE Department of Energy

DP Drainage Pump

DTP Drain Tank Pump

EC European Commission

ECCS Emergency Core Cooling System

ECP Emergency Condensate Pump (AKN)

ECR Emergency Control Room

EDG Emergency Diesel Generator

EFWP Emergency Feed Water Pump

EFWSP Emergency Feed Water Seismic Pump

EMP Emergency Make-up Pump

EMS Emergency Monitoring System

EOP Emergency Operating Procedures

EP (AZ) emergency protection

EPSS Emergency Power Supply System

ERG Emergency Response Guidelines

ERT Emergency Response Team

ESWP Essential Service Water Pump

ESWS Essential Service Water System

FFWP Fire Fighting Water Pump

FFFP Fire Fighting Foam Pump

FRS Floor Response Spectra

HAEK CJSC Acronym of the organization operating the ANPP

HCLPF High Confidence Low Probability of Failure

I&C Instrumentation and Control

IAEA International Atomic Energy Agency

IEP Internal Emergency Plan

IVR In-vessel (debris) Retention

LBB Leak-Before-Break

LOCA Loss of Coolant Accident

LOOP Loss of Off-Site Power

MB Main Building

MCC Main Crisis Centre

MCCI Molten Corium Concrete Interactions

MCE Maximum Considered Earthquake

MCL Main Circulation Loop

MCP Main Circulation Pump

MCR Main Control Room

MES Ministry of Emergency Situations

MLIV Main Loop Isolation Valve

MP Make-up Pump

MRZ Russian abbreviation for the maximum design earthquake (analogue of Safe Shutdown Earthquake, SSE or SL-2)

MSIV Main Steam line Isolation Valve

MSL Main Steam Line

NFMS Neutron Flux Monitoring System

PAMS Post Accident Monitoring System

PAR Passive Autocatalytic Recombiner

PCR Plant Control Room

PGA Peak Ground Acceleration

PMP Potassium Metaborate Pump

PNAE Russian nuclear standard

PORV Power Operated Relief Valve

PRV Pressurizer Relief Valve

PSA Probabilistic Safety Assessment

PSHA Probabilistic Seismic Hazard Analysis

PWR Pressurized Water Reactor

PZ Pressurizer

PZ Russian abbreviation for design earthquake

RA Republic of Armenia

RDGB Reserve Diesel Generator Building

RLE Review Level Earthquake

RMG Reversible Motor Generator

SAR Safety Analysis Report

SAMG Severe Accident Management Guidelines

SBO Station Blackout

SDV-A Steam Dump Valve to the Atmosphere

SDV-C Steam Dump Valve to the Condenser

SEC Second Emergency Cooling

SFDSF Spent Fuel Dry Storage Facility

SFP Spent Fuel Pool

SFP CP Spent Fuel Pool Cooling Pump

SFP CSP Spent Fuel Pool Cleaning System Pump

SFP HE Spent Fuel Pool Heat Exchanger

SFP MP Spent Fuel Pool Make-up Pump

SG Steam Generator

SG AFS Steam Generator Auxiliary Feed water System

SG FWDP Steam Generator Feed Water Diesel Pump

SG SV Steam Generator Safety Valve

SMA Seismic Margin Assessment

SNiP Russian civil code

SPS Seismic Protection System

SSC Structures, Systems, and Components

SSEL Safe Shutdown Equipment List

SS HE Spray System Heat Exchanger

TACIS Technical Assistance to the Commonwealth of Independent States

UDCB Unit Direct Current Board

UPS Unit Pump Station

VSN Temporary Russian civil code

WWER Water (moderated) Water (cooled) Energy Reactor

1. GENERAL DATA ABOUT THE SITE AND PLANT

1.1. Brief Description of the Site Characteristics

The ANPP site is situated in the western part of the Ararat Valley, 10 km northeast of the regional center of Armavir, in 28 kilometers west of Yerevan and 16 km from the border with Turkey. The site is surrounded by the mountains from the north and northwest, from the east and south by the Zangu irrigation channel.
The ANPP site has a slope of 1.5% to 5% to the south towards the Ararat Valley. The zero level of the site corresponds to the 934.5 meters above the sea level.
Both in the subregion (R=25 km) and in the site area (R=5 km) there are no any inflammable and explosive materials accumulated, as well as any industrial facilities observed that could be potential sources of explosions and fires.

The ANPP site is connected to the transportation network by three roads (from north, east and south-west) and one railway line from the southwest, which reduces the probability of simultaneous loss of all access roads during extreme natural occurrences.

The geological structure of the ANPP site area is characterized by a wide expansion (up to 200-300) complex of fractured lavas and slags, volcanic emissions, and tuffs. Intersection and hilly terrain, good water permeability of lava and wide distribution on the surface of block lounging create favorable conditions for infiltration of precipitation into the depths. Water infiltrating into lava on the slopes of the Mount Aragats massif, reaches a relative water pressure move further to the east - southeast side of the Ararat Valley, forming the subterranean streams. The groundwater level at the site of an underground stream on the ANPP site lies at the depths 86-95 m from the surface. The ground water streams flow into the lake Aygrlich and the Kulibeklinsky and Aygrlich springs at the level 850-840m above the sea level, forming a flow channel of the river Sevjur.

The river Sevjur is a source of service water supply of the ANPP. Drinking and fire water supply of the ANPP is provided from the spray pond, built on the left bank of the river Sevjur. Water intakes are located at 6-6.5km distance from the ANPP site. All waterways and ponds existing at the site area are located south of the site at a considerable distance and well below the site.

The ANPP site is characterized by two main features: seismic and remote source of service water supply.

1.2. Main Characteristics of the Unit

The ANPP consists of two identical units with VVER-440 (V-270) type reactors, which are two-circuit reactors on thermal neutrons. VVER-440 has been conceived as twin units, in mirror spatial arrangement. A part of equipment and systems is common for both units. Among the common part of systems and structures are reactor hall, refueling machine, spent fuel transport, radioactive waste handling, receipt, storage and transport of fresh fuel, vent stack, access to controlled area, demineralized water treatment system, service water system, cooling water system, diesel generator building.

The Unit №1 is in conservation mode, there is no fuel in the core. The cooling pond of the Unit №1 is used to store spent nuclear fuel of the Unit №2 in operation.
The primary circuit of each unit has six loops, two isolation valves on each loop and horizontal steam generators with large coolant volume on secondary side of the steam generators. The reactor core is composed of 349 hexagonal fuel assemblies with 126 fuel rod positions each. 37 control rod assemblies have fuel followers underneath their neutron absorbing parts so that efficiency of scram is increased by removal of the part of fuel from the core together with the insertion of the absorber rods. Those units use two steam turbines. Electricity is generated in main synchronous generators on a common shaft with turbine and excitation generator.

The reactor installations of each unit are located in separate containments of the reactor building and the turbine generators - in the common turbine hall. The design thermal output of each reactor is 1375 MW, and the electric power of the Unit №2 in operation is 407.5MW.

ANPP design positively differs comparing with standard VVER-440/230 units.

The site seismicity (magnitude 8 according to MSK-64 scale, see more in chapter 2) caused root design improvements related not only to structures but also to the safety related systems leading to placing new serial number V-270.

The ANPP was the first plant in the Soviet Union built in an area with high seismic activity. The structures of the reactor building, DG station were designed taking into account the high seismicity of the site. The following differences from the design V-230 had been implemented at the design stage and in the first years of operation:

- SGs are designed take into account the seismic loads,
- MCP similar to V-213 type with fly-wheels were installed,
- to reduce the seismic loads to the systems and supporting structures from seismic impact, the main equipment of the primary circuit (SG, MCP, isolation valves, Pressuriser surge lines) are additionally fastened by hydraulic snubbers;
- Additional fastening of the upper part of the Pressuriser with support rods;
- The industrial anti-seismic protection system is installed which includes additional check valve installed on pipelines of the seismic category 1 of the primary and secondary circuits and the cooling water systems, which is closed during seismic event,
- Annular dry tank, filled with serpentinit instead water (V-213 type),
- Walls instead of columns in the SG box,
- Additional “antiseismic” emergency feedwater system located in the boron compartment,
- Additional secondary cooldown system in the boron compartment is installed.

The design of structures is essentially differ from the design of V-230. The transversal stacks of electrical equipment are cut-off from the reactor hall structures. The carcass frames of the turbine hall, electrical equipment and roof of the reactor hall above the elevation 10.5m are made in structural steel, roofing coating is made with profiled steel sheet and light sealings. Construction of the rigid part of the reactor hall, exhaust ventilation center, radwaste building, cable tunnels are made in reinforced concrete. For design of reactor building, boron compartment, radwaste building the seismic load was taken by one point higher than the geographic seismicity of the site and additional multiplying factor γ=2,7 was applied. For the premises of reactor hall and radwaste building where radioactive liquids are possible to store for a long-term, the lining is made of stainless steel, i.e., they are performed on a “tank in the tank” principle.

In 1982 a major fire occurred at ANPP which initiated safety re-assessment and implementation of safety improvements:
- installation of additional shutdown control panel;
- installation of additional emergency power cable network which can feed the safety-related 6kV and 0,4kV equipment;
- separation of power cables for redundant safety-related equipment;
- destroyed cables were replaced with larger capacity (oversized) cables;
- and many other fire protection improvements were introduced.

The nominal values of the main technological parameters of ANPP Unit №2 are specified in the Table 1.2-1.

Table 1.2-1. Main technological parameters of ANPP Unit 2

Parameter	*Value*
Reactor thermal power	1375MW
Primary circuit coolant flow rate	42000m^3/h
Pressure in the primary circuit	12.26MPa
Coolant temperature on the reactor inlet	267°C
Reactor coolant average heating	29.3°C
Shutdown concentration of boron acid in primary coolant	12.00 g/kg
Main steamline pressure	4.5MPa
SGs steam flow rate	2700t/h
Maximum linear load of fuel cladding	325W/cm
Quantity of fuel assemblies in the core	349

The ANPP is connected to the power grid of Armenia by five power transmission lines of 220 kV and six power transmission lines of 110kV. Through the aforementioned lines the ANPP is connected to five nodal substations. The lines 220 and 110kV is wired on the switchboard buses 220 and 110kV of the station. The switchboard busses 220 and 110kV are interconnected by the coupling autotransformers. The operating power of the distribution devices in the ANPP is accomplished by two operating transformers 15.75/6kV. The power of each transformer is 25MVA. The redundancy is provided by two reserve transformers 110/6 kV and 35/6kV. The capacity of each transformer is 32MVA. The presence of two switchboards with autotransformer connection provides a reliable connection to the power grid.

Spent fuel pools separate for each of the units are located adjacent to the reactor vessels. Spent fuel is cooled at least 5 years in the spent fuel pool in a storage grid in a pool filled with borated water. Fuel is stored in a storage grid in vertical position enabling cooling by circulation of boric acid solution with concentration corresponding to requirements derived from neutron-physical characteristics of fuel. The storage grid consists of hexagonal grid in which spent fuel assemblies or leak tight canisters (for assemblies with damaged cladding) are inserted. There are two grids placed in the pool. The lower (operating) grid is fixed, the upper grid (reserve) is removable, and common for both twin units. The basic grid has capacity of 359 spent fuel assemblies and 13 hermetic cases for untight fuel. In case of short-term storage of fuel assemblies transported out of the reactor during inspections and repairs of the reactor internals, a reserve storage grid is used. It is placed above the basic grid, and it can accommodate 351 fuel assemblies.

The pool, which is open during refueling, is connected through a transport passage to the refueling pool (the area above the open reactor). Outside of fuel manipulation periods, the top of the spent fuel pool is covered and it is isolated from the refueling pool by a slide gate that blocks the transport passage. This gate forms part of the hermetic confinement boundary during operation.

In 1997-2000 the first phase of a long-term dry spent fuel storage facility (DSFS) was constructed on the basis of the NUHOMS concept at the ANPP site. Fuel is shipped in canisters (56 assemblies in each canister) and placed in the dry storage facility after storing in the cooling pond at least 5 years. The design and construction of the second extension of DSFS was initiated in 2006. The construction was completed in 2012. The second extension is now filled to 84% percent. The residual heat of the spent fuel canisters in the storage facility is removed by natural air circulation.

1.3. Description of the systems for implementation of main safety functions

Reactivity control

Reactor. VVER-440 (type V270) reactor type assumes two independent reactivity control methods. In proposed fuel cycles, total core reactivity inventory in cold non-poisoned condition at the beginning of the campaign beginning equals to 14.57-15.19% depending to the configuration of fresh and burned fuel assemblies. To compensate this reactivity inventory, and to control the reactor, there are two independent systems affecting reactivity based on various physical and technical principles:

- reactivity control using the control rods (mechanic method),
- reactivity control by boric acid concentration change (liquid absorber) in RCS coolant.

The mechanical reactivity control system is used for compensation of fast reactivity changes during the campaign, reactor power regulation, reactor trip, gradual reactor shutdown and reactor maintaining in sub-critical conditions. The system is based on vertical movement of absorber (with fuel followers) in the reactor core. It consists of 37 control rods connected with drives installed in the upper reactor block via interconnection rods. The control assemblies function by gravity and reliable reactor shutdown by their insertion can be assumed. The assemblies themselves are sufficient for maintaining sub criticality of the core without injection of borated water except cold shutdown state. Insertion of the control rods to the core causes displacement of the fuel part of the control rods out of the core and insertion of the upper absorption part to the core. Volume of fission material in the core decreases and absorber volume increases, thus attenuating the fission reaction (introduction of negative reactivity).

***Reactivity control by boric acid concentration increase (liquid absorber) in RCS coolant*.** The system affecting the reactivity is based on absorber concentration in RCS coolant (boric acid being the coolant). The systems implementing this function are as follows: Emergency core cooling system (ECCS), Primary side make-up system and a system for boron solution injection.

High pressure emergency core cooling systems (HP ECCS)

HP ECCS is classified as safety system and is capable of injecting boron concentrate from the emergency cooling water storage tank (boron tank with concentration of 12g/kg) to the reactor core in all operational modes including accidental situations thus increasing the boric acid concentration in the primary circuit to level needed for reaching the sub-critical state in the core. Considering flow rate of one HP ECCS pump, required RCS boron concentration can be expected 30 minutes

Primary side make-up system and Boron system

The primary circuit make-up system (the system important to safety) includes four plunger pumps and performs the function of boron control during normal operation modes as determined in the technological specification for ANPP Unit №2 operation, such as full power operation, “hot stand-by”, minimum controlled level by providing transition of the reactor and maintaining it in subcritical condition. Considering the flow rate of four makeup pumps, required boron concentration for safe shutdown can be achieved within 15 minutes using the boron water with concentration of 40g/kg stored in boron tanks. The boron solution preparation and injection system (important to safety) is designed for preparation of a boric acid solution with the required concentration and for injection it to the primary circuit make-up system. Additionally, the system provides with injection of the high concentration boric acid 40g/kg to the spent fuel storage ponds. Conservatively, about 8m^3 of high concentration borated water is sufficient to ensure subcriticality even for the case of natural circulation of coolant in the primary circuit. H3BO3 inventory in the boron system tanks is sufficient for provision of required core sub-criticality in any operational regime and in any fuel campaign time point.

Additional systems in ANPP design

As the unit has to be evaluated also with regard to beyond-design basis events, in provision of the safety function “Core sub-criticality” also are considered the following systems exceeding beyond their standard design use:

- Provision of boron concentration 40g/kg from the system of preparation and injection of boric acid solution in the emergency boric acid tank (B-8/2) is provided for at the ANPP.

- The following pumps can be used to inject boric acid from the emergency boric acid tank (B-8/2) in the primary circuit: NZB (spent fuel make-up pumps) and NBO (SFP purification pump). This configuration can be used only in the conditions when the reactor is connected with the cooling pond.

Spent fuel pool (SFP)

Sub-criticality in the SFP is provided in two independent ways:

- Geometry of the storage grid – fuel assemblies are placed in a triangular mesh with a pitch of 225mm
- Minimal boric acid concentration of 12g/kg.

In normal operational conditions, spent nuclear fuel (in operational conditions: COLD SHUTDOWN and REFUELLING also fuel temporarily unloaded from the core) is stored in the spent fuel pool racks and is cooled with borated water solution with concentration of 12 g/kg. According to the current regulatory requirements sub-criticality in SFP must be min. 0.05 (K_{ef}< 0.95) without taking credit of boric acid in the coolant and with filling the grids with fresh nuclear fuel assumption.

The requirement – K_{ef}< 0.95 – is fulfilled in normal operational conditions of SFP rack, at accident with complete failure of NPP power supply accompanied by coolant circulation interruption in SFP rack resulting in change of coolant density. The requirement on K_{ef}<0.98 is fulfilled also for beyond-design basis accident accompanied by complete water discharge from SFP.

Analyses results prove that the sub-criticality of the spent fuel pool is ensured even the fuel is covered by non borated water, or partially covered with H_3BO_3 solution with concentration at least 12g/kg for the whole range of coolant density. Fuel sub-criticality in SFP is always guaranteed when heat removal is provided.

Heat transport from reactor to UHS

At deviation from normal operation for heat removal from the core following systems are used:

- Safety systems:
  - steam generators safety valves system and SG steam dump system to atmosphere (BRU-A)
  - High pressure emergency cooling system of reactor (ASN)
  - Low pressure emergency cooling system of reactor (AKN)
  - High-pressure emergency core cooling system.
- Auxiliary systems:
  - Emergency feedwater diesel driven pumps (DFWP).
  - AFWPs (auxiliary feedwater pumps) located in Turbine Hall.

Beside the mentioned systems normal operating systems could also be used (e.g. main feed water system, residual heat-removal system) – see Fig. 1.3-1. However their operation is not taken into account within the current investigation because of their vulnerability to external hazards.

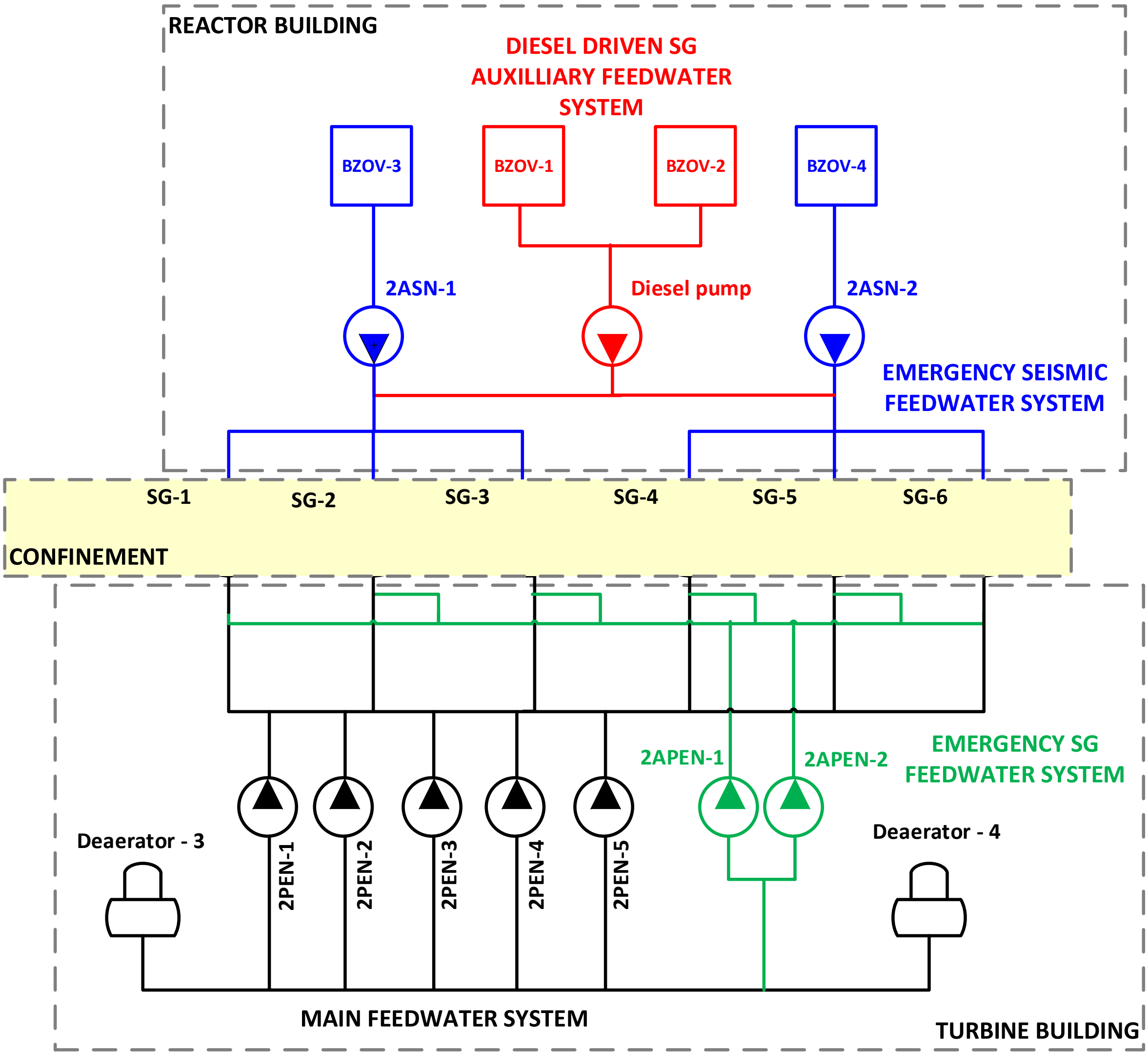


Figure 1.3-1. Principal scheme of high pressure SG feedwater systems

Steam generators SVs and steam dump station to atmosphere (BRU-A)

In case of heat removal via BRU-A or SG SV the secondary circuit is not closed. Heat removal on RCS side is performed in natural or forced circulation. Heat transfer from RCS to SC takes place in SG subject to maintaining sufficient water level in SG. Water is supplemented by DFWP, AFWP or ASN and steam can be removed from SG via the BRU-A or SG SV. In case of steam removal via BRU-A, water can be supplemented to SG also by normal or auxiliary SG feeding systems. All mentioned cases represent secondary side Feed &Bleed strategy.

High pressure emergency cooling system (ASN)

The high pressure emergency core cooling system is designed for high-pressure water supply to SG in case of failure in water supply to the SG from the systems in the turbine hall (SG normal and emergency make-up water system). The high pressure emergency cooling system consists of two identical channels that are completely independent. Each channel provides for a seismic resistant pump - ASN (nominal pressure of 56kg/cm^2 5.5 MPa), which takes in demineralized water from the corresponding tank BZOV-3 or BZOV-4 and supplies to SGs 1÷3 (channel 1) and SGs 4÷6 (channel 2) respectively. The main equipment of the system is located in the boron compartment of the Unit №2 (seismic safety category I), BZOV tanks are located outside the main building (has seismic classification level 1 after strengthening and re-evaluation made in 2011).

The high pressure emergency cooling system is actuated automatically in case of seismic event at actuation of the industrial antiseismic protection system interlocks; in other cases the system is activated manually by the operator from the MCR-2. The emergency make-up water system components are consumers of the second category of reliable electric power supply and there is ability to supply power both from the regular power supply sources, and from the DGs.

Low pressure emergency cooling system (AKN)
The low pressure emergency core cooling system is designed to maintain the parameters of the primary circuit through the secondary circuit at the low pressure (no more than 5 kg/cm^2) in the main steam header (MSH) and the residual heat removal of the reactor core. Low pressure emergency core cooling system consists of two identical channels. Each channel includes an emergency condensate pump (AKN) with nominal pressure of 1.2 MPa, which pumps cooling water in a closed loop: the emergency condenser (AK) – AKN – SGs 1÷6)-MSH-AK. Service water to AK is provided from the Essential Service Water System. There is also possibility to supply water to the emergency condenser pump from the BZOV-3, 4 located outside of the reactor building (has seismic classification level 1 after strengthening and re-evaluation made in 2011). The main equipment of the system is located in the boron compartment of the Unit №2 (has seismic classification level 1). The reactor low pressure emergency cooling system is activated manually by the operator from the MCR.

The system components are consumers of category 2 reliability and are capable to supply power both from the regular power sources and from the DGs.

High-pressure emergency core cooling system can be used for heat removal from the core in the primary Feed & Bleed regime. Cold coolant is injected to RCS by HP ECCS pumps. Hot coolant is removed from RCS via PRZ SV to the confinement. After condensation of expanded RCS coolant in the confinement using the containment spray system, the coolant is drained by gravity force to the emergency coolant tank from the confinement floor and supplied via the confinement sprinkler system (CSS) heat exchanger by HP ECCS pumps back to the primary circuit and to the spray system nozzles. Heat is removed in the CSS heat exchangers. Cooling medium is water supplied from Essential suppliers cooling system. ECCS has the seismic classification level 1.

Additional systems

Emergency feedwater diesel driven pump
Additional feedwater system of steam generators is designed to supply demineralized water to steam generators during total loss of power to the unit when diesel generators are inoperable, or in emergency situations with failure of all normal and emergency SG make-up systems. System is a single channel system. The system consists of diesel pump (DNP SG) which takes in demineralized water from the tanks BZOV-1.2 located on the outside of the main building (seismic classification level 1) and supply it to the steam generators through

the make-up header of the emergency cooling system of high and low pressure. The main equipment of the system is located in the boron compartment of the Unit №1 (seismic classification level 1).

Additional feedwater system of steam generators is activated manually by the operator. To start-up the DNP SG two groups of autonomous batteries have been installed, one of which is “operating”, the second one is “reserve”. Supply of diesel fuel for the diesel pump provides with the diesel pump operation within one day.

Emergency feedwater system

The emergency feedwater system is designed to supply water into the steam generators during off-site power lose and/or failure of the normal feedwater system. The system consists of two channels with suction from feedwater deaerators. Each channel has one pump (APEN) which takes water from two deaerators and supplies to the steam generators by the emergency feedwater pipeline through emergency level controllers of SGs. The main equipment of the system is located in the turbine hall.

SG emergency feedwater system is actuated automatically at off-site power loss. In other cases the system is activated by the operator manually from MCR-2.
The emergency feedwater system components are consumers of the second category of reliability and can be powered also both from the regular power supply sources and from the DG.

Heat transfer from SFP to the ultimate heat sink

Heat is removed from the spent fuel cooling pond by spent fuel cooling system which consists of two channels connected by the suction and discharge collectors. Each channel provides for the cooling pump (NRB), which supplies cooling water in a closed circuit: cooling pond heat exchanger (TOBV) - pump – cooling pond - TOBV. Water is cooled by service water supply from the essential service water cooling system to TOBV. The main equipment of the system is located in the reactor hall (and has seismic category level 1).

One NRB pump of the cooling system is permanently operated. When disconnecting the operating pump, the second channel of the system is connected automatically, in case of failure of automatics it is activated by the operator.

Besides the mentioned system the cooling pond can be supplied from the boric acid storage tank (B-8/2) by NZB pumps (cooling pond filling pump) and NBO (boric purification pump) can be used. According to the subcriticality calculations the cooling pond can be filled by water without boron. Given this, it is also possible to make-up the cooling pond with fire carriages.

Alternating power supply sources

Off-site power supply

Connections of the plant with external power grid

The ANPP is connected to the power grid of Armenia by five power transmission lines 220 kV and six power transmission lines 110kV. Through the above-mentioned lines, the NPP is connected to the five nodal substations. The lines 220 and 110 kV are wired on the switchboard buses 220 and 110kV of the ANPP. The switchyard buses 220 and 110kV are interconnected by the connection autotransformer.

Back-up power supply sources
There are transmission lines 220 and 110kV that assure redundant power supply to 220 kV and 110 kV switchyard. The power supply is provided both from hydroelectric and thermal power stations. Power can be supplied to the sections 6/0.4kV of house loads both by the regular circuit through the main transformers 3T, 4T and tapped transformers 23T, 24T, and through the redundant circuit by the start-up transformers 1TR or 2TR.

All diverse sources that can be used for the same task as the main back-up sources
In accordance with the "Regulations of grid dispatching service of the RA" in the case of total blackout of the energy system of Armenia the recovery of the ANPP power supply from the backup hydroelectric power station "Argel" with the capacity of 50MW is provided for. In case of a loss of external power supply to the NPP from the power transmission lines of 220kV and 110kV (for example, the collapse of the energy system of Armenia) the dispatcher of the RA grid gives a command for actuation of "Argel" HPP and through the direct transmission line 110kV ("Bjni") the power is supplied to the ANPP. "Argel" HPP in accordance with the requirements established in the regulation enforced in Armenia "Seismic stable construction- Design standards" II-6.02-2006, the seismic impact on the level of 8 points (~ 0.2 g) is estimated.
The regulation provides for the recovery of power supply to the ANPP in case of total blackout of the grid system of Armenia within 3-5 minutes both in the presence and absence of communication of the ANPP with the central dispatcher controlled energy system of the RA.

DGS

The emergency power supply system of category 2 is a redundant power source of the site (diesel-generator station).
The emergency power supply system of category 2 is designed to supply power to essential consumers of the reliability category 2 during loss of off-site power. The system consists of two independent channels. Each channel includes:
- Two diesel-generators with capacity 1500kW each;
- Section 6 kV (3 RB-2 and 4 RB-2);
- Section 0.4 kV (25 BNN, 26 BNN, 69 NO and 70 NO);

- Operating transformers 6/0.4 kV (65T, 66T, 69T and 70T);
- A battery with a capacity of 450A per hour is installed for each two channels of DGs.
- All electrical equipment of the emergency power supply systems of category 2 is designed for seismic impact with the ground acceleration 0,4g (seismic category I), except for the reserve diesel fuel tank (300m^3) relating to the seismic stability category 2, and its failure during the maximum design level earthquake cannot be excluded.

Fire fighting in all of cable tunnel and shafts of the ANPP is provided by the automatic foam extinguishing system.

Cable tunnels are equipped with rainwater trays to remove water trapped in the tunnels.

An additional source of power to prevent a severe accident at the ANPP is an "Additional emergency cooling system" (DAR).
The DAR is designed to supply power to essential consumers of the unit during beyond design basis accidents, and in the absence of power on sections 6kV and 0.4kV and the inability to power it from operating and backup power sources including the failure of the emergency power supply system of the category 2.

The DAR consists of:
- Reserve transformer (1TR);
- Diesel generator (DG) with the capacity of 1500kW;
- Outdoor switchboard (KRUN-6 kV);
- Outdoor transformer substation (KTPN-0.4kV);
- Additional cables to individual consumers.

The system is put in operation manually at command of the ANPP shift supervisor.

Batteries for DC power supply

The system is designed to supply power to the systems important to safety of the power supply reliability group 1.

The emergency power supply system (EPS) of category 1 has two independent channels, and each channel supplies 100% power required for the safety system channels.

Each channel includes:
- Seismic resistant batteries "VARTA" with capacity of 1500A per hour;
- Two reversible motor generators (ODG), one - operating, the other - reserve;
- Two thyristor devices for reserve activation;
- Section 0.4 kV (28NA, 29NA);
- DC panel (BSCHPT-1, 2);
- Rectifier for battery charging.

During emergencies is charging is not available the system can supply power during 3-3.5 hours at maximum load.

The batteries are recharged:

- At power operation of the unit - from operating generators of the unit through the auxiliary transformers of house loads;
- At shutdown condition of the unit - from the external power grid through the redundant transformers;
- At total loss of power at the unit - from the diesel generators of the emergency power supply of category 2.

All electrical equipment of the emergency power supply systems of category 1 is designed for seismic impact with the ground acceleration 0,4g.

1.4. Plant safety assessments and modernization

1.4.1. Safety reassessment and modernization

Since the second half of 80s the safety of ANPP was upgraded continuously, including assessment of the safety level and implementation of technical and organizational measures aimed at improving the safety, reliability and operating culture the ANPP Unit 2. The most important measures aimed at improving the design level of safety, reliability and operating culture of the ANPP Unit №2 can be summarized as follows:

1. Development and approval in 1987 of "Integrated measures to improve the reliability and safety of existing and constructed nuclear power plants with WWER type reactors" after the Chernobyl accident in 1986. On the basis of this document there was developed "The schedule of implementation of measures to improve the reliability and safety of the ANPP units 1 and 2" which was approved by the Ministry of Atomic Energy of the USSR, the Ministry of Medium Machine-Building Industry of the USSR and the Gosatomnadzor of the USSR.
2. In the period from December 1988 to 1990 after the Spitak earthquake and the decision to shutdown the units 1 and 2 and in accordance with the "Integrated of organizational and technical measures to strengthen the operation regime, seismic stability, and also shutdown and transfer ANPP units 1 and 2 to long-term shutdown condition" there were performed measures on the anti-seismic strengthening and enforcement of structures and metal frames of:
    - Non-hermetic part (tent) in the turbine hall by the row "B";
    - electrical equipment shelves;
    - RDGS building;
    - diesel pump station at the discharge channel;
    - the main building by the row "B" and by the axes 33,35;
    - wall panels of the main building;
    - ceilings and wall panels of the circulation pump station.
3. Based on the order № 06-79 of the Minister of Atomic Energy and Industry of the USSR as of 06.03.1991 "On examination of the Armenian NPP units to assess the technical condition of the ANPP systems and equipment and to determine the concept of a possible start-up and temporary operation of the ANPP units until alternative sources are introduced" the survey of conditions of the NPP units was performed in 1991 -1992 under the "Technical Assistance for the safety assessment of Unit 2" by FRAMATOME and EDF experts.
4. In 1993 the decision was made to restart the ANPP Unit №2. In the implementation of measures related to the ANPP Unit №2 restart there have been developed: "The concept of the Armenian nuclear power plant units restart" in 1993 and "The list of safety improvement measures for the ANPP Unit №2" in 1995, the Ministry of Energy of the RA, Yerevan, Armenia, 1995, the majority of which were implemented before the ANPP Unit №2 restart and included the measures:

- to address the findings identified during examination of buildings, structures and equipment;
- to process the results of measurements of sediments, tilts of buildings and structures before and after the earthquake occurred on 07.12.88;
- inlet and outlet channels executive survey;
- data analysis of research report on “Calculation and experimental verification of seismic stability of ANPP units №1 and №2” performed by VNIIK with implementation of the recommendations of GOAEP
- examination of the technical condition of cooling towers and building structures of irrigation devices;

5. In the period 1993 ÷ 1994 at the stage of the repair works, measures were implemented to improve the safety of the Unit №2 including:
   - preventive maintenance of the equipment in the scale of capital repairs;
   - in-service inspection of 100% equipment and piping of the primary and secondary circuits;
   - technical inspection of equipment and pipelines according to the new rules;
   - recovery of systems and components in the design scale, replacement or recovery and extending the resource of the systems elements with the exhausted lifetime;
   - Implementation of measures to improve the reliability and safety (upgrade) of equipment and systems, including confinement;
   - detaching equipment and piping of the Unit №2 to ensure their seismic stability in accordance with the existing technical documentation;
   - Installation of seismic stabile equipment;
   - Anti-seismic strengthening of buildings and structures after establishment of new standards on construction of seismic stable nuclear power plants.

6. In 1995, after 6.5 years of shutdown the ANPP Unit №2 was restarted with the safety level established in the “Concept on the ANPP restart”. To ensure the process of continuous safety improvement of the ANPP Unit №2 there was developed a program of measures on strengthening safety, reliability and operation culture of the Armenian NPP Unit №2 for 1997 ÷ 2000”, which was approved by the RA Ministry of Energy and agreed with the ANRA.

7. On the basis of the Law of the RA on Safe Utilization of Atomic Energy for Peaceful purposes” the ANRA requested the operating organization to perform a safety assessment of the ANPP based on modern approaches in accordance with the documents of the IAEA and to develop a “Safety analysis report”. In 2006 the “Safety analysis report of the ANPP Unit №2”, was developed on the basis of the “Requirements for format and content of the safety analysis report of the Armenian NPP Unit 2“ Yerevan, 2002, approved by the Government of Armenia and ratified the President of the Republic of Armenia to include the results of the deterministic safety analysis, and information sufficient for an adequate understanding of the current design of the unit, the safety concept and also to identify technical and organizational measures to ensure the ANPP safety.

8. After the Armenian NPP Unit №2 restart in 1995 the Armenian Government adopted a decree on implementation of safety improvement measures at the unit № 2. Within the framework of these activities the re-evaluation of seismic safety was assigned with the highest priority and included as a separate task in the “List of ANPP Unit №2 safety improvement measures for the period from 1998 to 2004 “. This task also includes the two main recommendations of the IAEA:
   - Additional study of certain issues related to seismic risk of the ANPP site - verification of the seismotectonic model of the territory adjacent to the ANPP site.
   - Analysis of volcanic risks of the ANPP site (the results are provided in the chapter 2).

9. The ANRA also required the operating organization to perform probabilistic safety analysis of the ANPP based on the “Requirements to format and content of the safety analysis report of the ANPP Unit 2“ Yerevan, 2002, approved by the Government of Armenia and ratified by the President of the Republic of Armenia. The document establishes a requirement to provide the results of probabilistic safety analysis as a separate annex to the “Safety analysis report of ANPP Unit 2”. The probabilistic safety analysis methods are generally accepted tool for analyzing the NPP safety. The main advantage of PSA is an in-depth qualitative and quantitative analysis of the actual configuration of the NPP with definition of the factors contributing most to the overall risk of the reactor core damage.

10. From 2010 to 2011 under contract No ARM9022-86943V the IAEA consortium composed of four organizations: NRI (Rez Research Institute of Atomic Energy), VUJE (Research Institute of NPP Operations of Slovakia) Armatom and NRSC developed a comprehensive program on ANPP modernization. The comprehensive modernization program considered safety issues identified both in the “Safety analysis report of the ANPP Unit 2” and several missions of the IAEA to the ANPP, and suggested ways of resolving issues with the high safety category. The comprehensive modernization program on the basis of deterministic and probabilistic safety analyzes covered all safety deficiencies, which are defined for the design of the given reactor.

11. In the operation of the NPP a specific attention is paid to the analysis and improvement of operational documentation. The aim of this work is to improve safety in normal operation of the Unit №2 through development of guidelines and methods of safe operations and implementation of symptom-oriented emergency operating procedures designed to assist the operator in diagnosing events and implementing appropriate emergency actions, as well as to develop guidelines for management of beyond design basis accidents.

1.4.2. Seismic safety reevaluation and upgrading

The ANPP is the first NPP in the former USSR, built in a seismically active region. Absence during designing of a regulatory framework for the construction of nuclear power plants in seismically active region has resulted to paying much attention to improving the seismic safety of the ANPP from the design phase.

According to the seismic zoning map existing at the initial stages of designing, the ANPP site was assigned to a 7-point scale seismic zone (~ 0.1g). After the earthquake in Vrancea (Romania, 1977), the level of the seismic effect for nuclear power plants was redefined and made 8 points (~ 0.2g). However, in 1972 9 points (0.4g) was adopted for the design seismicity for the reactor shaft, and 8 points (0.2 g) - for the reactor compartment. In addition to the assessment the increasing factor 2.7 was established (see Chapter 2) according to the standards existing at the time of construction.

Since during designing of the ANPP there was a lack for special earthquake-resistant electrical equipment, electrical equipment were tested for the seismic stability in two phases between 1975 and 1978. Tests were performed alternately in two mutually perpendicular directions in the horizontal plane with sinusoidal oscillations in the frequency range from 0.5 to 30 Hz.

The temporary standard VSN-15-78 was issued in 1979; it established requirements for 2 levels of seismic hazard, DE and MDE:

- DE - design earthquake: a maximum intensity earthquake at the NPP site with a frequency once every 100 years.
- MDE - maximum design earthquake: a maximum intensity earthquake at the NPP site with a frequency once every 10000 years.

According to these standards, the seismic stable is the NPP, which during an earthquake up to DE and inclusively is able to produce electricity, while during MDE safe shutdown of the NPP is ensured.

Based on the VSN-15-78 from 1983 to 1985 comprehensive measures to identify the normative seismicity of the ANPP site were performed, as a result of which the maximum acceleration values 0,2g have been recommended for MDE and 0,1g - for DE.

In 1988 the first floor response spectra have been obtained for the calculation of seismic stability of buildings, systems and equipment which showed that the buildings of categories I and II fully meet the requirements of the mentioned standard. Separate buildings of the category III need to be strengthened, for which the ANPP reconstruction design was developed, which included also implementation a new essential consumers cooling system (SOOP) with two spray ponds, which were to have the seismic stability category I (0,4g).

After restart of the ANPP Unit №2 in 1995 the leadership of the Republic of Armenia, giving a paramount importance to reassessment of seismic safety of nuclear power plants, made a decision on increasing of the earthquake design level to a value corresponding to 84% confidence, for which the IAEA had developed a document RU-5869 “Technical Guideline for seismic re-evaluation program of the Armenian NPP Unit №2“(TG).

Thus, the initial level of design basis earthquake for the ANPP, corresponding to 7-point magnitude earthquake has repeatedly revised, justified and is currently 0,35g for the horizontal direction.
Detailed information on the work carried out to strengthen the framework and detaching equipment on building structures and facilities of the ANPP to reduce the impact of seismic effects on the equipment to the level of impact of the earthquake spectrum are provided in the Chapter 2, “Earthquake” of this report.

1.4.3. Use of PSA as part of the safety assessment

The PSA activities for the ANPP Unit №2 have been initiated in 2002. Over the past period there have been developed a number of PSA models, which cover the following aspects:

- Undesirable event - damage to the ANPP Unit №2 reactor core (the ANPP developed models for the PSA level 1)
- Considered initiating events - internal events, external hazards and fires in the NPP premises
- Considered operating conditions/modes – reactor power operation (50-100% of the nominal power, both TGs are in operation).

During this period, a series of expertises including the expertise of the PSA model by the IAEA IPSART mission (2007) have been performed. The PSA models were continually improved according to the expertise results. Currently the third version of the PSA including both internal events and fire and external factors is valid. The current PSA model reflects statistical data on equipment failures, and configuration of the NPP as of December 2010.

Summarized results of the PSA are demonstrated in the Figure 1.4.3-1.

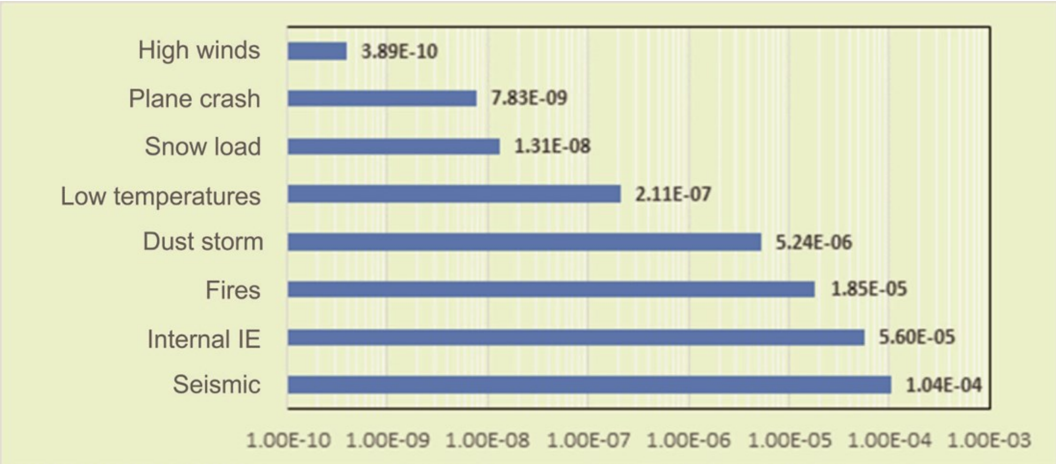


Figure 1.4.3-1. Contribution to ANPP core damage risk from various initiating events [1/year]

Analysis of the dominant contributors to the core damage risk allowed to determine the NPP risk profile and to identify the main problematic nodes.

Internal initiating events. The total contribution to the ANPP core damage risk from internal initiating events is 5.60E-05 [1/year]. Analysis of the results of quantification of the PSA model for internal initiating events allowed coming to the following conclusions:

- The main contributor to the core damage risk from internal initiating events are:
    - The primary circuit leaks
    - breaks in the secondary circuit
    - transients.
- Criticality accident scenarios associated with the primary circuit leaks are failures of sprinkler system to cool the tank B-8/2. The main contributors to the system failure is failure of the valve supplying water to the SGs-RCP deck to close and the failure of valve on the recirculation line. On the basis of the above mentioned, it is recommended to improve the reliability of the sprinkler system.
- The main problem associated with breaks on the secondary circuit is a failure of equipment of the turbine hall as a result of steaming in the premise. On the basis of the above mentioned it was recommended to perform qualification of critical equipment of the turbine hall for its functioning during steaming in the premise.
- Significant contribution to the core damage risk from transient is conditioned by the high frequency of occurrence of this IE. Critical accident scenarios associated with the transient is a failure of turbine generators to shutdown and failure of BZOK to close. This scenario results in sharp cooling of the primary circuit and the acceleration of the turbine take place.

Fires. The total contribution to the core damage risk from fires in the ANPP premises is 1.85E-05 [1/year]. Analysis of the results of quantification of the PSA model for fires in the ANPP premises allowed coming to the following conclusions:

- The fires risk in 17 out of the 102 considered sections of the ANPP Unit №2 is about 90% from the total core damage frequency due to fire. The main contributors to the fire risk are switchboard rooms, ventilation chamber of the SG-RCP deck (A-013/2), turbine hall, and control panels.
- Criticality scenarios of fire inside switchboard are scenarios leading to a loss of power to essential consumers of the reactor and/or a turbine hall in the propagation of fire between the switchboard rooms. In such fire scenarios, dominant contributors are the secondary feed and bleed system failures not requiring power supply - the additional make-up system of SG and IPU SG. Based on the above mentioned it was proposed to include measures to improve the reliability of the additional make-up system of SG, as well as to develop and implement a system of automatic fire extinguishing in the switchboard premises in the ANPP program of modernization.
- For the SG-RCP deck a critical is a fire scenario associated with the leakage from the oil system piping of the RCP in the room A-013/2, and the failure of oil pumps to disconnect. The calculations have shown that this scenario leads to the continuous oil supply into the room A-013/2, temperature rise and damage of the primary make-up pipelines which in turn leads to the core damage. Based on the above mentioned, it was proposed to include a set of measures to reduce a fire risk in the Room A-013/2 (improving the reliability of the pump disconnection circuit of the oil system, the implementation of fire detection system in the containment, reducing the likelihood

of oil leakage from the oil pipeline on the MNR (oil pumps) discharge, etc.) in the ANPP modernization program.

- The analysis of the cable routing allowed determining the points of intersection of control cables by the equipment of the reactor and turbine halls involved in the procedures of the reactor cooling (room E-202/9 and E-202/10 under the MCR-2). Based on the above mentioned it was proposed to include activities on implementation of reserve control panel in the ANPP modernization program. At this stage, the emergency shutdown panel is developed and implemented, which can partially solve the above problem.
- The considered fire scenario in the turbine hall leads to failure of the main equipment of the turbine hall, therefore, the failure of water supply to the SG by the systems APEN, PEN and NP. Core damage occurs when applying feed-and-bleed procedure on the primary and secondary circuits. The results of the analysis of importance and sensitivity shown that the results are sensitive to the assumption on maintaining the operability of passive components IPU PG in case of a fire at the turbine hall. Based on the above mentioned it was proposed to include implementation of additional studies to justify the operability of IPU SG for protracted fire loads in the ANPP modernization program. According to the results of studies, if necessary, to implement measures to protect the IPU SG from fires.

External events. The total contribution to the core damage risk of the ANPP from external events (excluding seismic effects) is 5.47E-06 [1/year]. The analysis of the results of quantification of the PSA model for external events allowed coming to the following conclusions:

- Most of the external events can be screened out by the following parameters:
  - External events do not affect the ANPP safety functions
  - External events have a negligible frequency of occurrence (F <1E-06 [1/g])
  - Occurrence of external event considered in the ANPP is not possible.

  As a result of a systematic screening of the full list of external events, the following external impacts to be implemented in the PSA model were selected: strong wind, dust storm, low temperature, air crash, and snow load. According to the results shown in the Figure 1.4.3-1 the dominant contributors to the core damage risk are accident scenarios associated with sandstorms and low temperatures. The influence of other external factors is negligible.
- Considered accident scenarios associated with a dust storm involves failure of ORU and raising the dust concentration in the air. According to the calculations, the dust concentration in the air may exceed the limits set for the air collectors of DG. Taking into account the non-hermeticity of premises, a failure of DG was considered in this scenario due to increase of dust concentration in the room. Based on the above mentioned it was proposed to include activities to protect DG from higher concentrations of dust (filter installation, ensuring the tightness of DGS premises, etc.) in the ANPP modernization program.
- Considered accident scenarios associated with the extremely low temperatures in the ANPP implies failure of ORU and icing of ASN suction pipe and DNP SG. In this accident scenario the only possibility to prevent core damage is the feed-and-bleed of the primary circuit or water supply to the SG from the emergency make-up system

(not meeting the principle of single failure). Based on the above mentioned it was proposed to include the protection of the suction piping ASN and DNP SG against extremely low temperatures in the ANPP modernization program.

Seismic hazard. The total contribution to the core damage risk of the ANPP from seismic effects is 1.04E-04 [1 / year]. The analysis of the results of quantification of the PSA model for seismic effects allowed coming to the following conclusions:

- 85% of core damage occurs due to seismic events with the ground acceleration to 0.5 g. Contribution of seismic events with the ground acceleration above 0.5 g is comparatively less due to the low frequency of occurrence of such events.
- The following groups of scenarios that have been found as having the greatest contributions to the seismic core damage risk:
    - 56% - groups of scenarios in which the seismic event causes a loss of power supply, without a break in the safety systems of the primary or secondary circuits. Approximately 90% of these scenarios cause a failure of diesel generators for various reasons.
    - 14% - scenarios with the structural destruction of buildings
    - 12% - scenarios in which a seismic event causes a break in the primary circuit
    - 9% - scenarios in which a seismic event causes insolated break of the secondary circuit.
    - 5% - scenarios in which a seismic event causes transient with possible additional failure, including the loss of the essential consumers cooling system and/or failure of the reactor protection.
    - 3% - strong seismic events (> 0.8g) where integrity or operation of the SSC cannot be guaranteed.
- Since the main risk contributors are groups of scenarios associated with failure of DG and total loss of power, the most important system is the DG additional makeup (diesel pump). Therefore, it was suggested to include measures to improve the system reliability (for example, the installation of the second channel of the system) in the ANPP modernization program. The seismic stability of BZOV tanks supporting DNP SG system was also upgraded. These measures allow reducing the CDF from seismic effects by 15%.
- The main risk contributors associated with the DG support structures is the gravity tank supplying service water to DG (BTV). To reduce the risk associated with BTV there was provided an analytical justification for the successful start-up of DG in the absence of BTV. The analysis showed that even in case of failure of BTV the DG successfully starts-up before the essential consumers cooling system actuates. Thus, the performed analytical justifications and the dismantling of the tank BTV allow reducing the CDF from seismic effects by more than 25%.

The IAEA IPSART mission is planned for October 2014 to make an independent examination of existing PSA models.

At present, the ANPP performs measures to expand the scope of the PSA models. In particular, the PSA model for internal initiating events at reduced power level and the shutdown reactor, and the level II PSA model are being developed

1.4.4. Modification in Progress

The analysis results for compliance of the units to the requirements of modern safety regulations, along with the probabilistic safety analysis and accident analysis and operating experience, formed the basis of the concept of modernization of the units with the purpose to significantly improve their safety.

The “list of technical measures for the ANPP Unit №2 safety improvement for 2006÷2016” is currently applied at the ANPP; it was approved by the Minister of Energy of the RA and agreed with the ANRA and includes the following key technical measures, the implementation of which will allow eliminating the remaining safety deficiencies, or greatly reduce their impact on safety.

- Continued improvements in the separation and independence of channels and logic actuation of the reactor protections, including the possibility to check them during normal operation;
- Implementation of the remaining recommendations on the program of ANPP Unit №2 seismic revaluation such as:
  - installation of additional fuel tank (50÷100 m) to fill the reserve tanks of DG, which will allow providing emergency power for 72 hours.
  - gradual modernization of the seismic monitoring system and seismic monitoring instruments.
- Implementation of a program on qualification of equipment important to safety, in accordance with the applicable standards;
- Evaluation of safety systems media in the premises where the temperature conditions may change during the expansion of the list of design basis accidents;
- Implementation of a complex of measures for inclusion the primary circuit leaks with an equivalent diameter DN 200 mm in the design basis accidents;
- Reconstruction of spray system
- Modernization of the in-core monitoring system ICMS;
- Installation of the system for monitoring presence of hydrogen generated in emergency situations, and installation of hydrogen recombiners;
- Development and introduction of symptom-oriented emergency operating procedures;
- Development and introduction of symptom-oriented emergency operating procedures for the shutdown reactor when the reactor vessel and confinement are open and the physical barriers between fuel assemblies and environment are not available;
- Analyzes of severe accidents;
- Development of severe accidents management guidelines.
- On the basis of the results of ANPP examination, the ANRA will prepare a national report (Stress test) where it will assess the current safety level of ANPP, and the conclusions made in the ANPP survey report. The national report will identify the necessary measures aimed to improve the ANPP safety.

1.4.5. Implemented technical measures

The most significant technical measures aimed at improving the design level of safety, reliability and operating culture of the ANPP Unit №2, and implemented between 1996÷2012 are listed below:

- Replacement of PRZ SV to improve the reliability of valves and the possibility to implement the feed and bleed procedure, i.e. to enable operation of valves in an aqueous medium;
- Installation of BZOKs for reliable shutoff of steam generators during breaks in the steam lines and MSH;
- Replacement of PRZ SV to enhance the reliability of system and the possibility of operation of valves in the water-water regime;
- Replacing electric motors at all main gate valves by the qualified ones;
- Installation of equipment to detect the primary to secondary leaks on the basis of the nitrogen-16 principle;
- Reconstruction of the make-up water distribution collectors in order to improve the process of heat removal from the tube;
- Commissioning of the acoustic leak detection system of the primary circuit and the system for detection of loose items in the primary circuit;
- A new system for the management and control of technological parameters;
- Modernization of the ANPP fire protection system;
- Replacement of SG electronic level regulators;
- Replacement of the obsolete radiation monitoring system;
- Installation of the equipment diagnostic system at the ANPP, according to the leak-before-break concept (LBB) results;
- Reconstruction of the sequential start-up automatics;
- Reconstruction of the system of permanent power supply of the category I;
- Assessment of environmental conditions for safety in the boron compartment;
- Reconstruction of the SG nozzle protection against clogging;
- Replacement of the obsolete neutron flux monitoring system;
- Installation on borometers on the primary circuit pipelines;
- Replacement of the automatic power control;
- Modernization of the automatic level control of SG;
- The neutron flux is monitored using new sensors and systems;
- Providing with the adequacy of filtration of air supplied to the MCR-2, and the control room protection against flying objects;
- Establishment of reserve control room;
- The implementation of I&C for post accident monitoring at the ANPP Unit 2;
- Implementation of a system of water level control in the reactor vessel;
- Completion of the seismic evaluation: the classification and definition of the functions of systems, structures and components, response spectra analysis and strength analysis of the main building and reserve diesel generator station, analysis of seismic stability of buildings and equipment related to safety;
- Completion of measures for seismic strengthening of buildings and constructions pipelines and equipment.

2. EARTHQUAKES

2.1 Design basis

2.1.1 Earthquake against which the plant is designed

2.1.1.1 Characteristics of the Design Basis Earthquake (DBE)

From the seismic point of view, the ANPP is located in the central, relatively low-active part of the Mediterranean-Trans-Asian seismic belt. The significant historical seismic event in the region is the Ararat earthquake of 1840 with 7.4 magnitude and the instrumentally observed - the Spitak earthquake of 1988 with 7.0 magnitude. Analysis of the historical and instrumental seismicity allows assuming that the seismic focuses of many strong (with 5.5 or higher magnitudes) earthquakes of the region are timed to the zones of active faults or are located in their direct vicinity.

From the volcanic point of view, the Armenian NPP is located on the Shamiram peripheral plateau of the Aragats volcanic region, which was formed during the Upper Pliocene (2.5 million years) - Upper Quaternary (0.4 - 1 million years).

The Armenian NPP site (R=5 km) is covered with a thick (400 m) mantle of Pliocene-Quaternary lava basalt-andesite rock composition, which serves as the foundation for the main buildings and structures of the plant. Three aquifers (to a depth of 400 m) are observed under the site, one of which is located at 85m depth below the surface. There are no landslide slopes, mudflow areas, faults, mining, subsidence soils, karst areas, soils with a load capacity of less than 2 kg/cm2 in the region where the Armenian NPP is located.

Geotechnical section of the site is alternation of the layers of the rock (approximately 70%) and soil (about 30%). Hard soils (greater than 10 m) are mostly monolithic basalt, but there are also small layers of the developed basalts. The following are the average values of the rocky layer's characteristics:

- volume density – 2,3÷2,6gram/cm^3
- bearing capacity – 800 ÷ 1000kg/cm^2
- V_p – 2500÷3000m/s
- V_s – 1800÷1960m/s.

Soil (depth is 3 - 5 m) consists mostly from the land waste rock and rock debris materials. There are also a loam, sandy loam, sand, and scoria. The average values of the soil layers are:

- volume density – 1,7 ÷ 2,1gram/cm^3
- bearing capacity – 40 ÷ 80kg/cm^2
- V – 600÷900m/s
- Vs – 350÷750m/s.

ANPP is the first nuclear power plant built in seismically active region of the former Soviet Union. Since no specific seismic regulation for Nuclear Power Plants existed during design of the plant, improving of ANPP seismic safety was constantly in the focus during following years. Table 2.1.1.1-1 shows the chronology of events and activities related to ANPP seismic design.

Table 2.1.1.1-1. Chronology of events and activities related to ANPP Seismic Design

N	*Period*	*Activities*	*Notes*
1	1966 - 1969	Decision to build ANPP	
2	1968-1972	Site selection, the initial stage of design and construction	(1)
3	1970	Start of Unit №1 construction	
4	1972	Categorization of SSC on three seismic categories	(2)
5	1975	Start of Unit №2 construction	
6	1974-1975	Seismic analyses and experimental studies for Seismic Category I equipment and piping for seismic demand 0,4 g	(3)
7	1975-1976	Seismic tests of the individual components	(4)
8	1977	After the Vrancea earthquake (Romania), the level of seismic hazard for ANPP was installed to be 8 points (~ 0.2 g)	
9	1977	Commissioning of Unit №1	
10	1977-1978	Seismic Tests of the electrical equipment	(5)
11	1972-1978	Complex engineering seismological studies	(2)
12	1979	Issued a temporary standard VSN-15-78	(6)
13	1980	Commissioning of the Unit №2	
14	1983-1985	Complex studies for the Regulatory definition of Seismic Hazard	(7)
15	1987	Project of Reconstruction to meet requirements of PNAE G-05-006-87. Revision of Seismic Categorization	(8)
16	1988	Issue of the first Floor Response Spectra for the main buildings and structures	(9)
17	07.12.1988	Spitak Earthquake	
18	1989	Both Units are shut down	
19	07.04.1993	Armenian government's decision for "Beginning of the recovery and resume operation of the second Unit of the Armenian NPP"	
20	1989 - 1995	Installation and construction works for the seismic strengthening of the Reactor and Reserve Diesel Generator Buildings. Seismic qualification of the Primary Circuit equipment and piping for seismic load 0,4g in accordance with the PNAE G-05-006-87 requirements, an experimental verification of the seismic capacities of the equipment and piping for the new-developed floor response spectra.	
21	1993-1995	Seismic Hazard Evaluation of the ANPP site with participation of international organizations and IAEA	(10)
22	05.11.1995	Re-commissioning of ANPP Unit №2	
23	1995	Armenian government's decision to increase PGA level up to 0,35	(11)

N	Period	Activities	Notes
		g	
24	1996 - 1998	Analysis of Seismic Category I and II piping for Floor Response Spectra corresponded to PGA = 0,21 g	(12)
25	1997	IAEA, 1997. Technical Guidelines for Seismic Re-Evaluation and Upgrading Programme of the Armenian Nuclear Power Plant, Unit-2, IAEA/RU-5869	(13)
26	1998-1999	Implementation of recommendations for piping upgrading	
27	2000	First Seismic Plant Walkdown	(14)
28	2001 - 2002	Developing the procedure for the Plant Safe Shutdown for SSEL compilation	(15)
29	1998 - 2004	Implementation of the seismic upgrading program	
30	2004	Preliminary report for PSHA of ANPP site	(16)
31	2004	Commissioning of the new system DAP for feeding of SG from the tanks with chemically desalinated water (BZOV-500) in case of ANPP blackout	
32	2004	Analysis of the bearing capacity for the Reactor Building Foundation	
33	2005	Modernization of the systems for the neutron flux monitoring and SG automatic step loading	
34	2006	Second PSHA study	(17)
35	2006	Final approval of floor response spectra for Reactor Building and Reserve Diesel Generator Building	(18)
36	2007	Final approval of SSEL	(19)
37	2007	Assessment of seismic capacities of ANPP civil structures (RB, RDGB and Ventilation Stack) under seismic loads corresponded to RLE with 84th percentile PGA = 0,35g	
38	2007 - 2008	ANPP Unit №2 Final Seismic Walkdown, Project N ARM9014-89032	
39	2009-2011	PSHA for new Unit	
40	2010 - 2012	Implementation of the IAEA Project N ARM9022-84188 for seismic qualification of the ANPP Unit №2 safety equipment	
41	2010 - 2012	Implementation of SSC upgrades based on recommendations issued in the frame of ARM9014-89032 and ARM9022-84188 Projects	
42	2012	Beginning of Stress Test Program, Seismic Qualification of SSC included in extended SSEL	

Notes:

(1) According to the seismic zoning maps that were valid during time of the ANPP initial design (1968-1970), plant's site was attributed to an 8-point scale seismic zone. After Vrancea earthquake (Romania, 1977) level of seismic loads for NPP was reviewed and

installed to be equal 8 points (~0,2g). However, in 1972 in accordance with the CNIISK recommendations seismic design for the Reactor Shaft was adopted for seismic level of 9 points (0.4g) and for the Reactor Building - 8 points (0.2 g). Additionally amplification factor equal to 2,7 was installed for the assessment according to SNiP.

(2) Later, comprehensive engineering - seismological studies undertaken in 1972-1978 have shown that maximal PGA at the ANPP site could reach values of 0,35g - 0,4g. Following to that seismic load for ANPP buildings, structures and systems were differentiated in function of their importance for safety in different seismic categories: 0,4 g - Category I; 0,2g - Category II, 0,1g - Category III.

(3) During the period 1974-75 the analytical and experimental studies of seismic category I equipment and piping were carried out. Dynamic studies were undertaken on the models, and dynamic calculations were performed for seismic input defined in terms of Eureka (1954 California) accelerogram scaled to PGA = 0,4g.

(4) Since the electrical equipment was not specifically assessed to earthquakes during ANPP design, it was decided to perform seismic tests for the samples of existing equipment. The first phase of testing was completed in the period from 1975 to 1976. Tests were conducted step by step for two orthogonal horizontal directions under sinusoidal input in the frequency range from 0.5 to 30 Hz. Circuits were checked for all operational and design states of the tested equipment: on/off/during switching on and off). Based on these results, in 1976 a design measures for strengthening of the civil frames and restraining of the electrical equipment were developed to reduce seismic loads up to the level corresponded to the floor response spectra and for excluding equipment's resonance frequencies. These upgrading were extended also for the cabinets and panels, which were not included in the list to be tested.

(5) During second phase of testing (1977-1978) other electrical equipment was tested on the shaking table taking into account performed strengthening and restraining. As result, this equipment was confirmed to be adequate for the specified frequency range and acceleration. Since no specific seismic requirements were defined in the factory technical specifications for the Instrumentation and Control Equipment installed at the Plant, it was decided to perform I&C seismic testing. The tests were conducted in 1977. Two levels of sinusoidal seismic input were applied in the frequency range 1-50 Hz: 0,5g and 1,0g. Some devices that demonstrated during testing short-term false alarms were replaced (totally 889 items).

(6) A temporary standard VSN-15-78 issued in 1979 has required consideration of two seismic hazard levels:

- PZ - Design Earthquake with maximal acceleration for 100-years return period;
- MRZ - Maximal Design Earthquake for 10000 years return period.

According to this standard at PZ level NPP should be capable for power generation, while under MRZ level should be ensured NPP's Safe Shutdown.

(7) According to VSN provisions from 1983 to 1985 there were performed additional studies which resulted to definition of two PGA levels: 0,1g for PZ and 0,2g for MRZ.

(8) In 1987, to achieve compliance with requirements of PNAE G-05-006-87 standard, there was compiled a project for the plant's reconstruction. The project included as

well analysis and testing of equipment that previously was not considered for seismic loads.

(9) First floor response spectra were obtained in 1988. Seismic analyses of buildings assigned to seismic categories I and II have demonstrated that these structures are capable to withstand prescribed seismic input. Only for some Seismic Category III buildings there were issued recommendations for strengthening. Such project was developed and included new system for cooling of Essential Service Systems and Components: two spray cooling ponds that were attributed to Seismic Category I (0,4g).

(10) Deterministic Seismic Hazard Assessment for ANPP site was carried out in 1993-1995 by Armenian Institutions and international companies (ISMES, Italia and Atomenergoproject, Russia) with assistance of the IAEA.

(11) After restarting the ANPP Unit №2 in 1995, Armenian Government, giving a primary importance to the ANPP safety issues, has decided to increase seismic design level up to the value that corresponds to 84^{th} percentile.

(12) Seismic Category I and II piping have been analyzed in period 1996-1998 according to the modern Standards and requirements. Seismic input was defined in terms median shaped FRS with ZPGA = 0,21g.

(13) After restarting the ANPP Unit №2 it was decided to reevaluate Plant's seismic design for Review Level Earthquake with use 84^{th} percentile peak ground acceleration. In response to this decision IAEA has developed document RU-5869 “Technical Guidelines for Seismic Re-Evaluation and Upgrading Programme of the Armenian Nuclear Power Plant" (TG).

(14) First seismic walkdown was undertaken during 2000 outage. Experts that participated in this walkdown concluded that ANPP Unit №2 has many seismically robust features. At the same time found seismic deficiencies were discussed in details and fixed according to issued recommendations. Later IAEA mission has confirmed the implemented fixes.

(15) The procedure for ANPP Safe Shutdown was developed in the period 2001-2002 for SSEL compilation. This procedure considered three variants of Safe Shutdown: hot, semi-cold and cold. The procedure presents seven major functions that must be performed to ensure the criterion of 72 hour period safe shutdown:

- reactivity control;
- reactor coolant system pressure control;
- reactor coolant system inventory control;
- reactor decay heat removal;
- providing reliable cooling of fuel elements;
- specific safety functions associated with a particular technological algorithm for reactor cooling after the earthquake;
- localization and confinement of radioactive products; determination of boundaries for systems responsible for performing seismic functions.

SSEL has been developed based on the semi-cold procedure for shutdown.

(16) In 2004 Armenian institutions NSSZ and Georisk developed preliminary report for ANPP site PSHA. However, after IAEA review, experts have concluded that the work performed does not fully meet modern requirements, and recommended to perform PSHA according NS-G-3.3.

(17) In the framework of DTI project NSPA18 (Great Britain) in 2006 Aspinal and ass Co. implemented the ANPP PSHA according to new requirements of IAEA NS-G 3/3. They received corresponding probabilistic seismic input characteristics (PSI) for probabilistic safety analysis (PSA). The implemented analysis was based only on currently available information provided by Armenian experts, and the activity scope did not include additional studies. The analysis resulted in specification of seismic zones with M_{max} 5,8÷8,0, and also dispersed seismicity zones with M_{max} 5,6÷7,6 in the main seismotectonic model (R=150km). The nearest Sardarapat seismic zone is in about 13 km south to the ANPP, and it is assessed as having M_{max} = 6,2. The ANPP site is located in M_{max} = 6,1 dispersed seismicity zone. This analysis uses Ambraseys and attenuation formula (2005). PRISK and AARISK codes were used for the NPP site PSHA. As a result the seismic hazard curves with 10^{-2} - 10^{-7} probability was received and corresponding response spectra were received of 0,4-40Hz frequency for 5% damping. The output seismic hazard curves were presented both as 10%-90% confidence and for mean assessment according to which assessed value of the ANPP site peak horizontal ground acceleration (PGA) is 0,28g for mean and 0,32g for 85% confidence with increase probability once per 10000 years (10^{-4}). Having reviewed these materials the IAEA experts stated that RLE remains as official seismic safety level. They recommended reviewing PSHA using other code, FRISK88MTM, and several attenuation damping formula.

(18) In 2006 the IAEA experts agreed and ANRA approved the floor response spectra for Main Building (MB) and Reserve Diesel Generation Station (RDGS). That allows performance of the final seismic walkdown.

(19) A number of the IAEA missions resulted in recommendations on SSEL correction, and its final version was approved in 2007.

As could be seen from the Table presented above, the initial DBE level for the ANPP corresponding 7 point level earthquake (PGA ~ 0.1g), have repeatedly been revised and now stands at 0.35g for the horizontal direction.

2.1.1.2 Methodology used to evaluate the design basis earthquake

Following to the decision of the Armenian Government on the ANPP Unit №2 restart the IAEA Mission (1993) recommended paying attention to the three following important aspects:

- Assessment of geological stability of the site, i.e., lack of a contemporary active fault, which can cause permanent displacement of ground under or in the vicinity of buildings and structures important for seismic safety.
- Specification of intensity of seismic ground motion, i.e., parameters associated with seismic design (such as maximum ground acceleration, ground response spectrum, duration, historical accelerograms, etc.) needed for reassessment of the plant buildings and components.
- Developing of complete reassessment program for structures, systems and components important for safety in regard to new data and advanced special methods and criteria recognized in international practice, and implement improvements, if needed.

Based on IAEA Guide 50-SG-S1 a program was developed aimed at implementation of the first two items.
These activities (see Table 2.1.1.1-1, item 21) were resulted in Regional seismotectonic model (R=150 km) identifying the areas of potential earthquake foci (PEF). Deterministic analysis of PEF area maximum magnitude (M_{Max}) was performed with following methods:

- Maximal historical earthquake ($M_{Maxhist}$);
- Segmentation;
- The overall length of the fault;
- The fractional length of the fault.

As a generalized PEF area Mмax quantity the methods arithmetical average value was assumed equal to 6,5-7,4. Also, a dispersed seismicity area was identified with M_{Max} = 5,5, including the ANPP site.

The catalog of earthquakes within 150km was specified.

The Sadych attenuation equation developed on the basis of numerous records of earthquakes recorded on typical sites with rock soil in California was chosen in this analysis.

Calculated accelerograms were obtained by the method of accelerograms generation with fixed response spectrum. For 50^{th} percentile accelerogram Cerro Prieto from Mexicali Valley earthquake, 1980 was used. For 84th percentile - accelerogram Vasquez Rock Park from Northridge earthquake of 1994.

To specify the physical, mechanical and dynamic properties of the soil six bores with depth of about 50m each were drilled near the ANPP main building. Special seismic works were performed. Selected drill samples were analyzed in laboratory.

ANPP peak ground acceleration was determined by a deterministic method, i.e. attenuation equation was applied for each PEF separately.

As a result, in 1995 the following data were received and approved by IAEA experts:

- the ANPP site is located in the center of one-piece tectonic block, there are no tectonic disturbance under the site;
- maximum accelerations of 0,21g for 50% and 0,34g for 84% confidence are expected at the ANPP site;
- fixed response spectra were received for horizontal and vertical component, and based on them there were recommended 3-component design accelerograms of 50% and 84% confidence;
- volcanic hazard probability in the ANPP area was judged to be exceptionally small, and resumption of volcanic activity in near future is not expected.

For seismic verification of SSC, before ANPP restart, it was recommended to use 0,21g (for 50% confidence) horizontal acceleration with following increase up to 0,34g (for 84% confidence) in future.

Seismic verification of systems, structures and components (SSC) for 0,21g (50% confidence) was completed before the ANPP Unit №2 restart.

Since the spectral and frequency characteristics of the existing response spectra didn't meet in full scope the IAEA requirements, then in development of Technical Guide for seismic reassessment №RU-5869 it was recommended to apply RLE as a new level of seismic input with the following characteristics:

- A free field surface horizontal peak ground acceleration of 0.35g which corresponds to the 84^{th} percentile;
- A 50th percentile response spectra shape for rock site, as provided in USA-NUREG/CR-0098;
- The vertical acceleration component should be equal to 2/3 of the horizontal acceleration throughout the entire frequency range.

In addition it was recommended in future to implement Probabilistic Seismic Hazard Analysis (PSHA) needed for confirmation of the RLE defined for the ANPP site.

2.1.1.3 Conclusion on the adequacy of the design basis for the earthquake

In order to implement these recommendations within activities on assessment of new unit construction site seismic safety in 2009-2011 complex activities (including field ones) were performed by Noratom Consortium involving foreign organizations and leading experts.
As a result of these activities 2 alternative seismotectonic models (R=300km) were developed specifying seismic zones with M_{max} 6,0÷7,9 and dispersed seismicity zones with M_{max} = 6,0÷7,7.

The nearest seismic zone is located in 6km north to ANPP and it is assessed as M_{max}=6,7, and the ANPP site itself is located in a dispersed seismicity zone with M_{max}=6,5.

For performance of NPP PSHA a well-known FRISK88MTM code was selected which was used for assessment of seismic hazard of many US nuclear facilities. It resulted in development of seismic hazard curves with $10^{-1} \div 10^{-7}$ probability, and corresponding response spectra were estimated at 0,5÷40Hz frequency for 5% damping. The output seismic hazard curves were presented both for 5%; 15%; 50%; 85%; 95% confidence and for mean assessment (see Figure 2.1.1.3-1, Figure 2.1.1.3-2 and Table 2.1.1.3-1, Table 2.1.1.3-2).

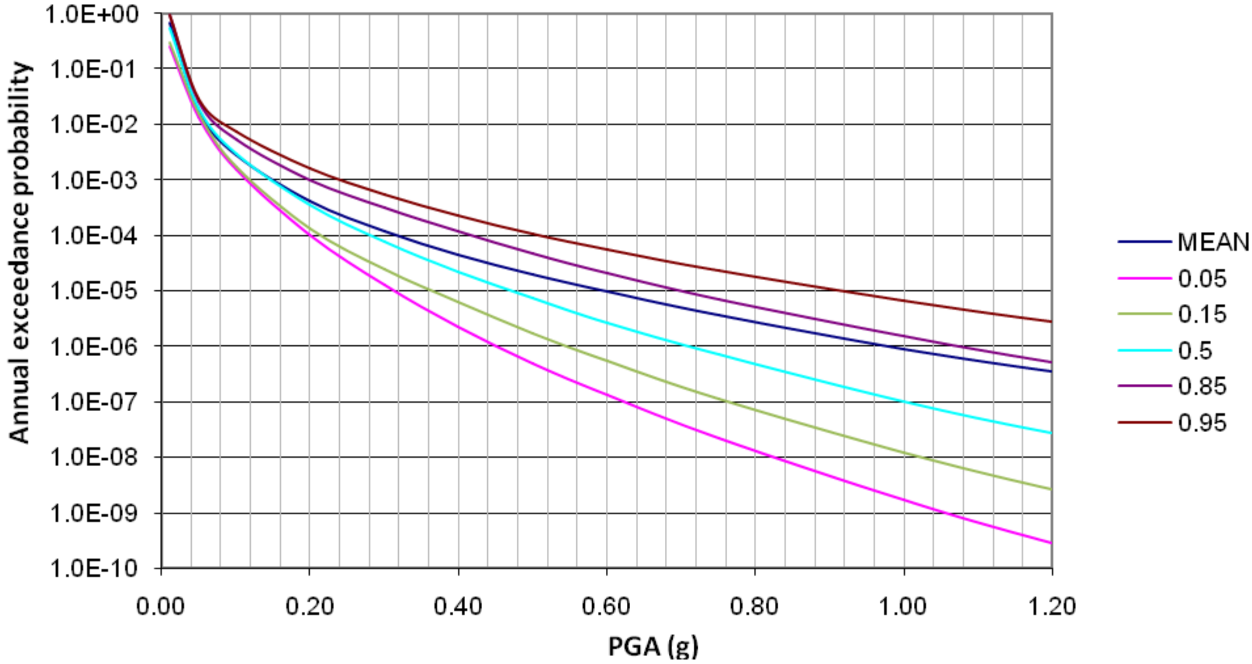


Figure 2.1.1.3-1. Seismic Hazard Curves for ANPP site (horizontal direction).

Table 2.1.1.3-1. Digitization of seismic hazard curves.

Амплитуда	**Уровень достоверности**					
PGA (g)	**Среднее**	**0.05**	**0.15**	**0.5**	**0.85**	**0.95**
1.00E-02	6.89E-01	2.51E-01	3.09E-01	5.37E-01	9.33E-01	9.33E-01
5.00E-02	1.79E-02	1.38E-02	1.59E-02	1.95E-02	2.57E-02	2.95E-02
1.00E-01	2.83E-03	1.51E-03	1.74E-03	3.02E-03	5.25E-03	7.41E-03
2.00E-01	4.12E-04	1.02E-04	1.35E-04	3.55E-04	1.00E-03	1.62E-03
3.00E-01	1.20E-04	1.29E-05	2.40E-05	7.76E-05	3.09E-04	5.56E-04
4.00E-01	4.43E-05	2.29E-06	6.46E-06	2.24E-05	1.18E-04	2.27E-04
5.00E-01	1.97E-05	5.01E-07	1.74E-06	7.41E-06	4.62E-05	1.06E-04
6.00E-01	9.50E-06	1.35E-07	5.37E-07	2.63E-06	2.09E-05	5.69E-05
7.50E-01	3.57E-06	2.24E-08	1.14E-07	7.08E-07	7.16E-06	2.32E-05
1.00E+00	9.11E-07	1.68E-09	1.25E-08	1.02E-07	1.51E-06	6.68E-06
1.20E+00	3.49E-07	2.88E-10	2.63E-09	2.75E-08	5.19E-07	2.72E-06
1.50E+00	1.00E-07	2.75E-11	3.43E-10	5.25E-09	1.22E-07	8.41E-07

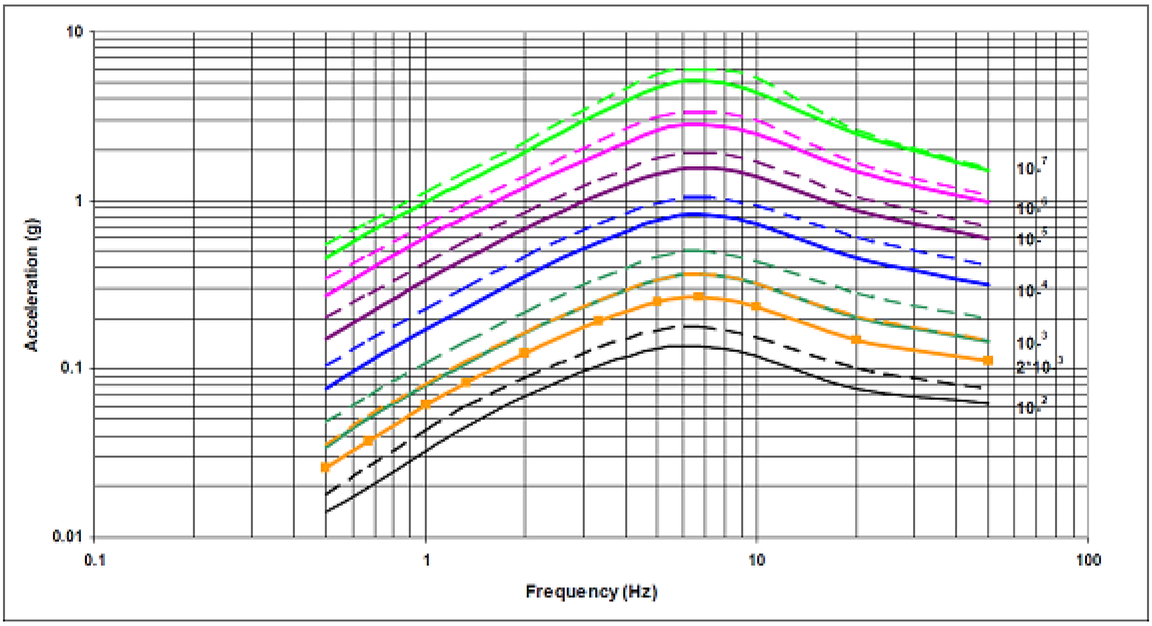


Figure 2.1.1.3-2. Horizontal Uniform Hazard Response Spectra (UHRS) for ANPP site.

Table 2.1.1.3-2. Digitization of Horizontal Uniform Hazard Response Spectra.

Гц	Годовая вероятность 10^{-2} (период повторяемости 100 лет)			Годовая вероятность $2*10^{-3}$ (период повторяемости 500 лет)			Годовая вероятность 10^{-3} (период повторяемости 1000 лет)			Годовая вероятность 10^{-4} (период повторяемости 10,000 лет)			Годовая вероятность 10^{-5} (период повторяемости 100,000 лет)			Годовая вероятность 10^{-6} (период повторяемости 10^{6} лет)			Годовая вероятность 10^{-7} (период повторяемости 10^{7} лет)		
	15%	mean	85%	15%	mean	85%	15%	mean	85%	15%	mean	85%	15%	mean	85%	15%	mean	85%	15%	mean	85%
0.5	0.012	0.014	0.018	0.020	0.026	0.035	0.025	0.034	0.049	0.052	0.076	0.105	0.086	0.150	0.202	0.132	0.272	0.343	0.198	0.458	0.543
0.67	0.017	0.020	0.026	0.029	0.037	0.053	0.038	0.051	0.069	0.072	0.109	0.147	0.120	0.214	0.278	0.185	0.388	0.475	0.270	0.648	0.744
1	0.027	0.032	0.043	0.049	0.061	0.081	0.061	0.079	0.107	0.112	0.170	0.227	0.185	0.335	0.426	0.288	0.601	0.717	0.419	0.984	1.113
1.33	0.036	0.045	0.060	0.063	0.082	0.111	0.079	0.108	0.145	0.147	0.231	0.308	0.249	0.453	0.571	0.390	0.800	0.956	0.563	1.302	1.487
2	0.057	0.068	0.088	0.098	0.123	0.162	0.121	0.161	0.216	0.225	0.353	0.464	0.374	0.682	0.844	0.579	1.188	1.380	0.860	1.930	2.196
3.33	0.089	0.104	0.134	0.152	0.192	0.256	0.195	0.255	0.342	0.363	0.561	0.725	0.623	1.077	1.313	0.974	1.883	2.234	1.506	3.269	3.801
5	0.114	0.131	0.168	0.201	0.252	0.338	0.253	0.340	0.460	0.477	0.761	0.966	0.805	1.444	1.771	1.323	2.598	3.138	2.106	4.674	5.561
6.67	0.119	0.135	0.177	0.210	0.268	0.366	0.268	0.364	0.500	0.512	0.827	1.048	0.881	1.567	1.897	1.456	2.846	3.361	2.391	5.169	5.956
10	0.107	0.119	0.153	0.191	0.233	0.320	0.242	0.319	0.435	0.467	0.722	0.927	0.808	1.383	1.682	1.342	2.460	2.980	2.136	4.376	5.282
20	0.066	0.076	0.101	0.111	0.148	0.207	0.138	0.202	0.283	0.251	0.459	0.604	0.398	0.868	1.047	0.581	1.480	1.660	0.796	2.458	2.606
PGA	0.055	0.062	0.075	0.094	0.111	0.146	0.116	0.145	0.200	0.215	0.316	0.416	0.363	0.592	0.700	0.545	0.981	1.073	0.762	1.501	1.601

The obtained values of peak horizontal acceleration for 50% (0,28g), "mean" (0,32g) and 85% (0,42g) with return period 10 000 years (10^{-4}) confirm the sufficiency of the considered RLE (0,35g) to ensure the ANPP seismic safety.

Detailed structural and geological, geophysical, archeological and paleoseismological studies were carried out aimed at assessment of capable superficial fault. As a result, no traces of a capable fault were identified within the ANPP site 5km radius.

The ANPP site probabilistic volcanic hazard assessment was carried out in compliance with DS-405 standards and IAEA recommendations. Field and laboratory activities were performed (radiometric rock dating with Ar/Ar method implemented in Great Britain and Japan) aimed at data collection and processing for volcanic products. Assessment of volcanic hazard was carried out with different methods, and the output was probability of volcanic activity resumption of $0,5x10^{-6}$.
Final mission of IAEA in September 2011 confirmed completeness of the implemented activities and provided data within the format of geological informational system (GIS), and

approved both the applied methodology and main results, being agreed by regulatory authority.

2.1.2 Provisions to protect the plant against the design basis earthquake

The ANPP with V-270 reactor facility initially had differences compared with its analogues (V-230). First of all they were the measures for seismic protection:

- V-270 type reactor facility was installed at ANPP. It was improved version of V-230 prototype seismically resistant up to the level of 0,4g;
- the Pressurizer and SGs were seismically designed (0,4g);
- in order to decrease the stresses occurring in support structures from seismic effect the main components of primary circuit, i.e., steam generators (SG), reactor coolant pumps (RCP), main stop valves (MSV), were additionally fixed with hydraulic snubbers;
- a number of valves were seismically qualified by testing;
- industrial seismic protection system (SIAZ) was installed. It generates a signal (actuation threshold is 50cm/s^2) for unit trip, crane and refueling machine breaking, closure of valves located on seismic category I pipes of primary and secondary circuits and gate valves of cooling water system channels.

Table 2.1.1.1-1 shows the appropriate measures to improve the earthquake resistance of the ANPP, conducted both during the design and construction of the plant and for the time of its operation.

2.1.2.1 SSC’s required for DBE

During the period from 2001 to 2002 a procedure for the Reactor Safe Shutdown under seismic impact was developed aimed at development of SSEL. The procedure describes and suggests 3 options of safe shutdown: hot, semi cold, and cold. Also, it describes 7 important functions to be implemented for ensuring of criterion of 72-hour safe shutdown period:

- reactivity control and monitoring;
- primary pressure control and monitoring;
- primary coolant inventory control;
- core residual heat and accumulated heat removal;
- ensuring reliable fuel element cooling;
- specific safety functions, functions related to special technological mode of reactor cooldown after earthquake;
- localization and confinement of radioactive products.

The SSEL was developed based on semi cold shutdown procedure.

A number of the IAEA missions resulted in recommendations on SSEL correction, and its final version was approved in 2007. 29 systems were included in SSEL:

1. Reactor;
2. Main coolant circuit;
3. Control and protection system (CPS);
4. Unit №2 main control room;

5. Neutron flux monitoring system;
6. Control and monitoring system;
7. Pressurizer system;
8. Primary circuit overpressure protection system;
9. RCP system;
10. Primary circuit auxiliary feedwater system;
11. Spray system;
12. Hermetic rooms system;
13. System of spent fuel storage pond;
14. Industrial seismic protection system (SIAZ);
15. SG blowdown system;
16. Main steam pipeline system;
17. SG feedwater system;
18. SG emergency feedwater system;
19. Unit №2 SG additional emergency makeup system;
20. Emergency high pressure core cooldown system;
21. Emergency gas removal system;
22. Essential loads cooling system (ELCS);
23. Secondary circuit overpressure protection system;
24. Emergency loads power supply system;
25. Diesel generator automatic load sequencer;
26. Ventilation system;
27. Radiation monitoring system;
28. Buildings and structures;
29. Operative communication.

SSEL components are divided into two groups: A and B. The first group includes the SSC which shall implement their safety functions during and/or after RLE, and the second group includes only the SSC which have no safety functions but shall maintain integrity.

According to the IAEA Guide (TG) a seismic reassessment shall include a plant walkdown that allows preventing in future assessment of a great number of seismically robust components. This assessment process is based on experience of earthquakes and studies (General Implementation Procedure - GIP) for verification of seismic compliance of systems and equipment included in SSEL at operating NPPs.

First plant seismic walkdown was implemented in 2000 (see Table 2.1.1.1-1, item 27). The experts concluded that the ANPP Unit №2 had many features of seismic robustness, and the identified deficiencies were discussed in detail and removed according to the provided recommendations, which was confirmed by the IAEA Mission.

During implementation of the safety upgrading program in 1998-2004 the seismic capacity of Pressurizer safety valves, SG safety valves, main steam isolation valves and stop valve of reversible engine generator was confirmed with use of comparatively conservative response spectra (at PGA = 0,35g).

In 2004 a new system DAP was developed for SG makeup from chemically demineralized water tanks (BZOV-500) in case of the ANPP blackout. It involved installation of additional emergency diesel pump in the boron compartment (Unit №1).

In 2005 during modification of neutron flux monitoring system and DG automatic load sequencer their seismic capacity up to RLE was confirmed with tests using corresponding floor response spectra provided by the ANPP.

Seismic qualification of ANPP Buildings and Structures was confirmed by analyses:
Analysis of the ANPP main building were performed and approved by the regulatory authority in 2004. It resulted in conclusion that the soil under MB basement is sufficiently strong for the applied seismic loads.

In 2007 assessment of the ANPP civil structures (MB, RDGS, VS) capacities were made at RLE (0,35g).

In the frame of Main Building structural analysis two models were considered: dynamic SUPERSASSI model, that takes into account 3D seismic input and soil-structure interaction effects and static STARDYNE model. Verification was performed for structural elements where combined forces (seismic and static) reached maximum values.

Seismic capacity of the Reserve Diesel Generator Building was confirmed by analysis under RLE seismic input with consequent assessment of reinforced concrete columns and metal structures.

In the frame of Peer Review made for Final Seismic Walkdown Dr. Stevenson suggested to perform additional study for evaluation of structural capacity of turbine hall structures interface with auxiliary building (transversal rack) and, also, for testing of internal wall slabs of RDGS for own weight.

These recommendations were implemented in the framework of seismic qualification project ARM/9/022. According to the provided analysis of turbine hall structure and transversal rack interface, the beams have large strength margin for RLE. Also, the RDGS internal wall slabs were tested for 1 g load (own weight) compliance, and according to the analysis the slabs have double strength margin.

Despite the fact that the vent stack (VS) was not included in SSEL, it was also assessed for RLE effect so as to consider its interface with MB. The design of VS (I seismic category) used conservative approach, and though the design PGA was 0,1g the calculations made with advanced methods demonstrated that it completely met RLE requirements (0,35g) and had large strength margin.

For ELCS and spray ponds no additional analyses were performed because they were new structures and were designed for explosion loads. As far as the design explosive loads were assessed as significantly exceeding loads at RLE, these buildings, according to the analyses, had large seismic strength margin.

The boron compartment being a separate underground building was not included in the SSEL because it is an integral component of the reactor compartment, and in the design process starting from 1972 the same requirements was applied to it as to the hermetic compartment (amplification factor 2,7 applied).

However, according to IAEA recommendation verification of the boron compartment was also included in seismic qualification project ARM/9/022, and in the framework of that project the analysis was completed concluding that the boron compartment has sufficient seismic margin.

No special seismic analyses for RLE were performed for the Auxiliary Building and Spent Fuel Dry Storage Facility because:

- Initially the Auxiliary Building was designed at the same seismic level as Hermetic Box of the Main Building (monolith reinforced concrete structure), and, though the design PGA was 0,2g, the analysis used amplification factor 2,7. Verification made after issue of PNAEG–5–006–87 standards confirmed that seismic capacity of the Auxiliary Building completely met those requirements.
- Spent Fuel Dry Storage Facility (SFDSF) has large strength margin for RLE since it is a new structure (2000-2005) and was designed according to contemporary requirements using design PGA=0,46g, and corresponding spectra that exceeded RLE spectra for about 2,5 times, according to “Safety report – Spent fuel storage”, 2000.

Taking into account all said above it could be stated that all ANPP buildings and structures which should maintain their integrity during safe shutdown comply with RLE (PGA=0,35g).
In 2006 the IAEA experts confirmed and ANRA approved the floor response spectra for MB and RDGS of the ANPP, allowing performance of the final seismic walkdown.

According to “ToR for detailed seismic walkdown of ANPP Unit №2” developed by IAEA the CKTI-Vibroseism Co. (CVS) and Czech company Stevenson and Associates performed final seismic walkdown during 2007 outage.

All 2589 SSC included in SSEL were inspected and documented. 23 components were inaccessible for walkdown (currently they are qualified in the framework of the ANPP seismic qualification project ARM/9/022-84188), 655 components were considered as parts of other components, 1193 items passed seismic qualification according to GIP methodology and were excluded from further review.

To provide seismic capacity for the rest of SSEL elements it was recommended to perform additional activities separated into the following three groups:

- "easy fix" measures for seismic upgrading of 446 components;
- more intensive and work-consuming seismic upgrading ("not easy fix"), which may require development of special design and analysis for 29 systems (components);
- additional thorough analysis and/or tests for 26 components.

The walkdown results were summarized in CVS report, which was later provided to the IAEA experts. Dr. J.D. Stevenson performed Peer Review of this Project.

At the present time all easy fixes for 446 components are completed: the ceilings of Unit №2 Main Control Room and Control Protection System room are modified. All identified deficiencies were fixed: it was provided additional upper restraining for number of cabinets and panels, doors of cabinets were secured, adjacent cabinets (panels) were bolted together, I&C racks were stiffened, etc.

The seismic walkdown included also review of the equipment not included in the SSEL but located in the vicinity of the SSEL SSC for their interactions, and review of non-seismic piping for possible flooding. If the interaction problem was found during walkdown, then appropriate recommendation were issued for those components.

In the frame of final seismic walkdown project ARM9014-89032 analyses of specific systems identified as the weakest ones were also performed.

The purpose of performed analyses was to define seismic margin for each of selected components in terms of High Confidence Low Probability of Failure (HCLPF). If HCLPF value appeared to be below 0,35g, then additional upgrading of this component was proposed with consequent reevaluation of seismic margin.

Seismic capacities of the Primary Circuit Piping and Pressurizer System against RLE were evaluated in the framework of leak-before-break (LBB) concept project, according to which surge lines (Pressurizer system) didn’t meet RLE requirements. Therefore, it was recommended to install dampers. Currently 3 viscous dampers are installed.

It was recognized that seismic improvement of the ventilation system located at high elevation (+21,5m) requires a substantial reinforcements. Then, it was decided to modify the whole system. Modification project INSC A1/08T4 “Reinforcement of ANPP Unit №2 main control room functions” was approved and currently in the framework of this project new seismically resistant ventilation system KLM-16 is installed.

Improvement of the seismic performance of piping systems located at el. +14,7m was included in TACIS project. Resulting recommendations for upgrading of Steam Lines, Feed Water and Emergency Feed Water piping are currently implemented: it led to installation of the additional hydraulic snubbers, piping rigid struts and whip restraints; the existing supporting system was partially modified.

The Emergency Gas Removing system is currently being modernized according to the project "Gas removing system from under the Reactor's Lid, ANPP Unit 2". All the design studies are completed, the necessary materials are procured, and implementation is envisaged during 2013 outage.

The IAEA Project ARM/9/022 for ANPP seismic qualification was launched to cover all remaining seismic issues that were discovered during the 2007 seismic walkdown. 109 components were considered in the frame of this work. Depending on existing issues for each component the following activities were carried out: analysis and testing, developing of design solutions and expertise of fulfilled measures.

Seismic resistance was confirmed by SMA for the following systems and components: Control Rod Drive System, PCLS temperature sensors, piping hermetic penetrations for the Main Steam and Feedwater lines, as well as for number of piping systems.

Recommended upgrading (installation of 27 viscous dampers and additional piping restraints) were implemented on the basis of the developed specifications during 2012 Outage for the following piping systems: Pressurizer Cold Injection line, Main Steam and Feedwater lines, piping of spent fuel cooling pond, emergency makeup of the primary circuit and additional emergency makeup. Some components (assembly set 2804HHA) were reinforced as well.

As result of seismic margin assessment the Demineralized Water Tank was seismically upgraded by installation of 48 anchor bolts to prevent its overturning during seismic event.

Finally it could be concluded, that all SSC included in the initial SSEL are seismically qualified for RLE.

In 2012 in frame of Stress Tests implementation program an initial SSEL was extended by inclusion of additional components required for the cooling of Spent Fuel Pool. Seismic Walkdown was undertaken in November 2012 and shows that these components are qualified.

2.1.2.2 Main operating contingencies in case of damage that could be caused by an earthquake and could threaten achieving safe shutdown state

ANPP is equipped with Industrial Seismic Protection System (SIAZ) that provides monitoring of the seismic activity and can actuate signal for the Plant's Safe Shutdown if the Earthquake with magnitude above 6 balls will registered.

SIAZ consists of three stations equipped with seismic detectors. If any of SIAZ seismometers detects earthquake above 6 balls (threshold is ground oscillations with acceleration above 50cm/s2) then acoustic and visual alarm actuates warning for increase of seismic activity at the site.

If simultaneous actuation of two SIAZ stations occurs, then Plant is automatically put in EP-I mode, the refueling machine and reactor hall crane controls are stopped due to SIAZ interlocks.

ANPP has developed and put in the action an operational instruction "ANPP Operational Staff actions during Earthquake". According to this document, all responsible operational ANPP staff in the case of earthquake and after appropriate order of the ANPP Shift Supervisor should organize inspections to check the state of the equipment and systems, detect any visible damages of the buildings and structures, cut off any damaged equipment and systems. Staff permanently keep in contact with ANPP Shift Supervisor informing him about operational situation and follows his recommendations.

ANPP Shift Supervisor takes decision for the NPP mode of operation based on the operational information and according to the requirements of the "Operating Procedures for ANPP Unit 2". At the same time he informs plant's management on the operational situation at ANPP.

2.1.2.3 Protection against indirect effects of the earthquake (seismic interaction issues)

2.1.2.3.1 Internal flooding

The possibility of the internal flooding as result of failure of non-seismically qualified piping and vessels during earthquake was considered in frame of performed seismic walkdown. During this study an ability of the drainage system to mitigate flooding consequences also was considered.

Most of the safety-related equipment and piping located in the boron compartment were seismically qualified for RLE level. However, even in case of the hypothetical rupture of DN400 piping from ELCS (room B-001/2), the volume of the spilled water would be equal to 88 m3. Such volume could not affect operability of the emergency pumps (APN, ASN, NBS, etc.), since a real threat for APN is free volume of $120m^3$ and three drainage pumps located in this compartment provide a total draining capacity $115m^3$/hour.

A potential source of flooding for the Turbine Hall are an open concrete channels. Since in the original design they were considered as important structures they were designed for seismic intensity of 9 balls (twice higher than other structures). As result, their design contains a number of anti-seismic measures:

- Channels are rigid in the horizontal plane due to connection joints of plates and filling joints with mortar;
- Plates are connected with vertical load-bearing structures of the framework;
- Channels are equipped with anti-seismic reinforced concrete belts: vertical elements provide stiffness of the walls;
- Settlement joints are located along opened and closed channels with span 20 - 25 meters. Design of these joints allows the relative movements of the channel's sections without loss of their tightness.

Disk shutters installed on the water-pipes should automatically prevent access of the water from the discharge channels to the turbine hall when SIAZ triggered. These valves are designed for 2g. Concrete channels were considered during Stress Test seismic walkdown and were judged to be robust under RLE. In case of the hypothetical failure of one of the disk shutters, there is also a water board. The main equipment of emergency power supply is located at ground level, and in the Turbine hall there is an open basement (-4.2m). Assuming that the basement is completely flooded, the water through the existing gates may spread to the site, and can not threaten the components of the emergency power supply. Additionally all significant compartments located at ground level are equipped with hermetic doors. Measures for water drainage from the discharge channel are developed to prevent flooding of RDGS building.

Despite the fact that SSEL boundaries on the +14,7m level were limited by BZOK valve, all piping located at this level were seismically upgraded. It should be noted also that at the elevation of 14,7m there is a waterproof layer that protects against flooding important cabinets located below in the room with intermediate relay panel.

2.1.2.3.2 Loss of external power supply that could impair the impact of seismically induced internal damage at the plant.

The following emergency power supply systems designed for 0,4g will be put in operation in case of loss of ANPP external power supply due to the earthquake:

- accumulator storage batteries that can provide electricity to the first category consumers for 7 hours;
- Diesel-generator station, which together with oil-fuel facility can provide electricity for at least 5 days. In case of loss of oil-fuel facility (these components are not qualified for RLE) diesel generator station by its own oil and fuel supply can provide electricity for about 30 hours.

2.1.2.3.3 Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

In the case of extensive damages of the roads during earthquake or formation of the traffic jams, transportation of the ANPP emergency rescue workers will be carried out by vehicles (off-road vehicles and armored fighting vehicles) of the internal forces of Armenia. They provide an ANPP physical protection on the basis of the bilateral agreement on cooperation with the NPP. The transportation route is chosen based on the actual situation on roads.

2.1.2.3.4 Other potential indirect effects induced by the earthquake (e.g. fire or explosion).

ANPP firefighting system is originally designed for 8 balls seismic level. Since components of this system have not been included in SSEL, the system could not be considered as seismically adequate to RLE. Since explosive materials and substances (flammable liquids and combustible) are used at the plant, ANPP is equipped with systems and equipment for fire-fighting, including fire-fighting water supply, installation of fire detection and suppression systems, fire fighting foam, gas fire, stationary fire extinguishing installation and the primary fire protection.

In addition, within the ANPP there are two permanently open entrances for the fire engines, special areas for fire hydrants and water sources. Way to all buildings and facilities are readily accessible.

Firefighting is supposed to be performed by ANPP fire-trucks, some of which are located in the garage, and a few are in the yard, ready to operate in case of fire.

2.1.3 Compliance of the plant with its current licensing basis

2.1.3.1 Licensee's processes to ensure that plant Systems, Structures, And Components that are needed for achieving safe shutdown after earthquake, or that might cause indirect effects discussed under 2.1.2.3 remain in faultless condition and operational

When license is issued the main requirement of ANRA in regard to seismic safety is implementation of seismic reassessment program for RLE, and development and implementation of seismic safety improvement activities. Seismic reassessment was performed in compliance with TG (IAEA №RU-5869). 12 main tasks were defined according to the Technical Guide (see flowchart in Figure 2.1.3.1-1). This figure also shows the consistency and coherence of different tasks that conventionally divided into three phases.

At the first stage, the list of systems and components required for seismic re-evaluation was defined, relevant design and as-built input data were collected, floor response spectra for the main ANPP buildings and structures were calculated (tasks 1-5 and 11).

At the second step seismic qualification of SSEL items has been undertaken on the basis of the data obtained in step 1. Seismic qualification process included plant seismic walkdown and screening of seismically robust components. Necessary seismic analyses and tests were carried out accordingly (tasks 6-9). Recommendations for seismic upgrading were issued as result of this stage.

In the third stage the necessary design solutions for seismic upgrading were developed: installation of additional supports and restraints, reinforcement of structures, etc… Also technical specifications were drawn up for further modification (task 10). Implementation of almost all of the planned activities was completed in 2012.

Spatial seismic interactions were in the focus of 2007 Walkdown. If any of such problems were identified during inspections (falling, proximity, flooding), then appropriate recommendations were issued. Implementation of these recommendations was realized in the frame of IAEA Project N ARM9022-84188 for seismic qualification of the ANPP Unit №2 safety equipment.

To realize the objectives identified by TG, ANRA and ANPP have approved the document "Procedure of ANPP Unit №2 seismic reassessment control process", which specified the methodology and regulatory framework, as well as the format for tasks and subtasks submission (task 11)

According to ANRA request for implementation of this project a number of IAEA experts were involved to provide scientific consultations, review of performed works and for evaluation of the obtained results through filling the appropriate forms (approval status of tasks) and with issue of recommendations if any.

The task was considered to be fully completed when after ANRA inspection it was assessed with the degree of completion "4" (according to the checklist).

Currently, a program of gradual modernization of the seismic monitoring and seismic monitoring equipment is developed.

Other License requirement was realization of IAEA recommendations for implementation of PSHA for ANPP site. These works have been completed and approved by the ANRA.

One more ANRA license requirement also includes performance of monitoring for the control of soil settlement and tilts of important buildings and structures.

During design stage the following monitoring systems were created and installed on the ANPP site:

- High altitude precision leveling network consisting of three cluster, six deep, six ground and three time frames was created at ANPP site to monitor the stability of the soil. Based on the results of the previous cycle and the last (2009) survey measurements, it was concluded that the soil at the ANPP site is stable, elevation changes occurred within the frames of the accuracy of their definitions, and they can serve as input for the determination of settlements (tilts) of buildings and structures.
- According to observations made for the structures settlements and tilts the following ANPP buildings are classified as class I leveling: Main Building, Auxiliary Building, Vent Stack Tower, Sanitary Laboratory Building, cooling tower, water-supply and outlet channels. Analysis of these results led to conclusion that main ANPP buildings and structures are stable.
- In order to determine the impact of the ANPP on groundwater, from 1970 to the present, a hydro-geological monitoring is conducted for 23 bores for the level and mechanical and chemical properties of groundwater from the surface aquifer (depth under the platform of ANPP about 85 m).

These studies confirmed that for the entire period of operation there is no impact of the ANPP on the quantitative and qualitative characteristics of the groundwater.

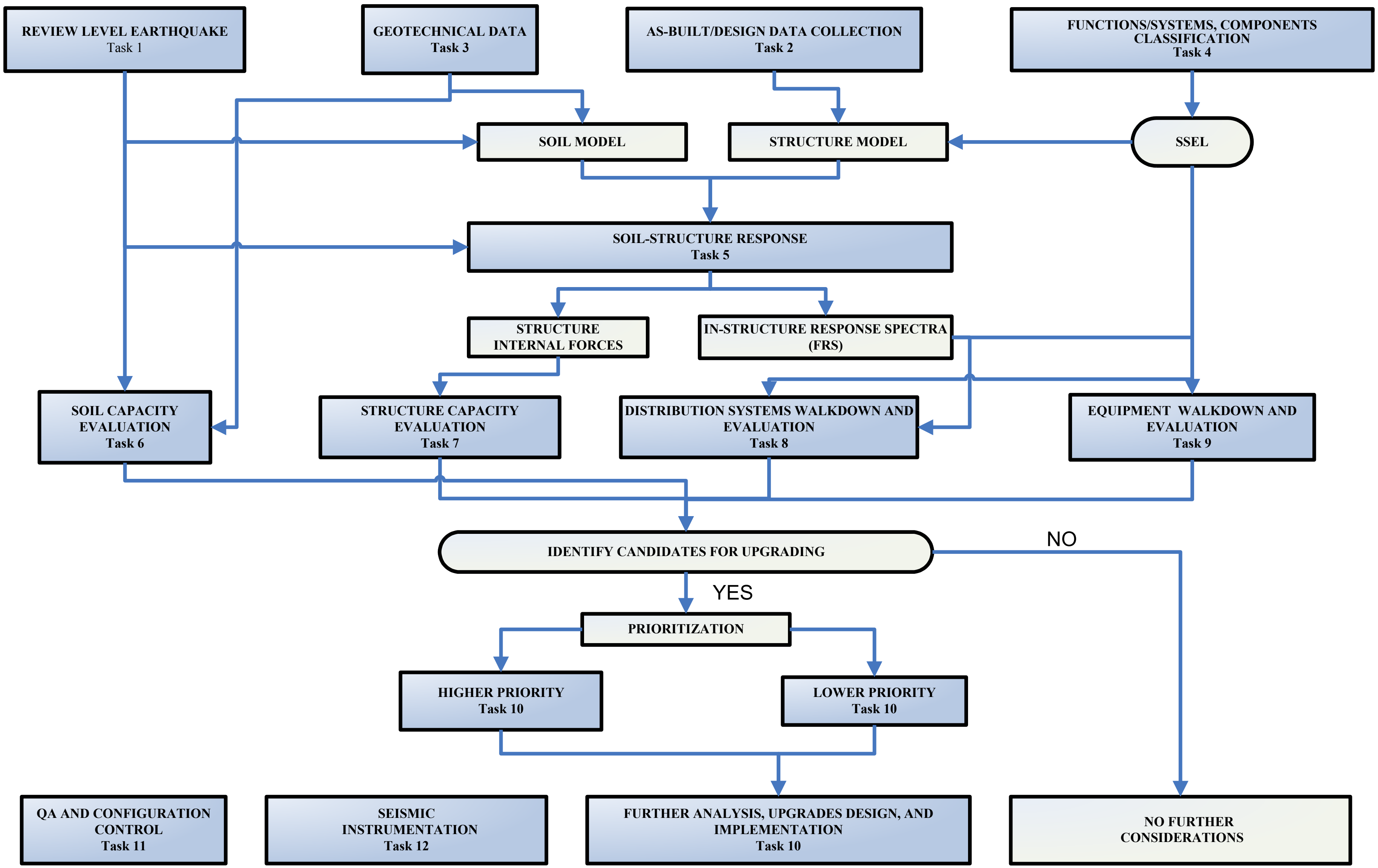


Figure 2.1.3.1-1. ANPP Seismic Reevaluation Tasks.

2.1.3.2 Licensee's processes to ensure that mobile equipment and supplies that are planned to be available after an earthquake are in continuous preparedness to be used

In case of the Diesel Generator's failure under Beyond Design Basis Earthquake it is envisaged to provide Steam Generators make-up from BZOV 1 and 2 tanks with the diesel pump installed in more seismically robust boron compartment (Unit №1).

BZOV tanks will be equipped with hydrants allowing them to be filled with water using fire-tracks located at the ANPP site if loss of power at plant occurs after earthquake.

Due to short time of the independent DG's operation it's planned to design an additional fuel tank with capacity of ≅50÷100 tons that could provide filling of DG reserve tanks and consequently provide emergency power supply in period of 72 hours.

2.1.3.3 Potential deviations from licensing basis and actions to address those deviations

As it was noted above, the main requirement of ANRA in regard to seismic safety was implementation of the seismic reassessment program for RLE, and development and implementation of seismic safety improvement activities.

For example, as part of this re-evaluation old SK-40 batteries were found not-complying to seismic demand. For this reason they were replaced to new set of seismically-resistant batteries VARTA.

Other ANRA licensing requirement was to develop and approve full-scale program for seismic upgrading of I&C. At present, this program has been developed and approved, and the implementation of the program is planned in the near future.

Thus, in view of the above activities, there are currently no identified deviations from the NPP licensing requirements.

2.2 Evaluation of safety margins

2.2.1 Range of earthquake leading to severe fuel damage

Probabilistic Seismic Hazard Assessment (PSHA) for ANPP site has been performed in the frame of works for new ANPP unit. This study met all IAEA requirements, was reviewed by IAEA experts and approved by ANRA. Thus, the data of this project reflect the level of current knowledge of seismic hazard for the ANPP site in the best degree and can be used to perform stress tests.

In conformance with the obtained results a PGA of 0,47g can be taken as reference for Beyond Design Basis Earthquake (BDBE), Figure 2.1.1.3-1. This PGA corresponds to the probability of the annual exceeding 10-5 on the median seismic hazard curve.

It should be noted that selected level of BDBE exceeds PGA = 0,42g with annual exceeding probability 10^{-4} on the 84^{th} percentile seismic hazard curve. This values is also in the good agreement with increased in 1,5 times PGA = 0,32 taken for the annual exceeding probability 10^{-4} on the median curve: 0,32g x 1,5 = 0,48g.

Within the framework of ANPP Unit №2 seismic re-evaluation program for RLE with PGA = 0,35g a Seismic Margin Assessment has been performed for the main components included in SSEL. As it follows from the flowchart presented in Figure 2.2.1-1 seismic capacity was evaluated in terms of HCLPF values. Under this approach for elements that were qualified by means of analysis or testing the specific HCLPF values were obtained using CDFM approach. For items that were qualified with use of indirect methods HCLPF values were estimated as HCLPF > ASL, where ASL = 0,35g - a screening level corresponding to the conservative GIP-WWER procedure that was applied for seismic walkdown. However, it seems to be appropriate, that within Stress Test program it is possible to redefine screening level to value ASL = 0,5g, taking into account the following circumstances:

1. 0,5g screening level is applicable for the existing GIP procedures: GIP-DOE and for the Seismic Margin procedure presented in the documents EPRI;
2. ANPP equipment, when properly anchored and evaluated for potentially damaging spatial interactions and with some caveats, has an inherent seismic ruggedness. For some classes of equipment, the Earthquake Experience Procedure may be used to verify seismic adequacy up to PGA=0,5g;
3. Most of ANPP electrical equipment (cabinets, panels and instrumentation on the racks) has an upper restraining supports performed at different stages of the seismic upgrading;
4. Large seismic upgrading of SSEL items was performed according to recommendations issued during reevaluation program.

Seismic Margin Assessment results are given in the summary Table 2.2.1-1. Appendix A presents a list of reports and documents with analyses and studies made for assessment.

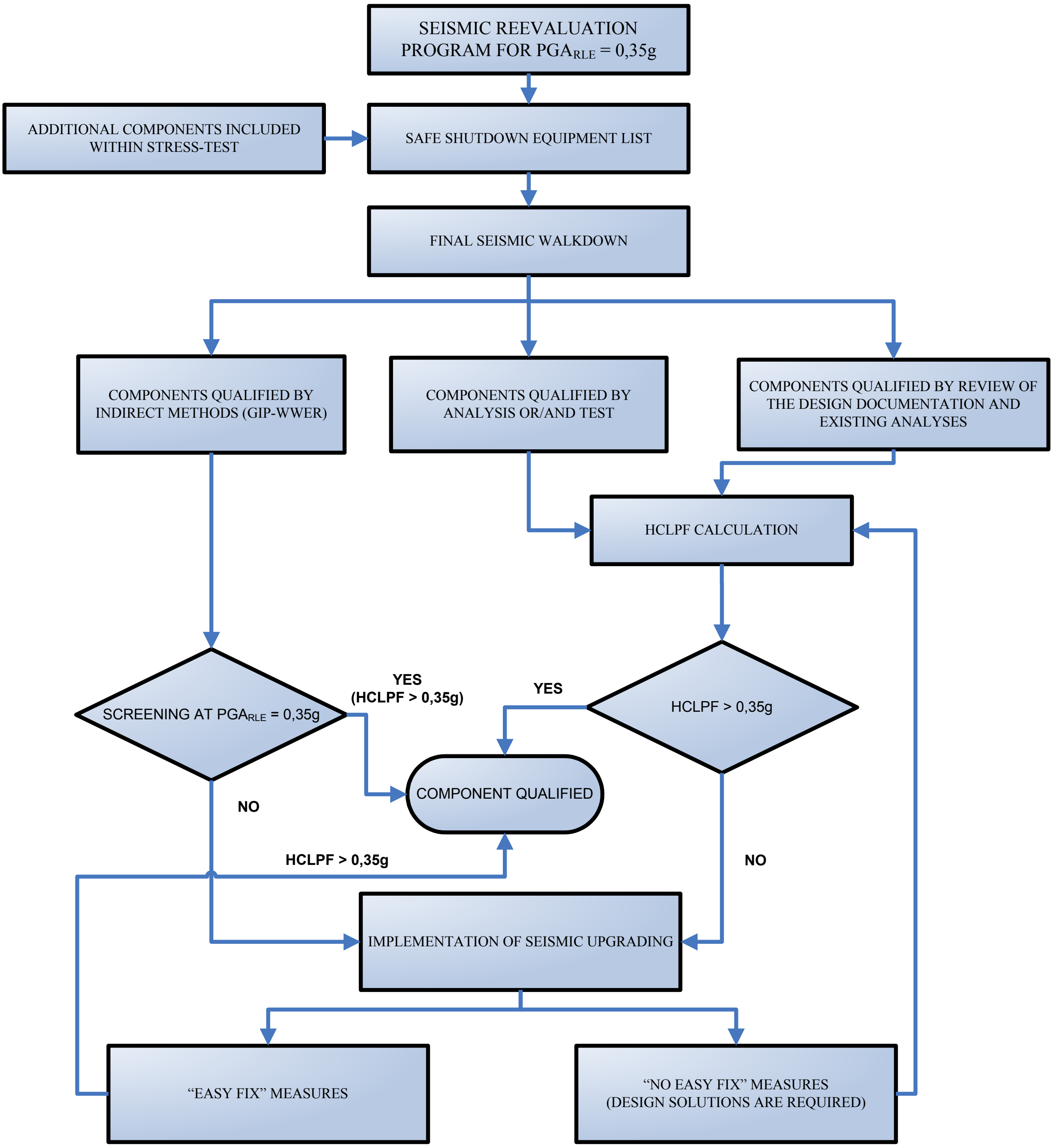


Figure 2.2.1-1. ANPP Seismic reevaluation flowchart.

Table 2.2.1-1. Summary Table of Seismic Margin Assessment for ANPP SSEL components

N	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Notes
1	Reactor Facility	Strength of Reactor Supporting Structure	1,6	0,63	
		Strength of Upper Unit's rods	0,63		
		Control Rods insertion time in AZ mode	0,7		
2	Primary Circuit Loop	Piping, loops 1, 6 DN 500	0,49	0,48	(1)
		Piping, loops 2, 5 DN 500	0,48		
		Piping, loops 3, 4 DN 500	0,50		
	2,1 Pipelines of blowdown, blowdown return and primary circuit drainage	from MCP №1	1,26	0,46	
		from MCP №2	0,46		
		from MCP №3	0,61		
		from MCP №4	0,85		
		from MCP №5	0,59		
		from MCP №6	0,54		
3	Control and Protection System			0,5	(2)
4	Unit №2 main control room			0,5	(2)
5	Neutron Flux Monitoring System			0,5	(2)
6	Control and monitoring system			0,5	(2)
7	Pressurizer system	Pressurizer Support structure	0,82	0,43	
		Piping (DN200)	0,47		
		Pressurizer cold injection piping (DN100)	0,43		(3)
		Discharge piping from Pressurizer to Bubbler tank	0,6		
8	Primary circuit overpressure protection system			0,5	(2)
9	RCP system			0,5	(2)
10	Primary circuit auxiliary feedwater system	EPF1 (Emergency Feed Pump)	0,58	0,58	
		EFP2	0,71		
		EFP3	0,62		

N	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Notes
		EFP4	0,98		
		EFP5	0,94		
		EFP6	0,73		
11	Spray system	Sprinkler Pump 1	0,38	0,38	
		Sprinkler Pump 2	0,38		
		Sprinkler Pump 3	0,38		
		Suction Line	0,86		
		Discharge Line	0,54		(4)
		Return piping	0,96		
		Boron solution pipeline for cleaning	0,94		
12	Hermetic rooms system		0,49	0,49	
13	System of spent fuel storage pool	Piping	0,85	0,5	
		NRB-1 (Spent fuel pit cooling circuit pump)	0,5		
		NRB-2	0,56		
		NZB (Pool Infilling Pump)	0,54		
14	Industrial seismic protection system (SIAZ)		0,5		(2)
15	SG blowdown system	SG blowdown lines in A014 Room	0,42		
		SG blowdown lines in SG compartment, loops 1 - 3	0,54		(5)
		SG blowdown lines in SG compartment, loops 4 - 6	0,46		
16	Main steam pipeline system	from SG1	0.40	0,39	
		from SG2	0.5		
		from SG3	0.5		
		from SG4	0.45		
		from SG5	0.48		
		from SG6	0.39		
17	SG feedwater system	from SG1	0.65	0,40	
		from SG2	0.40		
		from SG3	0.44		
		from SG4	0.43		
		from SG5	0.42		
		from SG6	0.55		

N	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Notes
18	SG emergency feedwater system	piping at level +14,7m	0,62	0,62	
19	Unit №2 SG additional emergency makeup system	piping in Reactor Compartment	0,81	0,55	
		piping in the boron compartment	0,66		
		weld joints of supports	0,55		
		BZOV -1,2 (Demineralized Water Tanks)	0,55		(6)
20	Emergency high pressure core cooldown system	piping in Reactor Compartment	0,81	0,55	
		piping in the boron compartment	0,66		
		weld joints of supports	0,55		
		1ASN (Emergency Seismic Pump)	0,77		
		2ASN	0,62		
		BZOV-3,4	0,55		(6)
21	Emergency gas removal system	piping	0,78	0,78	
		supports	1,6		
22	Essential loads cooling system (ELCS)	ELCS piping in the boron compartment	0,46	0,46	
		ELCS piping (Pump Station)	0,64		
		ECLS buried pipes	0,55		(7)
23	Secondary circuit overpressure protection system		0,5		(2)
24	Emergency loads power supply system	Diesel Generator	0,6	0,45	
		Cable trays (mostly loaded span)	0,45		
		Accumulator Storage Battery VARTA	0,9		
		Return Diesel Generator	0,6		
		ECLS buried pipes	0,55		(7)
25	Diesel generator automatic load sequencer			0,5	(2)
26	Ventilation system	Frame of the pedestal	0,57	0,57	
		load-bearing frame	1,1		

N	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Notes
		side panels	0,62		
		testing of active components	0,62		
27	Radiation monitoring system			0,5	
28	Buildings and structures	Main Building (Turbine Hall)	0,51		(8)
		Reactor Pit	15		(8)
		Reactor Box Compartment	0,98g		(8)
		Cooling Pools 1 and 2	0.52g		(8)
		Reactor Compartment, Reinforced Concrete Wall, row V, level 10,5-14,7	0,88		
		Boron Tank	0,98		
		RDGS	0,44		(8)
		ECLS Pump Stations 1 and 2	0,621		
		Dry Spent Fuel Storage	1,18		
		Auxiliary Building	0,77		
		Ventilation Stack	2,3		(8)
		Intake Water Building № 2	0,46		(8)
29	Operative communication			0,5	(2)

Notes:

(1) Load capacity of the hydraulic snubbers governs the value of HCLPF for Primary Circuit (HCLPF = 0,36g). The values shown in the table correspond to seismic capacity of the Primary Coolant Piping assuming the loss of all hydraulic snubbers.

(2) HCLPF is defined according to the screening procedure.

(3) HCLPF value is revised in comparison with data presented in report Rep02.ARM9014-89032S, the rationale is given in the report.

(4) HCLPF value is revised in comparison with data presented in report Rep08.ARM9014-89032S, the rationale is given in the report.

(5) Rep09.ARM9022-84188 presents value of HCLPF = 0,42g. This value is determined by the load capacity of viscous damper installed on piping. However, short-term exceeding of the permissible load is not a critical failure for the system. Value given in the table corresponds to seismic capacity of the piping.

(6) HCLPF value is revised in comparison with data presented in report Rep17.ARM9022-84188, the rationale is given in the report.

(7) HCLPF value is revised in comparison with data presented in report Rep12.ARM9014-89032S, the rationale is given in the report.

(8) HCLPF assessment is given in the report.

The following conclusions could be drawn from the results presented in Table 2.2.1-1:

- No cliff edge effects were identified for the items included in SSEL.
- The minimum HCLPF value defined among all SSEL components is 0,38g (Sprinkler Pumps located in boron compartment). However, it should be noted, that the sprinkler system, although was included in SSEL, is not directly involved in the Safe Shutdown procedure and, thus, its failure may not be an initiating event for a severe fuel damage.
- It should be understood that presented HCLPF values were obtained from a fairly conservative estimates, and according to applied probabilistic procedure they mean the 50th percentile of 1% probability of failure of considered element or system under design seismic event with a probability of the annual exceeding 10^{-4}. At the same time, events considered in the process of NPP Stress Tests are beyond design basis external events and have a much lower probability of occurrence (10^{-5}-10^{-6}). In accordance with ASCE 43-05 for the beyond design basis events a Target Performance Goal is less than about a 10% probability of unacceptable performance, that means that corresponding seismic margin for ANPP is about 1.5 times higher than HCLPF values defined for RLE. In terms of PGA this value is 1,5*0,38 = 0,57g, that more than 20% above the reference seismic level set in the procedure of the NPP Stress Tests.
- Thus, as it follows from the said above, ANPP has a sufficient seismic margin for the beyond design basis earthquakes with PGA = 0,47g.

2.2.2 Range of earthquake leading to loss of containment integrity

From the Table 2.2.1-1 follows, that the minimum HCLPF value for the systems provided sealing and integrity of the Confinement is 0,49g. This value corresponds to the strength of the piping containment wall penetrations. At the same time, the HCLPF value for the massive box part of the reactor compartment is 0,98g. Thus, containment integrity is provided with 10% probability of unacceptable performance at seismic event with PGA = 0,74g.

2.2.3 Earthquake exceeding the Design Basis Earthquake and consequent flooding exceeding Design Basis Flood

Because of the geographical location of the ANPP territory, scenario for flooding exceeding Design Basis Flood is not applicable.

2.2.4 Measures which can be envisaged to increase robustness of the plant against earthquakes

1. Assessment of ANPP seismic margins provided in this report is based primarily on the deterministic CDFM analyses performed in the framework of SMA procedure.

Performing of seismic PSA will allow identifying systems and components that may require further seismic improvement.

2. The project "Gas removing system from under the Reactor's Lid, ANPP Unit №2" should be implemented.
3. To provide an emergency power supply for period of 72 hours, an additional fuel tank with capacity of ≅50÷100 tons should be installed.
4. The programs for seismic upgrading of I&C equipment and seismic monitoring system should be implemented.
5. Perform seismic margin evaluation of the fire extinguishing system and implement recommendations for seismic safety improvement up to DBE.

3. FLOODING

The following description covers the Unit №2 reactor installation, the spent fuel pools of Units №1 and №2 and dry spent fuel storage facilities available at ANPP site.

3.1. Design basis

3.1.1. Flooding against which the plants are designed

The Armenian NPP is situated on the northern slope of the Ararat Valley. The ANPP site is sloped 1,5 to 5,0 % southward towards the Ararat Valley. The site is located about 934,5m above sea level.

All the waterways and basins existing in the site region are located to the south of the plant site. The Araks River is 16 km away from the site; the bed of the Araks River is located at 820 m above sea level. The Sevdzhur River, an inflow of the Araks River, is flowing south at a distance of 8 km from the site; the river bed is located at 860m above sea level. The Aknalich Lake is 5 km away from the site; the lake is located at 880m above sea level.

Thus, the water basins existing near the plant site are more than 50m lower than the absolute ground elevation of the plant.

The underground water infiltrate into the volcanic rocks at the sides of the Aragats Mount, moves to east/southeast towards the Ararat valley, forming subsurface flows. In the vicinity of ANPP the groundwater is at 86÷95 meters depth below the surface. The underground water flows in the Aygrlich Lake and through the set of Kulibeklinski and Aygrlichski water springs at the level of 840 to 850 m above the sea level, creating the flow channel of the Sevdzhur River. The monitoring shows that there is a tendency for groundwater level decrease during last years. Submersion of the site due to groundwater upswing is excluded.

The site is on the northern slope of the Ararat Valley. Mudflows coming from the upper part of the slope could be considered as a potential flooding source. However on the northern part, ANPP site is protected by hills. A walk-down of the peripheral area has shown that the area relief forms favorable conditions for water flow by-passing the plant site. The area between the hills and the plant site has little sloping and there are no conditions favorable for formation of intense mudflows (based on engineering judgment). On the north-western part a mountain canal for disposal of possible mud flows is foreseen. The site is also protected against mud flows from the neighboring rising grounds by the site fence concrete wall (about 2m high). However there is a lack of detailed calculations aimed to prove above mentioned justification.

During the design of ANPP flooding of the site was not considered as an essential hazard, therefore design basis flood was not properly evaluated. Based on the description of the geographical location, relief and water reservoirs of the site it can be stated that the plant is not subject to flooding due to high level water or other external natural phenomena which could lead to water level increase. Taking into account this statement, the flooding of the plant site is considered to be possible only in case of heavy rainfall and outlet canal overflow or breach.

When there is an inflow of significant amount of water into the outlet canal and that water level increases, the water will overflow from canal where a special overflow zone has been created by decreasing the height of the canal wall (see Figure 3.1). So in case of outlet canal overflow the water will bypass the DGS, due to the slope (overflow zone) as indicated in Figure 3.1. However there is still a danger to have a breach in outlet canal wall, which will lead to water absorption in the soil in vicinity of DG building. This could lead to relatively slow water penetration to basement of DG building (see analysis of DG building flooding in section 3.1.2).

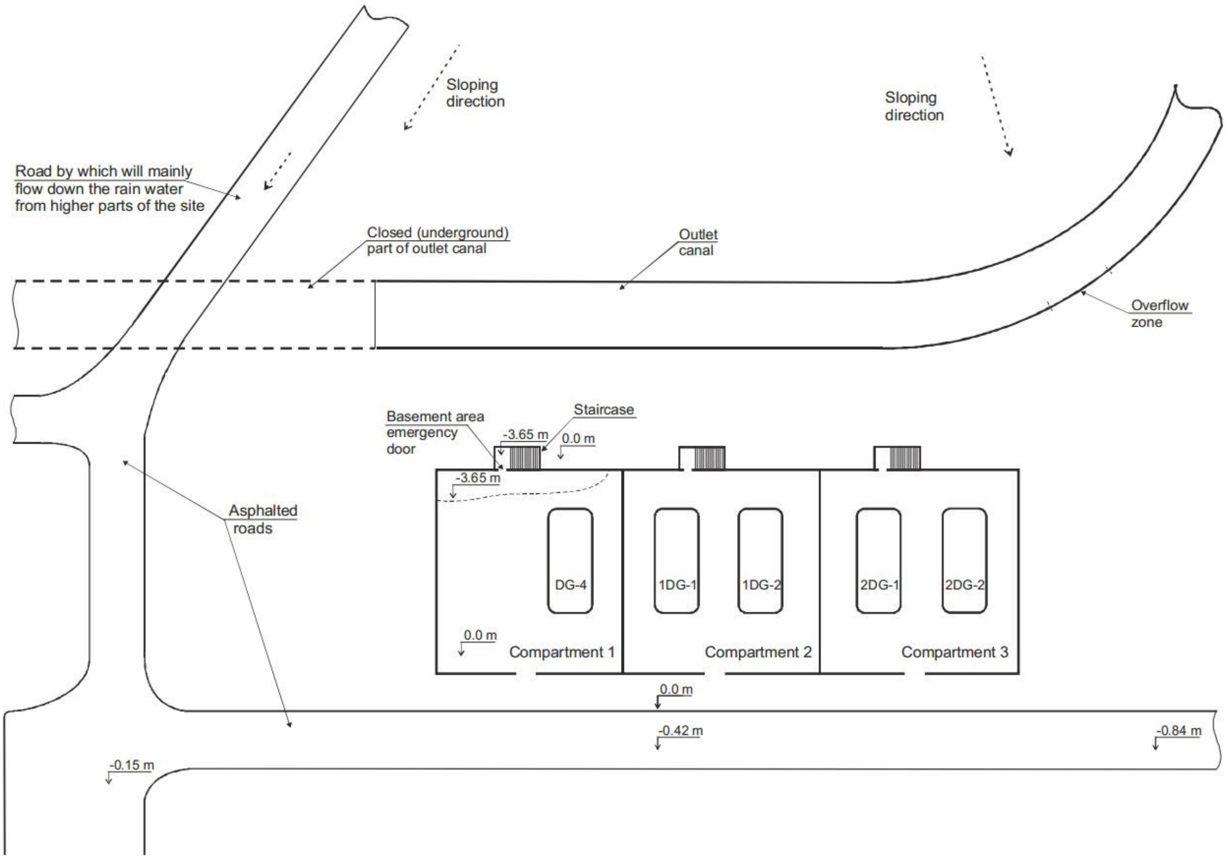


Figure 3.1. Layout of DG building and outlet canal overflow zone location.

The extreme rainfall is specified and addressed in the design of ANPP. The design basis rainfall for ANPP (buildings of reactor installation including spent fuel pools) was set to 145 l/sec/ha with the duration of 5 minutes (~4.35mm) and 65 l/sec/ha with the duration of 20 minutes (~7.8mm). In both cases values corresponds with a return period of 1 year. According to historical data the largest values of precipitation typically take place from April to June.

At ANPP site an Independent Spent Fuel Storage Installation (ISFSI) was constructed in the 1990-2000ies which contains horizontal storage modules (HSM) with dry shielded canisters (DSC). The DSC and HSM are designed for an enveloping design basis flood, postulated to result from natural phenomena such as a tsunami, and seiches. For the purpose of this bounding generic evaluation, a 15 m flood height is used.

Rainfall water removal from the plant site is organized by special sewer system which consists of network of underground piping. The collection of water from the site is done by special catch basins. Rainfall water from building roofs is entering the sewer network through the special outlets from the buildings.

The evaluation of design basis for plant sewer system was implemented using the method of limit of rainfall intensities. The method for identification of maximal flow is based on the assumption that for any water collector available at the site, the time to concentrate the flow equals to time that is necessary for flow to reach the final cutset. This assumption is used as a base for the method of rainfall intensities, that could be formulated in following manner: the flow rate of sewage water in the specific cutset will reach maximal value in case if duration of rainfall equals to time that is necessary for flow to reach this particular cutset from the farthest point of the investigated slope.

The design basis amount of rainfall water (Q) was calculated based on correspondent Soviet standards (SNiP) using following empirical equation:

$$Q=\Psi*A*F/(\tau+T)n$$

where Ψ - average coefficient of rainfall water flow, F – calculated area of rainfall water flow, τ – time of rainfall water flow concentration, T – total duration of rainfall water flow up to the farthest point of sewer system, A – coefficient of rainfall intensity depending on the geographical location (A depends on q20 parameter which is set to 65 l/sec/ha with the duration of 20 minutes taking into account geographical location of ANPP), n is a coefficient depended on geographical location of the plant (for ANPP site n equals to 0.57). Implementation of mentioned analysis allowed concluding that the design basis for ANPP sewer system should be not less than 1200m^3/h.

In the frame of ANPP external events PSA study the sewer system was re-assessed for 286 l/sec/ha (which corresponds to return period 10000 years). Analysis showed that sewer system is capable to prevent site flooding even in case of rainfall with intensity 286 l/sec/ha which is higher than design value by factor of 4.4. The analysis of rainfall with higher intensity was not performed.

Independent Spent Fuel Storage Installation (ISFSI) is evaluated for the design basis 15 meter hydrostatic head of water producing external pressure on the DSC shell and outer cover plates. A uniform pressure equal to the static head of 15 meters of water (0.15MPa) is applied to the external surfaces of the axisymmetric model of DSC. The maximum stress intensities in the DSC components resulting from the flood load condition were combined with the appropriate loads to formulate load combinations. The resulting total stresses for the DSC were compared with allowable stresses and it was concluded that calculated load stresses are much lower than allowable stresses.

As explained above the protection against site flooding is assured by plant sewer system.

Reassessment of the plant sewer system shows that existing sewer system is designed with large margins. According to the re-assessment results no flooding problem encountered even in case of rainfall intensity higher than design value by factor of 4.4. Mentioned analysis allows concluding that existing means of protection of reactor installation against rainfall are adequate.

As to Independent Spent Fuel Storage Installation (ISFSI) the design base flooding value equals 15 meter which is extremely unlikely to happen taking into account geographical location of ISFSI. The analysis performed for ISFSI allows concluding that additional protection against external sources of the site flooding is not needed.

3.1.2. Provisions to protect the plants against the design basis flood

The analyses of external flooding possibilities have been mainly performed during design stage (reactor design and fuel storage design) and later in the frame of external hazard PSA implementation. In addition, after Fukushima accident the Armenian NPP together with Nuclear and Radiation Safety Center (NRSC) launched a co-operation project with IAEA aimed to apply fault sequence analysis (FSA) method for ANPP. The main objective of project’s objective is to perform a complementary analysis of plant robustness by assessing potential impact of external hazards and their credible combinations using the FSA method and the software Fault Sequence Tool for Extreme Events (FAST-EE) developed by the IAEA. So the analysis presented below is mainly based on the results of mentioned studies and expert judgment.

Vulnerability to flooding is different for each type of structures, systems and components. Mainly mechanical, electrical and I&C components were considered vulnerable against external flooding. The analysis is done only for the civil structures housing safety significant structures, systems and components.

Following buildings are to be considered in the assessment of vulnerability to flooding:

- Reactor building and it’s auxiliary building (primary make-up system, boron preparation and supply system, spray system, SG make-up emergency seismic systems, spent fuel pools and their cooling systems)
- Turbine hall (SG feedwater systems, steam dump valves)
- Intermediate building (electric switchgears, cable tunnels, control rooms)
- DG station (diesel generators and supporting equipment)
- Essential service water system pump houses (essential service water pumps)
- Auxiliary building (boron preparation and supply system)
- Chemical treatment building (demineralized water pumps which are necessary to replenish water to assure long-term cooling)
- Dry spent fuel storage.

In addition open yard equipment also included in the scope of the analysis (i.e. transformers). Schematic representation of geodesic elevations of plant buildings is shown on Figure 3.2

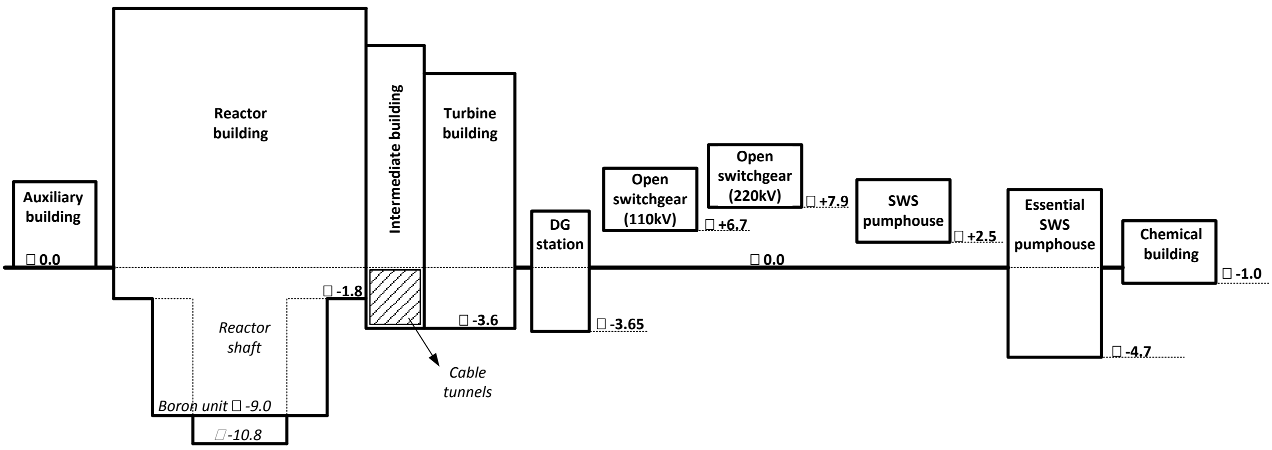


Figure 3.2. Schematic representation of geodesic elevations of plant buildings.

Reactor installation and spent fuel pool. Flooding safety margin assessment was not performed for reactor installation and spent fuel pool. However, implementation of above mentioned projects (e.g. PSA, FSA-ANPP) allowed revealing main vulnerabilities against heavy rainfall. Direct impact of water on equipment in case of water penetration through the buildings roofs was investigated (turbine building, reactor building, DGS, etc). Drains have been installed on the roofs of buildings that route the rainwater off the roof. To prevent roof collapse discharge pipes have been installed on the roofs. Discharge pipes route the rainwater out along the facade. Investigation of roofs structural properties allowed concluding that large amounts of water cannot collect on the roofs, even if the drains are clogged, therefore roof collapse due to water accumulation was considered non-applicable.

- **Turbine Hall.** The majority of turbine hall is not vulnerable for flooding. The water can penetrate the turbine hall through the doors at the ground floor in conditions of accumulation of water in surrounding area or through the windows (left open or broken by the high wind associated with the rainfall). Water penetrating through the windows can not affect equipment located at elevation ∇+14.7 of the intermediate building, because windows are located at axis “A” of TH building, whereas elevation ∇+14.7 is located between axis “Б” and “В” (distance between axis “A” and “Б” is 40 meters). Water penetrated from windows could affect steam dump valves to the condenser (BRU-K) which are located at ∇+6.3 elevation close to the axis “A” and start collecting at basement elevations.
  At the lower elevation of TH water can penetrate and affect safety-related equipment at:
  - ground elevation ∇0.0 where several electrical cabinets and RHR systems pumps are located
  - underground elevation ∇-1.8 auxiliary feedwater pumps are located,
  - underground elevation ∇-3.6, where main SG feedwater systems pumps are located.

  In addition it is possible to have water penetration from TH to boron compartment (see below description of reactor building flooding) and to cable tunnels. The water entering into the turbine hall will be drained to the basement area. Failure of equipment due to the flooding is possible only in case of continued increase of water level at elevation ∇-3.6. Rough estimation shows that about 2.500m^3 of water is needed to reach the critical level inside the Turbine Hall. The basement area of the turbine hall is equipped with conduits of "dirty" drainage system. The water in this system is dumped through a collector into the well of the receiver tank and then into the receiver tank from where the water is sent to purification by DFP (“dirty” flows pumps) pumps (see Figure 3.3).

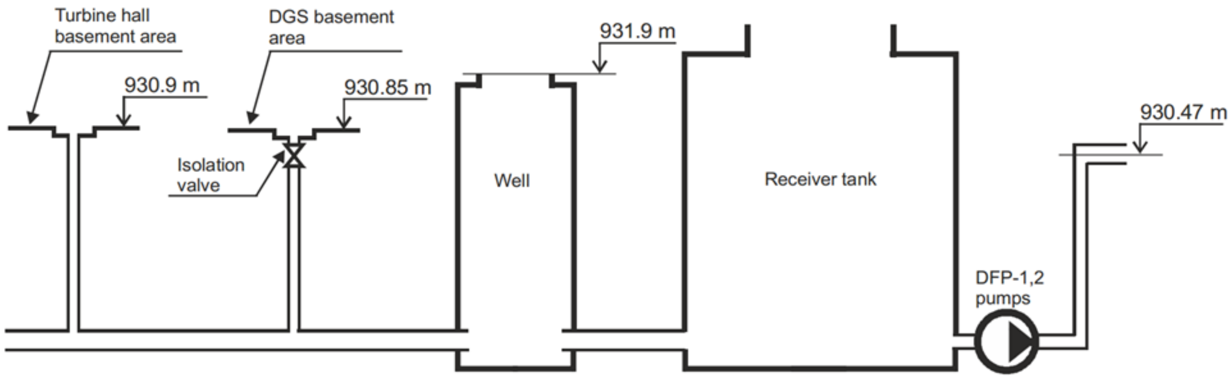


Figure 3.3. TH and DGS drainage system layout.

In case of DFP pumps failure the accumulation of water and increase of the level in the system will take place. This will result in accumulation of water in the whole system as it consists of communicating vessels. The well of the receiver tank is an underground tank which communicates with atmosphere; its upper part is 931.9 m above sea level (level -2.6 m). When water reaches this level an overflow from the well into the surrounding area will take place, and water will go in direction of sloping without any threat for plant safety related systems. The overflow phenomenon limits the maximum possible level in the system. Taking into consideration the possible inaccuracies of the levels as well as the need of some hydraulic pressure for water flow, the maximum level is assumed to be 30 cm higher, which corresponds to 932.2 m above sea level. This corresponds to a level of 1.3 m above the floor in the elevation ∇-3.6. This water level is sufficient to affect pumps of main feedwater oil system, which will directly lead to the failure of main feedwater pumps. The emergency feedwater (EFW) pumps are installed at 1.8 m from the floor of the elevation ∇-3.6, the RHR pumps are installed at ∇+0.2 elevation. Performed estimation allowed concluding that even in case of conservative assumptions the water accumulation in the basement of Turbine Hall level cannot lead to the failure of EFW and RHR pumps. The analysis performed allows concluding that the flooding of the turbine hall during heavy rainfalls will lead to failure of main feedwater system and the possible water overflow to the boron compartments and cable tunnels. The auxiliary feedwater pumps however will not be challenged by the water level.

- **Reactor building.** Most of the compartments in the reactor building are not vulnerable to flooding. The only possible locations in reactor building where water can penetrate and affect safety-related equipment are boron compartment #1 and boron compartment #2 which are located at elevation ∇-9.0. Despite the fact that both boron compartments are isolated by doors the water can principally penetrate into the boron compartments through leakiness of the non-waterproof emergency door (see Figure 3.4). The emergency door is exiting to the staircase connected to the turbine hall by another non-waterproof door. The door is opening in direction of staircase, thus the opening of door under water static pressure is excluded. Though in case of staircase flooding the water can penetrate to the boron compartment under static pressure of the water, the flow rate is considered to be low due to the limited opening surfaces on the doors between staircase and boron compartment.

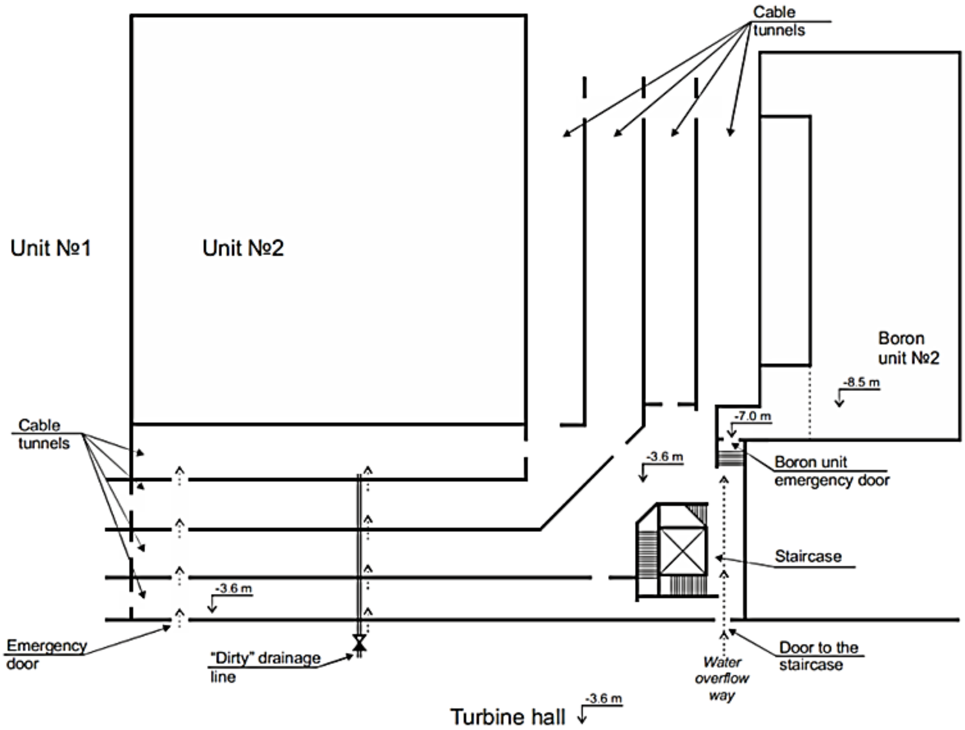


Figure 3.4. Possible water overflow way from Turbine Hall to boron compartment 2[1].

Even in case of water penetration the boron compartment is protected from flooding by special water drainage system. The boron compartment of Unit №2 is equipped with three drain pumps (capacity of first and second pump is 20 m^3/h (each) and third pump – 65m^3/h) to remove water from the boron compartment sump. The pumps are starting automatically in case of flood signal and can be powered by DGs. The boron compartment of Unit №1 is equipped with two drain pumps (capacity of 2 of the pumps is 20m^3/h). Even in case of pumps failure the rising of water up to an unsafe level for equipment operation should take a very long time (according to the rough estimation about 350m^3 of water should penetrate to the boron compartment room to lead to an equipment failure). However in case of hypothetical flooding of both boron compartments the plant will lose the possibilities to supply water from emergency SG make-up systems as well as to implement primary feed-and-bleed procedure. Taking into account that the water could penetrate to boron compartment only in case of flooding of TH lower elevations, the operability of main and emergency feedwater systems located in TH could not be credited in such hypothetical flooding scenario. Based on the information given above mentioned

1

scenario is very unlikely due to the double protection doors and special drainage system installed in each boron compartment. However since the hypothetical scenario has significant safety impact the efforts should be spend on investigation of the doors between TH and boron compartment in order to assure their leaktightness or to replace them by waterproof doors.

- **Diesel generator station.** The DGS has three separate compartments each of them has its own basement area. Water can penetrate the basement area through the external non-waterproof emergency door at the level of basement. The emergency door is always closed and located significantly below the building blind area in the cavity where theoretically water can penetrate and accumulate, and then overflow through the non-waterproof door. In the basement area DG supporting equipment is installed: DG air cooling fans, DG cooling water valves, oil pumps to keep DG in "hot standby" mode and DG start-up oil pumps. With the water level rise in basement area mentioned equipment could fail which will lead to the failure of all DG simultaneously. The DGS basement areas are equipped with conduits of "dirty" drainage system which is connected with turbine hall as it is described above (see Figure 3.3). As it was shown above in case of the failure of drainage system the water level could reach 932.2m above sea level which corresponds to a level of 1.35 m above the floor of the DG basement. This level is enough to reach all of the mentioned DG support equipment. Reaching the water level of about 45 cm in the DGS basement area can be considered as cliff edge effect, as the failure of generator air cooling system will result in sharp degradation of temperature conditions of generator and can lead to DG failure. Thus, the level increase up to 45cm in the basement is possible only in case of failure of drainage system DFP pumps and existence of a large amount of water (approximately 2500m^3) which could flood both DGS and TH as communicating vessels. The likelihood of such scenario is extremely low taking into account that the flow rate is considered to be low due to the limited opening surfaces on the emergency door at basement of DGS. However in order to eliminate mentioned hypothetical scenario the efforts should be spend on investigation of the emergency doors at basement of DGS in order to assure their leaktightness or to replace them by waterproof doors.
- **Essential service water system pump houses.** The flooding of pump stations of Essential Service Water System (ESWS) related to heavy rainfall is excluded from the analysis due to the protective measures taken in account in the design of the plant. The relief of the area surrounding the pump station excludes any water accumulation near the doors. The doors to the pump station are hermetic and located about 10 cm above the surrounding area. In addition ESWS pump stations don’t have any doors/penetrations that are located lower than the surrounding area level.
- **Intermediate building.** The most of the compartments in the intermediate building are not vulnerable to the flooding. The only possible locations in reactor building where water can penetrate and affect safety-related equipment are cable tunnels located at elevation ∇-3.6 (see Figures 3.2 and 3.3). The water can flow from the basement area of the turbine hall to the cable tunnels through a non-waterproof emergency door (see Fig. 3.4). The water flow between these areas is also possible through the "dirty" drainage system. On this line a check valve is installed in order to prevent the flow from turbine hall to the cable tunnels in case of turbine hall flooding. Water can penetrate to cable tunnel in case of check valve failure. When

assuming the sticking in open position of the check valve, it can be considered that the water rising in these areas will be simultaneous. The flooding in cable tunnels could lead to multiple spurious activations and failures of safety related components. Therefore it is important to spend efforts in order to assure leaktightness of the door between TH and cable tunnels. In addition it is important to increase reliability of check valves installed to prevent water flow from TH to cable tunnels by drainage system.
- **Chemical building.** The main safety-related equipment in the chemical building are demineralized water pumps that are located in the central hall of chemical department. The water can penetrate through non-waterproof doors. It was concluded that pumps failure will occur in case of water increase up to 20cm. The pumps are important for accident scenarios that require long-term cooling. Therefore it is essential to spend efforts in order to assure leaktighness of the chemical building doors.

On-site spent fuel storage. Flooding of the ISFSI greater than 0.46m above grade results in blockage of the HSM inlet vents. Flooding of the ISFSI greater than 1.7m above grade results in wetting of the DSC. Greater flood heights result in submersion of the DSC and blockage of the HSM outlet vents. Although the ISFSI site is not subject to flooding even for extreme conditions, the DSC and HSM are generically designed for an enveloping design basis flood, postulated to result from natural phenomena such as a tsunami, and seiches. As it was described above the design base flooding value equals 15 meter which is extremely unlikely to happen taking into account geographical location of ISFSI. The analysis performed for ISFSI allows concluding that additional protection against external sources of the site flooding is not needed.

Summary:
The analysis shows that during continuous heavy rainfalls with very small probability, when considering very conservative assumptions, the following rooms are under risk of flooding:
- Basement area of turbine hall with possible water penetration to cable tunnels and boron compartments,
- Basement areas of diesel-generator station.

As it was already mentioned these areas are connected by the "dirty" drainage system, but their flooding will simultaneously occur only, if the DGS is not isolated from the system by closing of the corresponding isolation valve in the "dirty" drainage system. If loss of offsite power accident is postulated then in case of simultaneous flooding of TH and DGS the only possibility to perform reactor cooling function is activation of diesel pump feeding the Steam Generators, located in boron compartment of Unit №1. However, as it has been mentioned there could be water propagation from TH to boron compartments through non-water proof doors. In this case the flooding scenario could lead to reactor core damage, because of inoperability of diesel driven pump due to boron compartment 1 flooding.

The mentioned scenario is very unlikely due to the double protection doors and special drainage system installed in each boron compartment. In addition the time window available for operator to implement recovery actions is large. However, since the hypothetical scenario has significant safety impact the efforts should be spend on investigation of the doors between TH and boron compartment in order to assure their leaktightness or to replace them by waterproof doors.

3.1.3. Plants compliance with its current licensing basis

For maintaining the operability of systems, structures and components that are needed for achieving and maintaining the safe shutdown state, regular testing, examinations and inspections are performed at the ANPP. Periodic surveillance tests and maintenance allows to assure operability of the systems and to monitor performance of components.

Plant's systems, structures, and components are categorized based on their safety significance. The graded approach is used for systems surveillance test and maintenance. The general approach is that the systems which have higher safety class are tested with higher periodicity than the lower safety class systems. In order to improve the feedback from surveillance test and maintenance activities ANRA is in the process of development of maintenance effectiveness monitoring requirements for ANPP.

All the equipment used for protection against water accumulation on-site and in-building is stationary no any portable equipment is foreseen at ANPP. So far, deviations from the licensing basis have not been observed.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

Investigation of the geographical location and relief of the ANPP site shows that the water basins existing near the plant site are more than 50m lower than the absolute ground elevation of the plant. The groundwater is about 86 to 95m below the plant site. The monitoring shows that there is a tendency for groundwater level decrease during last years. Submersion of the site due to groundwater upswing is also excluded. The considered flooding possibilities were only due to heavy rainfall and outlet canal destruction.

The design basis rainfall for ANPP (including reactor installation and spent fuel pools) was set to 145 l/sec/ha with the duration of 5 minutes (~4.35mm) and 65 l/sec/ha with the duration of 20 minutes (~7.8mm). The re-assessment of plant sewer system was performed for 286 l/sec/ha (which corresponds to return period 10000 years). Re-assessment showed that sewer system is capable to prevent site flooding even in case of rainfall with intensity higher than design value by factor of 4.4. Thus, existing sewer system is designed with large margins. The analysis of consequences of rainfall with higher intensity was not implemented.

Within safety margin assessment the main plant vulnerabilities against rainfall were investigated. Analysis was performed using over-conservative assumptions:

- it was considered that heavy rainfall could lead to loss of offsite power
- failure of TH and DGS drainage system pump was assumed
- credit was not given to staff actions to recover them

- it was assumed that water can penetrate from TH to boron compartment through doors leakiness with enough capacity to flood both boron compartments.

Only in case of all mentioned conditions are fulfilled the scenario could lead to loss of reactor and spent fuel pool cooling function, therefore the rainfall intensity which is enough to create mentioned conditions should be defined in future by detailed safety margin assessment study.

3.2.2. Measures which can be envisaged to increase robustness of the plants against flooding.

In spite of the extremely low probability of site flooding and measures already available, additional provisions are considered to further increase safety level of the plants as follows:

1. Equip the emergency doors of the staircases of DGS basement areas with a border in such a way that the penetration of water to the basement area can be excluded.
2. Equip DGS with alarms indicating occurrence of water level in basement area with output of light signals in MCR, central control panel and DGS operator room.
3. Foresee mobile equipment devoted to water pumping out from DGS and its basement.
4. Develop a procedure for operators for the case of water inflow in the DGS basement area.
5. Develop and implement measures with the purpose of building barriers on the way of water flow to the turbine hall gates.
6. Assure leaktightness of the doors located between TH and boron compartment or to replace them by waterproof doors
7. Enhance reliability of drainage system elements aimed to prevent water penetration from TH to cable tunnels
8. Perform detailed safety margin assessment in terms of rainfall flooding of ANPP site.
9. Perform detailed calculations aimed to prove that mudflow protection measures are enough to prevent mudflows impact on ANPP systems.

4. EXTREME WEATHER CONDITIONS

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

The ANPP site is located within the Lesser Caucasus in the Ararat plain. This region is characterized by a sharply continental and semi-desert climate. A distinctive feature of the climate of this region is the abundant sunshine and the warmth. The duration of sunshine is about 2600-2700 hours per year. The average solar radiation for the year on a flat surface reaches 6343 MJ/m^2. From north and north-east, Ararat valley is protected by the main Caucasian ridge that prevents direct intrusion of cold air masses from the north. From south, Agridags Ridge and Mount Ararat protect the plain from the southern winds.

Data from long-term records (since 1920) have been used to define the climate characteristics in the area of the ANPP site. These records come from meteorological stations and monitoring stations of the Armenian Department of Hydrometeorology within a radius of 40 km around the ANPP site.

Winter in the Ararat valley is moderately cold. Statistically the coldest period is the end of January. The minimum temperatures reach -30°C, during snowy winters the snow depth could reach up to 40 cm. The cloudy condition of the sky prevails in the winter season. The spring is wet in the plain of Ararat, with a large variability of the weather conditions. Spring temperature is highly variable (average temperature in March is about 4,5-5°C; average temperature in May is about 15,5 - 17°C). The highest rainfalls of the year occur during spring. The cold front to the back of cyclones produces heavy rainfall. Sometimes, the daily rainfall could be higher than the monthly rainfall. Summers are hot, dry and rather long. The average duration of the summers is about 4 to 4,5 months. The start and end of the summer season depend on the atmospheric conditions and could greatly vary from one year to another. The monthly average temperature in July and August exceeds 25°C with a maximum temperature of 40°C. Some years the soil surface is heated above 70°C. Early summer is characterized by thunderstorm activity with an average of 11 days of thunderstorm in June and a maximum up to 20 days. The prevailing wind direction is east and north-east.

The information about average and maximum parameters of main site characteristics are presented in Table 4.1.

Table 4.1. Average and maximum parameters of ANPP site characteristics.

#	Parameter	Average	Maximum
1.	Snow depth (mm)	120	240
2.	Hail (number of days)	2.1	2.4
3.	Rainfall (mm)	298	340

4.	High temperature (^{O}C)	+26.2[2]	+40.8
5.	Low temperature (^{O}C)	-1.7	-30
6.	Humidity (%)	56	-
7.	Wind speed (m/sec)	1.1	24
8.	Gust speed (m/s)	-	32

The design of the ANPP was carried out on the basis of Soviet construction norms and rules. The following parameters were taken into account in the design basis of the plant[3]:

- Wind load: wind speed of 27 m/s
- Snow load: 80kg/m^2 (0.8kN/m^2)
- High temperature: +42°C
- Low temperature: - 40°C.

Independent Spent Fuel Storage Installation (ISFSI) is also analyzed in regard with external hazards impact. The following parameters were taken into account in the design basis of the plant

- Wind load: wind speed 160 m/s
- Snow load: 10kN/m^2 (including snow and ice loading)
- High temperature: +52°C
- Low temperature: - 40°C.

Several external hazards originally were not explicitly considered in Armenian NPP design.

However, continued safety assessments at ANPP allowed to revisit this assumption and try to re-evaluate the necessity to expand the scope of external hazards analysis. Main sources of the updated information are ANPP safety assessment report and external hazards PSA. A broad list of possible external hazards to be considered is recommended by international standards (e.g. IAEA documents). Screening analysis performed within ANPP external hazards PSA allowed concluding that the list of hazards considered in the design documentation is comprehensive, no other natural hazards are found significant for ANPP. The only point, which was highlighted by additional safety assessments, is that not all of the wind and rainfall interactions were considered during ANPP design stage. Particularly it was identified that wind induced dust level increase could be challenging for plant safety. Thus many of the potential hazards were considered as non-applicable for ANPP site (such as tropical cyclone, tsunami, inflow and big waves, etc.). In addition many hazards were considered as having extremely low likelihood of occurrence. Particularly:

- Tornado was not considered in the design of the plant. According to current building standards and maps of the intensity of the tornado, for the mountain regions of

[2] For high and low temperatures the average value is presented correspondingly for hottest and coldest months

[3] Seismic and Rainfall hazards are discussed separately in Chapter 2 and Chapter 3 respectively

Armenia it was considered as a rather improbable phenomenon (1.9x10-6[1/y]). Due to the low probability of occurrence, the tornado phenomenon could be neglected.

- The ANPP site probabilistic volcanic hazard assessment was carried out in compliance with DS-405 standards and IAEA recommendations. Field and laboratory activities were performed (radiometric rock dating with Ar/Ar method implemented in Great Britain and Japan) aimed at data collection and processing for volcanic products. Assessment of volcanic hazard was carried out with different methods, and the output was probability of volcanic activity resumption of 5x10-7[1/y]. Final mission of IAEA in September 2011 confirmed completeness of the implemented activities and provided data within the format of geological informational system (GIS), and approved both the applied methodology and main results.

In the frame of external hazards PSA hazards curves were identified for selected hazards. According to the results the frequencies of design base parameters was identified (see Table 4.2)

Table 4.2. Frequency of occurrence of originally postulated design base parameters.

N	Hazard	Value	Frequency [1/y]
ANPP buildings and SSC			
1.	Wind load	27 m/s	2.0E-01
2.	Snow load	0.8 kN/m2	6.3E-02
3.	High temperature	+42°C	1.0E-02
4.	Low temperature	- 40°C	1.0E-04
Independent Spent Fuel Storage Installation			
1.	Wind load	160 m/s	<<1.0E-07[4]
2.	Snow load	10 kN/m^2	<<1.0E-07
3.	High temperature	+52°C	<<1.0E-07
4.	Low temperature	- 40°C	1.0E-04

However external hazards were analyzed in more detail for selected ANPP structures and buildings. Safety margins identified for selected ANPP structures are presented in chapter 4.2.

In the initial safety assessment studies a combination of hazards was screened out from consideration, however lessons learned from Fukushima accident shows that this topic could be essential especially for correlated hazards and combinations of high frequency / low magnitude hazards. In order to address this issue ANRA's technical support organization (NRSC) together with Armenian NPP launched a co-operation project with IAEA aimed to apply fault sequence analysis (FSA) method for ANPP. The FSA method was developed recently by IAEA and already successfully applied for Goesgen NPP in Switzerland. In this

[4] The hazard curve is not analyzed in detail for extremely low likelihoods, hence the exact frequency value is not available

method, critical combinations of the components, structures and human errors leading to core damage are analyzed in terms of different external hazards.
The main outcome of the mentioned above activity was identification of combinations of external hazards which could be critical for NPP design. In order to analyze different combinations of external hazards the operational limits of safety-related components have been thoroughly collected for following external hazards:

- Seismic[5]
- External flood[6]
- High winds (including dust storm)
- Snow load
- High temperatures
- Low temperatures.
- Lightning.

The results of the project show that following combinations of external hazards could be challenging for ANPP safety.

- **Seismic event during long lasting period with high temperature.** Long lasting high air temperature could not lead to significant challenge of ANPP safety if ventilation systems are operable (see details in chapter 4.2). However in case if at the same time seismic event could affect ventilation systems then it could lead to temperature increase in switchgear compartments and consequential loss of offsite power. In such a scenario switchgear equipment could become unavailable and even operability of DGs will not allow to supply electrical power to consumers. In such scenario there is still a possibility to perform reactor cooling function using diesel pump located in boron compartment of Unit №1 and SG safety valves that have high seismic resistance and could be operated without electrical power supply. Therefore it is important to assure operability of diesel-pump and SG safety valves in case of high air temperature.
- **Seismic event during long lasting period with low temperature.** Long lasting low air temperature could lead to loss of offsite power accident with unavailability of emergency cooldown system and diesel driven pump (see details in chapter 4.2). In such case operability of DG is considered to be critical for implementation of reactor cooling. Low air temperature could not affect DG operability by itself. The most vulnerable DG equipment is local diesel fuel tank. Freezing of the fuel in the tank or associated pipes could lead to DG failure. Fuel freezing temperature is -35OC, whereas the diesel fuel tank is located in DG building where the heating system maintains the air temperature above 5OC. However in case of heating system failure the freezing of diesel fuel lines could lead to failure of diesel generators. Heating system failure could occur in case of seismic event. Thus seismic event during long

[5] Seismic is discussed in chapter 2 of stress-test report
[6] External flood is discussed in chapter 3 of stress-test report

lasting period with low temperature could be challenging combination of external hazards for ANPP safety. It is necessary to mention that mentioned scenario could occur in case of simultaneous impact of low temperature and seismic event. According to plant procedures the impact of low temperature on emergency cooldown system and diesel driven pump would be detected within 24 hours. So in order for this scenario to happen seismic event should occur within those 24 hours. Therefore the likelihood of such combination is considered to be extremely low.

- **Low temperature and heavy snow load.** Low air temperature could lead to loss of offsite power accident with unavailability of emergency cooldown system and diesel driven pump (see details in chapter 4.2). In such emergency feedwater system is the only possibility to supply SG and perform reactor cooling function. However if low temperature will be combined with snow cover formation then the snow load could theoretically affect roof of turbine building with consequential failure of emergency feedwater system. In such scenario there is still a possibility to perform reactor cooling function using primary feed & bleed procedure with the electrical power supply from DGs. Hence for mentioned scenario operability of DG is considered to be critical. DG building has high capacity in terms of snow load, therefore it is also important to assure operability DGs in case of extremely low air temperature.
- **High wind associated with dust concentration increase.** According to the analysis performed high wind could lead to loss of offsite power. In case if high wind occurs in the summertime it is possible to have significant increase of dust concentration in the air. Increase of dust concentration in the air could lead to failure of DGs due to violation of allowable dust level at the DG air intake. Failure of DGs due to wind induced dust will lead to station black-out. In such scenario there is still a possibility to perform reactor cooling function using diesel pump located in boron compartment of Unit №1 and SG safety valves which could be operated without electrical power supply. Therefore it is important to assure operability of diesel-pump and SG safety valves in case of high wind and high dust concentration.

The original design of the plant unit did not take into account all the possible weather phenomena or other natural conditions. However the plant robustness against external hazards has been improved by continued plant modifications (e.g. installation of diesel driven pump to supply water to SG). External hazards analysis is mainly addressed in SAR and PSA. Analyses show that impact of different external hazards as well as their combinations is not leading to the core damage. It was found that most of the external hazards could lead to loss of offsite power or station black-out conditions which could be resolved using existing systems and procedures (see details in chapter 4.2). Therefore it could be concluded that existing means of protection of reactor installation against external hazards are adequate.

As to Independent Spent Fuel Storage Installation (ISFSI) the design base values correspond to external hazards with extremely low frequency of occurrence (see Table 4.2). So, the analysis performed for ISFSI allows concluding that additional protection against external hazards is not needed.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

As it was noted before the main sources of information on external hazard analysis for ANPP are ANPP safety assessment report and external hazards (EH) PSA study.

High wind. The effect of high winds includes consideration of wind pressure impact and possible increase of dust concentration in the air. Within EHPSA the possible wind pressure impact was analyzed. It was concluded that wind with 59 m/s speed could lead to collapse of ventilation stack (see Figure 4.1), which could lead to serious damage to auxiliary building where boron preparation tanks are located. It was concluded that wind is not leading directly to serious effect on buildings. The only effect is damage of window glasses and open switchgear in case of wind speed equals to 33 m/s.

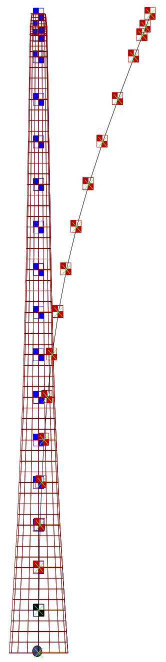


Figure 4.1. Mechanical model of ventilation stack for analysis of wind.

The other type of wind impact is possible increase of dust concentration in the air and wind-induced missiles that could affect safety-related equipment. It was assumed that equipment in the vicinity of turbine building windows could be damaged in case of major glass break which could occur when wind speed is above 33 m/s. Missiles also could affect open yard transformers therefore loss of offsite power is postulated. Increase of dust concentration in the air was found to be dangerous for DG air intake system. According to the analysis the critical level of dust concentration for diesel generator air intake system is 0.08 g/m3 (the frequency of scenario with DG failure due to high dust concentration is estimated as 1.7E-04 [1/y]). The loss of grid due to the high wind associated with the dust storm is also postulated. So external hazards PSA study consider mentioned scenario as a station black-out with frequency 1.7E-04 [1/y]. However mentioned scenario does not lead to core damage, because there is still a possibility to perform reactor cooling function using diesel pump located in boron compartment of Unit №1 and SG safety valves that are passive and could be operated without electrical power supply.

Snow loads. Design basis for ANPP site on snow load is based on Soviet SNiP standards (~0.8 kPa/m^2). The mentioned value is reflecting the snow load which is typical for that particular site. However realistic calculations show that the building roofs have much more capacity than given in the design basis. The realistic estimation of the roof fragility is available for following buildings (see Figure 4.2):

- Reactor building – 2.96kPa/m^2 (the hazard curve is not analyzed in detail for events with extremely low likelihood, hence the exact frequency value is not available)
- Turbine building - 1.78kPa/m^2 (the frequency corresponds to 3.57E-05 [1/y])
- Diesel Generator Station - 2.74kPa/m^2 (the hazard curve is not analyzed in detail for extremely low likelihood events, hence the exact frequency value is not available)

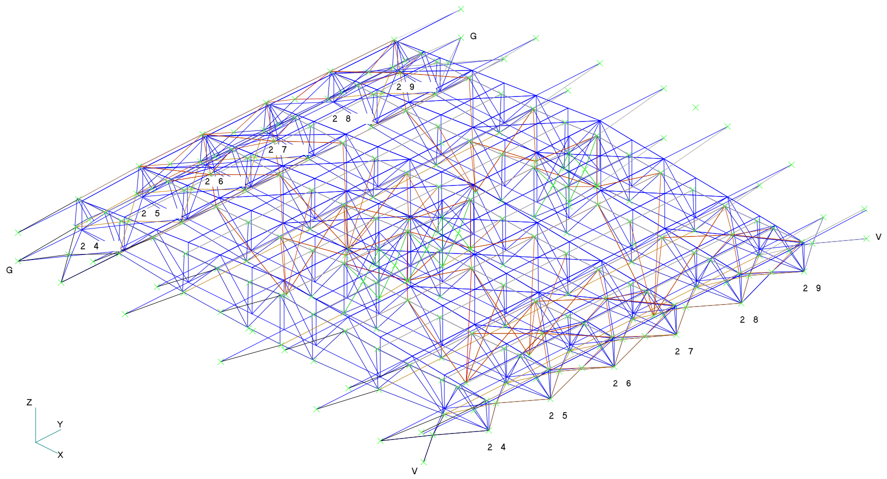


Figure 4.2. Mechanical model of reactor building roof for analysis of snow load.

The collapse of reactor building roof could affect spent fuel pool located in reactor building central hall. During normal operation the spent fuel pool is covered by concrete plates which are considered to be sufficient for fuel protection against roof collapse taking into account that reactor building roof consist of light structural elements. In case reactor building roof collapse will occur during outage then spent fuel damage could be postulated. However such kind of scenario is considered to be very unlikely because of high capacity of the reactor building roof against snow (2.96kPa/m^2 snow loading could occur with the frequency lower than 1E-07 [1/y]) and the fact that this scenario is only possible during plant outage which covers only 15% of the year. In addition it should be noted that outage period is usually scheduled for September – October, whereas the earliest snow cover formation is registered on November 25 (data of Hoktemberyan meteostation). Therefore this scenario is considered to be negligible for ANPP.

The collapse of turbine building roof could theoretically affect all the equipment located at turbine building. This will lead to the failure of main and emergency SG feedwater systems. In this case reactor cooldown still could be arranged by providing feedwater to SG from emergency cooldown system or diesel driven pump. Even in case of failure of mentioned SG feedwater systems, cold shutdown state could be achieved and maintained using primary feed & bleed procedure using emergency make-up system pumps. All mentioned equipment is located in boron compartments of reactor buildings of Units 1 and 2 and could not be affected by snow load. The frequency of snowfall which could potentially lead to turbine building collapse is quite low (3.57E-05 [1/y]), taking into account that there are still 3 different systems that could provide reactor cooling the scenario with core damage in case of turbine building roof collapse is considered to be unlikely.

The collapse of diesel building roof could all DGs located under it. The loss of grid is typically postulated for extreme weather conditions like a heavy snow. It could be stated that snow load that could damage the DGS roof could lead to station black-out. However such kind of scenario is considered to be very unlikely because of high capacity of the DGS roof against snow (2.74kPa/m^2 snow loading could occur with the frequency lower than 1E-07 [1/y]). However even mentioned scenario does not lead to core damage, because there is still a possibility to perform reactor cooling function using diesel pump located in boron compartment of Unit №1.

Extreme temperatures. Both extremely high and low temperatures could be challenging for ANPP safety. The nature of the hazard connected with extreme air temperatures is that it is long lasting process with sufficient time available for operator to take actions. Generally it was concluded that in case of operability of heating, ventilation and air conditioning (HVAC) systems the extreme values are not dangerous for Armenian NPP.

High temperature.
Most vulnerable points of ANPP in terms of high air temperatures are compartments with electrical and I&C systems. Conservative calculations performed for those compartments allowed concluding that most of the compartments are not susceptible for significant temperature increase even in case of ventilation system failure. The process of temperature increase is considered to be slow. According to calculation results equipment of some switchgear compartments could fail if external air temperatures are in the range of 55-60^{O}C (the frequency of such air temperature is extremely low <<1.0E-07). Thus even in case of long-term temperature increase and unavailability of ventilation system the worst scenario that could appear is station black-out due to unavailability of switchgear. However even this scenario does not lead to core damage, because there is still a possibility to perform reactor cooling function using diesel pump located in boron compartment of Unit №1 and SG safety valves that are passive and could be operated without electrical power supply.

Low temperatures.
Possible impact of low temperatures was also analyzed within external hazards PSA study. According to external hazards PSA analysis most vulnerable points of ANPP in terms of low air temperatures are:

- Outlet pipes of demineralized water tanks (BZOV). Those pipes provide water to emergency cooldown system and diesel driven pump. According to the conservative calculations[7] outlet pipes of demineralized water tanks (BZOV) could freeze if air temperature is -31.5°C with the frequency of occurrence 9.1E-03 [1/y].

[7] Several conservative assumptions were made during calculation of pipes' freezing, therefore presented value is expected to be conservative. More precise calculations are planned in order to identify realistic air temperature at which freezing of pipes of demineralized water tanks (BZOV) is expected.

- Open switchgear. It is assumed that the operability limits of open switchgear corresponds to plant design value which is -45°C with the frequency of occurrence approximately 1.0E-06 [1/y].
- Main transformers. Main transformers minimal design temperature is -45°C with the frequency of occurrence approximately 1.0E-06 [1/y].

Thus in case of air temperature decrease up to -42°C the plant can face loss of offsite power accident with unavailability of emergency cooldown system and diesel driven pump. In this case there are still two possibilities to implement reactor cooling function:

- Use of emergency feedwater system to supply SG. However for long-term water supply continues water replenishment to deaerator is necessary. Replenishment of deaerators is organized by: 1) NBZK pumps from BZOV demineralized water tanks or 2) from chemical department using NPRO pumps. In case of mentioned scenario the 1st option will be unavailable due to freeze outlet pipes of demineralized water tanks (BZOV). 2nd option also will be unavailable because NPRO pumps are not powered from DGs. Therefore long-term SG supply by emergency feedwater system could not be credited.
- Implementation of primary feed-and-bleed procedure.

It could be noted that in mentioned scenario the power supply from DG is a critical function which allows implementing reactor cooldown. The importance of DGs is also conditioned by the failure of diesel-driven pump system due to freezing of outlet pipes of demineralized water tanks (BZOV), which is the alternative way to cooldown the reactor in case of station black-out. Therefore operability of DGs in case of extreme low temperatures is critical. Data on the fuel used for the diesel generators indicates that it is suitable for operation in temperatures upper than –35 °C, at which temperature the fuel starts to solidify. The local diesel fuel tank is located in DG building where the heating system maintains the air temperature above 5°C. However in case of heating system failure the freezing of diesel fuel lines could lead to failure of diesel generators and consequential core damage.

Lightning. Operating experience, both the Armenian nuclear power plant, and other power units, shows that the maximum negative impact from a lightning stroke consists in disconnection of the power unit from an external network, with the subsequent transfer of the power unit to a safe condition. Lightning also could be a possible reason for an electromagnetic impulse due to overvoltage, it can have negative influence on the I&C system of the Power plant. However there is a lack of detailed analysis for lightning impact on ANPP, particularly following aspects should be covered:

- quantitative values of lightning stoke considered in lightning protective system,
- determination of vulnerability of safety equipment under electromagnetic field,
- objectives of lightning protection in terms of residual electromagnetic field on safety equipment (direct and indirect effects),
- justification of external and internal protection meets the requirements coming from lightning protection objectives.

The information presented above describes potential impact of different extreme weather conditions to the plant operation. Vulnerabilities are identified based on available analytical base. Though safety margins (excluding seismic events) are not precisely estimated for ANPP, it was also shown that even extreme weather conditions with extremely low likelihood of occurrence do not directly lead to core damage.

4.2.2. Measures which can be envisaged to increase robustness of the plants against extreme weather conditions

Performed investigation allowed to conclude that impact of considerable[8] external hazards as well as their combinations is not directly leading to the core damage. It was found that most of the external hazards could lead to loss of offsite power or station black-out conditions which could be resolved using existing systems and procedures. Critical systems and safety functions challenged by external hazards were identified.

In spite of the safety measures already available, additional provisions are considered to further increase safety level of the plants as follows:

1. To develop and implement measures aimed to protect DG from dust. Those measures could include improvement of DG compartments leaktighness and/or installation of special air filtering at DG air intake system.
2. To foresee measures aimed to remove snow from TH building roof in case of snow accumulation.
3. To implement realistic re-evaluation of air temperature which could lead to freezing of outlet pipes of demineralized water tanks (BZOV). Depending on the results of realistic calculations the measures could be necessary for protection of BZOV pipes against extremely low temperatures.
4. Analysis shows that diesel-driven pump system is critical in terms of protection against various external hazards. It is recommended to implement measures aimed to increase the reliability of the system and check adequacy of diesel pump exhaust pipe protection against external hazards. Particularly it is recommended to re-analyze diesel pump exhaust pipe protection against high winds and dust.
5. Analysis shows high importance of DG auxiliary systems (e.g. HVAC) in case of external hazards. It is recommended to implement measures aimed to protect DG auxiliary systems (e.g. HVAC) against external hazards (e.g. combination of seismic and low temperatures hazards).
6. To implement detailed analysis for lightning impact on ANPP
7. To improve plant specific PSA model in regard with following aspects:
    - to re-visit external hazards screening process taking into account information reflected in stress-test report
    - to complement PSA by critical combinations of external hazards identified within FSA-ANPP project
    - to re-visit hazard curves for different hazards taking into account updated meteodata.

[8] The impact of external hazards with extremely low frequency (<<1E-07) of occurrence was not taken into account

5 LOSS OF ELECTRICAL POWER AND LOSS OFULTIMATE HEAT SINK

5.1 Loss of electrical power

5.1.1 Loss of off-site power

The connection of ANPP to the RA power grid is realized by independent 220 kV and 110 kV lines. These lines are connected with ANPP via bus bars of 220 kV and 110 kV switchyards which are interconnected through a coupling autotransformer. Damage of the 220 kV overhead line will not cause the loss of grid connection. ANPP will receive power from 110 kV power grid and vice versa.

The main defense lines against loss of AC power are emergency power supply system and alternate emergency power supply system. Emergency power supply system consists with four individual 1.5MW EDGs and corresponding safety switchgear systems. Four EDGs are divided in two independent trains with two EDGs in each train. The capacity of one of the EDG is enough to supply all necessary consumers in case of LOOP. Alternate emergency power supply system consist with one 1.5MW DAR-DG. DGs of emergency power supply system and alternate emergency power supply system are located in DG building which have 3 DGs compartments. Two trains of EDGs located in two compartments with two EDG in each. The third compartment houses the DAR-DG (see figure 5-1).

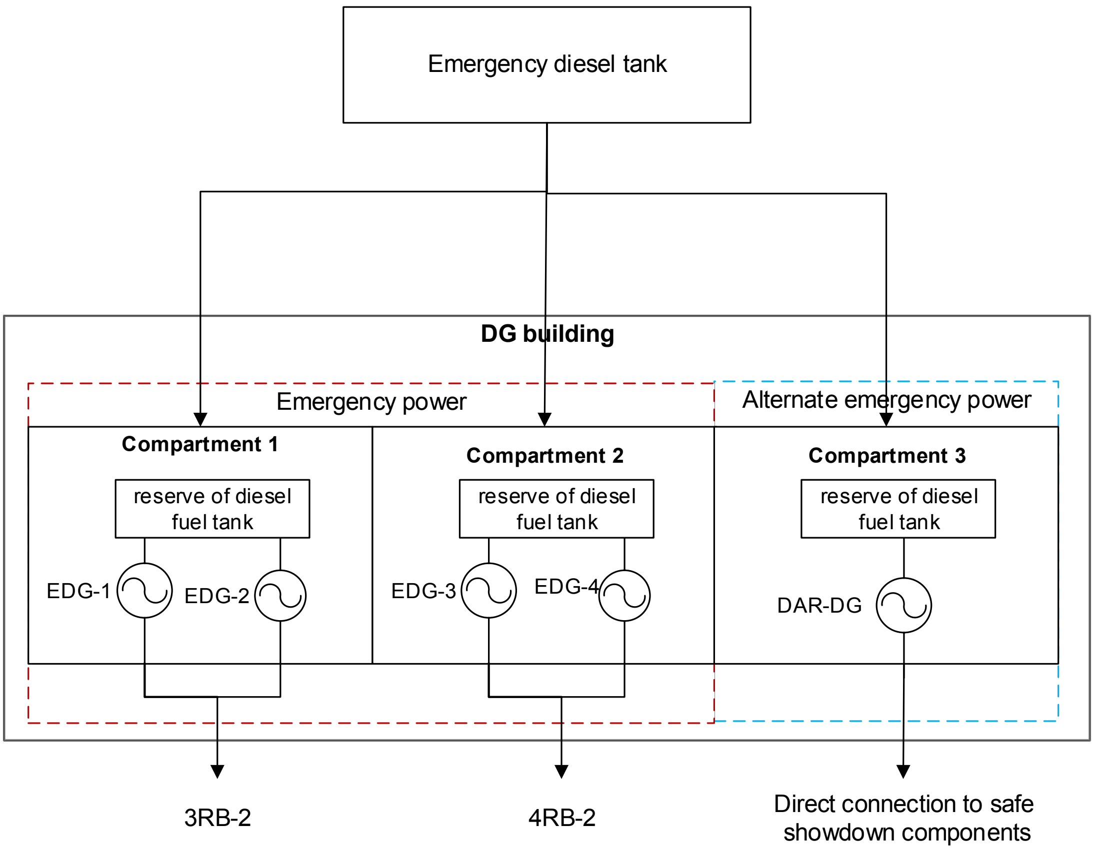


Figure 5-1. Emergency power supply system and alternate emergency power supply system.

Each train of the emergency power supply system is equipped with autonomous digital DG automatics and control system (DG I&C), system of the diesel generator load sequencer (DGLS) and system of the DG excitation and voltage regulation. Each DG diesel compartment has compressed air systems and a direct current board (for each channel), which ensure start-up of diesels in case of loss of NPP AC power supply, and an automatic fire suppression system. In addition, each compartment has a reserve of diesel fuel of about 25 m^3.

All elements that provide the start-up of EDG are powered from the direct current board of the given channel. The cooling of the EDGs is arranged by ESWS. The DGs of emergency power supply system are able to start without availability of Essential Service Water System (ESWS) pumps. After start and connection to 6kB bus-bars, by start-up program two of trains of ESWS put in operation which switches on two ESWS pumps in each train. According actual project one train with two pumps of ESWS can cool two EDGs.

If the off-site power is lost, EDGs will start automatically and provide power to the safety systems 30s after loss of off-site power. If automatic start-up is not successful, EDGs can be started and powered to bus bars 3RB-2 and 4RB-2 manually either from the main control room or from locally (see chapter 1).

5.1.1.1 Autonomy of the on-site power sources and provisions taken to prolong the service time of on-site AC power supply

Two 6 kV 3RB-2 and 4RB-2 bus bars are provided for the emergency AC power supply of ANPP. In the case of loss power on 6 kV bus bars 2 EDGs at each train are automatically started and connected.

DG of each channel (1DG-1, 1DG-2, I channel and 2DG-1, 2DG-2, II channel) are started-up only during the loss of power of its 6 kV bus bar. Voltage measurement transformer generates the signal of EDG startup if voltage on the 3RB-2 or 4RB-2 bus decreases bellow of 25% of nominal voltage level.

It is important to highlight that during the "cold" shutdown and refueling, DGLS program do not operate and there is no special program for these modes, but one train (at least) of the EPSS is always in stand-by mode. After the loss of off-site AC power and connection of DG to its bus-bars, operating personnel can manually switch on any of the pumps (in case of technological necessity) supplied from the corresponding EDG.

All situations mentioned above are instructed, trained and the operating staff is capable of performing them. The LOOP situations are a part of the standard operator training and they are rehearsed frequently at the training simulator.

As mentioned above each train of the emergency power supply system has a reserve of diesel fuel of about 25m^3, which is enough for the operation of both DG of the respective train with rated power during 27.7 hours. These tanks can be refilled from an emergency diesel tank (see figure 5.1). There is one emergency diesel tank at ANPP site. The emergency diesel tank holds enough fuel for 7 days of EDGs operation (four EDGs running at full nominal power). However, emergency diesel tank is not seismically verified and the refilling pump is not powered from reliable power supply. Therefore, it is not considered as a reliable source of diesel fuel in case of loss of off-site power.

Despite of two DGs are allocated in each train, only one DG is enough to fulfill safety functions assigned to each emergency power supply train in case of LOOP. This feature gives possibility to manually switch off one DG in each train to save fuel for longer operation.

Robustness of the emergency power supply system in connection with seismic events, flooding and other severe weather conditions are described in respective parts of Chapter 2, Chapter 3, and Chapter 4.

5.1.2 Loss of off-site power and loss of the ordinary back-up AC power source

At ANPP there is an additional, completely autonomous and independent alternate emergency power system which can provide electrical power from a dedicated additional DG to the 6kV bus-bars and through completely independent cabling system to safety systems. The DAR-DG of alternate emergency power supply system (DAR) is situated in the third compartment of the DG building and has to be manually put into operation by the operating personnel in accordance with the operating procedures of the system. Preparation and putting into operation of the alternate cabling system could be done within one hour in accordance with the operating procedures of the system. Consumers of alternate emergency power supply system are defined in order to ensure the safety functions of the Unit №1 & 2 SFP and the Unit №2 reactor in any mode of operation of the Unit 2. Particularly alternate emergency power system provides electrical power to the following systems:

- SFP cooling pumps (Unit №2 SFP cooling pumps-1, 2);
- One pump in each ESWS train;
- One pump in each HPI (ECCS) system train of unit 2
- Second circuit cooling pump of the reactor in the water-water cooling mode;
- The Unit №1 SFP make-up pump (for make-up of Unit №1 SFP from Unit №1 boron tank B-8/1);
- One fire-fighting water pump (FFWP);
- One fire-fighting foam pump (FFFP);
- One primary circuit make-up pump;
- One drainage tank pump;
- One pump of the recirculation ventilation system of containment cooling (2R-1);
- One Emergency Feedwater pump (EFWP);
- Lighting cabinet;
- One boron make-up pump;
- One drainage pump (DP) of boron compartment 2.

The DG of alternate emergency power supply system (DAR) is able to start without availability of Essential Service Water System (ESWS) pumps. After start and connection to 6kV bus-bars, operating personnel immediately switch on one pump in each train of the ESWS. The DAR-DG is located in a separate diesel compartment and has a reserve of diesel fuel of about 25m³, which is enough for the operation of DG with rated power during 55 hours.

The fuel tank of DAR-DG can be refilled again from emergency diesel tank (see figure 5.1). In this case amount of fuel in emergency diesel tank is enough for 27.7 days of the DG operation if assume that tanks of EDGs were not refilled before. However, as mentioned above emergency diesel tank is not seismically verified and the pump is not powered from reliable power supply. Therefore, it is not considered as a reliable source of diesel fuel in case of loss of off-site power.

During operation of additional emergency power supply system only, the batteries cannot be charged from the DG.

In addition to above mentioned systems of emergency power supply there is an alternative way to provide emergency back-up power to ANPP systems. Electricity can be provided from a small hydropower station (50 MW) using a direct 110 kV transmission line. Recovery power from “Gumush” Hydro power station can be supplied within 3-5 minutes regardless of availability of ANPP communication with the central dispatcher service of the RA power network.

5.1.3 Loss of off-site power and ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

In case of loss of off-site power, ordinary back-up AC power sources(EDG), and the independent alternate emergency power system (DAR) at the ANPP site still are batteries and the auxiliary steam generator feedwater system which is equipped with one diesel pump. SG auxiliary feedwater supply is ensured by the operating personnel using the diesel pump installed in the boron room (B-001) of the Unit №1. The SG auxiliary feedwater system is completely independent (operability does not depend on other systems or elements).

The demineralized water to the diesel pump suction is supplied from DMWTs 1&2 (Unit №1, total capacity of about 1000 m3). The reserve of diesel fuel is sufficient for 24 hours of operation. However, spent pool cooling system of Units 1 and 2 in these cases is not available.

5.1.3.1 Battery capacity, duration and possibilities to recharge batteries in this situation

Reliable power supply of 1st category consumers is ensured by two autonomous independent trains of 0.4kV bus-bars which powered by physically and galvanically separated rechargeable batteries. Each train provides 100% of the capacity required for assigned safety function. Capacity of each rechargeable battery in each train is sufficient for provide electrical power supply to 1st category consumers during 7 hours. This time can be considerably extended by switching off part of consumers (not important instrumentations, panels of the power supply of MCR-2, etc.). Specific analysis should be performed to determine minimal required number of components to be powered from rechargeable batteries for identification of maximal duration of batteries' operation.

There are no mobile diesel-generators available ANPP which could be used in case of total loss of external AC and DGs.

5.1.3.2 Time available to provide ac power and to restore core cooling before fuel damage: consideration of various examples of time delay (timeline analysis) from reactor shutdown and loss of normal reactor core cooling condition (e.g. start of water loss from the primary circuit)

The chapter does not consider water supply from external sources. Time intervals given are margins facilitated by internal feedwater and coolant stock in the plant only. In case of off-site support sources the time margin would be unlimited.

Reactor at any power mode and in the "hot shutdown" mode.
If all AC power is lost at the time of reactor scram (station blackout), it has been shown by calculations that water existing in all SGs could be used for 5 h for heat removal, even if there were no countermeasures. After this time period there is still 0.5 m of water in SGs and auxiliary fill up is possible without any threat to SG tube integrity. Once SG heat-exchange surface becomes ineffective for the core residual heat removal, temperature in the core outlet will start to increase together with RCS pressure. When the pressurizer safety valve opening pressure is reached, loss of the RCS coolant starts with deterioration of core cooling at a certain moment. The reactor residual power in this phase is removed by the RCS coolant evaporation to the containment. The long-term loss of heat removal from the primary circuit will gradually change to loss of the core cooling. If power supply of the unit is not recovered on time and water supply to the SG or RCS is not recovered, the initiating event of blackout type leads to fuel damage after 9 hours of accident initiation.

However, SG water inventory could be maintained by the diesel auxiliary feedwater system, and decay heat is removed with SG SV. Within this mode, the plant can be kept in a hot shutdown for long time without any increased heat-up in the primary circuit depending on several limiting factor discussed below. The limiting factor for this time frame is the exhaustion of demineralized water tanks (2 DMWTs with capacity 500 m^3) and 2 feed water tanks with capacity 120m^3 of each, MCP sealing heat up, diesel fuel amount and rechargeable battery capacity.

Heat removal from the secondary side by diesel auxiliary feedwater system can be maintained during 160 hours (6.6 days) if refilling of diesel fuel will be possible (see figure 5-2).

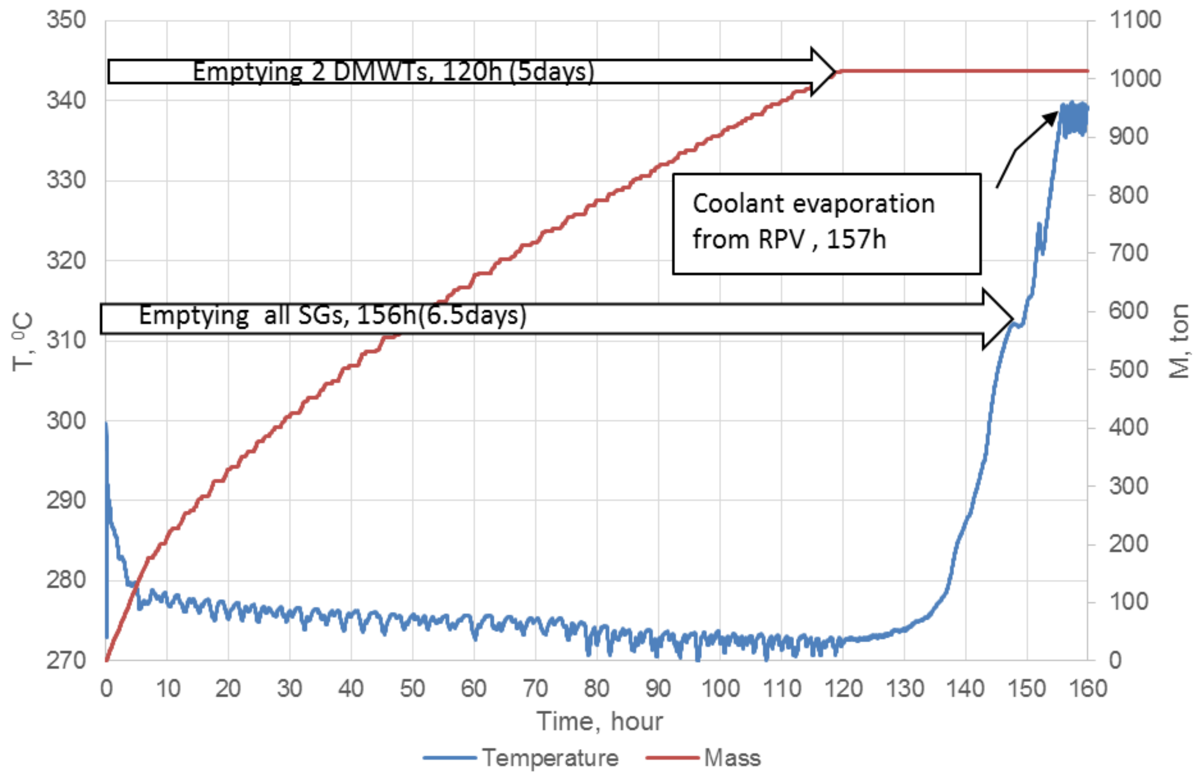


Figure 5-2. Temperature behavior in core outlet during SBO with DC power.

As mentioned above capacity of each rechargeable battery is sufficient for RMG operation during 7 hours. After this time, all control of ANPP such as level measurement in SGs will be lost. Therefore it is assumed that after 7 hour operator will be manually controlled mass flow rate of auxiliary feedwater diesel-pump to maintain pressure in primary side in nominal level (pressure measurement device doesn't need power supply). This ensures heat removal from the secondary side during 169 hours (7 days) if refilling of diesel fuel will be possible (see figure 5-3).

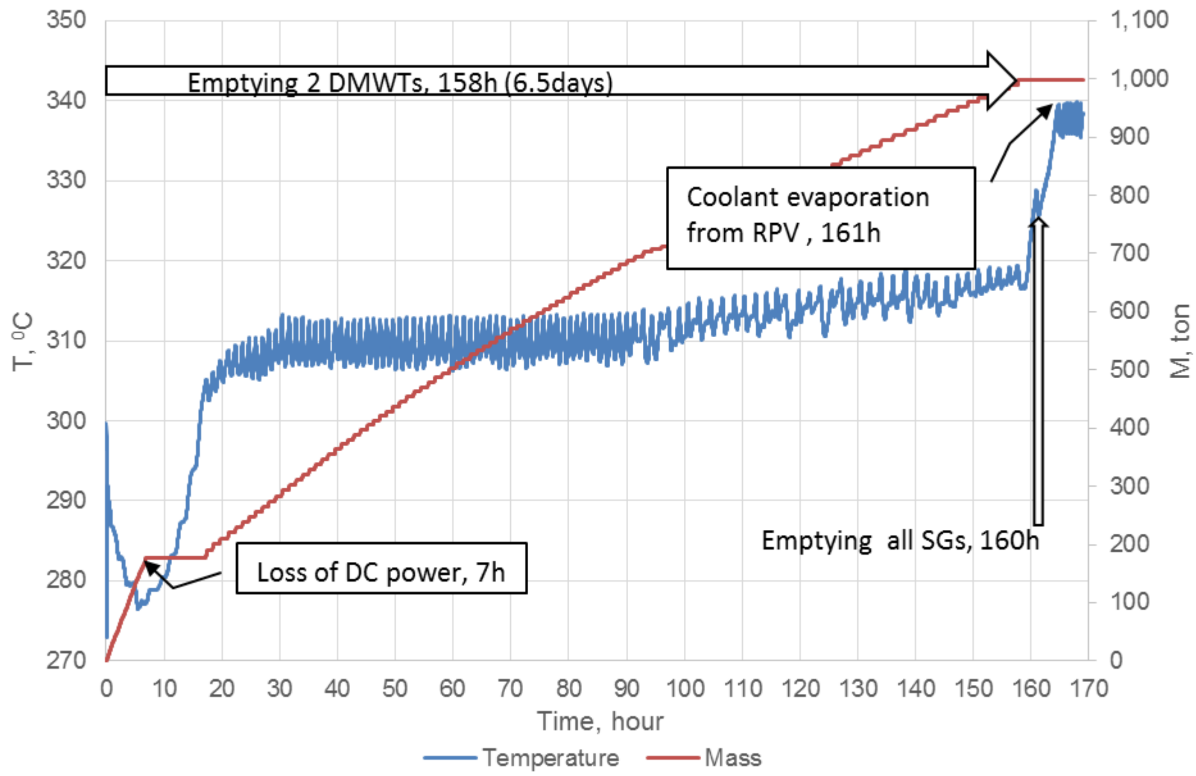


Figure 5-3. Temperature behavior in core outlet during SBO without DC power.

The reserve of the diesel fuel is sufficient for 24 hours of operation of diesel-pump (auxiliary feedwater system) without refilling of fuel tanks. Therefore in case impossibility to refilling of fuel tanks 24 hours heat removal from the secondary side will be possible using demineralized water of two tanks and additional 9 hours of heat removal by evaporating of remaining water in SGs (total time 33 hour).

During blackout, the RCS make-up is not available. In case of blackout, cooling of MCP seals will not be ensured due to loss of the MCP seal water flow, which, from the long-term point, may evoke the RCS coolant leakage through the MCP. Test results performed on full scope MCP seal model by “ENERGONASOS” showed that conditions endangering the seal integrity do not occur within 24 hours from loss of MCP cooling.

Currently amount of leak from MCP seals after 24 hours is not assessed. Special assessment should be performed to define amount of leak from MCP seals after 24 hours.

According analysis presented above minimal time margin in case of all AC power loss is 33 hours before fuel is damaged. During the first 24 hours heat removal from the secondary side will be possible using demineralized water of two tanks and additional 9 hours of heat removal by evaporating of remaining water in SGs.

Times for individual states are given in the following table:

Initiating Event	Time of emptying of 2 DMWTs	Time between feedwater of SG cease and core melting	Total time
SBO with possibility to re-fill diesel fuel of auxiliary feedwater system and without loss of DC power	120 hours / 5 days	40 hours	6.6 days
SBO with possibility to re-fill diesel fuel of auxiliary feedwater system and with loss of DC power	158 hours/ 6.5 days	11 hours	7 days
SBO without possibility to re-fill diesel fuel of auxiliary feedwater system	-	33 hours	33 hours

Open reactor, with natural circulation via two loops and normal reactor level.
If all AC power is lost, it has been calculated that coolant boiling in the reactor core is started after 4 hours. In case of station blackout there is no special system at ANPP, which can be used for cooling the core. The decay heat will be removed trough secondary side until stopping natural circulation in loops, which happens after 4 hours as coolant start to boil. After that, evaporation of coolant will lead to level decrease in reactor. At 5.5 hours after accident initiation, coolant level in reactor will reach hot leg nozzles. Fuel rods damages will take place after 18.5 hours.

According analysis performed above minimal time margin in case of all AC power is lost and open reactor condition is 18.5 hour before fuel is damaged.

Spent fuel.
In case of loss of all AC power, there is no special system at ANPP, which can be used for cooling spent fuel pools of units 1 and 2. The limiting worst-case situation for the available time is the evacuation of the whole reactor core to the unit-2 loading pool. If the pool cooling is lost in such a situation, the pool water will start boiling in about 3.2 h. After that, evaporation of coolant will be lead to level decrease in spent fuel pool. At 26.7 hours after accident initiation, coolant level in reactor will reach the top of assemblies. To prevent the water-level drop in the pools, make-up water supply of about 22.8 kg/s would be needed to compensate for the boiling. The results of the calculations for the SFP2 are shown in the Table 5-1.

Table 5-1. The results of the calculations for the SFP2.

	Full unloading	Fuel is located at the bottom tier only (actual configuration)
Time of boron solution heating in Unit №2 SFP by 1°C	5.3 min	34.8 min
Time of boron solution achieving the boiling temperature in Unit №2 SFP	3.2 hour	21.3 hour

Time for 1 m of the boron solution to evaporate from Unit №2 SFP assembly and container sections	5.0 hour	-
Time for 1 m of the boron solution to evaporate from assembly section	3.2 hour	17.1 hour
Time for the water level to reach the top of assemblies	26.7 hour	70.1 hour
Time for the water level to reach the top of fuel pins	27.7 hour	75.7 hour
Fuel damage onset time (rapid oxidation of shells with hydrogen generation)	33.0 hour	104.6 hour
Consumption of circulation water preventing boiling in Unit №2 SFP (T_{SFP}<95°C)	82.1 m^3/ h	4.8m^3/ h
Water flow rate to maintain constant level in Unit №2 SFP	6.1 m^3/ h	0.36m^3/ h

Since residual heat in Unit №1 SFP is much less than in Unit №2 SFP, the operating personnel has enough time for organizing the Unit №1 SFP cooling down.

Normally, the decay heat generated by the spent fuel elements stored in the fuel pools is much less during outage than in case when all fuel from core is fully unloaded and stored in SFP. In case of fuel is located at the bottom tier only the uncovery of the fuel elements at SFP-2 would occur after about 70.1 hour. Make-up water supply of about 1.3kg/s would be enough to compensate for the boiling in the normal situation.

It is important to highlight that without countermeasures after 33 hours fuel damage will occur in SFP with rapid oxidation of cladding with hydrogen generation. The release of hydrogen inherent in this type of accident would generate a particular risk to the integrity of the reactor building. In view of these risks, it is important to apply every possible measure to avoid accidents liable to cause significant damage to the spent fuel assemblies, and therefore prevention of uncovering of the fuel assemblies is among the requirements to be applied to attain this objective.

5.1.4 Conclusion on the adequacy of protection against loss of electrical power

The time margin for the recovery of the AC-power supply before fuel damage is 33 hours if a total loss of the AC-power occurs. Core heat-up would take place no earlier than 9 h after the start of the event, even if there were no countermeasures. Using the diesel auxiliary feed water pumps the total time before loss of fuel cladding integrity is lost can be extended to 6 days if refilling of diesel fuel will be possible.

The capacity of rechargeable batteries is enough for providing the AC power of 0,4kV reliable power supply of the 1st category bus-bars for 7 hours for each safety train.
Existing event based EOPs not fully reflected all conditions, which may arise in case of SBO.

Test results performed on full scale MCP seal model by “ENERGONASOS” showed that conditions endangering the seal integrity do not occur within 24 hours from loss of MCP cooling.

Additional fuel tanks can be used to hold enough fuel for 7 days of EDG operation (four EDGs running at full nominal power). However, these tanks are not seismically verified and the fuel pump is not powered from reliable power supply.

At ANPP four demineralized water tanks (DMWTs), which can be used to supply water to SGs however in case of SBO, only two of these tanks can be used. As mentioned above, the auxiliary feedwater system takes water from two DMWTs only, however these tanks in case of availability of AC power can be refilled from other two demineralized water tanks.

The stock of diesel in the fuel tank assures the auxiliary feedwater pump operation for 24 hours without refilling from emergency diesel tanks.

5.1.5 Measures envisaged to increase robustness of the plant in case of loss of electrical power

To increase the level of the safety of NPP in the case of simultaneous loss of regular and reserve AC power supply sources it is proposed:

- To review and revise new developed symptom based EOP to ensure that all states which may arise in case of SBO are analyzed;
- To install two physically separated fixed pipelines for make-up of the coolant inventory in SFP from a mobile source (fire pumps or diesel pumps) and external water source dedicated for SA;
- To develop additional measures to extend the operating time of reversible motor generators (ODGs) in an inventor mode that will lead to increase the time to provide I&C AC power supply;
- To replace all reversible motor generators (ODGs) with modern inverters with less energy losses;
- To provide mobile DGs for charging batteries and supplying selected unit consumers during SBO;
- To implement analysis of circuit diagram for consumers power supply from DAR.
- Develop and implement activities aimed at minimizing personnel manual actions to activate the DAR system;
- To implement the scheme for battery charging from DAR and/or the portable diesel generator;
- To install an additional tank of diesel fuel in the DG stations (for each train) in order to run the train I and II of emergency power supply during 72 hours at full load, independently on the of the fuel inventory in the oil-fuel facilities of the plant;
- To implement autonomous alternative means for make-up of the Unit №1 and Unit №2 SFP, SGs 1-6 of the Unit 2;
- To perform additional calculations in order to determine sufficient effectiveness of the diesel pump for steam generator feeding in various power modes of reactor (cold shutdown, refueling, closed reactor, open reactor);

- To obtain experimental evidence of MCP seals tightness after 24 hours of loss of cooling.

5.2 Loss of the ultimate heat sink

The primary UHS is the surrounding atmosphere. Heat removal from the core to the primary UHS is provided by two systems which are ESWS and System of circulation water supply ("Prud" and “Sevdjur” pumping stations, cooling towers, inlet and outlet channels). However, heat removal from SFP and from containment to the primary UHS is provided by only ESWS.

In case of immediate and long-term loss of operability of System of circulation water supply, the ESWS can provide heat removal from the core, SFP and containment to the primary UHS during unlimited time if coolant inventory in ESWS can be maintained.

As loss of the primary ultimate heat sink, the event with interruption of raw make-up water supply to NPP site and failure of heat exchangers of normal residual heat removal system will be considered.

Simultaneous failure of both two trains of ESWS will be considered in the following evaluation as an envelope case for loss of the primary ultimate heat sink and the alternate heat sink.

5.2.1 Design provisions to prevent the loss of the primary ultimate heat sink

5.2.1.1 System of circulation water supply

The system of circulation water supply ("Prud" and “Sevdjur” pumping stations, cooling towers, inlet and outlet channels) provide water to heat exchangers of cooldown system for normal operation (CSNO). The water to the CSNO heat exchanger is supplied from BTV tank through NTB pumps. The BTV tank filled from following sources:

- “Sevjur” pump station, which intake water directly from Sevjur River;
- “Prud” pump station which intake water from artificial water reservoir;
- Directly from Inlet Circulation Water Channel (see figure 5-4).

In addition, water can be supplied directly to heat exchangers of normal residual heat removal system from Diesel-Engine Pump Station.

Cooling water, after it is passed through heat exchangers of CSNO, is being discharged to Circulation Water Outlet Channel.

Service water diesel-engine pump station is designed to supply service water from the outlet channel for cooling the main equipment in reactor building (MCP components cooling loop) and provide for ESWS make-up when there is no outside power supplied to ANPP. Available at the pump station diesel fuel (five tanks of 230 liters each) should be sufficient to operate one diesel-engine pump with the maximum load for more than 50 hours.

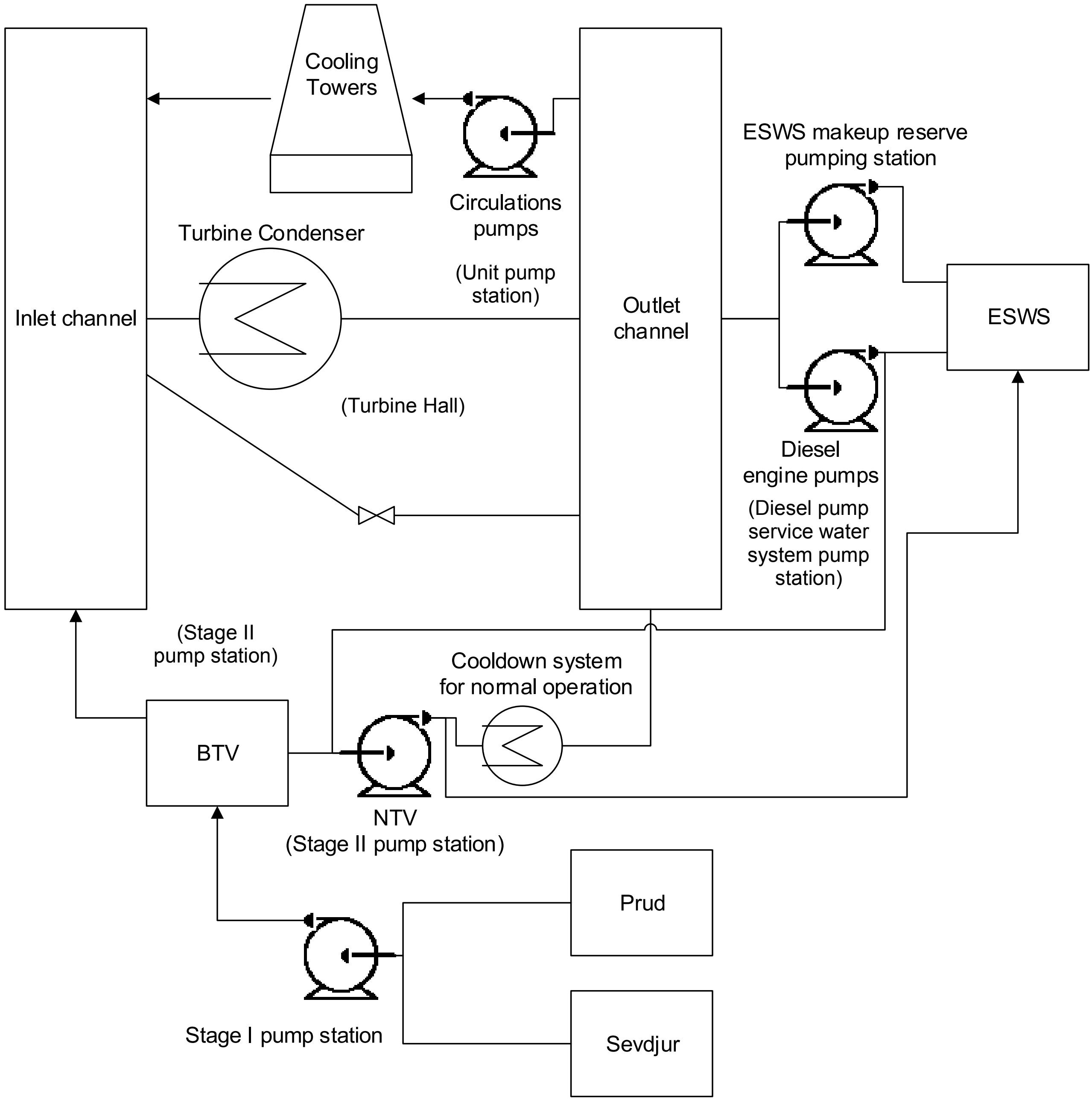


Figure 5-4. Schematic of residual heat removal systems of ANPP.

5.2.1.2 ESWS system

The ESWS system is a "supporting" safety system providing cooling water to systems responsible for core cooling. ESWS consists of two functionally and physically independent trains. Each train includes one 5020 m^3 spray pool, a pump station with three pumps (ESWP) of 802.8m^3/h flow rate, a second makeup pump station with two 56 m^3/h pumps, water lines, valves and instrumentation. In addition, there is one standby pool (5020 m^3) serving for two trains. All system components are designed as seismic resistance Class I. Pumps and electric operated valves are powered from Class II safe power supply system. ESWS fulfills the safety function of heat removal from safety systems to the primary UHS (atmosphere). ESWS should provide not only for the ultimate heat removal system, but also to cool all consumers requiring uninterrupted cooling water supply.

To restore ESWS losses two different lines are designed (see Figure 5.4). These are:

- main feedwater makeup lines of ESWS which take their water from the Stage II pump station;
- line from ESWS makeup reserve pumping station. These pumps are automatically actuated and supply water from the discharge line of the circulation water to ESWS.

In case of an accident at ANPP, when ESWS water losses cannot be covered using the main or ESWS makeup reserve pumping station line. ESWS will be supplied with water using diesel engine pumps from the outlet circulation water channel.

Outlet circulation water channel is used for the ESWS makeup via the ESWS makeup reserve pumping station or using diesel engine pumps; this water volume ensures ESWS operation during more than one month.

At a rated level of 3.25m the ESWS spray pools inventory comprises 5020m^3. The minimum operating level in spray pools is 3.0 m. Note that the amount of water contained in the spray pools between the minimum 3.0 m level and the level of 1.2 m when ESWS pumps operation will fail is 3480m^3.

5.2.2 Loss of the primary ultimate heat sink

As enveloping event for loss of the primary ultimate heat sink, the event with interruption of raw make-up water supply to NPP site and failure of heat exchangers of CSNO was considered. It means that loss of following systems was considered, as they are vulnerable to seismic hazards:

- “Sevjur” pump station, which take water directly from Sevjur River;
- “Prud” pump station which take water from artificial water reservoir;
- Lines from Circulation Water Channel to BTV tank;
- Diesel-Engine Pump Station which supply water directly to heat exchangers of normal residual heat removal system.

In this case, heat removal from the core, SFP and containment to the primary UHS is provided by ESWS.

The coolant inventory in the ESWS system decreases due to evaporation and water carry-over (40m3/hour). The required coolant inventory in ESWS can be maintained as mentioned above by water make-up by Diesel-Engine Pump or from ESWS makeup reserve pumping station. ESWS makeup reserve pumping station has two independent channels with one pump in each channel and is powered by EDG bus-bars. Mass flow of ESWS emergency make-up pumps is 55 m^3/hour.

Outlet circulation water channel is used for the ESWS makeup via the ESWS makeup reserve pumping station line as well as for makeup using diesel engine pumps; this water volume ensures ESWS operation during more than one month.

5.2.2.1 Integrity of MCP seals

Cooling of the MCP seals is organized through system of circulation water supply. Taking into account that MCP seals cannot be cooled through ESWS, hence in case of loss of primary ultimate heat sink the failure of MCP sealing cooling will occur. Loss of cooling water through coolers of the MCP seals in long-term may evoke the MCS leakage through the MCP seals letdown lines. Test results performed on full scale MCP seal model by manufacturer ("ENERGONASOS") showed that conditions endangering the seal integrity do not occur within 24 hours from loss of MCP cooling.

5.2.2.2 Availability of an alternate heat sink

In case of loss of ESWS, residual heat from the core can be removed directly to atmosphere by steam (SG SVs, BRU-A) or via other systems independent on ESWS particularly following system can be used (detailed description of mentioned system is reflected in chapter 1):

- SG auxiliary feedwater system (diesel pump, DMWTs 1&2);
- High pressure SG emergency cooling system (ASN);
- Low pressure SG emergency cooling system (AKN);
- Emergency SG feedwater system.

Feedwater flow to SG is provided from four demineralized emergency water tanks. However, heat removal via the alternative systems does not ensure residual power removal from auxiliary technological systems (cooling of intermediate circuits, vent systems, etc.), confinement and SFP and thus, it is not a full scope heat sink compared to ESWS. However, it will ensure core heat sink without external FW sources for more than 16 days if all water from four tanks of demineralized water will be used (Four demineralized water tanks can be used to supply water to SGs only in case of existing AC power at ANPP otherwise only two demineralized water tanks can be used).

5.2.2.3 Possible time constraints for availability of alternate heat sink and possibilities to increase the available time

In case of complete loss of ESWS, residual heat from the core can be removed directly by steam release (via BRU-A or SG SVs) to the atmosphere while ensuring feedwater flow to SG. The time of maintaining this regime is limited. The power plant has coolant inventory for additional 16 days. After this time, make-up of these tanks from ESWS pools must be provided. In case of refilling demineralized water tanks from two ESWS pools, core heat sink will ensure for more than one month.

5.2.3 Loss of the primary ultimate heat sink and the alternate heat sink

After loss raw water and ESWS systems residual heat from the core can be removed directly by steam to atmosphere (SG SVs, BRU-A) or via other systems independent on ESWS particularly following system can be used:

- SG auxiliary feedwater system (diesel pump, DMWTs 1&2);
- High pressure emergency cooling system;
- Low pressure emergency cooling system;
- Emergency SG feedwater system.

Feedwater flow to SG is provided from four demineralized water tanks. However, heat removal via the alternative system does not ensure residual power removal from auxiliary technological systems (cooling of intermediate circuits, vent systems, etc.), confinement and SFP and thus, it is not a full scope heat sink method to ESWS. However, it will ensure core heat sink without external FW sources for 16 days if all water from 4 tanks of demineralized water will be used (four demineralized water tanks can be used to supply water to SGs only in case of existing the AC power at ANPP otherwise only two demineralized water tanks can be used).

5.2.3.1 Spent fuel pool

Residual heat from spent fuel is removed in heat exchangers of SFP cooling systems to ESWS system. Loss of heat removal from SFP occurs in case of failure of both channels of ESWS system. No other system is available for heat removal from SFP then SFP cooling systems. An additional possibility to cool SFP is using recirculation trough boron tank B-8/2 (sump). Using SFP make-up pumps (SFP make-up pumps, and pumps to supply boron solution for cleaning) water from B-8/2 can be supplied to SFP and returned back to B-8/2. Time within the temperature in the Unit №2 SFP and B-8/2 tank reaches about 800C is ~3,3 days in case of complete core unload.

The results of calculations for the Unit №2 SFP without use of the make-up are listed in table 1 of section 5.1.3.2.

5.2.3.2 External actions foreseen to prevent fuel degradation

External actions have to focus on ESWS make-up, demineralized water make-up and supply logistics. There are several alternative methods for compensation of ESWS circuit water losses either from internal or external sources. The basic time reserve till initiation of ESWS make-up because of insufficient water inventory in ESWS pools is 72 hours. Another internal water source in the NPP is the Circulation Water Channels (outlet and inlet) that contain water for more days depending on the event scenario (see chapter 5.2.2). This water can be pumped to ESWS pools by Diesel-Engine Pump or from ESWS emergency make-up pumps.

5.2.3.3 Time available to recover one of the lost heat sinks or to initiate external actions and to restore core and spent fuel pool cooling before fuel damage: consideration of various examples of time delay from reactor shutdown to loss of normal reactor core and spent fuel pool cooling condition

Times for individual states are given in the following table; the detailed description including analysis is provided in chapter 5.2.3:

	Initiating Event	ESWS loss - time	Time between ESWS loss and core melting	Total time
1	Raw make-up water loss without internal and external intervention	72 hours / 3 days	16 days	19 days
2	Raw make-up water loss with internal intervention – available from Outlet Circulation Water Channel by Diesel-Engine Pump or from ESWS emergency make-up pumps.	30 days	16 days	46 days
3	Raw make-up water loss with internal and external intervention	unlimited	-	unlimited
4	ESWS loss	-	16 days	16 days

5.2.4 Conclusion on the adequacy of protection against loss of ultimate heat sink

- The resistance of SFP-2 in loss of UHS conditions (loss of make-up raw water) is adequate;
- Current procedures do not fully prescribe operating personnel actions in case of loss of the ultimate heat sink with various failures. The following actions are needed:
    - Additional calculations for various common failures of ultimate heat sink systems and in various emergency scenarios;
    - Improvement of existing calculations;
- The water volume of ESWS ensures at least 72 hrs of ESWS availability after loss of raw make-up water;
- Additional water inventory available on the site in Circulation Water Channel (outlet and inlet) provide for additional 30 days of ESWS availability;
- After loss of all ESWS trains the basic design provides minimum margin of 16 day to fuel damage in the reactor;
- After loss of all ESWS trains the basic design provides minimum margin of 3 days to fuel damage in SFP-2.

5.2.5 Measures which can be envisaged to increase robustness of the plants in case of loss of ultimate heat sink

Evaluation of safety margins of ANPP in the case of UHS loss proved the plant design ability to ensure protection of safety barriers for given type of events during considerably long time, thus providing sufficient time margin for accident management interventions to recover UHS. Despite the robustness of power plant design, its safety can be improved by the following modifications:

- To provide for mobile pumps for ESWS make-up from Circulation Water Channel;
- To provide results of analysis of MCP seals' behavior at long-term failure of cooling (more than 24 hours);
- To construct a fixed line for maintaining the coolant inventory in SFP from a mobile source (fire pumps);
- Develop and implement additional measures to use the large reserves of service water in the inlet and outlet channels, as an alternative heat sink.

5.3 Loss of the primary ultimate heat sink, combined with station black out (i.e., loss of off-site power and on-site ordinary and backup power source)

Loss of the primary ultimate heat sink with simultaneous loss of off-site power and loss of on-site back-up AC power sources create similar conditions as loss of off-site power, ordinary back-up AC power sources(EDG), and the independent alternate emergency power system (DAR) see 5.1.2.

In case of loss of ultimate heat sink with simultaneous loss of off-site power and loss of all on-site stand-by and back-up power sources, the following still remains operable to ensure decay heat removal for at least 33 hours:

- SG auxiliary feedwater system (diesel pump from the DMWT-1,2, see 5.1.3);
- Main steam pipelines (including SG SV, BRU-A);

5.3.1 Reliable I category power supply from rechargeable batteries (see 5.1.1.1). Time of autonomy of the site before loss of normal reactor core cooling condition

Results of all the available calculations are given in the section 5.1.3.

5.3.2 External actions foreseen to prevent fuel degradation

In case of loss of the primary ultimate heat sink with simultaneous loss of off-site power, with only on-site back-up power source available:

- The DAR-DG of alternate emergency power supply system (DAR) is situated in the third compartment of the DG building and has to be manually put into operation by the operating personnel in accordance with the operating procedures of the system. Preparation and putting into operation of the alternate cabling system could be done within one hour in accordance with the operating procedures of the system (see 5.1.2).

In case of loss of ultimate heat sink with simultaneous loss of off-site power and onsite stand-by and back-up power sources at the ANPP site still are batteries and the auxiliary steam generator feedwater system which is equipped with one diesel pump. SG auxiliary feedwater supply is ensured by the operating personnel using the diesel pump installed in the boron room of the Unit №1. The SG auxiliary feedwater system is completely independent (operability does not depend on other systems or elements) (see 5.1.3).

5.3.3 Measures envisaged to increase robustness of the plant in case of loss of primary ultimate heat sink, combined with station black out

- To review and revise new developed symptom based EOP to ensure that all states which may arise in case of SBO are analyzed;
- To install two physically separated fixed pipelines for make-up of the coolant inventory in SFP from a mobile source (fire pumps or diesel pumps) and external water source dedicated for SA;
- To develop additional measures to extend the operating time of reversible motor generators (ODGs) in an inventor mode that will lead to increase the time to provide I&C AC power supply;
- To replace all reversible motor generators (ODGs) with modern inverters with less energy losses;
- To provide mobile DGs for charging batteries and supplying selected unit consumers during SBO;
- To implement analysis of circuit diagram for consumers power supply from DAR and develop and implement activities aimed at minimizing personnel manual actions to activate the DAR system;
- To implement the scheme for battery charging from DAR and/or the portable diesel generator;
- To install an additional tank of diesel fuel in the DG stations (for each train) in order to run the train I and II of emergency power supply during 72 hours at full load, independently on the of the fuel inventory in the oil-fuel facilities of the plant;
- To implement autonomous alternative means for make-up of the Unit №1&2 SFP, SGs 1-6 of the Unit 2;
- To perform additional calculations in order to determine sufficient effectiveness of the diesel pump for steam generator feeding in various power modes of reactor (cold shutdown, refueling, closed reactor, open reactor);
- To obtain experimental evidence of MCP seals tightness after 24h of loss of cooling.
- To provide for mobile pumps for ESWS make-up from Circulation Water Channel.

6. SEVERE ACCIDENT MANAGEMENT

In every circumstance, being normal operation, incidental or accidental conditions, the operators' actions are described in procedures that are reviewed on a regular basis in function of Return of Experience (REX). As soon as a crisis situation is identified, the Internal Emergency Plan (IEP) is activated, meaning that the Emergency Response Team (ERT) is formed and that stand-by crews are called on- site. The IEP of ANPP also foresees the possibility to rely upon external people, competences or logistic means. At present, the accident management of the ANPP Unit №2 is solely based on existing Emergency Operating Procedures (EOPs), which have the main objective to restore core cooling. Some new EOPs are presently under development. However, in order to mitigate more specifically a severe accident, being defined as the significant degradation and partial or total melt down of the core, for which the focus needs to be made upon the containment and minimization of fission product releases, some additional documents called SAMG, or "Severe Accident Management Guidance", are currently being developed by ANPP for the Unit №2. It is therefore slightly premature to present specifically the management of severe accidents as those SAMG are not fully available yet. Meanwhile, the general organization and the strategies presented in the following pages are those currently applicable in the occurrence of broader emergency situations. Because the SAMG will be based on WOG approach the description of future situation and some recommendations are based on analogy with the WOG SAMG.

There is an extensive plant modernization project underway based on the results of recommendations from previous missions and plant assessment. When not mentioned, the plant status in July 2012 was used as the reference one. In some cases, the actions that are underway or planned in the near future have been also taken into account but it is explicitly mentioned that they do not represent the reference plant state. The findings in this report support all of such actions as their contribution to reducing the consequences of a severe accident has been indicated.

6.1 Organization and arrangements of the licensee to manage accidents

Due to the current development of the SAMG, the operators of ANPP are at present not yet fully educated to the mitigation of severe accidents. Meanwhile, they follow the current EOPs (including procedures for Beyond Design Basis Accidents) to manage the accident and avoid its progression towards a severe accident.

Besides the accident management by EOPs, the onset of a crisis situation leads also to the initiation of the IEP and therefore to the formation of the ERT, under the technical direction of the Plant Chief Engineer, himself supervised by the Plant Director. This team, whose main objective is to ensure the accident management for the entire ANPP site, can request the support of the people and equipment (classified or not) on site as well as additional external people, competences or logistic means. As such it can be stated that the IEP of ANPP is developed in order for the ANPP site to fulfill all its safety functions in all circumstances. The IEP is foreseen to handle whatever emergency situation that can arise, from incidental to accidental situations due to external hazards.

6.1.1 Organization of the licensee to manage the accident

In case of a nuclear accident, the management of the ANPP site can be ensured from 4 distinct locations, namely:

- two dedicated to the control of the reactor, being:
  - The Main Control Room (MCR) in which operators are located in normal and accidental situations;
  - The Back-up Control Panel located in the turbine hall, which is currently being implemented;
- two for the management of the whole IEP, being:
  - The Main Crisis Centre (MCC), located in a sheltered building on-site where the ERT initially gathers;
  - The Back-up Crisis Centre (BCC), located in the building of Armatom in the city of Yerevan.

The MCC is located in shelter №2. The shelter №2 was designed as a radiation protection shelter (Class III protection).
In case of accident or radiation hazard, information from the MCC is transferred with ANPP communication system: operative communication means, telecommunication means with cable and radio relay lines, plant communication means, special communication means, radio communication means (Trunk system), mobile communication, internet, electronic mail.

The MCC is provided with power supply systems, water supply and sewage system, heating system and tanks of emergency water reserve.

In case of ANPP blackout the MCC is powered with reserve DG of shelter №3.

The ventilation system of the MCC is operated in two modes: pure ventilation mode (I mode) and filtering ventilation mode (II mode).

The BCC is devoted to welcome the ERT in case the MCC becomes inhabitable, namely when:
- The dose rate at the plant territory exceeds 0.2 μSv/hour;
- It is subjected to too great damages (e.g.: earthquake, flooding, fire, etc.);
- The CO_2 concentration in the atmosphere of the MCC exceeds 4%.

6.1.1.1 Staffing and shift management in normal operation

Shift crew ensure a continuous presence on site and are always composed as described in table below.

<table>
<tr><th>Unit</th><th>Plant state</th><th>Number</th></tr>
<tr><td rowspan="2">ANPP Unit 2</td><td>Full power, hot stand-by and intermediate state</td><td>4 operators in MCR and
4 machinists on-site for local inspections</td></tr>
<tr><td>Cold shutdown</td><td>3-4 operators in MCR and
4-5 machinists on-site for local inspections</td></tr>
</table>

The shift crew is assisted by the radiation protection agents, the fire brigade and the site security.

The shift crew has the mission to perform the initial required actions for any event that could occur and at any time (24 hours a day, 7 days a week). In case the unit enters an emergency situation, the Internal Emergency Plan (IEP) is activated to guarantee that the required internal and external resources are mobilized to manage the event. Under such incidental or accidental operating conditions, the ERT can request additional stand-by crews of on-call technicians and staff available to support the shift crew.

The roles of each intervening person or group are defined in the IEP. The stand-by crew is composed of 10-12 persons, who will work in smaller groups and replace each other. These additional stand-by crews will be available wherever requested within 15 minutes during work time and 1 hour during off hours after the activation of the IEP.

The entire personnel of the ANPP site are gradually involved:

- Shift crew;
- Emergency Response Team (ERT);
- Stand-by crews;
- On-call managers;
- On-call technicians (mechanics, electricians, radiation protection officers, ICT);
- The security and medical people;
- The entire personnel of the ANPP site upon request, based on individual competences.

Contractors usually working on the ANPP site can be requested as well. From this point of view, some agreements exist with several organizations – Internal Forces and Ministry of Emergency Situations.

The following resources are included on the ANPP site:

- Fire response equipment to handle various kinds of events (fire, explosion, dangerous products releases, etc.);
- Specifically-equipped vehicles for on-site and off-site radiological measurements;
- Stocks of personal protection equipment;
- Guards to limit the access to the site to authorized people only.

6.1.1.2 Plans for strengthening the site organization for accident management

After introducing SAMG, the ERT staff will be extended by the personnel of Technical Support Centre, located in the MCC.

6.1.1.3 Measures taken to enable optimum intervention by personnel

See chapter 6.1.1.1 for the present state (July 2012). Extension of the personnel with introduction of SAMG is foreseen. Training on SAMG will be organized for the personnel of the Technical Support Centre and for operators.

Technical measures are being introduced – improvement of habitability of the main control room and installation of the backup control panel, details are in chapter 6.2.5.

6.1.1.4 Use of off-site technical support for accident management

The present state is described in this sub-chapter. Figure 6-1 “Organization of population protection planning system” demonstrates all authorities of the RA involved in the RA emergency response system and charged with functions and duties in the area of technical support to ANPP in case of accidents, in particular:

- Emergency Commission of the RA;
- Ministry of Territorial Administration and Emergency Situations of the RA (MTAES) (involving State Reserve Agency)
- Permanently acting committees on emergency situations of the RA regions;
- Regional Rescue Services of the RA;
- Ministry of Energy and Natural Resources;
- State Committee on Nuclear Safety Regulation of the RA;
- Ministry of Defense;
- Police of the RA;
- Ministry of Transport and Communications;
- Ministry of Nature Protection;
- Ministry of Health;
- Ministry of Economics;
- National Security Service;
- Ministry of Urban Development;
- Ministry of Diaspora;
- The RA Government Committee of State Revenue;
- Ministry of Foreign Affairs;
- Head Office of Civil Air Fleet;
- Ministry of Agriculture;
- Ministry of Justice.

The functions of these organizations in case of accidents at ANPP are regulated by the RA legislation on CD&ES (Civil Defense and Emergency Situations), “National plan of population protection in nuclear and/or radiation accidents at ANPP (External emergency plan)” (Decree of the RA Government №2328-H of 22.12.2005). In addition, each organization has its own emergency plan with detailed description of functions and protective activities, and forces and means required for their implementation, as well.

In addition to the RA authorities indicated in figure 1 ANPP cooperates with the Regional Crisis Centre (RCC) of WANO MC for WWER NPPs based on Rosenergoatom LLC Crisis Centre.

Medical Aid. In case of accidents at ANPP the medical response is implemented in regard with requirements of the following documents:

- Procedure "Organization of medical protection at ANPP in case of emergency situation" MA.ATD.12.SChS-006;
- "Response plan of HAEK CJSC for nuclear and/or radiation accidents" MA.ATD.41.SChS-001. Section "Medical protection".

Before arrival of emergency medical teams of the RA Ministry of Health the first aid (preliminary, medical) is organized and provided by ANPP sanitary team members and Medio Prophylactic department. The sanitary treatment and decontamination of affected personnel is performed jointly with decontamination and sanitary treatment group.

The sanitary group and Medio Prophylactic department (MPD) of HAEK CJSC coordinates joint activity with the RA Ministry of Health forces on rendering the first medical aid, qualified and dedicated medical aid to ANPP personnel and taking them to medical institutions.

The functions and duties of sanitary team and MPD members are described in "Regulation for sanitary team" and "Regulation for MPD".

The functions and duties of the Ministry of Health forces are regulated by the RA legislation on CD&ES, "National plan on protection of population in nuclear and/or radiation accidents at ANPP" (Decree of the RA Government №2328-H of 22.12.2005) and "Plan of medical security and protection of the RA Ministry of Health staff in nuclear and/or radiation accident at ANPP". Involved forces of medical response in case of emergency are listed in table below.

Involved forces of medical response

No.	Activity	Involved medical forces
1.	First medical aid and sorting of affected personnel	MPD, Sanitary group
2.	Medical security of the sheltered personnel	MPD, Sanitary group
3.	Evacuation of affected personnel from ANPP site	Sanitary group, Special teams of the RA MH
4.	Dedicated aid to affected personnel	Specialized teams of the RA MA
5.	Prompt hospitalization of affected personnel	Specialized institutions of the RA MH
6.	Medical examination	Specialized hospitals of the RA MA

7.	Medical security of evacuated personnel	Hospitals, polyclinics of the RA MA

Engineering support of external organizations. The following organizations and institutions provide engineering support to ANPP:

- **State nuclear safety regulatory committee of the RA:**
  - Assessment and prediction of the event.

- **MTAES of the RA:**
  - Provision of fire safety;
  - Arrangement and implementation of off-site radiation monitoring;
  - Provision with required logistic supplies from the state reserve;
  - Provision with meteorological information.

- **Staff of the RA Police Unit №1043:**
  - Participation in ANPP physical protection;
  - Preservation of public order;
  - Participation in evacuation activity.

- **Regional Crisis Centre of WANO MC WWER NPPs:**
  - Analysis and prediction of emergency situation;
  - Development of recommendations on emergency situation management, localization, minimization of consequences and recovery of Unit safe condition;
  - Development of recommendations on personnel and population protection measures;
  - Consulting ANPP on issues of nuclear, fire safety, engineering, radiation and chemical protection, and design features of Units;
  - Development of conclusions on ANPP accident sequence and required measures at state (national) level;
  - Logistical supply of ANPP.

Support of contractors and other companies

Depending on accident scale and features during performance of protective measures for personnel, accident localization, performance of evacuation measures and accident mitigation the manager of HAEK CJSC emergency activities shall prepare (if needed) in an established order a Request to RCC and the RA authorities on involvement of additional forces and means of the RA.

The RCC shall ensure follow-up and coordination of logistical support in compliance with ANPP request. The request and logistical support to ANPP is implemented in compliance with existing national legislation and via corresponding governmental infrastructure.

The MTAES is the authorized institution on organization and performance of measures on protection of territory and population in case of accident at ANPP. The forces and means for accident mitigation are mobilized through MTAES.

The content of forces and means mobilized for mitigation of emergency situation at ANPP, and also need in their increase is defined depending on the emergency situation nature and scale, and their consequences, as well.

Emergency measures during emergency situations at ANPP are organized and performed by national organizations headed by the RA Prime-Minister (Head of the RA Civil Defense), acting as Chairman of Governmental Commission on emergency situations.

The interactions during emergency situation mitigation at ANPP are implemented in compliance with “National plan of population protection in nuclear and/or radiation accidents at ANPP”. The content of forces and means mobilized for accident mitigation are defined depending on each specific situation.
The interactions are planned and organized in case of threat and accident occurrence, and in the course of recovery activities.

The issues of interactions are periodically reviewed at the management meetings of interacting institutions, ES Commissions of executive authority.

Provision of emergency response with required missing equipment, including individual protection means, special clothes, tools and accessories, spare parts and repair materials, decontamination and sanitary treatment means, etc., is implemented from the State reserve of the RA, in compliance with “Regulation plan of state reserve in nuclear and/or radiation accident at ANPP”, and prepared emergency requests.

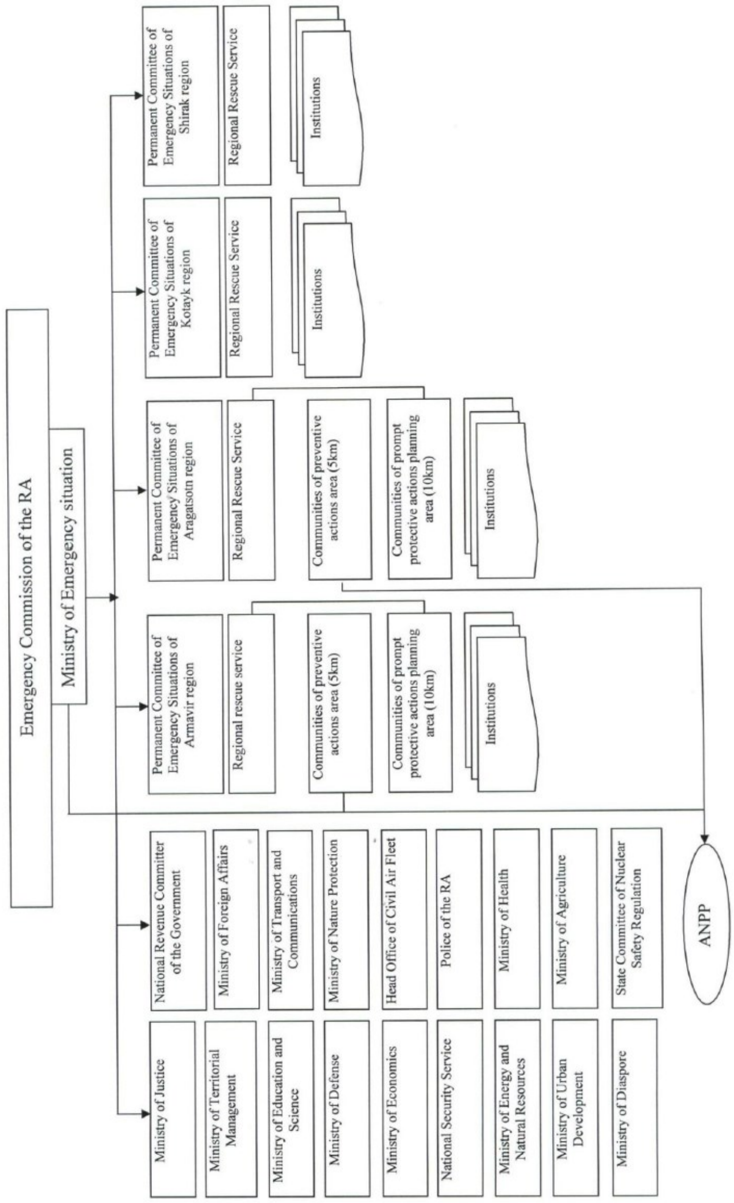


Figure 6-1. Structure of Population Protection Planning System.

6.1.1.5 Procedures, training and exercises

The present state is described in this sub-chapter. The training does not yet include SAMGs and will be extended after their introduction.

Internal emergency plan

- Procedures:
  The Internal Emergency Plan (IEP) set of procedures describes the structure and organization of the IEP for the ANPP site. It is comprised of 10 procedures which are updated on a regular basis.

- Training programs and exercises:
  Every person involved in the internal emergency plan receives initial training and periodic refresher training depending on his role. Training procedures define for each function or role which training program must be provided with regard to emergency responsiveness. This training also includes periodic exercises.

  Some exercises are planned at least two times a year (three exercises have been organized in 2012). They simulate real emergencies. One exercise on two deals with core melt scenarios. Everyone participate to these exercises. Each exercise is documented.

NPP operation

- Procedures:
  The existing operational documentation on emergency measures to be taken by the personnel is aimed at preventing design-basis accidents development into beyond-the-design-basis ones and at mitigation of consequences of beyond-the-design-basis accidents.

  An initial event of a blackout may assume a number of scenarios and appropriate responses by operating personnel with regard to recorded symptoms on systems and equipment condition. Operating personnel's activity in this regard is regulated by the following documents:
    - Procedure on mitigation of accidents in the reactor installation of ANPP Unit №2;
    - Guideline on beyond the design basis accidents management at ANPP;
    - Procedure on mitigation of accidents in the electric part of ANPP;
    - Procedure on recovery of ANPP auxiliaries' power supply in case of loss of RA power system;
    - Procedure on accident mitigation in the RA power network;
    - I category EPSS abnormal operation modes;
    - II category EPSS abnormal operation modes;
    - Procedure of Additional Cooling down System.

- Training programs and exercises:
  The operation staff is trained at ANPP Training Centre (TC), further they perform probation at a specific work place under supervision of an experienced operator. Following the probation at a workplace, the staff takes an exam on knowledge and practical use of rules, standards and procedures of operation at the Regulatory Authority or plant board (with regard to the position assigned). Once exams are successfully passed, the staff receives a right to work independently and is assigned as standby operators at a workplace supervised by an experienced operator. Following a successful standby work at a workplace, which confirms that the standby operator has practical knowledge and skills of performing maintenance of the equipment assigned, the standby operator is allowed to work independently. The personnel receive annual training on qualification maintenance and improvement at TC, periodically take exams on knowledge of rules, standards and procedures of

operation. Operating personnel receive monthly emergency trainings. Various (in terms of topics and way of conduct) emergency trainings demonstrate that the personnel are trained and ready for taking measures in non-standard situations. After the update of existing EOPs and introduction of SAMGs the whole training plan will be revised. Besides of requirement of knowledge of these procedures and guidelines, the training should aim to improve communication skill of the staff.

6.1.2 Possibility to use existing equipment

In the occurrence of a severe accident, it is fundamental to dispose of the following resources:

- Water for injection into primary (borated) and second (non-borated) circuits;
- Electrical supply to the vital equipment.

All existing equipment on site can be relied on to recover any abnormal situation along with the potential cross-connections whenever and wherever needed, as described within the incidental and accidental procedures.

As non-conventional means used in case of unavailability or failure of some of this equipment, the possible use of resources from the ANPP Unit №1 (which is in permanent shutdown without fuel in the reactor) can be considered, especially the water from the Borated Water Storage Tank (BWST).

6.1.2.1 Provisions to use mobile devices (currently under development)

In order to manage beyond-design events that could occur on the ANPP site, leading to the unavailability of the existing, fixed equipment, non-conventional equipment are foreseen to be used to fulfill several functions, such as:

- Feedwater supply to steam generators by fire pumps;
- Water supply for spent fuel pools;
- Emergency lighting;
- Diesel generators.

The mobile devices are foreseen to request only gas/power/water connections using flexible hoses stored at some location close to where they should be required.

6.1.2.2 Provisions for and management of supplies (fuel for diesel generators, water)

The amounts of supplies (fuel for diesel generator, boric acid for primary circuit, etc.) required for ensuring the correct operation of the safety equipment in the occurrence of an accident, excluding severe accidents conditions, are mentioned in the technical specifications. The respect of the minimum supplies and availability are described in and checked through specific procedures. Current situation is described here.

At the ANPP site, the following minimum amounts of supplies are available:

- Service tanks - 25m³ of fuel in each section of Diesel Generators Station, representing about 27 hours of operation of the diesel generators at full power. Under representative emergency conditions the tanks can be filled up from reserve tanks at the rate of 20m³/h or directly from a tank truck;
- There is an emergency diesel fuel tank at the plant ensuring DG operation for at least 15 days;
- Local diesel tanks at engine driven pumps, e.g. feedwater pumps, can be supplied for 24 hours;
- 800 + 800 m³ of borated water (Units №1 and №2), for injection into the primary circuit, but there may be a requirement to share this water with Unit №2 Spent Fuel Pool (SFP);
- 2000 m³ of normal water from the 4 demineralized water tanks, representing more than 15 days of supply to the steam generators;
- 2 sets of electrical batteries that can supply electrical power to vital equipment during at least 7-8 hours.

6.1.2.3 Management of radioactive releases, provisions to limit them

All mechanisms endangering the containment functions during severe accidents will be analyzed in frame of SAMG development. Currently SAMG for ANPP is under development process. The main safety functions for limiting radioactive releases are containment isolation and ensuring containment integrity. Especially at ANPP spray system is installed as part of localization system to reduce internal pressure and prevent opening of pressure relief valves and reduce the leak rate. In additional aerosols and iodine in-containment inventories which could deposit thanks to use of containment sprays. Using the containment spray system helps to retain radioactive aerosols and some of the iodine within the liquid phase. Moreover, potassium metaborate (K[B(OH)4].xH2O) can be added to the water spray to capture the iodine in a more efficient way, and then keep it in the containment and borated water storage tanks. As such, the containment spray system can in one day capture up to 99 % of the airborne iodine released from the primary circuit to the containment.

The confinement is made of reinforced concrete with thick walls (up to 1.5 m) to resist to some pressure increase, with a metallic liner on the inside to improve its leak tightness.

In case of radioactive release towards the confinement, the confinement structures and the spray system will guarantee that most of the radioactivity remains inside the confinement. As long as this radioactivity stays inside the containment, one can rely on the radioactive decay to limit the releases to the environment.

The internal ventilation is shut down in case of pressure rise.

The leak rate of the containment penetrations is checked on a regular basis (once a year). The current objective at the ANPP Unit №2 is to improve the leak tightness of all those penetrations in order to decrease the overall containment leak tightness by a factor of 10.

Some devices dedicated to the monitoring of the radioactivity level are also implemented at several locations on the ANPP site, including the reactor hall. These devices permit both fast detections of any leakage problem as well as constant measurements of the radioactive releases at the ventilation stack.

In case of radioactive releases due to some fuel handling operations carried out beyond the reactor hall, the confinement exhaust filtered ventilation system is switched into the mode of repair ventilation from reactor hall.

Finally, rather large storage capacities exist on the ANPP site in order to delay at most any potential liquid release towards the environment; their total net volume is 4354 m3 from which 1850 m3 is currently free.

Radiation monitoring is carried out at ANPP site and in controlled area rooms with use of RADACS facility and System 8004-01. This monitoring enables timely identification of worsening of radiological situation at the plant site, including penetrations failure.

Radioactive releases are monitored both on-site and within 5 km zone from the plant. However, monitoring system for off-site zone is not fully covered 5-km zone. Therefore, new system should be installed.

Monitoring of radioactive release and effluents at ANPP includes:

- Monitoring of gamma-aerosol releases activity from vent stack;
- Monitoring of activity of drained water of service and sanitary sewage (SSS), and industrial and storm sewage (ISS)

Water effluents to SSS and ISS are monitored with use of a number of instruments as multifunctional TRIATLER meter, beta-spectrometry device and others.
In addition, radiation monitoring at ANPP includes monitoring of essential loads cooling systems through which release of radioactivity to environment is possible.

6.1.2.4 Communication and information systems (internal and external)

The communication and information systems used at the ANPP site in case of an emergency situation are described in the Internal Emergency Plan. The description of current situation follows.

Internal communication.
The internal means of communication include:

- The on-site sirens and sirens for 5km zone are controlled from Central Control Board and Crisis Centre;
- Some radio wave devices for specific contacts (T.R.U.N.C. system);
- A radio-station that can run for several hours on back-up batteries in case of a station black-out event;
- Some internal phones that can run for 24 hours on back-up batteries in case of a station black-out event;

- Mobile phones with availability durations of 8 hours in stand-by mode or 2 hours in active communication;
- For emergency situations there are two mobile commutators for 10 subscribers each one. It is possible to create an internal operational communication at any location (from MCR or Crisis Centre).

The communication with the crisis centre is performed via a special communication system, similar to the one used by the army.

External communication.

- The external communication channels are executed on the base of radio-relay communication in two directions (ANPP – Armatom Institute, ANPP – MTAES, Yerevan). Each one has capacity of extension up to 30 channels;
- The external operational communications are redundant and are executed in Skada system, in high-frequency and 3 radio-relay communication (ANPP – Energy Grid Central Control Service). In total about 24 communication channels.

6.1.3 Evaluation of factors that may impede accident management and respective contingencies

6.1.3.1 Extensive destruction of infrastructure around the installation that hinders access to the site, including communication systems

The management of the damages that could occur to the installations of the ANPP site as well as its periphery is described within the Internal Emergency Plan (IEP). In order to ensure the initiation of the IEP and the formation of the Emergency Response Team (ERT), the Main Crisis Centre (MCC) is located in bunkered rooms in one on-site building. In case that these rooms are not habitable (conditions described in detail in IEP), the Backup Crisis Centre (BCC) rooms will be used: they are located sufficiently far from the plant.

Equipment are foreseen at the ANPP site to handle situations for which major obstacles should be removed or displaced to recover the site accessibility (e.g. extensive damages to the roads, fallen trees or pylons, etc…). If these proved to be insufficient, external support can also be requested, for instance from the army.

The infrastructure outside the site is not the responsibility of the plant operator. However, the IEP includes coordination with the relevant authorities (ministries, government and army) that will undertake the required measures needed to re-establish appropriate access to the power plant site (see chapter 6.1.1.4).

6.1.3.2 Loss of communication facilities / systems

See chapter 6.1.2.4

6.1.3.3 Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site

In order to perform the operations necessary to manage an accident, the intervention dedicated to the nuclear unit is concentrated in either the Main Control Room (MCR) or at the Back-up Control Panel (BCP) which is currently being implemented. Moreover, emergency response room is used to allow the formation of the ERT, whose objective is to support more globally the accident management, including on-site and off-site issues. Impairment of work performance in these locations mainly due to possible radioactive contamination has been indicated before and correction measures are being implemented described further for each location in chapter 6.2.5.

a. Main Control Room and Back-up Control Panel

This point is dealt with in chapter 6.2.5.

b. Emergency Response Team Centre

The emergency response team centre is located in the Main Crisis Centre (MCC), and its "habitable zone" is fitted with external air filtering equipment. In the presence of contamination on the site, existing procedures define a single access route to this "habitable zone". This access is equipped with a changing room, contamination measurement equipment and basic decontamination means designed to prevent the transfer of contamination to the "habitable zone". A single exit route is also defined with a stock of protective equipment allowing personnel leaving the emergency response team centre to wear according to the instructions issued by the radiation protection officers.

This emergency response team centre is equipped for a long stay, with a kitchen with a food supply and sanitary equipment (showers and toilet units). The emergency response team centre is equipped with indicators and recorders associated to the monitoring channels of the liquid and gaseous effluents exhausts of the unit. Basic equipment for measuring radioactivity is also provided in this centre. Two vans, parked close to it, are also fitted to measure radioactivity on the field. Indeed, in case of radioactive contamination or excessive dose rate, the radioactivity monitors at the exit of the nuclear zone of a unit may become inoperative. As a result, control may be transferred to the monitors of the access building at the site entrance. However, no radioactivity monitoring will be operational if the electricity is cut off.

Safe from flooding, resistant to the design basis earthquake, fitted with decontamination infrastructure, powered by a stand-alone generator and equipped with external air filtration devices, this emergency response team centre is suited to perform its expected functions. However, in case of loss or inaccessibility of the emergency response team centre, a fallback room is provided at the off-site centre located in the ARMATOM building in Yerevan, 28km from the plant”.

In regard to the results of WANO Peer Review carried out in June 2013, improvement of the MCC and BCC is planned; currently the reconstruction design is under development:

- MCC – create an airlock in the crisis centre;

- MCC - purchase additional dose metering instrumentation and equipment;
- Creating a new back-up crisis centre BCC of ANPP.

A room is assigned in the premises of the the Ministry of Territorial Administration and Emergency Situations of the RA to create ANPP BCC implying implementation of the following:

- Installation of communication and notification devices;
- Installation of computers, office equipment, special software for situational analysis, hearings and meetings;
- Provision with individual protection means.

c. Intervention capability in case of destruction of some on-site installations

Apart from the fact that some of the buildings are "bunkered" and could act as an on-site meeting and dispatching point if some installations are destroyed, some devices that can be used to clear the roads are available. However, the use of heavy equipment is planned only via contracts with companies in the vicinity of the power plant.

6.1.3.4 Impact on the accessibility and habitability of the main and second control rooms, measures to be taken to avoid or manage this situation

See chapter 6.2.5

6.1.3.5 Impact on the different premises used by the crisis teams or for which access would be necessary for management of the accident

See chapter 6.1.3.1

6.1.3.6 Feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods)

Majority of activities needed for severe accident management are controlled from the MCR or MCC that are not directly threatened by effects of extreme external events. “Severe Accident Management Guidance” for the Unit № 2 is currently developed by ANPP. In frame of SAM project implementation special assessment should be perform to assess feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods). In addition, list of equipment that will be installed for severe accident management, should be located in buildings resistant against external effects.

6.1.3.7 Unavailability of power supply

In frame of SAM project implementation it should take into account assumptions regarding unavailability of power supply sources during severe accident. Redundancy of power sources with robust design should be considered. All consumers, which will have special function during severe accident, should be in additional powered from the dedicated power sources with robust design. More detailed information concerning unavailability of power supply is presented in 6.3.1.6.

6.1.3.8 Failure of instrumentation

Currently accident procedures are based on a limited number of parameters which are monitored by safety rated instrumentation. During the implementation of accident procedures, a permanent connection between the shift crew and the emergency response team is established. The physical parameters measured represent an important part of this interchange. An assessment of the validity of the parameters supplied is performed to ensure there is no malfunction.
In the highly unlikely hypothesis that all instrumentation is down, necessary actions will be assessed and carried out on the basis of calculation tools and graphs implemented by the emergency response team.

However, when SAM will be implemented it is necessary to consider diverse measurements principles for I&C to decrease their vulnerability in case of failure of some measurements. Even though qualification of instruments for severe accident conditions is not required, it is necessary to prove its survivability in such conditions. Analyses should be performed for determination of thermal-hydraulic and radiation parameters which will be used in severe accident domain. This point is also dealt with in chapter 6.2.4, where a list of instrumentation needed for SAMG is included.

6.1.3.9 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 is only one unit in operation on the site (Unit 2). The other unit (Unit №1) is without fuel and serving as backup for reserves of water. It cannot negatively influence the accident management on Unit 2. There are no other installations on the site. Spent fuel storage pool of Unit №1 is not estimated to present a risk, fuel older than 5 years is stored here which implies very long times to fuel damage.

6.1.4 Conclusion on the adequacy of organizational issues for accident management

See previous subchapters for current status. Development of symptom oriented EOPs and SAMG is proceeding, they will much improve the accident management. The introduction of SAMGs will require also re-training the personnel.

6.1.5 Measures envisaged to enhance accident management capabilities

See previous subchapters. Especially introducing SAMG will enhance accident management capabilities. Some technical measures have already been planned or are added in this report, chapters 6.3, 6.4.4. The data from the environment radiation monitoring system should be made available to the MCC.

In frame of SAM project implementation special assessment should be perform to assess feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods).

All consumers, which will have special function during severe accident, should be in additional powered from the dedicated power sources with robust design.
Analyses should be performed for determination of thermal-hydraulic and radiation parameters which will be used in severe accident domain. For determined list of parameters it is necessary to prove its survivability in severe accident conditions.

6.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

6.2.1 Description of the accident management measures currently in place before occurrence of fuel damage in the reactor pressure vessel (including last resorts to prevent fuel damage), after occurrence of fuel damage and after failure of the reactor pressure vessel of a scenario of loss of the core cooling function

6.2.1.1 First stage: before occurrence of fuel damage in the reactor pressure vessel

In case of loss of core cooling, before core damage in the reactor pressure vessel, the guideline for beyond design basis accident management is applicable. Attending that the events dealt with in this guideline are beyond the design basis (failure of safety systems or equipment), all can lead to loss of core cooling and all are dealt with in this section.
The objective of this guideline is to ensure safety of the unit:

- By ensuring shutdown of the reactor and maintaining it subcritical;
- By ensuring core decay heat removal;
- By ensuring Reactor Coolant System heat removal and if necessary RCS cooldown;
- By controlling reactivity in case of Loss Of Coolant Accident by providing adequate boron concentration;
- By diagnosing the initiating event.

The guideline follows an event-based approach. It takes into account four families of events: loss of coolant accident (LOCA), steam generator tube or collector rupture (SGTR or SCTR), loss of feedwater accident and loss of off-site power (LOOP). Thirteen events have been identified among those four families:

- LOCA of an equivalent diameter ≤ 32mm coupled with failure of Safety Injection (APN) system;
- LOCA of an equivalent diameter > 32 mm and ≤ 200mm;

- Steam Generator Tube Rupture coupled with failure of Safety Injection system;
- Steam Generator Primary Collector Rupture (equivalent diameter 100mm) coupled with failure of isolation of safety valve of affected SG;
- Spurious opening and failed closure of pressurizer safety valve coupled with failure of Safety Injection system;
- SCRAM with failure to close turbine isolation valves;
- Main steam collector break (full section);
- Main steam line break (full section, not isolable);
- Spurious opening and failed closure of SG safety valve;
- Main feedwater line break (not isolable);
- Main feedwater collector break coupled with failure of auxiliary feedwater system;
- Total loss of feedwater (main and auxiliary);
- Complete Loss of Off-site Power as a consequence of "Maximal designed earthquake".

The guideline contains one specific tool to help for diagnosing the event. The diagnostic tool helps for the determination of the event, its confirmation based on main plant parameters, on the sequence of automatic protection signals and the status of systems important to safety. It will help detecting the failed equipment and the related safety function.

After this, for each event, the objectives, some indication on time window, the sequence of protection signals and the actions to be performed are indicated.

For all events, the final objective is to achieve controlled and stable cold shutdown by cooling the reactor system.

To achieve the first general objective of the guideline, the operators are instructed for all families of events a first set of actions to control and confirm manually all automatisms (including SCRAM), to control reactor parameters and control rods insertion and to switch to the source range of neutron flux measurement.

The specific actions for the four families of events are summarized hereafter.

More specifically, for LOCA events with loss of Safety Injection (APN) which is a typical event provoking loss of core cooling, in addition to the first set of actions, the recovery of Safety Injection is instructed (the plant is equipped with so called Accident Makeup System providing for High Pressure Safety Injection) and also the depressurization and cooldown of the primary circuit by the second circuit (at a maximum rate of 60°C/h). During RCS depressurization, the operators have to be careful with the risk of return to criticality (particularly at 140°C, they have to inject boron and stabilize boron concentration in RCS) and they have to maintain the operation of reactor coolant pumps (they should only be stopped in case of vibration). As soon as safety injection is recovered, the operators have to start safety injection pumps and inject water from the Borated Water Storage Tank (BWST), which already has high boron concentration. If Safety Injection is not recovered in the time window, RCS injection has to be started with alternate means such as injection from boric acid supply tank with normal make-up pumps. The operators have to locate the break with help of the system for leak control of reactor coolant, the information on noise, the follow-up of the reactor parameter or, in last resort, by using the method of loop isolation. Once the break is located, the affected loop is isolated and the related reactor coolant pump is stopped. It may be impossible to isolate the loop if the break is in the non-isolable part, then injection has to continue. The water escaping from the RCS is automatically drained to the BWST. The injection into RCS to recover primary inventory has to take into consideration the risk of solid RCS in case of overfill of pressurizer (discharge lines have to be used to control pressurizer water level) and the risk of return to criticality. RCS depressurization and cooldown has to be continued with second circuit.

In case of SGTR coupled with failure of the Safety Injection, in addition to the first set of actions, the operators have to detect the affected SG, isolate it on both water and steam lines and stop reactor coolant pumps of the related loop. In case of affected SG cannot be detected or isolated, the primary circuit has to be depressurized and cooled down with adequate injection of boron.

In case of SGCR (primary collector rupture) with failure of closing of affected SG safety valve, the specific action is the RCS depressurization and cooldown by the unaffected SG below the safety valve opening setpoint and isolation of the failed primary loop including stop of MCP.

In case of loss of feedwater accident, in addition to the first of actions, the recovery of feedwater is instructed. In case auxiliary feedwater is unavailable, other means for injection into SG have to be used: main feedwater, emergency feedwater (seismic pump, condensate pump) and second emergency feedwater (diesel pump). If feedwater to SG cannot be recovered and water level in SG is too low, then primary feed and bleed is instructed.

In case of Station Black-Out as a consequence of “Maximal designed earthquake”, in addition to the first set of actions, the status of safety functions to achieve cold shutdown has to be established in accordance with ANPP procedures. The loss of primary and second inventory has to be minimized. Three symmetrical loops have to be isolated; heat removal is performed with the other loops. It is instructed to recover electrical supply by starting diesel generators manually or by recovering external electrical supply. In case electrical supply is recovered but not for sections 3,4, the second emergency feedwater system with diesel pump has to be used. After recovery of electrical supply, it is instructed to slowly feed SG (to avoid thermal shock) with auxiliary feedwater, start reactor coolant pumps (taking care of boron concentration) and start cooling down with second feed and bleed.

Symptom based EOPs are under development to replace the Design Bases and Beyond Design Bases procedures at the time of SAMG implementation. More details concerning their implementation are in chapter 6.2.3. Adding low pressure ECCs is foreseen that would much enhance the cooling capacity.

6.2.1.2 Second stage: after occurrence of fuel damage in the reactor pressure vessel

In case of loss of core cooling, after core damage in the reactor pressure vessel (before reactor pressure vessel failure), the guideline for beyond design basis accident management is still applied. No specific instructions exist in case of occurrence of fuel damage in the reactor pressure vessel. The guideline contains a detailed description of the phenomenology of a severe accident with specificity for the same initiating events as for preventing core damage. The same recovery actions than before core damage will be continued after core damage. The description of severe accident phenomenology is considered insufficient to guide the operators, it should be replaced by SAMG. Besides, as the existing guidelines are oriented to core cooling recovery, some negative effects where SAMG provide guidance may be overlooked. SAMGs are being developed for the situation of a severe accident. Some technical measures have been also proposed in this report.

6.2.1.3 Third stage: after failure of the reactor pressure vessel

In case of loss of core cooling, after core damage in the reactor pressure vessel (after reactor pressure vessel failure), the same guideline for beyond design basis accident management is still applied. No specific instructions exist in case of reactor pressure vessel failure. The same recovery actions than before core damage will be continued after core damage and failure of reactor pressure vessel. Among these actions, the injection into the RCS will help cooling the core still inside the reactor pressure vessel but also the debris of the core present in the cavity through the failed reactor pressure vessel.

The present guideline is insufficient and will be replaced by SAMGs. They will be basically the same as those for the containment and the early phase. A feasibility study of in-vessel retention by external vessel cooling will start, when realized, this measure would with high probability exclude this stage of accident.

6.2.2 Identification of any cliff edge effects, and evaluation of the time before they occur for a scenario of loss of core cooling function (all stages)

For the assessment of the cliff edge effects related to time margin between reactor shutdown and core meltdown the analysis of blackout scenario was performed with the RELAP5 Mod3.2 code. The calculation showed that key aspect is operation of the diesel driven pump for steam generator emergency feedwater system. It can supply water up to the pressure of 5.26 MPa (53.6 kp/cm^2). If this system is in operation there is large time margin between reactor shutdown and core heating up. Heating up of fuel cladding beyond 1200°C (maximum acceptable value) was observed after 169 hours (7 days) from the initiation of the accident. Any alternative device which is able to fill in water into steam generators would result in the prolongation of time delay between reactor shutdown and the start of core degradation.

In the worst case when the diesel driven pump for SG FW supply is not available from the beginning, the time margin between reactor shutdown and start of core heating up was shortened to 6-11 hours depending on reactor power. The similar time margins can be expected during transient scenarios initiated by total loss of feedwater.

The combination of LOCA accident with plant blackout could lead to earlier core damage; nevertheless, on the basis of probabilistic analysis made for similar type of nuclear power plants (VVER-440), the probability of such event seems to be low.

Cliff Edge Effects Related to Time of Reactor Pressure Vessel Failure.
To assess the cliff edge effects related to time of Reactor Pressure Vessel Failure the MELCOR 1.8.6 analysis of long term station blackout scenario was performed. Steam generator emergency feedwater system with the diesel driven pump was not taken into consideration. Reactor pressure vessel lower head failed due to thermal creep rupture at 53847 seconds (14:57 hours = 14 hours 57 minutes) after the start of the accident. This relatively long time is connected with the assumed operating power 1265 MW thermal - 92% of the nominal power 1375 MW. Two other types of scenarios – complete loss of feedwater and SG primary collector break and lift-off have been analyzed more recently with MELCOR in the frame of SAMG development. The nominal power 1375 MW thermal is assumed for these scenarios. In the base case (without accident management) of the scenario with feedwater loss, vessel failed at 40530 s (11:16 hours). Similar vessel failure time would be the result of Station Blackout at the reactor nominal power.

In all the analyzed scenarios, corium is ejected to the reactor cavity and corium pool is formed in the reactor cavity. Temperature of melt pool is high enough to start corium-concrete interaction. Therefore there is a real threat of corium penetrating the cavity wall which is only 60 cm thick at the man shaft access to the cavity. The results of MELCOR analysis of Station Blackout Scenario showed that this penetration was reached at 120942 s (33:35 hours). The other scenarios are analyzed until vessel failure only. Though this penetration can be considered also a cliff-edge effect, it has less serious consequences as it does not lead directly to atmospheric release, fission products enter the soil and diffuse more slowly than in the atmosphere. Nevertheless, cavity base mat or shaft penetration should be postponed as much as possible. Therefore, it is suggested to study the possibility of in-vessel debris retention by external vessel cooling in order to even preclude vessel failure. This measure has been already realized at both units of the Loviisa VVER- 440 plant and it is foreseen for many other VVER-440 units.

A more serious cliff edge effect is the failure of vessel bottom head at high pressure. Though it probably would not cause sufficiently strong vessel missile effect to eject it from the confinement, it may rupture the confinement due to overpressure of gases and direct containment heating. It may also cause pressure relief valves misalignment and failure to close. To prevent these high pressure melt ejection effects, timely primary depressurization by pressurizer PORV will be included in developed new SAMG and it was already analyzed in frame of analytical justification of SAMG.

6.2.3 Assessment of the adequacy of the existing management measures for a scenario of loss of core cooling function (all stages), including the procedural guidance to cope with a severe accident

In case of loss of core cooling, the guideline on beyond the design basis accident management is presently applied. This guideline aims to prevent core damage and does not consider a possible occurrence of core damage. The lack of measures on beyond design accident management and of technical tools providing with mitigation of their consequences is recognized as a negative feature of ANPP2. The need for comprehensive operating procedures is also indicated in IAEA-TECDOC-640.

In the framework of the safety improvements of ANPP2, the implementation of improved Emergency Operating Procedures intended to assist the operator for diagnosis of events and for application of relevant emergency action has been scheduled. Accordingly, the development of a full set of Emergency Operating Procedures (at full power and for shutdown states) based on Westinghouse Owners Group Emergency Response Guidelines (WOG ERG) is under way. The project is supported by the US Department of Energy with Pacific North National Laboratories and Argonne National Laboratories and by Ukrainian experts for specific VVER-440/270 aspects.

The adequacy of WOG ERG as Emergency Operating Procedures is recognized internationally. They are used in many plants worldwide. The WOG ERG deal with preventive accident management and contain actions for dealing with both design basis accidents and beyond design basis events before the onset of core damage. Their application to VVER NPPs has already been performed in Ukraine and has been analytically justified. The ANPP2 plant-specific implementation is supervised by international experts and it should be adequate. Its adequacy will be validated by return of experience of their implementation. Return of experience of Fukushima which is or will be included in generic WOG ERG will be analyzed if any and implemented if applicable to ANPP Unit №2.

The development of Severe Accident Management Guidance (SAMG) has been recommended by IAEA mission in 2009. In the framework of the Comprehensive Modernization Program for ANPP-2, one task has been devoted to severe accident analysis aiming, among others, to support the development of SAMG. It has been recommended to develop the complete list of potential severe accidents, to perform the analysis of these accidents and to develop the guidelines for accident management accordingly.

Presently, SAMG at full power are under development. They are based on generic WOG SAMG philosophy. The WOG SAMG is applicable once core damage has occurred. Their priority is to contain and minimize fission product releases. They consist in strategies for the implementation of mitigate severe accident management measures. The generic WOG SAMG is adapted to VVER-440/270. The adaptation for ANPP Unit №2.is performed by Armatom Institute with the support of US Department of Energy. A first draft of some of the procedures is available at present. The objective is the implementation of SAMG at full power for ANPP2 by the end of 2015.

The adequacy of generic WOG SAMG is recognized internationally. Generic WOG SAMG has been validated during their development. Revision 0 has been issued in 1994. Return of experience of numerous plant-specific implementations has been taken into account in revision 1. The application of generic WOG SAMG to VVER has been studied in the framework of EC PHARE project and recommendations are provided for plant-specific VVER-440/213 implementation. Since then, the WOG SAMG has been successfully implemented on all the VVER-440/213 units in Czech Republic, Hungary and Slovakia. The implementation of plant-specific SAMG for ANPP2 considers those recommendations and the return of experience of the previous implementations if applicable to VVER-440/270.

The implementation of SAMG for shutdown states will be considered in a future revision.

Any other return of experience of Fukushima which will be included in generic WOG SAMG will be analyzed and implemented if applicable to ANPP Unit №2. Revision of post-Fukushima generic WOG SAMG is awaited at the end of 2015. All new means, features as a follow-up of the stress tests or resulting of the modernization program of ANPP Unit №2 will be included in SAMG.

Regarding the technical tools to manage a loss of core cooling, it is recognized that the original Emergency Core Cooling Systems (ECCS) are not able to provide an adequate safety function for short and long term cooling for the full spectrum of Loss Of Coolant Accident. The comprehensive modernization program of ANPP Unit №2 includes the modernization of ECCS to meet the requirements for the newly defined maximum Design Basis Accident and to increase the capability for long term cooling. The installation of the new ECCS will allow improving the core cooling capability.

6.2.4 Evaluation of the potential for additional measures for a scenario of loss of core cooling function (all stages). Part I: Suitability and availability of the required instrumentation

In case of loss of core cooling (even in case of core damage), the guideline on beyond the design basis accident management is presently applied. Seven plant parameters are used to manage the event in this guideline. Those plant parameters are:
- Pressure in the reactor building;
- Water level in Steam Generators;
- Status of pressurizer safety valves;
- Noise in the turbine building;
- Pressure in main steam collector;
- Pressure on feedwater pumps head.

It has been recognized during the different IAEA missions that I&C of ANPP Unit №2 were very old fashioned and might not be reliable. In the framework of the different related safety upgrading programs for ANPP Unit 2, the post-accident I&C has been and is still being improved. It consists among others in the implementation of the Post- Accident Monitoring System (PAMS, including the protection systems elements of this system). The implementation is supplemented with the description of the PAMS as implemented using both the new PAMS instruments and the upgraded protection system instruments. The demonstration that the overall system fully meets the identified system design criteria i.e., Regulatory Guide 1.97 revision 2, including the recommendations on variables displayed, measurement range, and design and qualification criteria has to be provided within the ANPP Comprehensive Modernization Program.

The implementation of such a program that meets the international recommendations and the consideration of the upgraded instrumentation in the under development plant-specific WOG Emergency Response Guidelines is adequate for accident management before core damage: the plant-specific PAMS are qualified for accident monitoring.

In addition, this upgrade program for instrumentation allows ANPP2 to adopt the generic WOGSAMG approach regarding instrumentation for their plant-specific SAMG. The generic WOGSAMG approach is based on the current characteristics of the unit and the existing equipment, including instrumentation. At international level, it is recognized that "concerning the instrumentation needs and capabilities, the experience has shown that this strategy to be workable for the following reasons:

- Analyses have shown that instrumentation environmentally qualified for design basis accidents in a conservative way, exhibits important capabilities to remain operational in severe accident conditions (analyzed in a best estimate way), especially given the reduced accuracy needs.
- The identification of redundancies and alternate means to obtain information on key parameters can increase the confidence in the capabilities of existing instrumentation in severe accident conditions. When several sensors measure the same parameter, it is easier to identify failed instruments. It is also often possible to obtain indirect information on a given parameter (e.g. the safety injection flow rate is an indication of the primary pressure). Graphical aids can be prepared to help interpret some indications (e.g. to obtain the level of water in the reactor building sumps from the level of water remaining in the reactor water storage tanks).
- In order to obtain an accurate picture of the accident and its progress, it is necessary to measure a large number of parameters. However, it has been shown that such a detailed picture was not needed to derive an effective severe accident management plan and that only a few key parameters were sufficient for this purpose, thereby reducing the instrumentation needs".

It has to be noted that the availability of the instrumentation depends on the availability of off-site power, of safety diesels and DC batteries (7-8 hours of autonomy). In case of total loss of instrumentation, severe accident management can be continued based on WOGSAMG philosophy by applying the different guidelines according to their priority.
At present, most of the measurement instrumentation for ANPP2 WOG SAMG is already in place, it is thus only needed to include them in PAMS accordingly to what is mentioned above. The instrumentation is used for the follow-up of the following seven key plant parameters:

- Core temperature;
- Pressure in the primary circuit;
- Water level in the steam generators;
- Pressure in the containment;
- Water level in the containment sump;
- Hydrogen concentration in the containment atmosphere;
- Dose rate on the site.

From this list, only the hydrogen concentration measurement in the containment atmosphere is missing. Some details concerning individual instrumentation at ANPP2 follow:

- Core thermocouples are located at the exit of selected assemblies. They can be supplemented by cold/hot legs thermocouples;
- RCS pressure is measured at several places in the RPV, primary circuit loops and pressurizer;
- Water level is measured in all 6 steam generators;
- Containment pressure is measured at several places in the containment rooms;
- Water level is measured in the borated water storage tank and in the similar tank of the first unit used as a reserve for Unit 2. The tank of Unit №2 serves also as a sump, the water level on the containment floor from which it is drained is always small and not measured;
- Hydrogen concentration in the atmosphere is not measured. Adding this measurement at least in the low concentration range (below 10% mole percent of hydrogen) is strongly recommended, as it should be used in the SAMG;
- Dose rates measurements are taken at several places on the plant premises.

In addition to these means, some other alternative means to measure or estimate the key plan parameters are provided in WOG SAMG. For all these measurements, an assessment of their survivability in Beyond Design Bases conditions including severe accident conditions is necessary.

Any return of experience of Fukushima regarding instrumentation which will be included in future generic WOG SAMG will be analyzed and implemented if applicable to ANPP2.

6.2.5 Evaluation of the potential for additional measures for a scenario of loss of core cooling function (all stages). Part II: Availability and habitability of the control room

The ANPP Unit №2 control room is situated at the level +9.6 m, i.e. at the same level as the reactor hall. The control room is between the reactor hall and the turbine hall. This is above and aside from the containment ceiling. The control room is equipped with overpressure ventilation which is considered insufficient for accident conditions for several reasons including lack of activity measurement or no prevention of smoke ingression. After recommendations from IAEA missions, a project of reconstruction was prepared, based also on overpressure venting but providing for filtered venting and conditioning of the intake air and measurement of activity. The delivered air is filtered on a battery of filters including de-humidifier, normal filters, absolute filter and gaseous iodine charcoal filter. The control room is designed to be used for 30 days in accident conditions. In case of failure of the air-conditioning (e.g. complete loss of electricity) or high activity of the intake air, the venting can be stopped and the control room is then hermetically closed. It can be used in this state for 30 hours, which is calculated from the concentration of CO2 exceeding permitted level.

The backup control panel enabling safe shutdown and replacement of the main control room in case of an accident is located in the turbine hall. Its connection to the plant systems but not take in operation because habitability of backup control panel is not satisfied the regulatory requirements. It has following capabilities (when coupled to Post Accident Monitoring System):

- Control of primary normal make-up system main equipment;
- Control of primary pressure system (pressurizer) main equipment;
- Control of primary emergency make-up system main equipment;
- Control of spray system main equipment;
- Control of fuel storage pools cooling systems main equipment;
- Control of primary coolant purification system main equipment;
- Control of boron compartment 2 drain pumps;
- Control of boron concentrate pumps;
- Control of primary emergency gas removal system main equipment;
- Essential loads cooling system train 1 and train 2 pumps; Control of SG emergency feedwater system equipment; Control of emergency cooling-down system equipment;
- Control of isolation valves on main steam lines and main steam header;
- Control of isolation valves on different ventilation systems;
- Emergency trip of reactor : AZ-1 button;
- Control of plant status main parameters.

After finishing the reconstruction of the main control room and putting into operation the backup panel, the availability and habitability of the control room can be generally considered as appropriate. For the case of a severe accident, the habitability of the main control room will be ensured by the new implemented ventilation, air conditioning and emergency filtration system; in case of habitability loss and evacuation, use of the backup control panel should be foreseen. For Design Basis Accident with loss of primary coolant, the load by radiation has been verified for different wind directions. On base of performed analysis new “HEPA” filters installed on MCR ventilation system. In the long term of severe accident, large release of activity to the environment could be possible. In the case of large release to the environment, the filtration system may not have sufficient capacity to retain the radioactive aerosol or gases (noble gases would not be retained at all).

In case of a shutdown accident with open reactor to the reactor hall, fission products from the reactor hall may enter the main control room.

For any case of reduced habitability of the main control room or of the backup panel room, it is proposed to assess if one should not better recommend to evacuate the personnel and to allow later on for temporary short re-entries of personnel wearing the adequate protective suits instead of keeping the personnel inside the isolated control room. Indeed, after vessel bottom head failure (or in case of its prevention) the actions foreseen in the SAMG to be performed by the personnel are usually not very frequent, once in many hours. Restoring control and returning the unit to a safe state may last several days or weeks.

6.2.6 Evaluation of the potential for additional measures for a scenario of loss of core cooling function (all stages). Part III: Evaluation of potential H2 accumulations in other buildings than containment

The evaluation of potential hydrogen accumulations in other buildings than containment for a scenario of loss of core cooling function is covered by the evaluation performed in chapter 6.3.6 for protecting containment integrity.

6.3 Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

6.3.1 Description of the accident management measures and plant design features for protecting containment integrity after occurrence of fuel damage

After core damage in the reactor pressure vessel, the guideline for beyond design basis accident management is still applied but no specific instructions exist for such a case. The guideline contains a detailed description of the phenomenology of a severe accident with specificity for the same initiating events as for preventing core damage (see chapter 6.2.1). Practically, the use of the beyond design basis accident management guideline will be continued after core damage with in addition, some actions to protect the containment integrity and reduce the fission product release to the environment, especially the verification of automatic and manual start of containment sprays which is already in the existing guideline.

It should be reminded that for ANPP2 large release of radioactivity might already occur from the intact containment due to:

- High natural leak;
- Possible opening of the pressure relief system in the early severe accident phase.

All the measures to protect containment integrity should not allow for permanent high pressure in the containment, especially above 0.75 bar overpressure that would open the containment pressure relief valves.

6.3.1.1 PART I: Elimination of fuel damage / meltdown in high pressure

Because of small containment volume and containment pressure relief valves, vessel bottom head failure at high pressure would lead to fast relief opening and it may lead to containment rupture or failure of the relief valves closure. Both would lead to very high release of radioactivity. Vessel bottom head failure at high pressure should be thus avoided by all means.

Fuel damage and especially vessel bottom head failure at high pressure are eliminated by timely opening of pressurizer safety valves that can be operated manually as PORVs. According to the present status, the personnel would use the guideline for beyond design basis accident management which does not directly connect this action to reaching the core exit temperature corresponding to the onset of core damage.

When taking into account the flow area of the valves, they have a sufficient capacity to fully depressurize the primary system within 30 minutes while the time between the onset of core damage and vessel failure is 4 – 5 hours (it was 4:30h between the start of high temperature Zr oxidation and vessel failure in the SBO scenario). This means that the high pressure melt ejection can be avoided even if the operator action to depressurize the RCS is much delayed. On the other hand, timely depressurization would also reduce the early hydrogen production because the fuel ends of the control rods below the core produce large quantity of steam impairing steam starvation. For this reason, the RCS depressurization should start as early as possible after the onset of core damage (if not instructed by EOP to start it before core damage). Because there are only two valves and both must be open to depressurize the RCS, their reliability to open and to be kept open should be re-assessed.

6.3.1.2 PART II: Management of hydrogen risks inside the containment: prevention of H2 deflagration or H2 detonation (inerting, recombiners, igniters)

Hydrogen deflagration or detonation is considered as a remaining issue in case of a severe accident (i.e. after core damage) for ANPP Unit №2. This situation is inherent to the specific design of such a plant for which:

- The containment[9] volume is relatively small (13500m^3);
- The total mass of Zr in core is relatively large, 18449kg.

Moreover, the in-containment hydrogen issue during severe accident is strongly depending on the sequence type of accident occurring and especially on the initiating event (e.g. history and timing of events, timing/rate of production and release of hydrogen):

- Large break Loss Of Coolant Accident (LOCA) or steam line break lead to very early opening of the pressure relief of the containment and loss of large part of atmosphere including oxygen.
- Station Black-Out (SBO) with partial inertization by steam.
- Other scenarios without primary system rupture like transients or a small break LOCA with sprays operating.

The case 1 was studied in detail in frame of Comprehensive Modernization Program for Armenian NPP Unit №2 for a large break LOCA 209 mm on the pressurizer surge line. The break leads to containment pressure relief opening and loss of oxygen rapidly after the initiating event, so that the hydrogen released later in the severe accident phase (i.e. after core damage) does not present a major risk for the containment. However, if containment sprays are started during the transient that leads to containment underpressure, air ingression from the environment occurs, therefore increasing the risk for hydrogen explosion, even if Passive Autocatalytic Recombiners (PARs) would be present. In frame of Comprehensive Modernization Program, it is recommended to manage spray system's mass flow to avoid any risk of hydrogen deflagration. Particularly, it is recommended to implement spray system interlock to reduce the risk of failure of the confinement liner or the ALS due to subatmospheric pressure and to reduce oxygen leakages from environment. Analysis results showed that spray system interlock implementation significantly reduces amount of oxygen in confinement and consequently decreases risk of hydrogen flammability. However, reduced amount of oxygen impacts hydrogen recombination rate and results of analysis showed that in the end of calculation hydrogen volume concentration reaches value of 76%. Therefore spray system interlock needs to be revised considering as flammability limits as well as PARs reaction rate.

[9] The ANPP has no containment of the usual type, i.e. leaktight and without possible opening to the environment. It is equipped with a confinement, but for the purposes of these stress tests, it will be denoted as a containment because it replaces its functions.

In frame of development of ANPP Unit №2 SAMG SBO analysis was performed. The results of calculations showed that due to partially inertisation of containment by steam and leak of hydrogen and oxygen to the environment risk of hydrogen explosion was suppressed. Because of the high natural leak rate of the ANPP Unit №2 containment, also a high amount of fission products is released to the environment before the end of calculation, i.e. 33 hours after the initiating event. To reduce fission products release rate from containment to the environment it is necessary to reduced pressure in containment by sprays system. However, as already stated in case 1, this action could lead to an increase of the hydrogen explosion risk.

The case 3 can lead to high risk of fast hydrogen deflagration or transition to detonation due to spray system operation, which will be led air ingression from the environment.

At present, the plant has no specific means (e.g. PARs or igniters) to control the hydrogen risk. Moreover it is shown that the action of reducing the sprays operation, which is a typical solution to avoid risk of hydrogen deflagration in case of air ingress from the environment, is difficult to manage as stopping sprays could lead to large release of activity to the environment. Besides, it is not certain that dangerous regimes like accelerated hydrogen deflagration or transition to detonation cannot occur in other scenarios without sprays.

To conclude:

- The hydrogen risk is well recognized for the ANPP Unit №2 and analyzed in frame of the Comprehensive Modernization Program development, ongoing PSA level 2 study and within the ongoing project aiming at developing a set of SAMG, specific for ANPP Unit №2 (see chapter 6.2.3).
- Providing hydrogen concentration measurement and installing PARs are listed in Comprehensive Modernization Program for Armenian NPP Unit №2 as a part of approved plant modernization.

It is recommended to continue further in these activities, but also to study the possibility to install hydrogen igniters as they could manage the hydrogen risk during the in-vessel phase while PARs are too slow. The number of PARs can be then decreased, they would reduce the in-containment hydrogen residual risk (e.g. in case of air ingress) in complement with igniters. Care should be taken for the distribution of both PARs and igniters inside the different compartments of the containment.

6.3.1.3 PART III: Prevention of over-pressurization of the containment

According of analysis performed in frame of ANPP SAMG development containment presents no risk of over-pressurization mainly because it is equipped with pressure relief to the atmosphere before reaching its design pressure. However, in case of failure all containment safety valves the spray system can't maintain pressure in confinement below of design pressure.

In addition, specific assessment of possible hydrogen deflagration/detonation in case of severe accident is not considered. Hydrogen deflagration/detonation can also lead to pressure pike in containment. It is recommended to continue assessment pressure pike in containment due to hydrogen deflagration/detonation.

6.3.1.4 PART IV: Prevention of re-criticality

The VVER-440 reactor of the ANPP Unit №2 is basically a PWR reactor not designed to operate in the boiling regime. After the core geometry is lost in a severe accident, re-criticality is highly improbable. To prevent re-criticality in the phase with intact geometry, following measures are taken:

- During an accident, the water for the primary makeup system is taken from the Borated Water Storage Tank (BWST) with the capacity of 800m^3 located below the containment. The minimum boric acid concentration in this tank is 12g/kg and the temperature 50°C, for which parameters a subcritical core is assured;
- Sprays also take water from this tank;
- As an additional source of water for the primary makeup or sprays, the use of identical tank of the Unit №1 (which unit is not in operation) is foreseen, the boron concentration and temperature being the same as in the BWST;
- No other source of water can be used inside the containment except of clean water used for the compensation of water mass during normal operation. This makeup has a low capacity and the operators are instructed to add boric acid solution to this water;
- There is no difference between the injection and the recirculation phase, the collected water is drained back to the BWST, so its temperature will slightly increase, which increases the margin to criticality;
- The possibility of fast decrease of the coolant temperature injected to the core is prevented by the limited capacity of containment heat removal.

6.3.1.5 PART V: Prevention of basemat melt-through

At present, there is no specific strategy in place for preventing basemat melt-through (BMT) at ANPP Unit №2 in case of a severe accident. It has to be noticed that the cavity is surrounded by soil and BMT would not lead to direct activity release to the atmosphere. A sound strategy should be developed and discussed for ANPP Unit №2 to be integrated into the whole plant strategy as represented in the future set of SAMG.

However, due to the current plant configuration, it seems already clear that only a few alternatives exist. Hereunder two alternative recommendations are proposed and discussed, that should require smaller to medium plant upgrade, they can be considered as additional measures to enhance containment integrity:

- Small – installing a movable concrete block that would close the access shaft from the cavity bottom and prevent early corium penetration of the thin wall of the access shaft;
- Medium – selecting the strategy of in-vessel retention by external cooling and prevent Molten Core/Concrete Interactions (MCCI).

When choosing the first upgrade, it should be also supported by SAMG proposing flooding the cavity by existing means (primary makeup system after vessel failure), which would slightly slow down the corium penetration and strongly support the retention of fission products from MCCI in the cavity. The first upgrade would already postpone the BMT to several days after core damage, because the total thickness of the concrete cavity bottom is 3.1 m and that of the wall 2.49 m.

The second upgrade (fully removing BMT) is suitable for smaller power reactors like ANPP 2 and it is foreseen in the list of planned plant upgrades. It is also installed on both units of the Loviisa VVER-440 NPP. It would require provision for removal of the shielding rings at the bottom of the vessel shortly after core damage (hydraulic device in Loviisa) to enable good access of water to this part of the vessel with strong thermal load by the debris. It would also require changing the water drainage scheme so that cavity can be fully flooded on demand without challenging future use of primary makeup (this should be possible to fulfill as the cavity volume is only 170 m^3). It also requires a heat removal strategy to prevent cavity water from boiling off (sprays for containment heat removal can be used).

ANPP Unit №2 supports the second upgrade and proposes to start a feasibility study.

6.3.1.6 PART VI: Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

Prevention of re-criticality and containment pressurization by gases is fully passive and does not need any electric system.

As mentioned above there is presently no strategy to cope with hydrogen deflagration / detonation and basemat melt-through.

Performing SAMG is mostly dependent on DC power; the autonomy of batteries is about 7 to 8 hours, which is insufficient. Proposal to develop and implement measures to recharge the batteries is included in chapter 5; it would be also needed after core damage.

Some of the additional measures proposed would need additional equipment with power supply, but this could be best solved by additional sources:

- Prevention of fuel damage / meltdown at high pressure – pressurizer PORV (or safety valves) have a dedicated source of DC from a battery that can last 24 hours;
- Early hydrogen (released from core before vessel failure): To implement recommendation concerning spray system interlock which will reduce the risk of failure of the confinement liner or the ALS due to subatmospheric pressure and to reduce oxygen leakages from environment additional measures should be proposed for supply power to spray system or alternative spray system using diesel engines as independent source., igniters using additional battery (the requirement would be about 4 hours of work at most to cover the early phase of the accident), hydrogen measurement – small additional requirement for DC power or sharing the battery with igniters;
- Late hydrogen (released from MCCI - after vessel failure -, or hydrogen that remained in the atmosphere from the early phase): PARs which do not require any source of energy;
- Base mat melt-through:

- Solution based on protecting concrete block: no energy needed during the accident, the block would be positioned after the inspection of the cavity;
- Solution based on external vessel cooling: very small amount of DC power would be needed to flood the cavity and remove the shielding ring (the removal itself can use gravity), then existing sprays using AC power can continue, for an alternative spray system, variant without recirculation is also viable, but heat removal from the containment has to be recovered after some hours.

The equipment used for protecting containment integrity does not use compressed air.

6.3.1.7 Measuring and control instrumentation needed for protecting containment

See chapter 6.3.4.

6.3.1.8 Conclusion on the adequacy of severe accident management systems for maintaining the confinement capabilities

See chapter 6.3.3.

6.3.1.9 Measures which can be envisaged to enhance capability to maintain confinement capabilities after occurrence of severe fuel damage

See chapter 6.3.

6.3.1.10 Identification of any cliff edge effects, and evaluation of the time before they occur for protecting containment integrity after occurrence of fuel damage (all aspects)

Several cliff-edge effects in relation with the protection of containment integrity that can have a strong impact especially on activity release towards the environment during a severe accident have been identified in the ANNP VVER-440/270 design. The first of them is the possibility to trigger the containment pressure relief valves opening. It was designed to protect the containment against overpressure failure during a design basis accident. The first valve opens at 0.75 bar overpressure, the remaining valves at 0.8 bar. It may open any time when the containment pressure reaches this level. The opening is less probable after core damage because the steam source from the primary system is reduced. The valves may, however, open due to deflagration of hydrogen coming from Zr-steam reaction, this risk being not so crucial because of the short duration (several seconds) of this phenomenon. Deflagration might hypothetically also cause another cliff-edge effect due to improper closing of the valves after the deflagration. This possibility should be analyzed as the valves have not been designed for such steep pressure increase. Later, short (but longer than those to hydrogen deflagration) openings can be caused by sudden steam source during the core degradation. Openings due to sudden steam sources can be probably eliminated or at least reduced by sprays.

The second strong cliff-edge effect is fast hydrogen deflagration or its transition to detonation that may fail the containment. According to SBO calculation, this might happen already during the Zr-steam reaction about 1 hour after the onset of core damage if spray system is operable. Developing SAMG for hydrogen together with technical solution – installing hydrogen concentration measurement and igniters – are proposed in chapter 6.3.1 to reduce this risk.

Similar cliff edge effects can occur later, several hours after the vessel bottom head failure. Vessel bottom head failure would occur about 4 hours after the onset of core damage. If sprays can be kept in operation, opening the containment pressure relief due to pressure of gases would occur after many hours and would not have the character of a cliff-edge effect on fission products release. This is because the basaltic cavity concrete leads to only small sources of non-condensable gases, i.e. carbon monoxide and carbon dioxide. However if sprays are not working, this late opening may be considered as a cliff-edge effect on radioactivity release.

Also the possibility of a cliff-edge effect due to hydrogen deflagration or detonation is much smaller in the late phase after the vessel bottom head failure in case of spray system inoperability, because of oxygen-lean atmosphere and pressure in confinement will stay above atmospheric pressure. In addition, SBO calculation has shown that in case of failure of confinement flaps to open and inoperability spray system or in case of available spray system deflagration or detonation risk in the late phase after the vessel bottom head failure is much higher due to presence of oxygen. Using PARs (e.g. of small capacity) and spray system automatic shut-off possibility is proposed in chapter 6.3.1 to avoid hydrogen deflagration by limiting air ingression from outside in case of containment reaching under pressure.

A typical cliff-edge effect during the phase after vessel bottom head failure is the cavity basemat melt-through or radial penetration by debris. The consequences of the cavity basemat melt-through or radial penetration are reduced by the fact that it does not lead to direct radioactivity release to the atmosphere because the whole cavity is buried in the sandy soil. At present, the cavity wall is much weakened close to the personnel access shaft so that a cliff edge effect at this place could happen shortly after vessel failure. Technical solution is proposed in chapter 6.3.1 to eliminate this risk.
The basemat melt-through and the late hydrogen risk can be completely eliminated by using the strategy of debris retention in the vessel by external vessel cooling as proposed in chapter 6.3.1. Because it would also require restoring containment heat removal, it would also reduce any cliff-edge effect due to containment pressure relief opening except during several hours for which the concept may be used without heat removal. Some technical changes (especially those granting free water access to the vessel surface and water delivery to the cavity) and SAMG adaptations would be needed.

6.3.2 Assessment of the adequacy of the existing management measures for protecting containment integrity after occurrence of fuel damage (all aspects), including the procedural guidance to cope with a severe accident

The guideline on beyond the design basis accident management is presently applied even after occurrence of fuel damage. The assessment of the adequacy of the procedural guidance to cope with a severe accident for protecting containment integrity after occurrence of fuel damage is covered by chapter 6.2.3. The adequacy of WOG SAMG is also recognized for protecting the containment integrity. However, for ANPP the maximal design pressure of confinement is reached during 2F500mm LOCA and there is not existing means to cope with this pressure. For smaller LOCA existing analysis showed that existing system such as spray system and safety flaps are able to protect confinement integrity. In addition to avoid hydrogen deflagration/detonation installation of PARs and spray system automatic shut-off possibility was proposed.

It is recognized that the containment plant design makes difficult to prevent the release of fission products during a severe accident. Specific features of the containment impacting those releases are the following:

- High natural leak corresponding to several hundred percent / day starting at the design pressure;
- Containment pressure relief valves to keep the integrity of the containment during large LOCA (pressurizer surge line break).

Concerning the first item, the plant is aware of the situation and measures to reduce this leak to about 100%/day are already proposed. Technical details have been elaborated by specialists who managed even higher reduction of leak for a plant of similar design (VVER-440/230 in Jaslovske Bohunice, Slovakia).

Regarding containment pressure relief opening in the phase of severe accident, this can be prevented or at least limited by using containment sprays. The improvement of the existing spray system is underway. The installation of an alternative spray system that would use independent sources of water and energy (e.g. mobile pump) will be assessed, as sprays are crucial to both reduce the radioactivity release to the environment and manage the risk for hydrogen explosion (see chapter 6.3.1).

Other recommendations have already been made in chapter 6.3.1 and 6.3.2. These should be analyzed independently and also by combining them in order to verify the adequacy of those potential measures that could be implemented as part of the whole severe accident management strategy and as such reflected into future SAMG developed for the ANPP 2 (see chapter 6.2.3).

6.3.3 Evaluation of the potential for additional measures for protecting containment integrity after occurrence of fuel damage (all aspects). Part I: Suitability and availability of the required instrumentation

The suitability and availability of the required instrumentation for protecting containment integrity after occurrence of core damage was estimated from chapter 6.2.4. There is no hydrogen measurement in the atmosphere that would be needed if the active way of its removal (igniters) is selected.

6.3.4 Evaluation of the potential for additional measures for protecting containment integrity after occurrence of fuel damage (all aspects). Part II: Availability and habitability of the control room

The availability and habitability of the control room for protecting containment integrity after occurrence of core damage is covered by the evaluation performed in chapter 6.2.5.

6.3.5 Evaluation of the potential for additional measures for protecting containment integrity after occurrence of fuel damage (all aspects). Part III: Evaluation of potential H2 accumulations in other buildings than containment

Because of large containment natural leak there is a high probability that containment atmosphere will leak outside together with hydrogen contained in it. After implementation of some measures to mitigate the hydrogen risk inside the containment – installing igniters and Passive Autocatalytic Recombiners (PARs) as proposed in chapter 6.3.1 - the possibility of hydrogen release during the early phase will much decrease. It will be not suppressed completely because the about 350 kg of hydrogen (corresponding to about 41% of core Zr oxidation) is enough to consume all oxygen and any additional hydrogen will accumulate inside the containment. Small additional amount of hydrogen consumed by oxygen ingression to the containment, which could be in underpressure caused by previous oxygen consumption and sprays operation, is not considered.

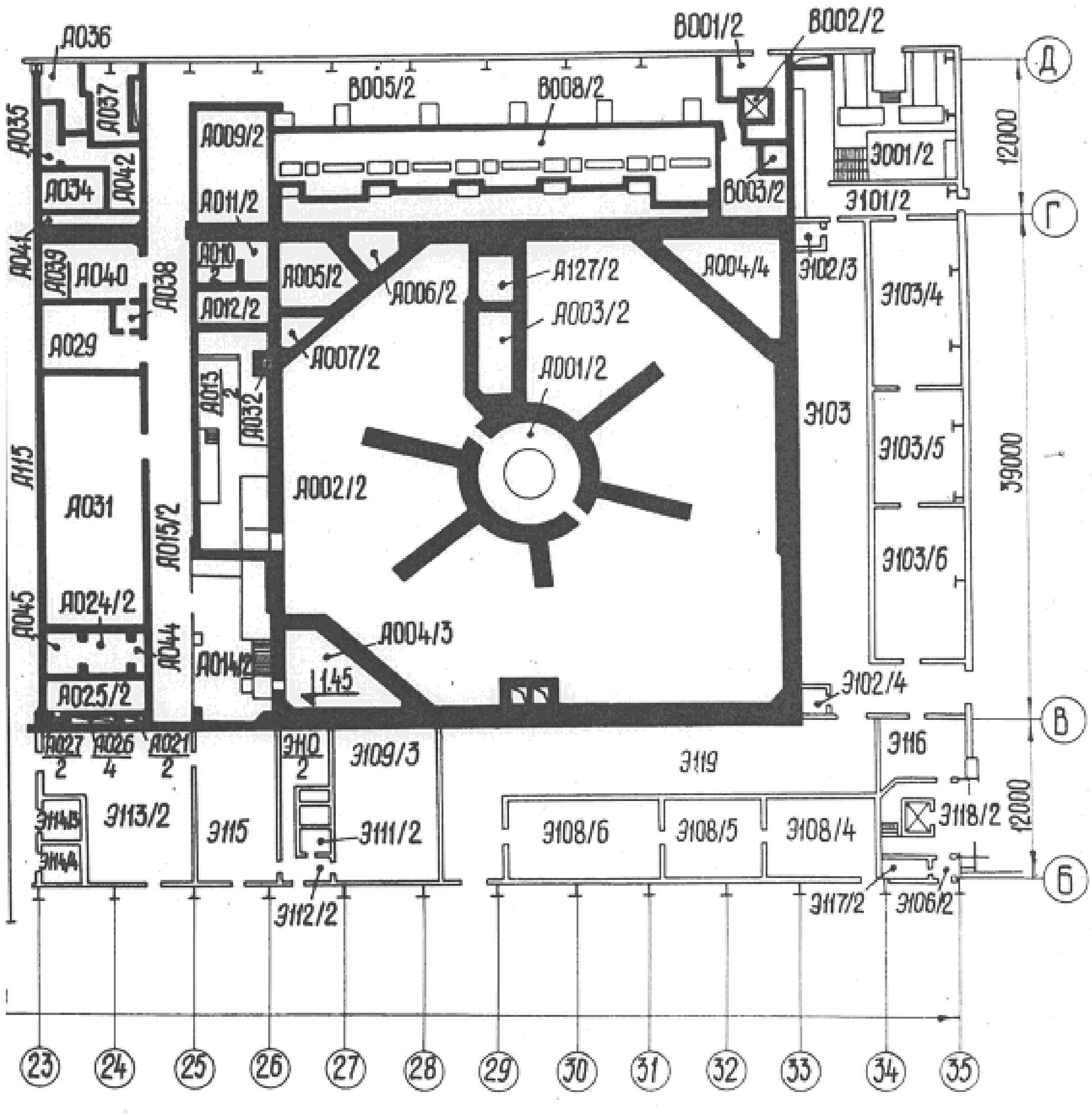
Much of the hydrogen coming in the containment during the early and the late phases of the severe accident will be released outside. Hydrogen is released mainly directly towards the environment via the containment pressure relief valves, which therefore prevails too much accumulation of hydrogen that can enhance leakages to the neighboring buildings.

The other main leaking path is probably the one to the reactor hall because of several material hatches in the floor of the hall (ceiling of the containment). The reactor hall volume is very large, about 100000m^3, so the accumulation would require a large quantity of hydrogen. Three hundred kg of hydrogen represent only 4% concentration, which is just at the limit of a possible ignition, while the hydrogen leak to the reactor hall will be probably much smaller. This estimate is based on the assumption of no stratification.

There is still a last possibility of hydrogen leak through the four side walls. There exist smaller rooms at the sides and between the two units, as can be seen from the figure hereafter, at the level close to the bottom of steam generator rooms. The thick walls are the containment boundary. Accumulation in some of these rooms cannot be excluded; the other rooms are open to larger rooms which are open to the environment.

Preference should be given, however, to the solution of in-containment hydrogen risk management described in chapter 6.3.1. This will automatically decrease the risk of hydrogen accumulation in other buildings than containment because of the consumption of the main quantity of hydrogen as soon as it is released in- containment. Also reducing the natural leak can be beneficial for avoiding hydrogen accumulation outside the containment.

It is therefore needed to assess more in details the risk of hydrogen accumulation outside the containment before to envisage any strategy or measure (e.g. improved venting of some rooms or installing PARs). PARs could be an effective solution in such circumstance as the accumulation of hydrogen that could occur in other buildings than containment would be slow.



6.4 Accident management measures to restrict the radioactive releases

6.4.1 Radioactive releases after loss of confinement capabilities

6.4.1.1 Design provisions

Uncontrolled fission products releases from ANPP Unit №2 are the real threat for public health and safety. Depending on the course of an accident, activity can be released into:

- Containment (hermetic rooms);
- Second circuit (through SGs);
- Environment after loss of containment integrity and
- Non-hermetic rooms.

Containment natural leakage rate is very high (effective leakage flow path area is about $0.02m^2$). This corresponds to containment failure of the leak type of a typical PWR containment. In order to reduce the fission product threat to the environment:

a) The natural leakage should be reduced to an equivalent of 60% volume/day.
b) Sprays should be available with very high probability to reduce the pressure in the confinement.
c) Appropriate steps in hydrogen management should be taken. Because of small containment volume and the use of sprays, recombiners would be insufficient for the in-vessel hydrogen production and igniters should be considered.

6.4.1.2 Operational provisions

No SAMG system including some procedures for FP releases reduction is implemented at the ANPP 2 at present nevertheless there are operator’s actions (procedures, measures) which are able to restrict the radioactive releases:

- To close main isolation valve on damaged loop in the case of FP leakage through damaged SG;
- To start containment spray system is the most effective measure to restrict FP release in the case of damaged or non-tight containment.

These two operator actions should be also foreseen in the SAMG being developed. Besides, the use of igniters will probably need other SAMG action(s) to be defined.

6.4.1.3 Hydrogen management

During the first day of the SBO accident about 425 kg of hydrogen was produced due to oxidation of Zr and 400 kg due to MCCI. Although the inflammable gas concentration high enough to start hydrogen combustion, no burns were identified in the hermetic rooms during the whole scenario due to inoperability of spray system. Inertization of atmosphere by steam and significant leakage of hydrogen due to very high operational leakage rate of the containment are reasons of it.

The containment sprays switching on could change the fractions of reactive gases in the atmosphere and start hydrogen burns or even transition to detonation.

6.4.2 Accident management after loss of cooling of the spent fuel pool in the long term

6.4.2.1 Hydrogen management (not limited in this situation)

Spent Fuel Pools (SFP) are located in the reactor hall which is common for both reactor units. The analysis of spent fuel pool behavior after loss of SFP cooling was performed. Two cases were calculated for SFP of Unit 2:

1. Standard state of the SFP with 349 spent fuel assemblies.
2. In addition, all fuel assemblies from active core are moved into SFP.

Time margin up to fuel damage was determined: 104 hours and 33 hours for the case 1 and 2 respectively. The SFP can be also cooled with boron acid solution taken from the B-8 tank by means of the NBO-1,2 pumps. The projects have been worked out to supply water into the SFP from the fire system or from the diesel driven SG feedwater pumps. The latter one is able to supply water into the SFP even during station blackout.

Subcriticality of the fuel in the SFP is ensured by boron concentration in the cooling water and use of borated steel for inner equipment in the SFP namely storage lattices.

Because the SFPs are located in non-hermetic rooms of the reactor building, severe damage of fuel would result in massive FP release into the reactor building and then into environment.

In the course of core and fuel degradation gaseous hydrogen is produced by zirconium-steam reaction in case of boil off scenarios. Because of large reactor hall volume (about 100000 m3), the hydrogen does not present an immediate risk, its concentration remains below the ignition limit if stratification is not assumed.

The SFP Unit №1 stores fuel assemblies at least 5 years after shutdown. Time till fuel damage was not analyzed, but significantly longer time is expected in comparison with the SFP Unit 2, i.e. many days. This enables even sharing of some of the resources with SFP Unit 2, but always respecting the assumption that larger help to the plant from outside (equipment, source of water) cannot be expected within 72 hours.

To enhance the safety of SFP Unit №2 in case of a total loss of offsite power and to avoid a cliff-edge effect of SFP fuel damage and radioactivity release, the possibility for the SFP Unit №2 to use cross-link (resources sharing) with SFP Unit №1 where the time window before to reach that cliff edge is much larger. This is described in chapter 5 where following measures are proposed: a) the possibility to use a mobile diesel pump for SFP makeup b) and c) using boric acid solution or diesel driven second makeup pump.

6.4.2.2 Providing adequate shielding against radiation

Damage of fuel in SFP would probably result in contamination of reactor hall common for both of the units. The FPs are released from damaged (degraded) fuel located in the SFP. In such situation, all staff (personnel) should leave the reactor hall. In the case of regimes when the containment is connected with reactor hall, during some regimes of outage, the containment can be contaminated as well.

Main control room and back-up control panel are situated in rooms adjacent to the reactor hall. This part could be exposed to radiation in the case of high radiation level in the reactor hall.

In the initial phase of loss of the SFP cooling up to the start of water boiling and drop of water level the reactor hall is accessible. Drop of water level in the SFP and start of water boiling would result in the growth of radiation dose in the vicinity of the SFP and its surroundings would not be accessible.

The activity could spread in whole reactor hall including the part corresponding to adjacent unit so that the all personnel should leave the reactor hall.

Even partial damage of the fuel would result in contamination of whole reactor hall. In this situation the reactor hall is not accessible. During the regimes with refueling or reactor unload the reactor hall could be connected with the containment therefore the containment is also contaminated.

The habitability of both of the control rooms can be also indirectly influenced by:

- activity released from the reactor hall and sucked by control room ventilation system or
- direct radiation from adjacent containment room (pressurizer room).

6.4.2.3 Instrumentation needed to monitor the spent fuel state and to manage the accident

Following instrumentation is needed to monitor and manage SFP state:

- Water temperature measurement,
- Water level measurement and
- Measurement of mass flow rate in the SFP cooling and make up system.

Reading of above mentioned measured parameters is available only in main control room.

Because of large volume of reactor hall, its low tightness, and low decay heat power it is assumed that the majority of measurements will be accessible. Measurement of radioactivity level in the reactor hall atmosphere is very important.

6.4.3 Conclusion on the adequacy of measures to restrict the radioactive releases

There are not any SAM strategies and SAMG implemented at the ANPP Unit №2. The implementation of SAMG is under way (see chapter 6.2.3). Leakage rate from containment is very high. That's why the measures to restrict radioactive releases into environment are limited. One of the possible effective measures is to start containment spray system which can:

- reduce containment pressure, and
- wash out radioactive aerosol particles from containment atmosphere into the sump.

Present measures to restrict the radioactive releases are not adequate. Because of high importance of the spray system, adding alternative sprays with independent source of energy (using a diesel driven spray pump) and water (borated water storage tank of Unit №1) should be foreseen.

6.4.4 Measures envisaged to enhance capability to restrict the radioactive releases

Several measures should be applied to enhance capability to restrict the radioactive releases.

The most effective measures are related to the prevention of the accident to progress into the severe one with the core damage (melting) and FP release. Such measures are related mainly to preventive ones, but are also a part of severe accident measures (e.g. injection to the RCS). In any time the reduction of releases from fuel is a key factor, so alternative core cooling system with independent power supply and water sources would significantly influence FP releases from core. Such alternative cooling system would be functional for medium and low pressure sequences. Effective and fast primary circuit depressurization is needed in case of high pressure sequences to allow water injection and restoration of decay heat removal (see also chapter 6.3.1.1). These measures would be effective mainly for early phase of core degradation.

In case of the severe accident progression into late phase, it is open issue what approach to be applied to melt cooling. Generally for the VVER-440 reactors, the in-vessel retention approach is implemented (Finland) or is to be implemented (Hungary, Slovak Republic, and Czech Republic). Such approach relying on gravity cavity flooding cannot be used for ANPP Unit №2 where sufficient amount of water exists in borated water storage tanks, but they are located too low. So the application of IVR approach would need to solve the open question of the water transport to the cavity (see also chapter 6.3.1.5). The second possibility is to start cooling during the MCCI, but such approach is effective only if melt is spread on sufficient area. Melt spreading is not possible at the ANPP Unit №2 due to the design provisions of the containment, so the IVR approach seems to be more realistic for the application. Appropriate technical solution is not in scope of this report.

The natural leakage of the containment is very high and capability of such containment to retain FP inside of the containment is unacceptably low. The reduction of natural leakage is the key measure. Because of low volume of the containment, decreasing the natural leak is insufficient; more leaktight containment will be pressurized during the course of the severe accident, mainly in case of spray system failure. Containment pressure reduction can be solved with several approaches:

- Filtered ventilation, but due to relatively low design pressure, such system has to be active with dedicated source of power;
- Alternative spray system with water supplied from outside or from the borated water storage tank of Unit №1 – the analytical verification is necessary especially for the case when this spray system would work in injection only.

The second solution is better, it is much more efficient. The first solution would be possible as an extension of the existing non-filtered venting, but then it would not mitigate the problem of large natural leak. In PHARE 94 project 2.06 the possibility to counterbalance high natural leak by filtered venting was studied and rejected because it would need vacuum pumps of large capacity and large filtration building. Sprays have been found much more efficient.

In case of enhanced leaktightness of the containment, hydrogen produced during the core degradation will be accumulated in the containment. It is obvious that some solution of the hydrogen issue is necessary, because in case of spray operation the probability of hydrogen deflagration and/or more dangerous regimes of deflagration like flame acceleration and transition to detonation must be eliminated. This is discussed in chapter 6.3.1.2.

Above proposed measures related to the accident initiated during reactor operation and have to be taken into account during the preparation of the SAMG to coordinate and optimize their functionality.

Procedures and guidelines to manage accidents with fuel melting in the SFP are not available so far. Nevertheless appropriate measures are well known: to restore water make up into the SFP and to ensure decay heat removal. Future revision of generic WOG SAMG will include SFP guidance. Such a SFP guidance should be implemented in future revision of ANPP Unit №2 SAMG.

There are sufficient time margins to restore the SFP cooling (see chapter 6.4.2). In the worst case (all fuel assemblies from the core are moved into the SFP during outage) the time margin is about 33 hours.

Large volume of the reactor hall results in effective dilution of FP. Reactor hall ventilation system should be immediately switched off to reduce the FP leakage in case of severe accidents initiated in the SFP and/or in reactor during the outage with opened reactor.

Hydrogen generation in case of accident initiated in SFP has to be analyzed taking into account also the case of refueling when the reactor is open directly to the reactor hall.

7. GENERAL CONCLUSION

This chapter concludes the results of assessment and analyses presented in previous chapters with focus on already implemented safety measures and further potential improvements of safety.

7.1. Key provisions enhancing robustness (already implemented)

In course of its operation, Armenian NPP was in process of systematic safety assessments and upgrades. Particularly, substantial safety upgrades were implanted in early 80's to increase robustness of ANPP with regard to station blackout. Before restart in 1995, thorough safety assessment and equipment qualification were performed that resulted to implementation of key safety upgrades in accordance with new Armenian legislation, IAEA safety standards and best international practice. In 2011 Comprehensive safety assessment of ANPP was carried out that led to development and prioritization of further safety upgrades. Since 1993, Armenian NPP hosted a number of international missions in different technical areas performing independent review of its safety.

The VVER 440 reactors have a number of inherent safety features favorable for plant recovery from events. These features include low core power density, plant layout with six loops isolable by valves on each loop and two turbines reducing impact of several transients. Use of horizontal steam generators facilitate transition to natural circulation in the primary circuit, large water inventory in the primary circuit provide appropriate time margins for plant operators.

RA voluntarily joined to EC Stress Test initiative. Within this initiative, ANPP carried out safety self-assessment of Unit №2 and of wet spent fuel pools of Unit №1 and №2 in accordance with the ENSREG specifications.

7.1.1. Earthquakes

Following the decision of the Armenian Government on the ANPP Unit №2 restart (1993) comprehensive seismic re-evaluation of ANPP was performed, including:

- Assessment of geological stability of the site, demonstrating i.e., lack of an active fault, which can cause permanent displacement of ground under or in the vicinity of buildings and structures important for seismic safety.
- Specification of intensity of seismic ground motion, i.e., parameters associated with seismic design (such as maximum ground acceleration, ground response spectrum, duration, historical accelerograms, etc.) needed for reassessment of the plant buildings and components.
- Development of complete reassessment program for structures, systems and components important for safety in regard to new data and advanced special methods and criteria recognized in international practice, and implementation of improvements, if needed.

- Compiled list of SSC's required for design basis earthquake (Maximal Design Earthquake for 10 000 years return period).
- Procedure for the Reactor Safe Shutdown under seismic impact was developed aimed at development of SSEL. The procedure describes and suggests 3 options of safe shutdown: hot, semi cold, and cold. In addition, it describes safety functions to be implemented for ensuring criterion of 72-hour safe shutdown period.
- Seismic qualification of ANPP Buildings and Structures was confirmed by analyses and calculations.
- Protection against indirect effects of the earthquake seismic interaction issues such as: Internal flooding, loss of external power supply, situation of preventing access of plant personnel, fire or explosion.

Seismic Margin Assessment for ANPP SSEL components shows that:
- No cliff edge effects were identified for the items included in SSEL.
- The minimum HCLPF value defined among all SSEL components is 0,38g (Sprinkler Pumps located in Boron compartment). However, it should be noted, that the Sprinkler System, although was included in SSEL, is not directly involved in the Safe Shutdown procedure and, thus, its failure may not be an initiating event for a severe fuel damage.
- Presented HCLPF values were obtained from fairly conservative estimates, and according to applied probabilistic procedure they mean the 50th percentile of 1% probability of failure of considered element or system under design seismic event with a probability of the annual exceeding 10^{-4}. At the same time, events considered in the process of NPP Stress Tests are beyond design basis external events and have a much lower probability of occurrence ($10^{-5} \div 10^{-6}$). In accordance with ASCE 43-05 for the beyond design basis events a Target Performance Goal is less than about a 10% probability of unacceptable performance, that means that corresponding seismic margin for ANPP is about 1.5 times higher than HCLPF values defined for RLE. In terms of PGA this value is 1,5*0,38 = 0,57g, that is more than 20% above the reference seismic level set in the procedure of the NPP Stress Tests.
- Thus, as it follows from the said above, ANPP has a sufficient seismic margin for the beyond design basis earthquakes with PGA = 0,47g.

7.1.2. Flooding

Plant robustness against flooding hazard was initially assured by appropriate site selection. ANPP site is selected in the way that the water basins existing near the plant site are more than 50 m lower than the absolute ground elevation of the plant. Investigation shows that the site characteristics allow eliminating hazard coming from ground water. The plant site was found to be also protected against mudflows by the site fence concrete wall (about 2 m high). However it was noted that there is still a lack of detailed calculations aimed to prove sufficiency of site fence concrete wall resistant against mudflows.

The rainfalls are considered as only possible flooding hazard for ANPP. Rainfall water removal from the plant site is organized by special sewer system which consists of network of underground piping. Analysis performed within plant specific PSA study showed that sewer system is capable to prevent site flooding even in case of rainfall with intensity higher

than design value by factor of 4.4. It is expected that sewer system has even bigger safety margin, however the analysis of rainfall with higher intensity was not yet performed.
In addition special flood protection means were analyzed individually for different plant buildings. As a result of the analysis it was concluded that the site flooding has extremely low probability and available flood protection measures are considered to be appropriate. However, additional provisions were recommended in order to further increase robustness of the plant against possible flooding.

7.1.3. Extreme meteorological conditions (other than extreme precipitation)

Analysis was performed for possible impact of different type of external hazards at ANPP. In addition special effort was put on systematic analysis and identification of critical combinations of hazards for ANPP. In order to address this issue ANRA's technical support organization (NRSC) together with Armenian NPP launched a co-operation project with IAEA aimed to apply fault sequence analysis (FSA) method developed by IAEA. The FSA method aimed to identify critical combinations of hazards based on thorough analysis of minimal cut-sets provided by PSA and plant equipment operability limits.

Performed investigation allowed concluding that impact of considerable[10] external hazards as well as their combinations are not directly leading to the core damage. It was found that most of the external hazards could lead to loss of offsite power or station black-out conditions which could be resolved using existing systems and procedures. Critical systems and safety functions challenged by external hazards were identified. Particularly it was noticed that diesel-driven pump system, DGs and DG auxiliary systems (HVAC, fuel supply etc.) have critical safety functions used in case of large variety of external hazards.

7.1.4. Loss of electric power and loss of ultimate heat sink

ANPP has a robust design with regard to the risk of loss of electric power. At ANPP there are 3 different options (with different vulnerability to external hazards) for providing power supply to plant home consumers (in addition to their redundancies); 2 of these options are independent on the electricity distribution grid. These various options can be activated either automatically or by plant staff. There is back-up power source capable to provide power supply for unlimited period of time by small hydropower station (50 MW) using a direct 110 kV transmission line. Recovery power from “Gumush” hydro power station can be supplied within 3-5 minutes regardless of availability of ANPP communication with the central dispatcher service of the RA power network.

Independent from the grid internal power sources are emergency power supply system and alternate emergency power supply system with fuel reserves for 3 days. In addition to 3-day fuel supply, there is one emergency diesel fuel tank at ANPP site. The emergency diesel fuel tank holds enough fuel for 27.7 days emergency power supply system operation. However,

[10] The impact of external hazards with extremely low frequency (<<1E-07) of occurrence was not taken into account

emergency diesel fuel tank is not seismically verified and the refilling pump is not powered from reliable power supply.

In case of loss of off-site power, ordinary back-up AC power sources and the independent alternate emergency power system, at the ANPP site still are available batteries and the auxiliary steam-generator feedwater system which is equipped with one diesel pump. The reserve of the diesel fuel for the diesel pump is sufficient for 24 hours of operation without refilling of fuel tanks. Therefore, even with blocked possibility of fuel delivery heat removal from the secondary side and corresponding core cooling will be possible during 24 hours. After fuel delivery is possible, auxiliary feedwater system is able to remove heat from the secondary side during 7 days using water amount of demineralized emergency water tanks installed at the site.

Time margin to core uncover in case of SBO occurring at full power, using only coolant inventory available in primary and secondary circuits is about 9 hours. For shutdown regimes, this time interval is extended at least to 18.5 hours. For loss of heat removal from the spent fuel pool, time margin up to fuel uncover without any operator actions are more than 26.7 hours for the most conservative case with complete off-loading of the core into the pool, or more than 70.1 hours for more realistic situations (for partial core unload).
The primary ultimate heat sink for Armenian NPP is the atmosphere. The alternate method of heat removal is steam dumping to the atmosphere. Therefore, as a loss of the primary ultimate heat sink in current assessment is considered the situation with unavailability of transport of heat to the UHS. If normal plant cooling through the secondary circuit and cooling towers is not available, remaining options include direct release of steam from steam generators to atmosphere through the steam dump system, or by primary circuit feed and bleed, or by heat removal through the essential service water system.

In the 90ies, after re-start an independent Essential Service Water Systems as ultimate heat sink with cooling ponds was constructed and implemented. By that a separation in cooling of safety and of operational systems was performed. ESWS make up is provided from condenser cooling channels and from independent source of water. Since failure of essential service water system could have serious consequences regarding heat removal from the core, from the spent fuel pool and from the containment, this case was analyzed in detail in the stress tests as the most conservative one. Results of assessment showed that in case of essential service water system loss, the amount of cooling water reserve at ANPP is sufficient for heat removal for about 16 day.

In case of loss of raw water supply, results of assessment showed that cooling water reserve at ANPP is sufficient for heat removal for about 19 day.
The case of a combined station black-out and loss of ultimate heat sink at ANPP is in fact covered by the station black-out only, since the station black-out is always accompanied with the loss of ultimate heat sink.

7.1.5. Severe accident management

Severe Accident Management Guidance for the Unit 2is currently developed by ANPP. ANPP employed general WESTINGAUSE Owner Group (WOG) methodology for SAMG development.

The most effective measures are related to the prevention of the accident to progress into the severe one with the core damage (melting) and FP release. In any time the reduction of releases from fuel is a key factor, so alternative core cooling system with independent power supply and water sources would significantly influence FP releases from core. Effective and fast primary circuit depressurization is needed in case of high pressure sequences to allow water injection and restoration of decay heat removal. These measures would be effective mainly for early phase of core degradation. Application of IVR approach must be studied.

The reduction of natural leakage is the key measure. Because of low volume of the containment, decreasing the natural leak is insufficient; more leaktight containment will be pressurized during the course of the severe accident, mainly in case of spray system failure. Containment pressure reduction can be solved with alternative spray system with water supplied from the borated water storage tank of Unit №1.

In case of enhanced leaktightness of the containment, hydrogen produced during the core degradation will be accumulated in the containment. A solution of the hydrogen issue is necessary, because in case of spray operation the probability of hydrogen deflagration and/or more dangerous regimes of deflagration like flame acceleration and transition to detonation must be eliminated.

Above proposed measures related to the accident initiated during reactor operation and have to be taken into account during the preparation of the SAMG to coordinate and optimize their functionality. Future revision of generic WOG SAMG will include SFP guidance. There are sufficient time margins to restore the SFP cooling. In the worst case (all fuel assemblies from the core are moved into the SFP during outage) the time margin is about 33 hours. Hydrogen generation in case of accident initiated in SFP has to be analyzed taking into account also the case of refueling when the reactor is open directly to the reactor hall.

7.2. Potential safety improvements and further work forecasted

The potential further improvements for safety and robustness of the ANPP against external phenomena are summarized below separately for different areas of assessment.

7.2.1. Earthquakes

Robustness of the plant against earthquakes has been significantly increased recently and it is considered to be adequate in accordance with the current requirements. Nevertheless, the following measures for further improvements are envisaged:

1. Assessment of ANPP seismic margins is based primarily on the deterministic CDFM analyses performed in the framework of SMA procedure. Application of combination of deterministic and probabilistic safety assessment will help to continue further seismic enhancement and identify systems and components that may require further seismic improvement.
2. Installation of additional fuel tank with capacity of ≅50÷100 tons to provide an emergency power supply for period of 72 hours.
3. Seismic margin evaluation of fire extinguishing system and implementation of measures to reinforce the system.
4. Analysis of impact of explosion of nitrogen recipients and hydrogen storage tanks.
5. Investigation of possible consequences in case of seismic induced flooding in Turbine Hall (TH), impact of safety-related systems in TH, interaction with adjacent compartments.
6. Completion of the program for seismic upgrading of I&C equipment and seismic monitoring system.

7.2.2. Flooding

In spite of the extremely low probability of site flooding and measures already available, additional provisions are considered to further increase safety level of the plants as follows:

1. During the analysis, it was found that rainfall water could penetrate to the basement area of DGS through the external non-waterproof emergency door at the level of basement. The emergency door is always closed and located significantly below the building blind area in the cavity where theoretically water can penetrate, accumulate and then overflow through the non-waterproof door.
2. Analysis shows that the majority of TH is not vulnerable for flooding. However the water still can penetrate to the turbine hall through the doors at the ground floor in conditions of accumulation of water in surrounding area. In case of unlikely scenario with continues water accumulation it is possible to have failure of equipment located at the elevations ∇-3.6 to ∇0.0 (main feedwater system, emergency feedwater system, etc.).
3. Based on the performed analysis it was concluded that there is a possibility to have water penetration from TH to boron unit №1 and boron unit №2 which are located at elevation ∇-9.0. Despite the fact that both boron units are isolated by doors the water can principally penetrate into the boron units through leakiness of the non-waterproof emergency door that is located at the staircase connected to the turbine hall by another non-waterproof door.
4. Analysis shows that there is a possibility to have water penetration from TH to cable tunnels in case of failure of check valve on the line of “dirty” drainage system.
5. Investigation shows that there is a lack of detailed safety margin assessment in terms of rainfall flooding of ANPP site.

6. During the analysis it was shown that mudflows could have an impact on plant safety. Though there are special measures foreseen against mudflows, there is also the lack of detailed mudflow analysis.

Following measures are recommended for implementation based on analysis performed within Chapter 3:

1. In order to resolve the issue related to possible water penetration to DGS it is recommended to:
   - Equip the emergency doors of the staircases of DGS basement areas with a border in such a way that the penetration of water to the basement area can be excluded,
   - Foresee mobile equipment devoted to water pumping out from DGS and its basement,
   - Equip DGS with alarms indicating occurrence of water level in basement area with output of light signals in MCR, central control panel and DGS operator room,
   - Develop a procedure for operators for the case of water inflow in the DGS basement area.
2. In order to resolve the issue related to possible water penetration to turbine hall it is recommended to seal the penetrations to the turbine hall from surrounding area and develop and implement measures with the purpose of building barriers on the way of water flow to the turbine hall gates.
3. In order to resolve the issue related to water penetration from TH to boron units it is recommended to assure leaktightness of the doors located between TH and boron unit or to replace them by waterproof doors.
4. In order to prevent water penetration from TH to cable tunnels it is recommended to enhance the reliability of drainage system elements (e.g. dedicated check valve).
5. Perform detailed safety margin assessment in terms of rainfall flooding of ANPP site and calculations aimed to prove that protection measures are enough to prevent mudflows impact on ANPP systems.

7.2.3. Extreme meteorological conditions (other than extreme precipitation)

In spite of the safety measures already available, some findings were identified during self assessment of ANPP:

- Increased dust concentration in the air was found to be dangerous for DG air intake system. According to the analysis the critical level of dust concentration for diesel generator air intake system could be reached in case of high winds during dry weather conditions.
- Analysis shows that the collapse of reactor building roof could affect spent fuel pool during outage mode.
- Analysis of possible impact of extreme temperatures shows that pipes of demineralized water tanks (BZOV) could be vulnerable against low temperatures.

However, it was found that existing calculations of low temperatures impact on BZOV could be over-conservative.

- Analysis shows that diesel-driven pump system is critical in terms of protection against various external hazards, whereas the system is implemented as one channel design and could be vulnerable to different external hazards (i.e. high dust concentration in the air could lead to clogging of diesel pump exhaust pipe).
- Analysis shows high importance of DG auxiliary systems (e.g. HVAC) in case of external hazards. It was also noted that DG auxiliary systems could be vulnerable to different external hazards (e.g. low temperature)
- Investigation shows that there is a lack of detailed analysis for lightning impact on ANPP
- Review shows that there are several aspects of plant specific PSA model which should be revisited based on the results of performed analysis (external hazards screening process, external hazards combinations, etc.).

Following measures were recommended for implementation based on analysis performed within Chapter 4:

1. In order to resolve the issue related to dust hazard it is recommended to develop and implement measures aimed to protect DG from dust. Those measures could include improvement of DG compartments leaktighness and/or installation of special air filtering at DG air intake system.
2. In order to resolve the issue related to snow accumulation on turbine building roof it is recommended to foresee measures aimed to remove snow from TH building roof in case of snow accumulation.
3. In order to reduce conservatism in decision-making it is recommended to implement realistic re-evaluation of air temperature which could lead to freezing of outlet pipes of demineralized water tanks (BZOV). Depending on the results of realistic calculations the additional measures could be necessary for protection of BZOV pipes against extremely low temperatures.
4. In order to resolve the issue related to diesel driven SG feedwater pump it is recommended to check adequacy of diesel pump exhaust pipe protection against external hazards. Particularly it is recommended to re-analyse diesel pump exhaust pipe protection against high winds and dust.
5. In order to resolve the issue related to DG auxiliary systems (e.g. HVAC) it is recommended to implement measures aimed to protect DG auxiliary systems (e.g. HVAC) against external hazards (e.g. combination of seismic and low temperatures hazards).
6. It is recommended to implement detailed analysis for lightning impact on ANPP.
7. In order to assure completeness and quality of plant specific PSA model for external hazards it is recommended:
    - to re-visit external hazards screening process taking into account information reflected in stress-test report,

- to complement PSA by critical combinations of external hazards identified within FSA-ANPP project,
- to revisit hazard curves for different hazards taking into account updated meteorological data.

7.2.4. Loss of electric power and loss of ultimate heat sink

ANPP design has number of provisions aimed at prevention of the core damage in case of loss of electric power and loss of ultimate heat sink. All these provisions ensure time margins for recovery actions. Failures of these provisions represent the potential cliff-edges, which could result in irreversible core damage. All available options either for loss of power supply or loss of ultimate heat sink, or both are sufficiently described in chapters of this report. Despite the robustness of power plant design, following safety improvements are recommended:

1. Implement additional measures to assure longer operation time of batteries during SBO including:
   - Provision of mobile DGs for charging batteries during SBO,
   - Develop and implement additional measures to extend the operating time of reversible motor generators (ODGs) in an inverter mode that will lead to increase the time to provide I&C AC power supply,
   - Replace all reversible motor generators (ODGs) with modern inverters with less energy losses,
   - Implement a new electrical scheme for charging batteries from DAR system and/or the portable diesel generators.
2. Assure long term heat removal from Unit №1 and 2 SFPs in case of SBO:
   - Implement two new separate lines for make-up of the coolant inventory in SFP from a mobile source (e.g. fire pumps or diesel pumps) and external water sources for SFPs emergency cooling.
3. The emergency diesel fuel tank holds enough fuel for operation of alternate emergency power supply system during 27.7 days. However, emergency diesel fuel tank is not seismically verified and the refilling pump is not powered from reliable power supply. Increase the reliability of the refueling process or to foresee measures to install additional fuel capacity in terms of seismically qualified and reliable fuel tank (see recommendation №2 in 7.2.1 and №8 in 7.2.4).
4. For activation of Alternate emergency power supply system (DAR) manual actions are needed and activation can take up to 1 hour. Implement analysis of circuit diagram for consumers power supply (from DAR). Develop and implement activities aimed at minimizing personnel manual actions to activate the DAR system.
5. During the "cold" shutdown and refueling it is not foreseen DGLS program. Implement DGLS program for "cold" shutdown and refueling modes.

6. Existing EOPs do not contain appropriate procedures for all emergency states, which may arise in case of loss of the primary ultimate heat sink, combined with station black out.
7. In case of station blackout, cooling of MCPs seals is not ensured due to loss of the MCP seal water flow which, in long lasting situations, may evoke the RCS coolant leakage through the MCP. Assure MCP seals long-term (more than 24 hours) operation in case of cooling failure.
8. Installation of an additional tank for diesel fuel in the DG stations (for each train) in order to provide emergency power supply to trains I and II during 72 hours at full load, independently on the fuel inventory in the oil-fuel facilities of the plant (see recommendation №2 in 7.2.1 and №3 in 7.2.4).
9. Perform additional calculations in order to demonstrate sufficient effectiveness of the SGs emergency feedwater diesel pump for modes of reactor (cold shutdown, refueling, closed reactor, open reactor). Implement autonomous alternative means for make-up of SGs 1-6 of the Unit 2.
10. Provide for mobile pumps for ESWS make-up from Circulation Water Channel.
11. Develop and implement additional measures to use a large reserve of service water in the inlet and outlet channels, as an alternative heat sink.

7.2.5. Severe accident

The following recommendations are formulated for prevention of core melt, restriction of radioactive releases and protection of the confinement integrity:

1. Development of a full set of severe accidents management guidelines covering also SFP.
2. Modernization of Emergency Core Cooling System to ensure long time operation and reliable compensation of higher leak rate. Introduction of alternative low pressure core cooling system with independent power supply and water sources.
3. Comprehensive analysis of hydrogen generation and implementation of measures to reduce hydrogen exposure probability. Implementation of measurement of hydrogen concentration in containment.
4. Modernization of the Spray System including implementation of interlocks to reduce the risk of depth subatmospheric pressure and reduce oxygen inflow from outside. It is recommended to foresee measures to supply spray system components using mobile DG equipment. It is also recommended to implement feasibility study for adding alternative sprays with independent source of energy (using a diesel driven spray pump) and water (borated water storage tank of Unit №1) should be done.
5. Feasibility study and development of measures aimed at maintaining melting fuel inside RPV via external cooling of the reactor vessel.
6. Further improvement of containment tightness. A detailed analysis of possibility of hydrogen accumulation in rooms outside the containment.

End of the report