CJSC «HAEK»

UNITS 1-2

STRESS TEST SELF-ASSESSMENT REPORT OF THE ARMENIAN NPP

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ACRONYMS

AC	Alternate Current	-	электропитание переменного тока
AMCP	Main Circulation Pump Automatic System	АГЦН	Система автоматики ГЦН
ANPP	Armenian Nuclear Power Plant	ААЭС	армянская атомная электростанция
ANRA	Armenian Nuclear Regulatory Authority	АГАН	армянский государственный атомный надзор
ASCE	American Society of Civil Engineers	-	американское общество инженеров-строителей
ATS	Automatic Transfer Switch	АВР	Автоматическое включение резрва
BCC	Backup Crisis Centre	ЗЦРАС	Запасной Центр Реагирования на Аварийную Ситуацию
BDBE	Beyond Design Basis Earthquake	ЗЗУ	землетрясения запроектного уровня
BMMT	Basemat MeltThrough	СППО	сквозное прожигания плиты-основания
BWST	Borated Water Storage Tank	БЗБВ	Бак запаса борированной воды
CD&ES	Civil Defence and Emergency Situations	ГО и ЧС	гражданская оборона и чрезвычайные ситуации
CDFM	Conservative Deterministic Failure Margin	-	метод анализа консервативного детерминистического запаса до отказа
CNIISK	Central Building Research Institute of former USSR	ЦНИИСК	Центральный научно-исследовательский институт строительных конст-рукций им. В.А. Кучеренко
CP	Cooling Pump	НР	Насос расхолаживания
CSBO	Complete Station Blackout, i.e LOOP and loss of all back-up AC power-sources		Полное обесточение станции
CSG	Complete Switchgear	КРУ	комплектное распределительное устройство

CVS	CKTI-Vibroseism	ЦКТИ	структурно-механическая консультационная инженерная фирма
CWPH	Cooling Water Pump House	БНС	блочная насосная станция
DAP	*additional emergency make-up*	ДАП	дополнительная аварийная подпитка
DBE	Design Basis Earthquake	ПЗ	проектное землетрясение
DC	Direct Current	-	электропитание постоянного тока
DFP (NZS)	“Dirty” Flows Pump	НЗС	насос замазученных стоков
DG	Diesel Generator	ДГ	дизель-генератор
DGLS	Diesel Generator Load Sequencer	АСП	автоматика ступенчатого пуска
DGS	Diesel Generator Station	ДГС	дизель-генераторная станция
DMWT	demi-water storage tank	БЗОВ	бак запаса химически обессоленной воды
DOE	Department of Energy	-	департамент энергетики США
DP	Drainage Pump	ДН	Дренажный насос
DTP	Drain Tank Pump	НДБ	насос дренажногобака
EC	European Commission	ЕС	Европейский союз
ECCS	Emergency Core Cooling System	САОАЗ	Система Аварийного Охлаждения Активной Зоны
ECP	Emergency Condensate Pump	АКН	Аварийный кондесатный насос
ECR	Emergency Control Room	РЩУ	резервный щит управления
EDG	Emergency Diesel Generator	ДГС	дизель-генераторная станция

EFWP	Emergency Feedwater pump	АПЭН	аварийный питательный электронасос
EFWSP	Emergency Feedwater Seismic Pump	АСН	аварийный сейсмический насос
EMP	Emergency Make-up Pump	АПН	Аварийный подпиточный насос
EMS	Emergency Monitoring System		
EOP	Emergency Operating Procedures	АЭП	Аварийные эксплуатационные процедуры
EP (AZ)	emergency protection	АЗ	аварийная защита
EPRI	Electric Power Research Institute	-	научно-исследовательский институт электрической энергии
EPSS	Emergency Power Supply System	САЭ	система аварийного электроснабжения
ERG	Emergency Response Guidelines	РУТА	Руководство по Управлению Тяжелыми Авариями
ERT	Emergency Response Team	ОАР, ГАР	Отряд Аварийного Реагирования, Группа аварийного реагирования
ESWP	Essential Service Water Pump	НСО	
ESWS	Essential Service Water System	СООП	система охлаждения ответственных потребителей
FA	Fuel Assembly	ТВС	тепловыделяющая сборка
FFWP	Fire Fighting Water Pump	НПВ	Насос пожарного водоснабжения
FFFP	Fire Fighting Foam Pump	НПП	Насос пенного пожаротушения
FP	Fission Products	ПД	продукты деления
FRS	Floor Response Spectra	-	поэтажные спектры ответа
FWP	Feedwater Pump	ПЭН	питательный электронасос

GIP	Generic Implementation Procedure	-	стандартная процедура сейсмической верификации оборудования АЭС
GSOP	Generator Shaft Seal Oil Pump	АМНУ	маслонасоса уплотнения вала генератора
HAEK CJSC	Acronym of the organisation operating the ANPP	ЗАО НАЭК	Закрытое акционерное общество “Армянская атомная электростанция”
HCLPF	High Confidence Low Probability of Failure	-	высокая достоверность низ-кой вероятности отказа
HP	High pressure	ВД	высокое давление
I&C	Instrumentation and Control	КИПиА	контрольно-измерительные приборы и автоматика
IAEA	International Atomic Energy Agency	МАГАТЭ	Международное агентство по использованию атомной энергии
ICT	Instrumentation and control Technician	-	сотрудник отдела ИТ
IEP	Internal Emergency Plan	ВАП	Внутренний Аварийный План
IVR	In-vessel (debris) Retention		удержание расплава внутри корпуса реактора
kp	Kilopond	кгс	Килограмм-сила
LBB	Leak-Before-Break	-	течь перед разрушением
LFT	Low Frequency Transducer	ПНЧ	Преобразователь низкой частоты
LOCA	Loss of Coolant Accident	АПТ	авария с потерей теплоносителя
LOOP	Loss of Off-Site Power	ПВЭ	авария с потерей внешнего энергоснабжения
LP	Low Pressure	НД	низкое давление
MB	Main Building	ГК	главный корпус
MCC	Main Crisis Centre	ОЦРАС	Основной Центр Реагирования на Аварийную Ситуацию

MCCI	Molten Corium Concrete Interactions	ВРАЗБ	взаимодействие расплава активной зоны с бетоном защитной оболочки
MCE	Maximum Considered Earthquake	МРЗ	максимальное расчетное землетрясение
MCL	Main Circulation Loop	ГЦК	главный циркуляционный контур
MCP	Main Circulation Pump	ГЦН	главный циркуляционный насос
MCR	Main Control Room	БЩУ	блочный щит управления
MES	Ministry of Emergency Situations	МЧС	Министерство по чрезвычайным ситуациям
MH	Ministry of Health	МА	Министерство здравоохранения
MLIV	Main Loop Isolation Valve	ГЗЗ	главная запорная задвижка
MP	Make-up Pump	НБ	Насос подпиточный
MPD	Medio Prophylactic Department	МПД	медико-профилактический департамент
MRZ	*russian abbreviation for the maximum design earthquake (analogue of Safe Shutdown Earthquake, SSE or SL-2)*	МРЗ	максимально-расчетное землетрясение повторяемостью раз в 10000 лет
MSIV	Main Steam line Isolation Valve	БЗОК	быстродействующий запорный отсечный клапан
MSL	Main Steam Line	ГПК	Главный паровой коллектор
MLIV	Main Stop Valves	ГПК	главный предохранительный клапан
NFE	Nuclear Fuel Element	ТВЭЛ	тепловыделяющий элемент
NFMS	Neutron Flux Monitoring System	АКНП	аппаратура контроля нейтронного потока
NOC	Normal Operating Conditions	НУЭ	нормальные условия эксплуатации
NPP	Nuclear Power Plant	АЭС	Атомная электростанция

NSSS	Nuclear Steam Supply System	ЯППУ	Ядерная паропроизводящая установка
NSSZ	*National Service for Seismic Protection*	НССЗ	национальная служба сейсмической защиты
PAMS	Post Accident Monitoring System	ПАСМ	Система мониторинга после аварии
PAR	Passive Autocatalytic Recombiner	ПАР	устройство пассивной автокаталитической рекомбинации
PC	Process Condenser	ТК	технологический кондесатор
PCC	Process Condenser Cooler	ОТК	охладитель технологического кондесатора
PC	Primary Circuit	-	первый контур
PCS	Primary Circuit System	-	система первого контура
PCR	Plant Control Room	ЦЩУ	Центральный щит управления
PEF	Potential Earthquake Foci	ВОЗ	возможные очаги землетрясений
PGA	Peak Ground Acceleration	МУГ	максимальное ускорение грунта
PMP	Potassium Metaborate Pump	НТС	-
PNAE	*Russian nuclear stanSECd*	ПНАЭГ	правила и нормы в атомной энергетике - общая часть аббревиатуры нормативных документов по безопасности в атомной энергетике.
PORV	Power Operated Relief Valve	ИПУ	-
PRV	Pressurizer Relief Valve	ПК КД	предохранительный клапан компенсатора давления
PSA	Probabilistic Safety Assessment	ВАБ	вероятностный анализ безопасности
PSHA	Probabilistic Seismic Hazard Analysis	ВОСО	вероятностная оценка сейсмической опасности
PSV	Pressurizer Safety Valve	ПК КД	предохранительный клапан компенсатора давления

PWR	Pressurized Water Reactor	PWR	-
PZ	Pressurizer	КД	Компенсатор давления
PZ	*russian abbreviation for design earthquake*	ПЗ	проектное землетрясение
PZ PORV	Pressurizer Pilot Operated Relief Valves	ИПУ- КД	импульсно-предохранительныt клапаны компенсатора даления
R	Reactor	РУ	реакторная установка
RA	Republic of Armenia	РА	Республика Армения
RB	Reactor Building	РО	реакторное отделение
RCC	Regional Crisis Centre	РЦКС	Региональный центр противодействия кризисным ситуациям
RCF	Reducing Cooling Facility	РУР	редукционное устройство расхолаживания
RCPS	Reactor Control and Protection System	СУЗ	система управления и защиты
RCS	Reactor Cooling System	-	Система охлаждения реактора
RCSL	Reactor Coolant System Loop	ГЦК	главный циркуляционный контур
RDGB	Reserve Diesel Generator Building	-	здание резервной дизель-генераторная электростанции
RDGS	Reserve Diesel Generator Standby	РДГЭС	резервная дизель-генераторная электростанция
REX	Return of Experience	-	Накопленный Опыт Эксплуатации
RI	Reactor Island	РУ	реакторная установка
RLE	Review Level Earthquake	ЗКУ	землетрясение контролируемого уровня
RMG	Reversible Motor Genrator	ОДГ	Обратимый двигатель генератор

RPC	Reactor Power Controller	АРК	автоматическое регулирование мощности и компенсация реактивности
RPV	Reactor Pressure Vessel	КАР	корпус реактора
RSA	Reactor Shim Assembly	ОР СУЗ	органы регулирования системы управления и защиты
RU	Rectifier Unit	ВУ	Выпрямительное устройство
S&A	Stevenson and Associates Company	-	фирма "Стивенсон и Компаньоны"
SAR	Safety Analysis Report	ООБ	Отчет по обоснованию безопасности
SAMG	Severe Accident Management Guidelines	РУТА	Руководство по Управлению Тяжелыми Авариями
SBO	Station Blackout	ОБ	Обесточивание электростанции
SDV-A	Steam Dump Valve to the Atmosphere	БРУ-А	быстродействующая редукционная установка сброса пара в атмосферу
SDV-C	Steam Dump Valve to the Condenser	БРУ-К	быстродействующая редукционная установка сброса пара в конденсатор
SEC	Second Emergency Cooling	ДАР	дополнительное аварийное расхолаживание
SFA	Spent Fuel Assembly	ОТВС	отработавшая топливная сборка
SFDSF	Spent Fuel Dry Storage Facility	СХОЯТ	сухое хранилище отработавшего топлива
SFP	Spent Fuel Pool	БВОТ	Бассейн выдержки отработанного топлива
SFP CP	Spent Fuel Pool Cooling Pump	НРБ	насос расхолаживания бассейна выдержки
SFP CSP	Spent Fuel Pool Cleaning System Pump	НБО	Насосов подачи борного раствора на очистку
SFP HE	Spent Fuel Pool Heat Exchanger	ТОБВ	теплообменник бассейна выдержки
SFP MP	Spent Fuel Pool Make-up Pump	НЗБ	насос заполнения бассейна

SG	Steam Generator	ПГ	парогенератор
SG AFS	Steam Generator Auxiliary Feedwater System	СДП ПГ	Система дополнительной подпитки парогенераторов
SG CR	Steam Generator (primary) Collector Rupture	РКГ	разгерметизация коллектора генератора
SG FWDP	Steam Generator FeedWater Diesel Pump	ДНППГ	Дизель насос подпитки парогенераторов
SG SV	Steam Generator Safety Valve	ПК ПГ	предохранительный клапан парогенератора
SG TR	Steam Generator Tube Rupture	РТГ	разгерметизация трубки генератора
SMA	Seismic Margin Assessment	-	Метод граничной сейсмостойкости
SNF	Spent Nuclear Fuel	ОЯТ	отработанное ядерное топливо
SNiP	*Russian Civil Code*	СНиП	строительные нормы и правила
SPS	Seismic Protection System	СИАЗ	система индустриальной антисейсмической защиты
SR	Startup Range	ДП	Диапазон пусковой
SRS	Safety Related System	СВБ	системы, важные для безопасности
SS	Safety System	СБ	система безопасности
SSC	Structures, Systems, and Components	СКК	системы, конструкции, компоненты
SSEL	Safe Shutdown Equipment List	ПОБО	перечень оборудования безопасного останова
SS HE	Spray System Heat Exchanger	ТОС	
SSP	Spray System Pump	НБС	спринклерный насос
SY	Switchyard	ОРУ	Открытое распредустройство

TACIS	Technical Assistance to the Commonwealth of Independent States	-	Европейская программа технической помощи странам СНГ
TC	Training Centre	ЦО	Учебный Центр
TG	Technical Guide	-	техническое руководство
TH	Turbine Hall	МЗ	машинный зал
TSA	Technical Safety Analysis	ТОБ	техническое обоснование безопасности
UDCB	Unit Direct Current Board	БЩПТ	Блочный щит постоянного тока
UPS	Unit Pump Station	БНС	блочная насосная станция
VS	Vent Stack	ВТ	вентиляционная труба
VSN	*Temporary Russian Civil Code*	ВСН	временные строительные нормы
WANO	World Association of Nuclear Operators	WANO	-
WOG	Westinghouse Owners Group	WOG	группа компаний с установленными энергоблоками компании Westinghouse
WR	Working Range	РД	Рабочий диапазон
WWER	Water(moderated)-Water(cooled) Energy Reactor	ВВЭР	водо-водяной энергетический реактор

INTRODUCTION

Following the accident at the Fukushima Nuclear Power Plant in March 2011 the European Union (EU) took the decision to conduct *"a targeted reassessment of the safety margins"* of the nuclear power plants operating on the EU Territory. This reassessment was named "Stress Test".

The European Nuclear Safety regulatory group (ENSREG) has prepared detailed technical specifications to conduct the Stress Test self-assessment.

The Stress Test aims at evaluating the plant robustness to natural events such as earthquake, flooding, extreme weather conditions…, that could lead to the loss of safety functions and consequently to severe accidents.

The targeted safety evaluation aims, amongst others, at clarifying the following issues:

- Are potential natural events that could occur at the plant site taken into account in the plant design basis?
- What is the response of the power plant when facing extreme situations and the safety margins and robustness of the plant beyond its design basis?
- Is there a risk of loss of ultimate heat sink and what would be the potential consequences of such loss?
- Is the plant able to manage severe accident involving nuclear reactor and spent fuel pools as well as to take the appropriate mitigative measures in case such accident occurs?
- Are the emergency response procedures available and ready to be used to keep control in case of emergency situations occurring as a result of single initiating events or combination of events and to properly manage accident which could involve all reactors and Spent Fuel Pools of the site?

As a result of the Stress Test self-assessment, potential improvements (hardware, procedures…) are identified and included in an action plan for realization.

EU invited the neighboring countries to implement the Stress Test of nuclear power plants operating on their territories.

The Government of the Republic of Armenia (RA) decided to take part in the EU Stress Test and requested ANPP to perform the Stress Test self-assessment.

The Armenian Nuclear Safety Authority, have decided to take part in the EU Stress test and requested ANPP to perform the Stress Test exercise consistently with the EU ENSREG specifications.

Because the complexity of the analysis required by the Stress Test goes beyond the ANPP technical capabilities, the European Commission (EC) has decided to launch a tender for the project that could provide ANPP with the technical assistance to carry out the Stress Test.

The technical assistance to ANPP has been provided by the following Consortium:

- Tractebel Engineering S.A.;
- Ustav jaderneho vyzkumu Rez a.s. (UJV);
- CJSC Armenian Scientific Research Institute for Nuclear power Plant Operation (ARMATOM);
- CKTI – Vibroseism (CVS).

The work on the Stress Test has started in August 2012.

The Consortium has written a methodology document, detailed guidance for ANPP to conduct the Stress Test and perform a self-assessment in accordance with the ENSREG Stress Test specifications. The proposed guidance has been approved by both ANPP and the State Nuclear Safety Regulatory Committee of the Republic of the Armenia.

1. CHAPTER 1: GENERAL DATA ABOUT SITE/PLANT

1.1. Brief description of the site characteristics

Location

The ANPP site is located in the western part of the Ararat valley about 10 km north-east from the regional centre of Armavir, 28 km west from Yerevan and 16 km away from the Turkish border.

On the north and north-west, the site is surrounded by mountains. On the east and on the south the site is bordered by the big Downside - Zangu channel.

The ANPP site is sloped 1,5 to 5,0 % southward towards the Ararat Valley.

The site is located about 934,5 m above sea level.

Around the site, up to 25 kilometers, there is no material, accumulated by man, which might induce fire or explosion hazard, as well as no human activity (industrial facilities) which could be considered as potential sources of explosion and fire.

There are three access roads to the site (from the north, east and south-west) and one railroad line from the south-west.

It is highly unlikely that all access roads will be simultaneously blocked in case of extreme natural events.

The broken and hilly relief of the region and the good permeability of the volcanic rock on the ground surface create favorable conditions for infiltration of atmospheric precipitation into the ground. The water infiltrate into the volcanic rocks at the sides of the Aragats Mount, moves to east/southeast towards the Ararat valley, forming strong subsurface flows. In the vicinity of ANPP the groundwater is at about 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 Sevdzhur River is the source of engineering water supply to ANPP. The potable water as well as fire-fighting water supply of ANPP is collected from the lake built on the left bank floodplain of the Sevdzhur River. The water intakes are situated at a distance of 6 to 6,5 km from the ANPP site.

All water ways and basins are located in the south of the ANPP site at a significant distance and significantly below the 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; the river bed is located at 860 m above sea level;
- The Aknalich Lake is 5 km away from the site; the lake is located at 880 m above sea level.

The site of ANPP is characterized by the two main following features:

- The seismicity of design basis earthquake level (DBE) is 7 points, and the maximum considered earthquake (MCE) is equal to 8 points;
- Remote location of the source of a service water supply.

The location of ANPP is shown in Fig. 1-1.

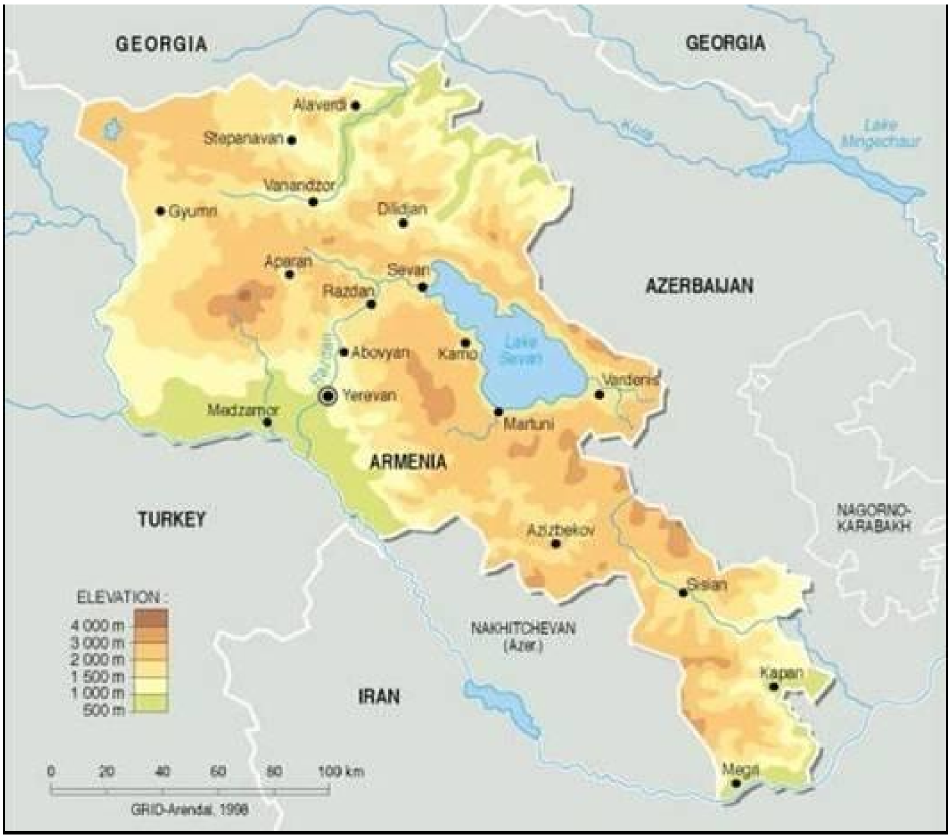

Fig. 1-1: Map showing the location of the nuclear power plant

Number of Units

The Armenian Nuclear Power Plant consists of two units of the WWER/440/270 type reactor. Both reactors are installed in the NSSS section of the building.

A picture of the plant is shown in Fig. 1-2.

Fig. 1-2: Picture of the Armenian Power Plant

After the Spitak earthquake in 1988 both units were shut down. In late 1995, the second unit was re-introduced into commercial operation (after maintenance and repair activities as well as implementation of modernization measures) and operates now at a level of 92% from the rated power.

In 2000 all nuclear fuel from the unit 1 core was unloaded into the spent fuel pools of units 1&2. Currently there is no more fuel in unit 1. The primary circuit of unit 1 is in preserved state.

The layout of the Armenian Power Plant is shown in Fig. 1-3.

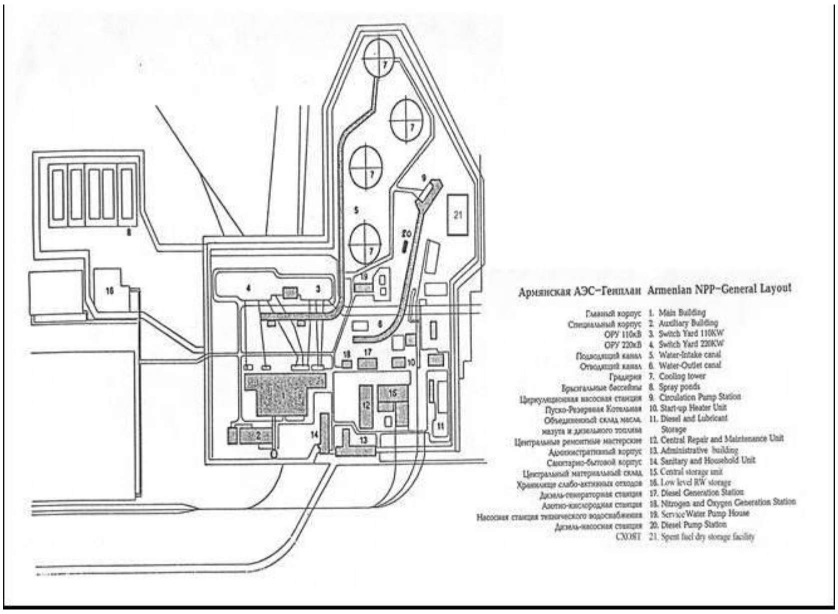

Fig. 1-3: Layout of the Armenian Power Plant

License's Owner

The license was granted to the Closed Joint Stock Company «ААЕК» which is performing its business activity starting with April 4th, 1996 as an operating company.

CJSC ААЕК remains the property of the Republic of Armenia.

1.2. Main characteristics of the units

Type and Power Output Rating of the Reactors

The Armenian Nuclear Power Plant consists of two power units of the WWER-440 (V-270) model type reactor.

The reactor WWER-440 (V-270) is a two-circuit thermal neutron power reactor operating with low enrichment sintered uranium dioxide fuel.

The V-270 reactor type is a V-230 reactor type modified in order to ensure the seismic resistance required for the ANPP site.

The SSC's seismic resistance is ensured by the implementation of the following main design features:

- The SGs are specially designed to be seismic resistant;
- The main equipment of the primary circuit (SG, MCP, MLIV, PZ piping) are additionally fixed with hydraulic snubbers in order to decrease the strengths occurring in support structures due to seismic effects;
- In addition to the support fixture to the basement, the top of the PZ is also fixed with four thrusts;
- The MCP/MCP-310 is replaced with MCP-317 (inertial seismic resistant);
- Seismic resistant two-train system of emergency make-up water supply to primary circuit is installed;
- Additional system (compared to WWER-230) of emergency high and low pressure cooldown through the second circuit is installed;
- SPS is installed;
- Additional isolation valves are installed on the pipelines of category I seismicity systems of primary and second circuits and on the systems of cooling water isolated from SPS.

Contrary to the similar VVER designs, the electrical transversal racks are isolated from the reactor compartment structures. The wire-frame structures of the turbine hall and the electrical racks, together with the vault of the reactor compartment (above the elevation of 10,5m) are designed with steel structures; the roof is covered with profiled steel sheets lightly sealed. The rigid structures of the reactor compartment, the exhaust ventilation centre, the auxiliary building, the tunnels and their connection structures were built in mass concrete. For structural design calculation of the rigid section of the reactor compartment (below the elevation of 10,5m), of the boron unit and the auxiliary building, the seismic load was assumed to be one point higher than the seismicity of the site, and additional rising factor γ = 2,7 was assumed. Concrete tanks of the reactor compartment and of the auxiliary building, which can be used for radioactive liquid long-term storage, are designed with stainless steel cladding; they are made on basis of the “tank-in-tank” principle.

The unit comprises a reactor, pressurizer, six circulation loops having horizontal steam generators PGV-4 (SG), main circulation pumps GSN-317 (MCP), main circulation loop (MCL) and main loop isolation valves (MLIV).

The main equipment of the power units 1 & 2 are,

- The light water power reactor WWER-440 (V-270);
- The turbine island K-220-44;
- The electric generator TVV-220-2A.

The reactor installation of each unit is located in the separate confinements inside the reactor building. The turbine generators are located in the same turbine building. The dematerialized light water with boric acid acts as a primary coolant and moderator in the reactor. The rated thermal power of the reactor equals to 1375 MW.

The installed electric power of each unit is equal to 407.5 MW.

Table 1-1 shows the nominal values of the main process parameters.

Parameter	**Value**
Reactor thermal power	1375 MW
Primary coolant flow	42000 m^3/h
Pressure in the primary circuit	12.26 MPa
Temperature of the primary coolant at the reactor core inlet	267°C
Average heat-up of the primary coolant	29.3°C
Concentration of the boric acid in the primary coolant	12.00 g/kg
Main steam pressure	4.5 MPa
Steam generator steam flow	2700 tonnes/h
Peak Linear Heat Rate	325 W/cm
Number of fuel assemblies	349 pcs

Table 1-1: The Main Process Parameters

Note:

Considering the weakness of the electric system, ANRA established, in 1995, the power limit of the ANPP at the level of 92% from the rated power. Consequently the allowed thermal power of the ANPP unit 2 is 1265 MW.

Date of physical startup

The first unit of the Armenian Power Plant was put into operation on the 22nd of December 1976 and the second unit on the 5th of January 1980.

After the Spitak earthquake on the 7th of December 1988, the Government of Armenia took the decision to shut down both units.

The first unit was shut down on the 25th of February 1989 and the second unit was shut down on the 18th of March 1989. However, due to lack in power supply in 1993 the Armenian Government decided to restart operation of the second unit. The comprehensive survey of the Armenian Nuclear Power Plant was conducted involving international experts and missions in order to ensure that the Armenian Nuclear Power Plant complies with requirements, standards and norms for safe operation of the nuclear power plants. According to the results of this survey the work program and the list of activities to bring the unit to the acceptable safety level, in compliance with IAEA recommendations (TECDOC-640) were developed.

Since 1993 a number of major measures have been implemented at ANPP, which largely contributed to ANPP safety upgrading, namely:

- Replacement of unreliable SG SV with new valves;
- Installation of MSIVs on steam lines downstream of the SGs and the MSL;
- Commissioning of Essential Loads Cooling System (ESWS);
- Installation of emergency diesel pump for water supply to SG in case of plant blackout;
- Modification and reconstruction of I and II category uninterruptible power supply systems;
- Improvement of the RCPS;
- Emergency protection of reactor, divided into two independent channels by technological parameters;
- "Leak-before-break" concept justified and three new systems for diagnostics of primary circuit integrity implemented;
- Fire protection activities implemented, decreasing ignition and fire propagation risk;
- Calculations for beyond design basis accidents and severe accidents performed; guidance for management of beyond design accidents developed; Unit 2 SAR done (first version); SAMG under development;
- FSS established for training of operators;
- Seismic qualification for SSC included in the SSEL;
- Etc.

The implemented and planned activities are detailed in Chapter 1.2 of the SAR.

The main activities to eliminate safety deficiencies were implemented during 2.5 years (1993-1995) at unit 2 in accordance with the approved list of activities.

On the 5th of November 1995, unit 2 achieved criticality and then commissioned in compliance with start-up multi-stage program. According to the license conditions the power output of the unit is limited at the level of 92% (1265 MW) from the rated power. The safety of the unit is substantiated for 100% (1375 MW) of the rated power. The limited power output is caused by the electric system of the Republic of Armenia.

Existing Facilities for Spent Fuel Storage

1) Spent fuel pool of the unit

In order to store spent fuel, each unit has a spent fuel pool next to the reactor cavity.

The spent fuel pool is located into the Reactor Central Hall. It is next to the transport corridor and the reactor fuel reloading pool (the area above the open reactor).

When the unit is operating the spent fuel pool is covered by reinforced concrete slabs and isolated from the fuel reloading pool by a sliding gate blocking the transportation passage. During operation this gate forms a part of isolating protection structure. The spent fuel pool is filled with a solution of a boric acid with a concentration of no less than 12 g/l.

The design of the fuel storage pools allows storage of the fuel assemblies (FA) in two different levels (racks). The rack to store FA under normal operation conditions is located in the bottom part of the pool; it can store the following amount of items:

- Unit 1: 334 FA and 38 leak-proof canisters (to store leaking fuel assemblies);
- Unit 2: 359 FA and 13 leak-proof canisters.

An additional rack has been designed and is used in case of complete unloading of the reactor core (when required). It is installed in the upper part of the pool above the bottom rack. The capacity of the additional rack allows to install 351 FA.

The main part of irradiated nuclear fuel of the units was transferred to Russia in 1990. The remaining irradiated fuel assemblies, i.e. 612 FA, were unloaded into the fuel storage pools of units 1&2.

After construction of the dry storage facility all these 612 SFA were sent to the SFDSF in the period of 2000-2003.

2) Spent Fuel Dry Storage Facility (SFDSF)

The first series of long-term dry storage facilities (Dry Casks for long term Storage of Spent Nuclear Fuel) was constructed at the ANPP site in 1997-2000. The design and construction has been carried out by the French company FRAMATOM (currently AREVA TN International) based on the NUHOMS concept.

From 2000 and until now there is no nuclear fuel in the core of unit 1.

Later the second series of the long-term storage facilities was constructed at the ANPP site. It is designed to house 12 canisters (each having capacity of 56 casings) after spent fuel is stored in SFP for 5 or more years.

Natural circulation of the air allows heat removal from the spent fuel canisters of the dray cask facility. The total residual decay heat in one canister yields in 3-10 kW (the permissible value is 14,5 kW).

The NUHOMS storage facility consists of reinforced concrete monolith structures (modules) which contain seal-welded sealed canisters, housing 56 assemblies. The modules are designed with a natural air circulation from the bottom to the top to remove the heat of the canisters. The canisters are filled with helium to prevent corrosion and to improve the heat transfer from the assemblies to the canister. The canisters are seal-welded tight and installed into the modules.

Fig. 1-4 and 1-5 show the storage modules with a spent fuel canister.

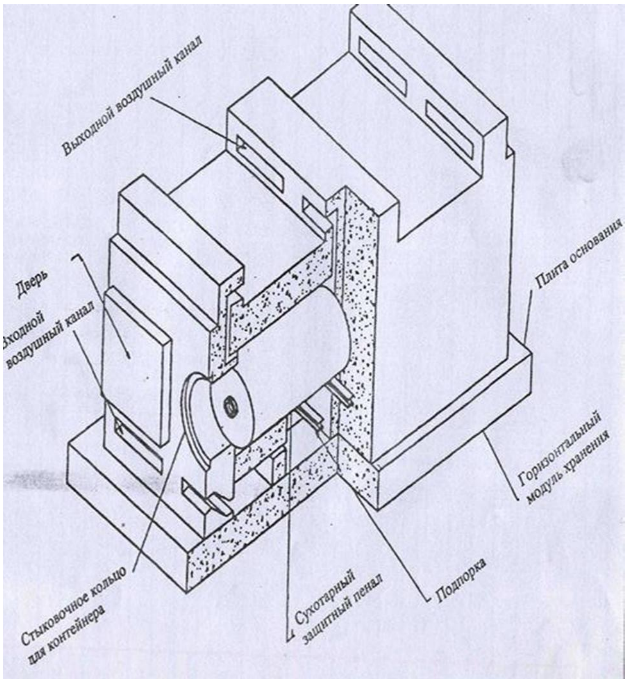

Fig. 1-4: Dry Cask - NUHOMS ® 56 V - Cross section

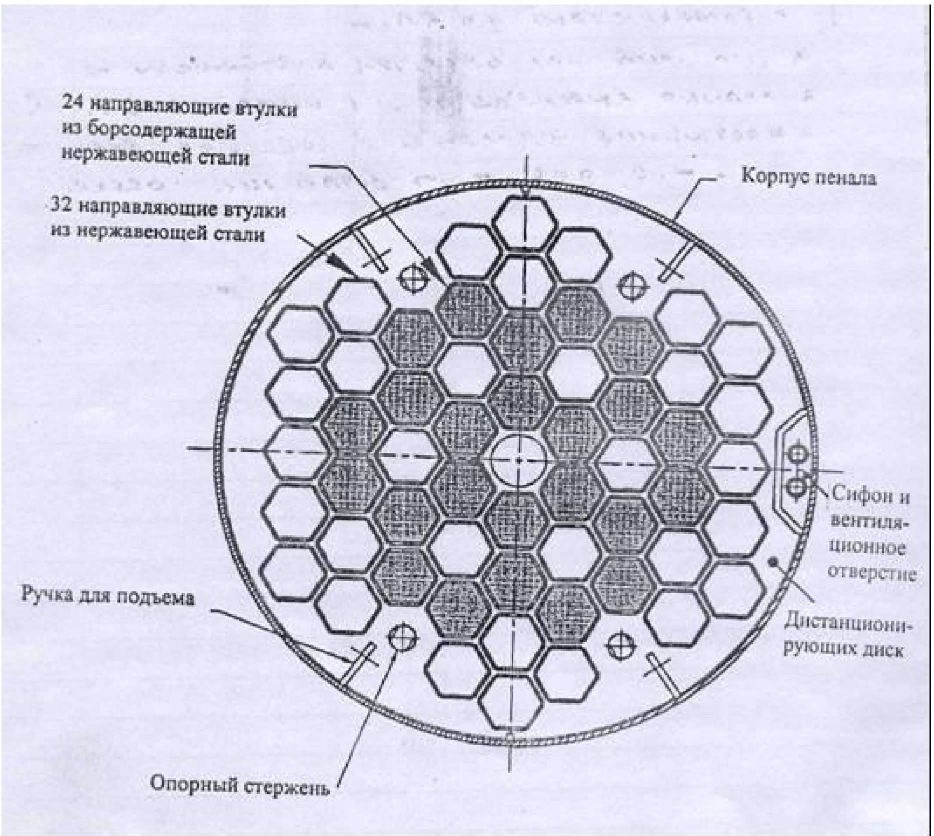

Fig. 1-5: NUHOMS canister with spent full elements

The amount of spent fuel elements in the spent fuel pools of units 1 and 2 between reloading is as follows:

- Spent fuel pool of unit 2: no more than 300-330 FA stored with a storage time < 3 year and a total power output not exceeding 250 kW;
- Spent fuel pool of unit 1: no more than 300-330 FA with a storage time > 3 years and a total residual power output not exceeding 60 kW.

The amount of spent fuel elements at the time of writing of the present report (October 10, 2012) is given in Table 1-2.

	SFP-1	**SFP-2**	**SFDSF**
Amount of SFA (pcs.)	288	295	1176 (21 canisters)
Total power (kW)	55	250	74

Table 1-2: Amount of spent fuel elements at the ANPP site on 2012-10-10.

Table 1-2 shows that, on 10 October 2012, there were 583 SFA in the storage pools. There were 82 free cells in SFP-1 and 75 free cells in SFP-2. The upper rack with a capacity of 351cells was also vacant.

From 1995 and until now (through the whole period of operation) there was no leaking FA found at the Armenian NPP.

1.3. Systems to ensure or maintain the main safety functions

The WWER-440 (V-270) is a pressurized light water power reactor vessel type.

The primary coolant (and moderator in the reactor core) is provided by demineralized water containing boric acid.

The reactor coolant system for WWER-440 (V-270) consists of six circulation loops.

Each loop consists of the following:

- The main circulation pipelines;
- The main circulation pump (MCP);
- Steam generator (SG);
- Main loop isolation valves (MLIV), providing isolation of any reactor loop.

Each of the six circulation loops consists of «hot» and «cold» legs. The «hot» leg delivers primary coolant from the reactor to the steam generator; the «cold» leg is used to carry the primary coolant cooled in steam generator to the inlet of MCP and afterwards into the reactor. In each loop in the «cold» and «hot» legs there are MLIV installed which are designed to isolate the loop with damaged component in case of emergency (leak in steam generator, leak isolated part of the main circulation pipelines, etc.).

The schematic diagram of heat transfer in the WWER-440 is shown in the Fig. 1-6 & 1-7.

The reactor core consists of 312 fixed fuel assemblies and 37 regulating and shim fuel assemblies (RSA) which can move in vertical direction. RSA are a working element of reactor control and protection system (RCPS). RSA consists of control rod (upper part) and fuel assembly (FA) which are connected by an intermediary rod. Using electromagnetic drive the RSA are moved (up or down) to control the reactor power output, or to compensate the excessive reactivity or to shut down the reactor. When the emergency protection (EP/SCRAM) is triggered, the power supply to the electric drives are de-energized, following which the RSA are moved down due to gravity force.

The main circulation piping system is designed in compliance with requirements set forth to prevent the release of radioactive substances into the environment. The design of the main circulation piping ensures normal operation of the reactor installation as well as safe shut-down and cooling down under impact of loads caused by the maximum design earthquake.

The difference of level between the core and the SG ensures coolant natural circulation in the primary circuit in case the MCP is disconnected, with the aim of removing core residual heat after reactor shutdown.

The steam generator PGV-4 is designed as a shell type horizontal heat-exchanger, with a submersible piping bundle designed to produce dry saturated steam.

The main circulation pumps MCP-317 maintains circulation of primary coolant in the primary circuit. It is a vertical single-stage centrifugal pump with hydrostatic sealing of the shaft; it has a single-sided impeller, axial water inlet and external three-phase asynchronous electric motor.

In order to ensure the main safety goal which is to prevent the release of radioactive substances outside the boundary of physical barriers, the design foresees the following systems and equipment are foreseen which ensure the main safety functions:

- Control of reactivity;
- Heat removal from reactor;
- Heat removal from the fuel pools;
- Retention of radioactive substances and ionizing radiation within the design boundaries.

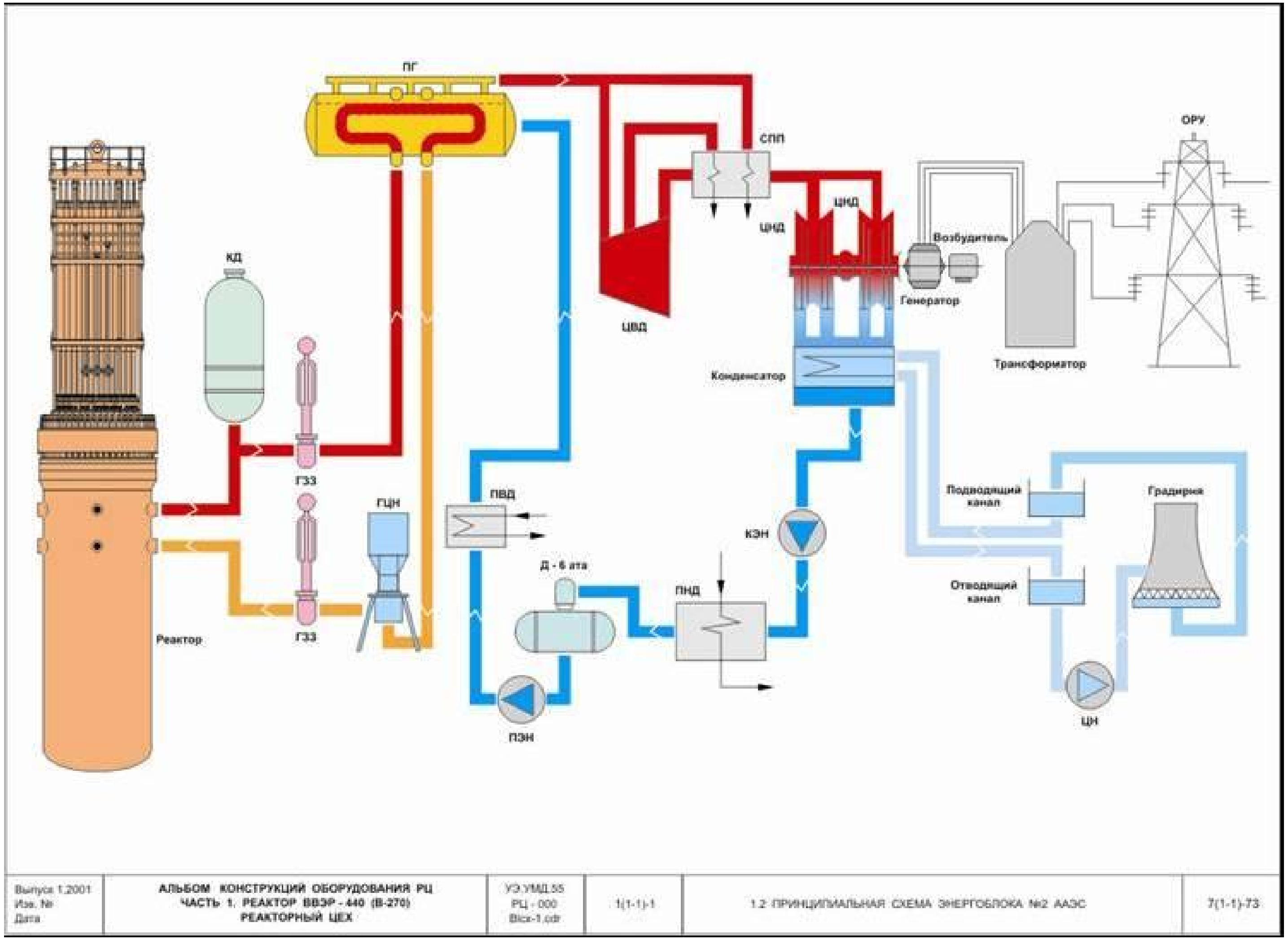

Fig. 1-6: Schematic diagram of heat transfer in WWER-440

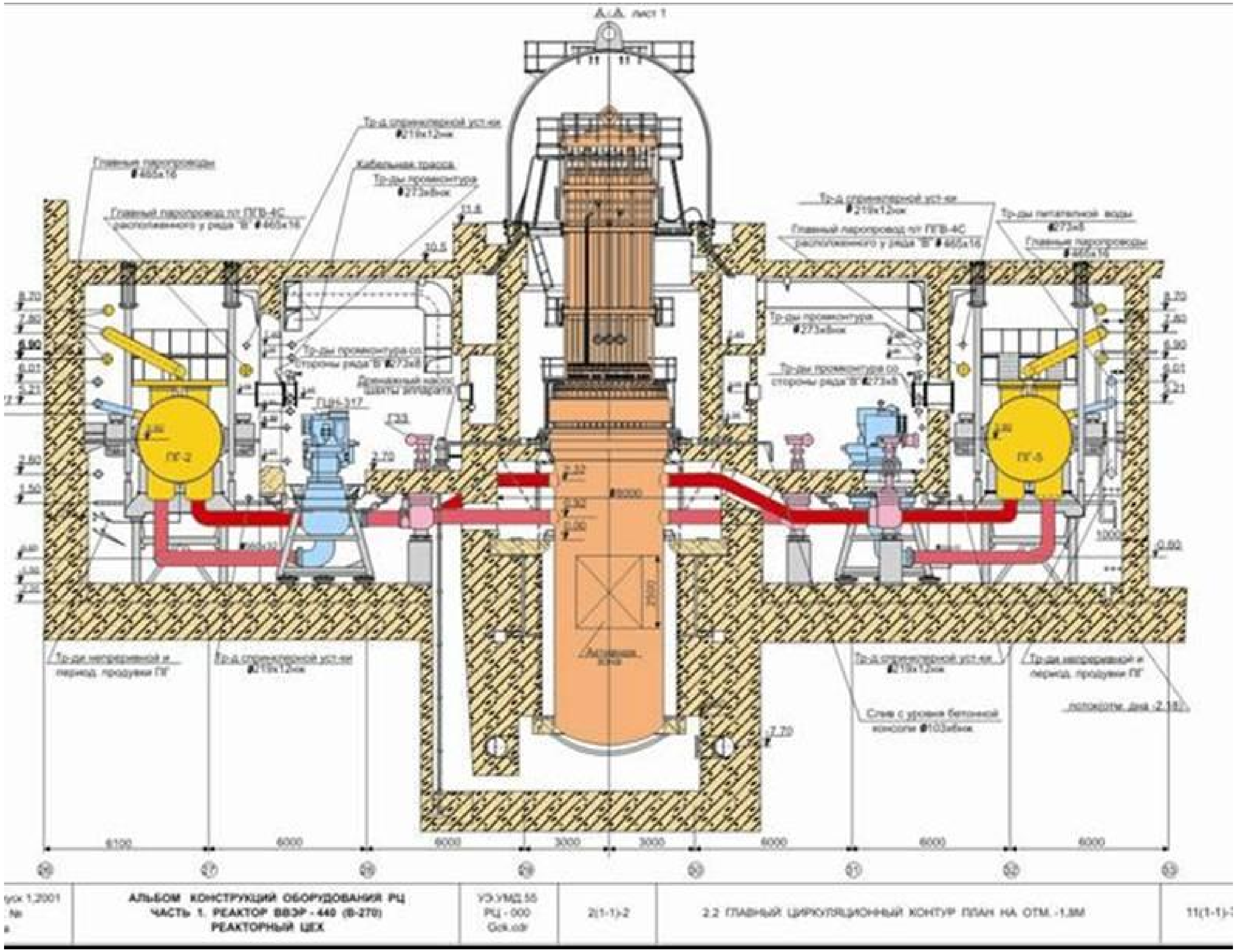

Fig. 1-7: Schematic diagram of heat transfer in WWER-440

SAFETY SYSTEMS

Protection Systems - Protection and control systems

The reactor control and protection system (RCPS) is designed to control the reactor operation during startup, normal operation, planned or emergency shut-down.

RCPS contain the following items:

- Mechanical part of RCPS system (37 RSA);
- Electric equipment of RCPS;
- Equipment to control and generate emergency protection alarms;
- Neutron flux monitoring equipment;
- Control and protection system with RSA electric drives.

The electric equipment of RCPS contains the following:

- Two sets of emergency protection equipment;
- Equipment to unload and restrict the power generated by reactor;
- Reactor power controller;
- The system for group or individual control of RSA;
- The system to control and indicate the position of the RSA;
- RSA power control system;
- Power supply system.

The emergency protection (EP) designed for the shut-down or slowing the intensity of the chain reaction in the core depending on the emergency control signals. All emergency control signals are divided into four types which differ each from other in effect by the rate for suppressing or decreasing the chain reaction or prohibition to increase its intensity.

EP I (quick reactor shut-down)

- Is ensured by gravity force moving down all RSA with a velocity of 20-30 cm/s;
- If the EP-1 signal disappears the RSA will continue to move down until they reach the final (bottom) position (the time to move RSA down is about 8 to 12 seconds).

The list of Armenian NPP V-270 reactor installation EP/SCRAM signals is given in Table 1-3.

Item	Conditions
1	Pressing any of three buttons «EP-I» (panel PA-3 MCR, panel 2 PCR)
2	Increase in power output of the reactor installation above the SCRAM level sets by setpoint adjustment device (SAD) when the in Working Range 2 (WR2) for two out of three in any NFMS train.
3	Period decrease to 10 seconds when in WR2 in two out of three channels for any NFMS train. Under condition that the power output of reactor increased by 5% Pn from the moment this setpoint was reached.
4	Period decrease to 10 seconds under when in WR1 in two out of three channels for any NFMS train. Under condition that the power output of reactor increased by one order of magnitude.
5	Period decrease to 20 seconds under active range of SR in two out of three channels for any NFMS train. Under condition that the power output of reactor increased by one order of magnitude.
6	Increase of the level in the pressurizer (PZ) by 1000 mm from the nominal value using two instruments out of three with confirming signal that at least one assembly stays above the «Bottom intermediate» zone.
7	Decrease of level in the pressurizer (PZ) by 2560 mm from the nominal value in two out of three instruments.
8	Increase of the pressure in the primary circuit above140 bar in two out of three instruments.
9	Decrease of the pressure in primary circuit below 95 bar in two out of three instruments with confirming signal that at least one assembly stays above «Bottom intermediate» zone.
10	Increase of the pressure differential above the core to 3.75 bar in two out of three instruments.
11	Decrease of the electric power to 0.55 from the nominal value or decreasing the MCP pressure differential to 1.5 bar for four or more MCP with actuation delay of 2 sec.
12	Decrease of the level in any two operating SG by 400 mm from the nominal value in to two out of three instruments.
13	Intermediate position of four turbine main stop valves or closed position of two out of four turbine main stop valves for the last operating turbine with actuation delay of 2 sec.
14	Increasing seismicity up to 50 cm/sec2 in two out of three monitoring stations.
15	Simultaneous loss of 220 V DC power supply at input terminals 1, 2 of the RCPS switch board.
16	Simultaneous loss of 220 V DC power supply at the input terminals 3, 4 on RCPS switch board.
17	The loss of DC 220 V power supply to 114 AV ÷ 116 AV intermediate relays of the reactor protection in two out of three inputs.
18	Simultaneous loss of AC ~380 V 50 Hz power supply at input terminals 1, 2 of RCPS switch board with actuation delay of 2.2 sec.

Item	Conditions
19	Simultaneous loss of power supply ~380 V 50 Hz at input terminals 3, 4 of RCPS switch board with actuation delay 2.2 sec.
20	The loss of AC power supply to the instrumentation equipment in two emergency protection channels out of three for this type of EP.
21	The plant of «Blackout» mode
22	Decrease of the pressure in any of Main Steam Lines half-manifold to 35 bar.

Table 1-3: List of emergency protection of the V-270 reactor of ANPP

EP-II (slow reactor shut-down)

- Is provided by the step-like movement of the control groups down according to predefined sequence with the velocity of 20-30 cm/s;
- When the EP-II signal disappear, the movement of the control groups down stops and RSA stop in positions which they reached at the moment when the signal disappeared.

EP-III (decrease of reactor output)

- Is provided by a sequential movement of the control groups down with a velocity of 2 cm/s;
- When the signal EP-III disappear, the movement of the control groups down stops and RSA stop in positions which they reached at the moment when the signal disappeared.

EP-IV (prohibition to increase reactor output)

- Is provided by prohibition of moving RSA up;
- When EP-IV signal disappear, the prohibition to move RSA up is removed.

In order to increase the reliability these two trains of the EP equipment are installed in different rooms.

The detailed list of emergency protection and the requirement to the operability of the chains forming the emergency protection of reactor installation are provided in the Armenian NPP Unit 2 Technical Specification.

Automatic reactor power controller is designed to maintain the power of reactor in consistency with the power output of the turbo-generators and to maintain the neutron power of reactor using signals generated by the neutron flux monitoring system (NFMS).

The neutron flux monitoring equipment monitors the density of the neutron flux in the ionizing channels installed in the biological protection of the reactor in all operational modes of the reactor installation except refuelling.

The whole range of the interval where the density of neutron flux is controlled is divided into the following ranges:

- Start-up range;
- Working logarithmic range (WR1);
- Working linear range (WR2).

The main technical features of the NFMS are given in Table 1-4.

Parameter	**Measurement range**		
	Startup (SR)	**Working logarithmic range (WR1)**	**Working linear range (WR2)**
Range to control the density of neutron flux n/(cm^2·s)	from 10^0 to 10^6	from 10^4 to 10^{10}	from 2.5x 10^5 to 1.2x10^{10}
Range to control power output, % $P_{n.}$	from 10^{-8} to 10^{-2}	from 10 $^{-4}$ to 10^2	from 2.5x 10^3 to 120
Range to control period (s)	from minus 999 to minus 5 and from +5 to +999		
Range to control reactivity, β_{eff}	from minus 25 to plus 0.75		
Setpoints for emergency protection	6 settings having one order of magnitude offset		109 settings with 1% P_n offset (ranging from 2 to 110% $P_{n.}$).
Settings of protections on period (s) EP PZ RM	10, 20, 40 20, 40, 80		10, 20, 40 20, 40, 80 40, 80, 160

Table 1-4: Main engineering features of the NFMS

NFMS fulfills the following functions:

- Control of power, period and reactivity of the reactor during start-up and operation;
- Setting up of power and period setpoints in individual channels;
- Forming discrete emergency protection alarms signals for power and period;
- Generate digital signals of emergency and preventive protection of power and period;
- Generate digital control signals for power and period;
- Generate digital signals for the current range;
- Generate digital signals on self-testing and verification of the measuring channels;
- Generating continuous analogous signals proportional to the power output and inversely proportional to the period;
- Generate digital signals indicating when the set power output levels are exceeded;
- Long-term documentation of the power, period and reactivity values;
- Indication the status of the main digital signals at the main control room panel;
- Automatic control of equipment operability with provision of failure signals for each channel;
- Manual and automatic verification of the neutron flux density monitoring channels.

The main NFMS equipment consists of two independent three-channel sets of NFMS for emergency and provisional protecting (EPP) and MCR Alarm Unit at the unit control panel.

The detailed descriptions and outline of operational status of the system components is given in Engineering Manual and Guidelines of the NFMS.

The system to control power of Nuclear Power Plant provides automatic maintaining of the present parameters of NPP at any power output level ranging from 3-110% from the rated value. It also ensures automatic unloading of the reactor installation and turbo-generators to allowable power output level in case of emergency situation which allow continuous operations of NPP under partial load.

When the emergency protection signals (EP) is generated, the controller automatically turns off the control of the reactor and turbine generator discharge control.

The electric motors in RCPS drives are fed from special power sources having non-standard output frequency and voltage; that allows simplifying the mechanics of RCPS and, consequently, making it more reliable in operations. A static low frequency transducer (LFT) is used as a special source of three-phase voltage that is supplied to electric motor in RCPS drive. At the input the LFT is supplied with power from 380/220 V, 50 Hz network; at the output the LFT generates a three-phase 120÷180 V with a frequency of 4 Hz.

Breaking force at the shaft of electric motor in the upper position of the RSA is ensured by supply of 100 V DC to the electric motor from rectifier unit (RU-2) which is also supplied from the 380/220 V, 50 Hz buses.

The low frequency transducers are used as follows:

- Two transducers are used to supply power to even and odd groups of RSA electric motors when CA are moving from or to the core;
- The third transducer is used as a hot backup (to replace one of the first two in case of failure);
- The forth transducer is used to supply power to electric motors operating in individual manual operations mode.

Rectifiers are used as follows:

- One remains in hot backup mode whereas the second is operational.

Power supply to LFT and RU is provided from two independent power sources consisting of 220 V power line using service transformer and special generator mounted on the same shaft as the main turbine generator.

The electric power is supplied from two sources onto two systems of buses using two independent feeders, each designed to handle 100% of the load. Failure of power supply to one feeder is allowed to be no longer than 1 second.

The load is connected to the electric buses as follows: two operating transducers and two rectifiers are permanently connected to their respective electric buses. The backup transducer and the transducer for individual RSA movement in automatic or manual mode is connected to any of the above two electric buses, using the splitter.

In order to avoid impact of the transducers on the operation of the instrumentation, the power supply of the instruments is designed by using two separate feeders from the two independent sources with automatic back-up.

The safety of the reactor in the power supply circuit is ensured by collecting 220 V DC from the buses to supply electric magnets. To do so the coils of contactors К1, К2 are included into schematics to control EP, whereas the buses to supply electric magnets are connected to the inlet of 220 V DC using the contacts of contactors К1& К2 to ensure reliable switch-off.

The circuit diagram to supply power to RCPS is given in Fig. 1-8.

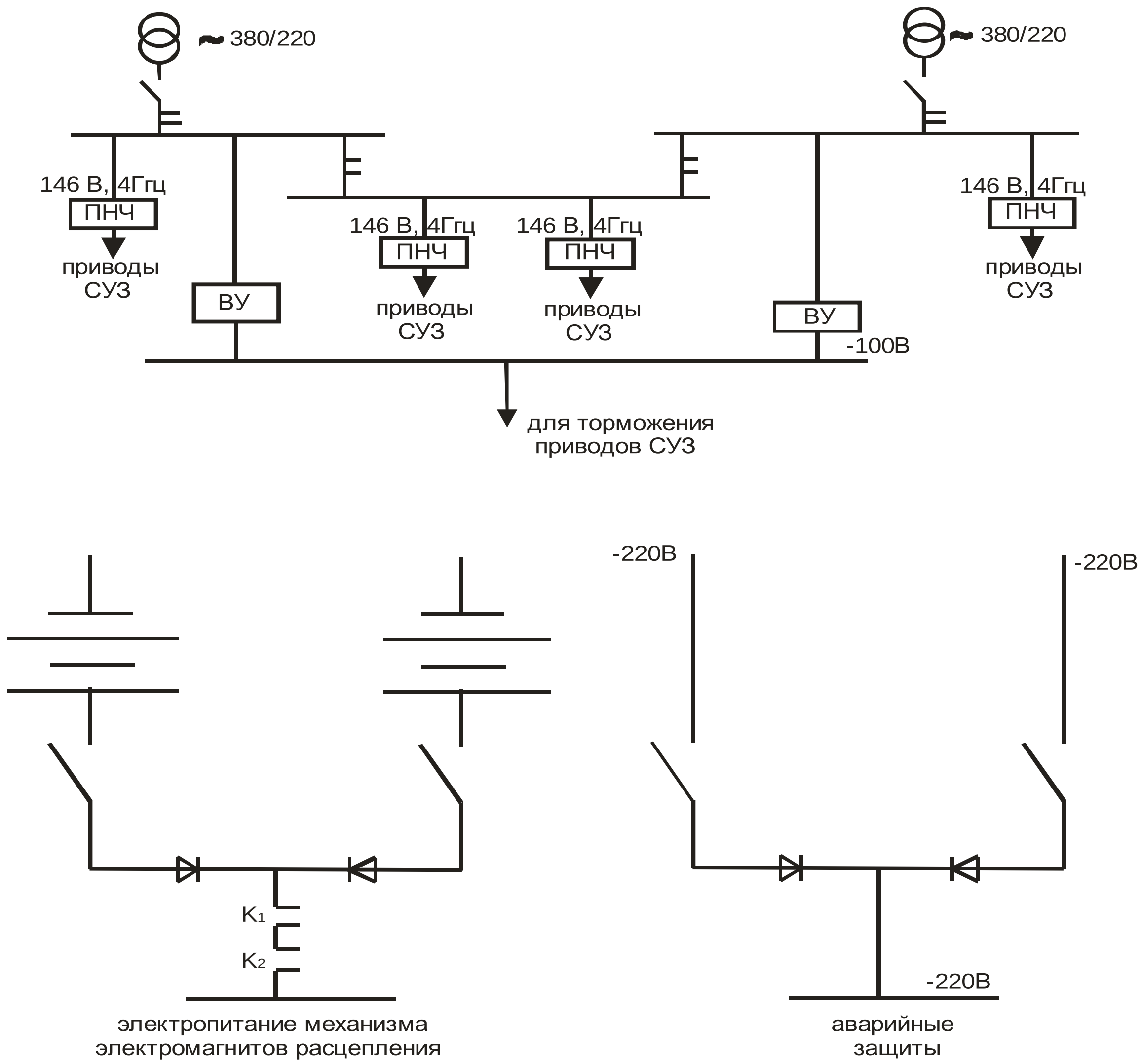

Fig. 1-8: Circuit diagram of power supply to RCPS

Primary Circuit Emergency Make-up System

Primary Circuit emergency make-up system is designed to inject boric acid solution with 12.0 g/kg concentration into the primary circuit in case of any initiating event requiring its performance (large LOCA).

The system ensures the following safety functions:

- To maintain coolant inventory in the reactor for emergency core cooling during and after accident with and without loss of primary circuit integrity;
- To transfer the reactor to subcritical state and to maintain it in this state.

Primary circuit emergency make-up system consists of two trains. Each train includes:

- 800 m^3 emergency boron tank with boron solution at a concentration no less than 12 g/kg;
- Three emergency make-up pumps;

- Piping, valves and I&C instrumentation.

System flow chart is shown in Appendix A.

The system is powered from the Safety Class 2 power supply buses and from the SEC as well. The system is constantly in standby and automatically put into operation when:

- The pressurizer level is 2560 mm below the nominal value or primary circuit pressure drops below 95 kg/cm^2 in the first loop. Automatic starts two pumps in each train;
- System start-up time (in case of the unit blackout) is 30 to 35 seconds.

Emergency make-up pumps flow rates versus primary circuit pressure are shown in Table 1-5 and 1-6.

Primary Circuit pressure, Pa	**minimum flow rate, kg/s**	**rated flow rate kg/s**	**maximum flow rate, kg/s**
0	34.3	36.11	37.92
$6.56x10^6$	34.3	36.11	37.92
$7.54x10^6$	30.34	31.94	33.54
$8.43x10^6$	27.71	29.17	30.63
$9.99x10^6$	22.43	23.61	24.79
$1.10x10^7$	18.47	19.44	20.41
$1.30x10^7$	9.23	9.72	10.21
$1.33x10^7$	7.13	7.50	7.88
$1.34x10^7$	0	0	0

Table 1-5: One EMP (Emergency Makeup Pump) flow parameters versus primary circuit pressure

Primary Circuit pressure, Pa	**minimum flow rate, kg/s**	**rated flow rate kg/s**	**maximum flow rate, kg/s**
0	52.78	55.56	58.34
$7.15x10^6$	52.78	55.56	58.34
$8.43x10^6$	47.50	50.00	52.50
$9.70x10^6$	42.22	44.44	46.66
1.11x107	36.95	38.89	40.83
$1.24x10^7$	29.03	30.56	32.09
$1.31x10^7$	21.11	22.22	23.33
$1.32x10^7$	20.59	21.67	22.75
$1.34x10^7$	0	0	0

Table 1-6: Two EMPs (working in same train) flow parameters versus primary circuit pressure

After upgrading, in 2007, of the Class 2 emergency power supply system with consideration of a single failure concept, minimum two emergency makeup pumps are started under any design initiating event.

Any emergency makeup train has an interface with a respective independent train of the essential service water system.

Unit 2 Safety Improvement Plan foresees upgrading of the emergency makeup system to ensure long-term operation of the emergency makeup system in case of the primary circuit LOCA and increased the design accident LOCA spectrum.

Basic diagram of the system is given in Appendix A.

Steam Generators Emergency Feedwater System

Steam generators emergency feedwater system is designed to provide emergency supply of the make-up water to the steam generators in case of malfunction (loss) of all feedwater pumps to ensure heat removal from the unit.

The system consists of:

- Two emergency feedwater pumps (2 EFWP-1&2) with a flow of 65 m^3/hour at rated head of 56 kg/cm^2;
- Two deaerators, each with a 120 m^3 effective volume;
- Piping, valves and I&C instrumentation.

Pumps 2 EFWP-1&2 are respectively powered from 0.4 kV bus of Safety Class II (26 BNN and 25 BNN).

The equipment is installed in the turbine building.

Basic diagram of the system is given in Appendix A.

High Pressure Reactor installation Emergency Cooling System

High pressure reactor installation emergency cooling system is designed for primary circuit cooling through second circuit in case of accidents and in case of earthquake until pressure in the Main Steam Line decrease to 5 kg/cm^2.

The system consists of two independent lines. Each train includes:

- Backup demi-water tank (DMWT) with effective volume of 500 m^3;
- Emergency feedwater seismic pump (EFWSP) with nominal capacity of 65 m^3/h at rated head of 56 kg/cm^2;
- Piping, valves, I&C, interlocks and alarms.

Each system train is connected to three steam generators.

System electrical equipment is supplied with 0.4 kV bus of Safety Group II (25 BNN, 26 BNN). The system is activated automatically in case of an earthquake (triggered by signals generated by SPS-I or SPS-II) with confirmation of 300 mm level drop in any SG against the nominal value. In case of complete unit black out the system (if required) can be put into operation by MCR operator.

Service water is not required for pumps operation as equipment is cooled by pumped demineralized makeup water.

Equipment of the system is installed in a “boron room” (B-001/2) and is designed as seismic resistance Class I.

Basic diagram of the system is given in Appendix A.

Low Pressure Reactor installation Emergency Cooling System

Low pressure reactor installation emergency cooling system is designed to maintain primary circuit parameters via second circuit when MSL pressure does not exceed 5 kgf/cm^2 and to remove the reactor core residual heat until the normal plant cooling system is recovered, or primary circuit cooling ensured using other means.

The system consists of two similar trains. Each train includes the following:

- Backup demi-water tank (DMWT) with effective volume of 500 m^3;
- Emergency condenser;
- Emergency condensate pump (ECP) with a nominal capacity of 32 m^3/h at rated head of 15 kg/cm^2;
- Piping, valves, I&C, interlocks and alarms.

Each system train is connected to three steam generators.

ESWS utility water is used as a cooling agent.

Pumps 2 ECP-1, 2 are powered from 0.4 kV bus of Safety Class II (25 BNN and 26 BNN). The system is started automatically when actuated by the SPS and when MSL pressure drops below 5 kg/cm^2.

System equipment is installed in the “boron room” (B-001/2) and is designed as Class I seismic resistance.

Primary Circuit Overpressure Protection System

Primary circuit overpressure protection system is designed to protect the reactor installation equipment and piping from excessive pressure in the primary circuit.

The system consists of two independent pressurizer pilot operation safety valves, piping, instrumentation and alarms.

Safety valves are installed in Room 503/2 of the reactor building and are designed as seismic resistance Class I.

Second Circuit Overpressure Protection System

Second circuit overpressure protection system is designed to protect the second circuit from excessive pressure during deviations from normal operation and accidents.

The system includes:

- Four steam dump valve to condenser (SDV-C) with steam release into turbine condenser;
- Two steam dump valve to atmosphere (SDV-A) with steam release to atmosphere;
- Twelve steam generators (two per SG) safety valve with SG steam release to atmosphere;
- I&C, interlocks and alarms.

Basic features of SG SDV-C, SDV-A and SG SV are given in Table 1-7.

Item	Consumption t/hour	Pressure of valve opening kgf/cm^2	Pressure of valve closing kgf/cm^2
SDV-C	2x2x440	50	46.5
SDV-A	2x2x440	53.6	52
SG SV	12x250	55.6 (reference)	50
		56.7 (operating)	51

Table 1-7: Basic features of SG SDV-C, SG SDV-A and SG SV

SDV-A are regarded as normal operation systems and perform heat removal function from the reactor installation in emergency modes related to unit blackout, design seismic impacts and other accidents caused by reactor installation heat removal failure.

Primary Circuit Emergency Gas Removal

Primary circuit emergency gas removal ensures full covering of the reactor core with coolant for reliable heat removal by removal of steam/gas bubbles from primary circuit top dead-end sections; the bubbles are released from the coolant when its temperature drops to the saturation margin in case of the emergency reduction of the pressure above the reactor core.

The system consists of pipelines connecting each SG headers and the space below reactor head with the bubble condenser.

During normal operation this system is considered as standby. The system is actuated by the operator in line with Technical Specification requirements.

The system is designed as seismic resistance Class I.

Main Steam Line Isolation Valves

The main steam line isolation valves (MSIV) are designed to isolate a damaged steam generator which has a steam line rupture in a non-isolable section from the steam lines of the intact steam generators in order to limit the primary overcooling and to limit the development of reactivity accident.

The system includes seven MSIVs (one valve per SG plus one valve on main steam line splitting the latter into two symmetrical half manifolds), instrumentation, and alarms.

The MSIV is triggered by the following signals:

- Pressure decrease in any semi-collector (I or II) below 35kgf/cm2, the MSIV on MSL is closed separating the MSL into two semi-collectors;
- Pressure decrease in I semi-collector to 30kgf/cm2 and below this value, the MSIV from three SGs (SG-1, SG-2, SG-3) are closed;
- Pressure decrease in II semi-collector to 30kgf/cm2 and below this value, the MSIV of corresponding SGs (SG-4, SG-5, SG-6) are closed.

MSIV closure time is 2 to 5 sec.

Note: The MSIV (upon closure) could be open remotely by operator. Opening time is 300 s.

The system is designed as seismic resistance Class.

Primary and Second circuit Hydraulic Shock Absorbers

Primary circuit and second circuit hydraulic shock absorbers are designed to absorb vibration of the equipment and pipelines of the primary and second circuits, and thus to prevent the risk of leaks which may be caused by earthquakes.

Primary circuit equipment is equipped with 93 hydraulic shock absorbers:

- 48 pcs for SG (8 pcs per SG);
- 18 pcs for MCP (3 pcs per MCP);
- 24 pcs for MLIV (2 pcs per MLIV);
- 3 pcs for pressurizer.

Second circuit pipelines (main steam, make-up water, SG blow down) located in the SG and MCP box are equipped with 27 dampers. There are also 17 vibration absorbers installed at the elevation 14,7 m of the turbine hall.

During normal operation the absorbers do not prevent movement of the equipment and piping induced by thermal expansion.

In case of seismic impact the shock absorbers work as rigid constraints.

Steam Generators Auxiliary Feedwater System

Steam generators auxiliary feedwater system (SG AFS) is designed to make-up steam generators in case of complete loss of outside power or in case of accident involving the failure of all normal and emergency systems foreseen for this purpose.

SG AFS consists of:

- Two tanks of demineralized makeup water;
- SG feedwater diesel pump;
- Automatic recirculation control valve;
- I&C;
- Piping and valves.

Equipment is cooled by air.

Equipment	Parameter	Unit	Value
Demineralized makeup water tanks	Useful volume	m^3	500
Diesel engine	Capacity	kW	207
	fuel consumption	l/hour	53
	fuel tank capacity	l	1462
Pump	rated pressure	kgf/cm^2	53.6
	rated flow rate	m^3/hour	65.9

Table 1-8: System Equipment Key Specifications

System equipment is installed in B-001/1 (Unit I, “boron room”).

Feedwater to steam generators is supplied via high and low pressure emergency cooling system pipeline.

The system is designed as seismic resistance Class I.

Basic diagram of the system is given in Appendix A.

LOCALIZING SAFETY SYSTEMS

Confinement

The system is designed to contain released radioactive substances within the accident localizing zones.

- To isolate the process systems and components from the environment;
- To protect personnel and the public from exposure to ionizing radiation and from radioactive hazard.

The system includes:

- Enclosing structures built of pre-stressed concrete with sealed steel lining;
- Penetrations (process, cable, ventilation, etc.);
- Ventilation system isolation valves;
- Man-holes, doors and their embedded parts; pressure-relief and safety valve**s.**

In normal operation the system is used to create and maintain under pressure in near-reactor rooms; the system is kept in standby position.

Spray System

The system is designed to reduce pressure in the confinement in case of accident accompanied with ruptures of primary circuit and second circuit pipelines (within the confinement) as well as to bind gaseous radioactive iodine contained in steam/gas (to transfer it into liquid phase).

The system includes:

- A tank of emergency boric acid solution of 12 g/kg concentration with an effective volume of 800 m^3;
- Three spray system pumps (SSP) of 280 m3/h rated capacity each at rated head of 4.2 kg/cm2;
- Two heat exchangers (2 SSHE-1, 2);
- Potassium metaborate 6.6 m^3 tank;
- Potassium metaborate pump (PMP) of rated capacity of 2.5 m^3/h at rated head of 10 kg/cm^2;
- Pipelines, sprays (jets, nozzles) and valves.

Suction and pressure pipelines of SSP are linked via common headers.

SSP and PMP pumps are powered from 0.4 kV of Safety Class II busses.

ESWS water is used as cooling agent.

The equipment is housed in B-001/2 (Unit 2, “boron room”).

During normal operation the system is on standby (automatic stand-by) mode.

The system is designed as seismic resistance Class I.

SUPPORTING SAFETY SYSTEMS

Secure Power Supply Systems

Secure power supply system includes:

- Class I emergency power supply system;
- Class II emergency power supply system;
- Emergency cooling system.

The systems are described in § 1.3.5.

Essential Service Water System (ESWS)

ESWS is designed to provide cooling water for the critical systems for the safety of the plant, which require continuous cooling water supply in normal operation conditions, in case of emergency, in case of power failure, during earthquakes, as well as during LOCA accidents.

ESWS provides supply of water to:

- Spray system heat exchangers;
- Spent fuel pools cooling heat exchangers;
- Primary circuit emergency makeup pumps;
- Diesel generators of a stand-by diesel generator power station and SEC systems;
- Emergency condensers and emergency cooling system pumps in low pressure unit;
- Demineralized makeup water tanks.

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 802.8 m^3/h flow rate, a second makeup pump station with two 56 m^3/h pumps, water lines, valves and instrumentation, and 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.

In any ANPP normal operation ESW pumps are in operation in each train with one pump in stand-by mode. Standby pump can be taken out of service for non-scheduled repair during maximum 72 hours. In case of failure of two ESW pumps in one train, the unit shall be transferred into cold shutdown mode [1].

At a rated level of 3.25 m the ESWS spray pools inventory comprises 5020 m^3. The minimum admissible level in spray pool is 3.0 m. If water level in the spray pool decreases to 3.0 m the operator shall transfer the unit into cold shutdown mode [1].

According to [2] and [3], after 72 hours after RLE the total water consumption from the ESWS spray pool will reach 3471 m3.

Note that water volume in spray pool between the minimum 3.0 m level and the level of 1,2 m when ESWS pumps will fail pumping is 3480 m^3.

Hence the conservative estimation shows that available minimum inventory of the water in the ESWS spray pools is sufficient for cooling the unit during 72 hours in case that pools makeup is not possible.

Two lines are designed to restore ESWS loses. Main water makeup lines for ESWS pools are the water lines of non-essential services water system (from the Stage II pump station).

In case of loss of outside power to ANPP the water will be (automatically) supplied from a discharge line of the circulation water supply system using an independent redundant line from the pump station for ESWS redundant makeup.

In case of an accident at ANPP, when ESWS water losses cannot be covered using the main or second 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 redundant line as well as for makeup using diesel engine pumps; this water volume ensures ESWS operation during more than one month.

Staff personnel actions and the process diagram for ESWS pools make-up with diesel pumps of circulating water system are given in the ESWS operating procedure.

System flow diagram is shown in Appendix A.

CONTROL SAFETY SYSTEMS

Instrumentation Systems Actuating Safety Systems

Instrumentation systems are designed to actuate, monitor and control safety systems while performing designed functions.

The systems comprise two independent trains; each train includes:

- Process parameters measurement circuit consisting of three independent measuring channels;
- Signals and alarms control circuit;
- Device safety or shutdown control circuit;
- Outlet relays power supply control circuit;
- Relay circuit for safety system actuation.

In normal operation conditions the system stays in the “ready to operate” mode (automatic standby).

Currently a new emergency monitoring system (EMS) is implemented. It completely meets the most recent requirements in this matter. The EMS is powered from two sections of the category I uninterruptible power supply. Regarding implementation of safety functions, it is designed in all emergency modes.

MCP Automatic System (AMCP)

AMCP system is designed to control the coolant flow rate through the reactor core and the primary circuit and the heat removal from the primary circuit via the second circuit; to generate signals acting though the reactor emergency protection system, to reduce the reactor neutron power in case the coolant flow rate decreases or the heat removal from primary circuit via second circuit is reduced.

AMCP system includes two similar instrumentation sets operating simultaneously and independently.

AMCP system determines the beginning of the loss of outside power (LOOP) mode and generates the following signals:

- EP-I is activated by the LOOP signal;
- Closing gate on inlet and outlet circulation water lines;
- Closing valves connecting the feedwater header and emergency feedwater header;
- SDV-C isolation and lock to open;
- LOOP alarm in the main control room.

Diesel Generator Load Sequencer (DGLS)

DGLS is designed for a step-by-step loading of the diesel generators connecting safety systems consumers to EDG of the emergency power supply system of Unit 2.

DGLS system includes two identical independent trains. Each train has a triple-set architecture that ensures functioning of the train in case of single failures and faults.

When started, DGLS forms the following signals (only in one train):

- Switching off, lock for switching on of the consumers in case of de-energizing of any (one) bus;
- Switching off, lock for switching on and automatic takeover in case of de-energizing of two buses;
- Switching off breakers at ZRB-1.2 (4RB-1, 2) buses;
- Start-up of diesel generators (1DG-1.2; 2DG-1, 2);
- Lock on operators' intervention for a 10 minutes period;
- Determines and implements a step-by-step-start up loading.

Seismic Protection System (SPS)

SPS is designed for automatic initiation of the reactor protection system in case of earthquake, switching off and restraining movement of the reactor building hoisters and fuel reloading machine, as well as for actuation of process equipment protections and interlocks.

SPS of Unit 2 is fitted with a set of sensors including six seismic sensors (installed in three stations).

Seismic sensors are sensitive to ground vibration in three orthogonal directions: two horizontal and one vertical. When the signal exceeds the value of ground acceleration of 50 cm/s^2 at any seismic detector in any direction, the signal generated by all three stations are sent to the emergency protection auxiliary relay. When the two signals out of three stations are simultaneously received, the actuating relay transfers signal to the reactor protection system.

OTHER CRITICAL SAFETY SYSTEMS

Primary circuit Make-up System

The primary circuit make-up system (in combination with boric acid solution make-up and supply system) is designed to perform the following:

- Preparation of boric acid solution of required concentration and supply to the consumers;
- Reactor transfer to subcritical state and maintaining it in this state;
- Reactor core reactivity control;
- Maintaining the rated level of the coolant in the reactor in transition modes, during and after an accident caused by break of the primary circuit integrity which does not caused by "large break";
- Supply of a boric acid solution to spent fuel pool.

The systems include:

- Two mixing tanks for the preparation and store of boric acid;
- Two boric acid solution service tanks (permanently filled with no less than 60 m^3 with minimum concentration of 40 g/kg);
- Three pumps for supply of boric acid solution from the service tanks;
- A heat exchanger to heat the boric acid solution;
- Four make-up pumps;
- A make-up water deaerator;
- A make- up water heater;
- Piping, valves and instrumentation.

System flow diagram is shown in Appendix A.

All system pumps are fed from Safety Class 2 power supply buses; power supply from SEC is also foreseen.

Boron and makeup water pumps specifications are shown in Tables 1-9 and 1-10.

Parameter	**MP-1**	**MP-2**	**MP-3**
Flow rate m^3/hour	8	20	20
Head kgf/cm^2	6.0	5.3	4.9
Power kW	13	13	11
Voltage V	380	380	380

Table 1-9: Boron pumps and their motors specifications

Parameter	Specification
Pump flow rate m^3/hour	1.6÷6.3
Suction pressure: min/max, kgf/cm^2	0.5/10
Discharge pressure: min/max, kgf/cm^2	160/195
Motor power, kW	55
Voltage, V	380

Table 1-10: Makeup pump specifications

Boron solution makeup and supply should be started by operator. Maximum time for the system startup is 5 to 7 minutes (based on operational experience). The make-up system switches to operation mode automatically.

Item No	Protection setpoints
1	Decrease of pressurizer level by 300 mm from the rated value initiates start-up of the make-up Pump I (one).
2	Decrease of pressurizer level by 500 mm from the rated value initiates start-up of the make-up Pump II (two).
3	Decrease of pressurizer level by 500 mm from the rated value initiates start-up of the make-up Pumps III and IV (all four pumps are in operation).
4	Increase of pressurizer level has by 300 mm above to the rated value results in initiation of the signal to switch off all make-up pumps and to lock switching the pumps on.
5	Temperature decrease in any loop below 130^0C with simultaneous increase of the pressure in primary up to 33 kgf/cm^2 initiates switching-off of all make-up pumps and locks switching the pumps on.
6	In case of decrease of the pressurizer level by 2560 mm from the rated value or decrease pressure in the primary circuit to 95 kgf/cm^2 ("large break") all make-up pumps are switch-off and with a lock to switch them on (emergency make-up pumps will start).
7	When SPS signals are generated (seismic impact) all operating make-up pumps are shutdown and power supply is locked.

Table 1-11: Makeup pump interlocks

Basic diagram of the system is given in Appendix A.

UNESSENTIAL SERVICES WATER SYSTEM

The system includes:

- Two Stage I water pump stations (“Sevzhur” and “Prud” stations);
- Cooling water pump house (CWPH);
- Inlet and outlet channels, water intake;

- Stage II service water pump station;
- Diesel engine pump station.

Main equipment cooling system uses the closed water circuit with water cooled in cooling towers. The system uses natural site height differential.

Cooling towers are located 6.5 m higher than the main building. The cooled is delivered from the tower to turbine condensers using an inlet channel and flows from the condenser under gravitational force through an outlet line to CWPH. Water losses from the system (evaporation, droplet entrainment, and blow down) are recovered by make-up using two lines coming from “Prud” and “Sevzhur” pump stations.

Stage I water pump stations are designed to make-up the inlet channel and to supply to the suction of Stage II pumps. Pump stations are equipped with the following:

- 3 pumps at “Sevzhur” pump station and 4 pumps at Prud pump station with a flow rate of 3000 m^3/h each. In normal operation and in case of emergency minimum 2 pumps remain operational at each pump station; one pump of each station is permanently operated. Pumps are powered by using 110 kV bus-bars. In case of disruption of the RA power grid, the pumps will be fed from a redundant hydraulic power station of the RA grid via a direct 110 kV transmission line.

Unit pump station (CWPH) is designed to supply circulation water from the outlet channel via cooling towers to the inlet channel.

Eight pumps are installed at the pump station. When unit is loaded the minimum of 6 pumps have to be available with 2 of them operated permanently. Pumps are powered from the auxiliary transformers (23; 24 TR).

The inlet and outlet channels of circulation water system are designed to ensure circulation supply of the cooling water through turbine condensers, to supply service water to the suction pumps of Stage II and to make-up ESWS spray pools. Rated level in the channels is 3.0 ± 0.2 m. Full water volume in the inlet channel is 77240 m3. Full water volume in the outlet channel is 36000 m3.

When the level in the channels drops to 2.5 m the operating personnel unloads the unit to the house load level.

When the level in the channels drops to 2.2 m, the operating personnel should transfer the unit into the cold shutdown mode (according to the requirement set in force by Technical Specification and Emergency Response Guidelines).

Water Intake

Four water treatment rotating screens (one per half condenser) are designed to remove algae and small objects from the water and to prevent condensers tubing plugging. There are sixteen gate valves installed (4 for each half condenser).

Three of four valves are installed in series upstream a respective half condenser and one is installed downstream a condenser. As the LOOP signal is generated and/or a seismic impact signal is registered (from SPS) three circulation valves installed in series in each half condenser (2 upstream and one downstream) will automatically close to prevent overfill of the outlet channel which might result in flooding of the turbine hall and the power plant site.

In case the first two circulation valves upstream a condenser fail the unit should be shutdown in 72 hours.

Circulation valves are fed from the reliable power supply bus of Class II. Circulation valves are qualified and certified as seismic resistant Class I equipment.

Stage II Service Water Pump Station

Stage II service water pump station designed to supply service water to unessential consumers in the reactor and turbine buildings and provide for ESWS make-up. There are four pumps installed at the pump station.

One pump is permanently in operation.

In case of emergency when all 4 Stage II service water pumps the cooling water to be supplied to unessential consumers in the reactor and turbine buildings can be delivered using Stage I pumps (this was successfully used during the 1982 fire at ANPP 2). The pumps are fed from ANPP auxiliary transformers.

Diesel-Engine Pump Station

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. In case ESWS fails the service water can be also supplied to SEC diesel-generator.

Five diesel engine pumps with a capacity of 580-1220 m^3/hour are installed at the pump station.

If four diesel-engine pumps fail it is allowed to operate the unit during 48 hours. After this period operator should transfer the unit into a cold shutdown mode. When the unit is in operation, a diesel-engine pump station should be available in standby mode. It should be started by the operators during 10-15 minutes in accordance with emergency response procedures.

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. Operation of diesel pumps could be required for make-up of essential loads cooling system and/or demineralized water tank.

No reserve tank or diesel fuel refilling (in case of accidents) is foreseen, as far as:

- Operation of one diesel pump for less than 8 hours is sufficient for the complete filling of one spray pond of ESWS;
- One diesel pump doesn't need more than 2 hours for complete filling of two tanks of demineralized water (DMWT-3 ,4 or DMWT-1,2).

Diesel equipment is cooled by air.

In case of emergency, when service water supply is necessary for ESWS make-up and to SEC diesel-generator, the required diesel-engine pump flow rate is about 140 m^3/h.

Diesel-engine pumps availability should be checked periodically with a frequency once per month.

Flow chart diagram of circulation water supply to Unit 1&2 turbine condensers and Stage II emergency service water supply is shown in Appendix A.

Information on seismic resistance of unessential loads service water system is detailed in §2.1.3.1.

Spent fuel pool cooling system

Cooling systems of the spent fuel pools of Unit 1 and Unit 2 are similar; they are designed to maintain the temperature regime in the SFP 1 and 2 and to clean suspended and ion-dispersed impurities from the SFP water.

The system (in each block) consists of:

- Two SFP cooling pumps (1, 2 SFP-CP-1, 2);
- Two heat exchangers (1, 2 SFP-HE-1, 2);
- Mechanical filter, anion and cation exchangers;
- SFP make-up pump (SFP-MP);
- Pumps to supply boron solution for cleaning (SFP-CSP): one pump at Unit 1 and two pumps at Unit 2;
- Piping, valves and instrumentation.

Pump motors are fed from 0.4 kV power buses of Safety Class II. In case of failure of auxiliary power supply system 2 SFP-CP -1, 2 and 1 SFP-MP pumps will be fed from SEC.

The ultimate heat sink for SFP water is ensured by the ESWS water supplied to SFP-HE heat exchangers.

In normal operation one SFP-CP pump and one SFP-HE heat exchanger are periodically switched on. The second pump and heat exchanger are in a stand-by mode.

In case of a complete failure of the SFP cooling system the SFP cooling is ensured using boron solution supplied from B-8/1(2) to 1, 2 SFP using pump 1(2) BSP – 1, 2 SFP-CSP-2 with discharge to B-8/1(2) via spent fuel pool overflow line.

The design does not foresee any SFP emergency cooling system.

<u>SFP-CP, SFP-CSP and SFP-MP pumps specifications</u>

SFP-CP and SFP-MP	SFP-CSP
Flow rate: 280 m^3/hour	Flow rate: 90 m^3/hour
Head: 4.2 kgf/cm^2	Head: 4.9 kgf/cm^2
$N_{el.motor}$: 75 kW	$N_{el.motor}$: 22-37 kW

The flow chart of the system is given in Appendix A.

Ventilation Systems

Ventilation systems are designed to maintain convenient conditions for personnel and to remove excessive heat emitted by operating equipment.

The essential ventilation systems from a safety standpoint include the following:

- V-1 and P-1, ventilation of central reactor building and SFP;
- V-2, ventilation of confinement;
- V-4 and P-4, ventilation of MCP and MLIV rooms;
- V-6 and P-6, ventilation of boron unit;

- R-1, recirculation ventilation of confinement;
- Ventilation of accumulator battery rooms;
- Ventilation of diesel-generator station rooms.

V-1 and P-1 systems work jointly. They are designed to ventilate the central reactor hall and spent fuel pool.

V-2 system is designed to create under pressure (~ 15-20 mm H_2O) in sealed unvisited rooms.

V-4 and P-4 systems work jointly. They are designed to remove excessive heat emitted in the room A-102 (MCP room) and to provide comfortable conditions for personnel working in these rooms.

V-6 and P-6 systems work jointly and are designed to remove excessive heat emitted by the equipment in boron room and to provide normal conditions for equipment in operation in the boron room.

R-1 recirculation system is designed to remove excessive heat emitted in the confinement (SG and MCP box) and to cool reactor cavity.

Air in R-1 system circulates in a closed loop as follows: SG and MCP box - suction duct - air coolers - fans - pressure duct - SG and MCP box. When cooling equipment and building structures in the SG and MCP room the air gets hot and the heat then transferred to the service water in the air coolers.

Ventilation of battery rooms is designed to remove excessive heat and aerosols from the battery rooms.

Ventilation of diesel-generator station rooms is designed to remove heat emitted by the operating equipment in DGPP building and to provide normal conditions for equipment operation.

Two sets of the abovementioned ventilation systems are installed: one being operational and another kept in a stand-by mode.

V-2 and V-4 systems are equipped with filtering units which clean the air by aerosol and iodine filters.

V-2, B-6, R-1 and DGPP ventilation systems are fed from Class II reliable power supply sources.

1.3.1. Reactivity control

The reactivity of unit 2 is controlled by using two independent reactivity control systems based on different operating principles.

The first reactivity control system is based on effect on movement of the absorber rods and fuel rods of the rector shim assembly (RSA). This is a system of mechanical movement of the RSA inside the core (for reactor power adjustment or SCRAM).

The second system affecting the reactivity is based on absorber concentration in primary circuit coolant (boric acid used as the absorber). The systems implementing this function are as follows: primary circuit make-up system and emergency core cooling system (ECCS).

RSA assemblies are controlled using reactor control and protection system (RCPS). When reactor is operated at rated power the RSAs assemblies are in the uppermost position with exception of the control group. Reactor emergency shutdown is ensured via insertion of all RSA assemblies into the reactor core with the maximum speed. In normal operation the boron control is ensured by the primary circuit make-up/blow-down system.

In case of emergency the boric acid concentration in primary circuit coolant should be adjusted using high pressure ECCS. ECCS is a two-train system automatically started with an alarm. Performance and sequence of the system startup is a function of an initiating event.

In case of emergency ECCS delivers boric acid solution to the primary circuit (this ensures transfer of the reactor core into subcritical condition also ensuring long term subcriticality after SCRAM) and performs a safety function of the reactor core cooling. Nuclear safety during reactivity control is ensured by permanent control of the neutron flux via monitoring regular parameters (neutron power, reactivity and period) and boric acid concentration in the reactor and in the ECCS trains as specified in the Technical Specification.

Reactivity Control at Unit Start-up

To achieve the reactor criticality a two-stage procedure is applied.

At Stage I all groups of RSA assemblies should be sequentially lifted to the uppermost position. To do this the boric acid concentration in the coolant should be no less than 12 g/kg; to ensure that after removal of all control assemblies to the uppermost position the reactor core remains deeply subcritical.

Simultaneously with removal of the control assemblies the operator should continuously monitor the position of control assembly groups, the boric acid concentration in the coolant, the rate of increase and intensity of the neutron flux and the reactivity.

At Stage II of start-up the criticality of the reactor is achieved via reducing the concentration of the boric acid in the coolant. In the course of decrease of the concentration of boric acid the working group of the controlling assemblies is used for potential reactivity change in a way that by the time of achieving the required power level the RSA working group is located some 100-150 cm from the bottom of reactor core. This stage should be also continuously monitored. The critical concentration of the boric acid is known in advance and the reactor safety is ensured both using RSA and also via boric acid control. The effectiveness of controlling assemblies working group is sufficient to reduce the reactor power from 100% to 0 without changing the concentration of boric acid in the coolant.

If any uncontrolled growth in reactivity occurs at the startup it will be immediately recorded or stopped either by reactor protection or by operator before any risk develops for reactor core.

Subcriticality Control in Spent Fuel Storage Pools

In spent fuel pools, subcriticality is ensured via two independent methods based on different concepts.

The first one is the design of spent fuel pool racks. Fuel assemblies installed under a protective water layer in the spent fuel pool racks are positioned with a fixed lattice of 225 mm entirely eliminating the chance of achieving criticality in a spent fuel pool in case the pool is filled with 100% pure water. The safety of this lattice for WWER-440 assemblies of maximum enrichment was verified via direct experiments using the critical assembly in Nuclear Power Institute named after Kurchatov.

To estimate the K_{eff} value for spent fuel pool racks when there is no boron in water a calculated breeding ratio was used for the infinite planar grid of "fresh" fuel assemblies using complete set of conservative material and geometrical parameters.

This system meets the condition of K_{eff} <0.95 [9, 10] for temperatures ranging from 27 to 127°C and, respectively, water density varying from 1 to 0.96 g/cm^3.

The calculation results are provided in publication [5].

The second method to maintain subcriticality in the spent fuel pools implies maintaining of the spent fuel pools with boric acid water solution, the acid concentration of no less than 12 g/kg.

Subcriticality Control at Fresh Fuel Handling

Within the NPP area the "fresh" fuel is transported in dry shipping containers designed for four assemblies in a set; the spacing between assemblies is 340 mm.

The assemblies to be loaded into reactor core are delivered from the "fresh" fuel storage to the reactor building in non-hermetic containers (with capacity 30 assemblies each) with a 225 mm lattice.

Publication [5] addresses all situations that may occur while handing "fresh" fuel.

The analysis showed that requirements of [9, 10] are met under all conditions.

Subcriticality Control in the Spent Fuel Dry Storage

Spent nuclear fuel of five and more years of storage is kept dry in NUHOMS 56 V storage facility.

Spent fuel (56 fuel assemblies) is placed in canister cells under a water layer in the spent fuel pool. Then the canister (in a shipping container) is lifted to the Central Hall of the reactor building; water is displaced from the canister using compressed air and the canister becomes completely dry. The canister lid is weld sealed and the canister is filled with helium and then transported for storage in a concrete module.

Subcriticality in the canister is ensured via special design of the canister.

The canister grid lattice is 180 mm. In accordance with calculations 24 canister cells are fabricated using the boron-alloy steel while 32 cells are made of stainless steel. Reference document [11] provides results of the analysis (to ensure subcriticality K_{eff} <0.95) far the cases that may happen when spent fuel is shipped to the storage facilities. In the most unfavorable case, when canister is filled with fresh maximum enrichment assemblies and canister is filled with 100% pure water the breeding ratio stays below 0.95

Protection of the plant personnel in case of radiation accident

There are four shelters at the plant site for protection of the plant personnel. The shelter Nr 2 (crisis centre) is dedicated to the managers of the emergency response system.

The shelters Nr 1 and 3 are for personnel working at the site during accidents and for emergency teams.

The shelter Nr 4 is for personnel of the plant security service.

All shelters are provided with ventilation, power supply, water supply, and sewage systems.

Ventilation systems of the shelters operate in two modes: ventilation mode (I mode) and filtration mode (II mode).

In shelter Nr 3 there is a 24 kW diesel generator ensuring power supply of the shelters.

In case of radiological issue, the accidents are managed from the crisis centre (shelter Nr 2) and the communication systems of ANPP are used: operative communication means, remote communication means using cable radio and relay lines, long-distance telephone lines, plant communication, special communication means, cellular communication, internet, email.

The crisis centre is provided with all required documentation and technical means for management of accident response.

1.3.2. Heat transfer from reactor to the ultimate heat sink

1.3.2.1. ALL EXISTING HEAT TRANSFER MEANS / CHAINS FROM THE REACTOR TO THE PRIMARY HEAT SINK (E.G., SEA WATER) AND TO THE SECOND HEAT SINKS (E.G., ATMOSPHERE OR DISTRICT HEATING SYSTEM) IN DIFFERENT REACTOR SHUTDOWN CONDITIONS: HOT SHUTDOWN, COOLING FROM HOT TO COLD SHUTDOWN, COLD SHUTDOWN WITH CLOSED PRIMARY CIRCUIT, AND COLD SHUTDOWN WITH OPEN PRIMARY CIRCUIT

The unit 2 of ANPP has two circuits WWER-440 (V-270 type) reactor installation. The reactor installation includes a reactor, pressurizer, six circulation loops with PGV-4 (SG) horizontal steam generators, GSN-317 (MCP) main circulation pumps, main circulation loop (MCL) and main loop isolation valves (MLIV).

During power unit operation, the heat is transferred from the reactor via MCL using MCP to steam generators, where it is transferred to second circuit water (coolant), transforming it to saturated steam. From the steam generators the steam is directed to the turbines. Downstream turbines steam enters turbine condensers, where during condensation the residual heat energy is transferred to the circulation water (the primary heat absorber). In its turn the circulation water is cooled in cooling towers exchanging heat energy with atmospheric air.

When the unit is shutdown for refuelling the unit is cooled in two stages. At Stage I cooling is ensured using steam discharge from SG into atmosphere via SDV-A (or steam discharge to condenser through SDV-C, if vacuum is maintained in the condenser) until pressure in the MSL becomes equal to 10 kg/cm^2 and then in accordance with the following pathway:

- SG - MSL - RCF - PC/PCC and afterwards returns to the SG using cooling pump (CP).
  RCF: reducing cooling facility

PC: process condenser

PCC: process condenser cooler

At Stage II (water-to-water heat exchange) cooling pathway is as follows:

- SG - MSL (or individual cooling pipeline) - RCF - PC (PCC) - CP - SG.

In case of hot shutdown heat is removed from the reactor using SG steam discharge to air via SDV-A and SG make-up from the normal or emergency water supply system.

In case of cold shutdown heat will be removed from the reactor core to the second circuit using primary circuit coolant natural circulation. Heat from the second circuit to the ultimate heat sink will be removed using Stage II pathway:

- SG - MSL (or individual cooling pipeline) - RCF - PC (PCC) - CP - SG.

In case of reloading the residual heat is removed from the core following the same approach, using two SG (one in operation and one stand-by).

In case the residual heat removal from reactor core using second circuit is lost or during reloading the reactor core heat is accumulated in water of boron solution tank (B-8/2) using the following procedure:

- Reload pool makeup is ensured using boron pumps or ECCS and water is discharged into B-8/2 through a discharge pipeline in the SFP.

For emergencies the design also provides for safety systems (components), which ensure residual heat removal from reactor core such as follows:

- Primary Circuit emergency make-up system (ECCS);
- Reactor core high pressure (HP) and low pressure (LP) emergency cooling system;
- Steam generators emergency make-up system;
- Primary Circuit overpressure protection system;
- Second Circuit overpressure protection system;
- Steam generators auxiliary feedwater system (using diesel-engine pump).

Reactor core HP and LP emergency cooling system is designed for cooling the reactor in case of the emergency modes which prevent operation of the steam generators water supply system caused by earthquakes and other hazards. During an earthquake the demineralized water will be supplied from the storage tanks to the steam generators by high pressure pumps while the steam is discharged to the atmosphere using SDV-A or SG SV (make-up/discharge). When SG pressure decrease to 5 kgf/cm2 make-up/discharge cooling is changed to close loop circulation of the second circuit coolant. Steam from SG enters the heat-exchanger of the LP emergency cooling system, gets cooled using ESWS water and then pumped back to SG by HP pumps.

In case of LOCA accidents that cannot be compensated by primary circuit make-up system, the emergency core cooling system (ECCS) is activated. The ECCS ensures supply of boric acid to reactor core in order to maintain the required inventory of the coolant in the primary circuit providing for reactor core cooling, residual heat removal and reactor core sub criticality control through make-up of boric acid solution into primary circuit.

If heat removal from the primary circuit using the second circuit is impossible, the emergency core cooling is ensured using the “feed and bleed” procedure feeding primary circuit by the ECCS, whereas primary circuit pilot operated relief valves (PZ PORV) installed at pressurizer are used for discharge.

The second steam generator makeup system is designed to make-up steam generators in case the unit is exposed to blackout as well as in case the steam generator water supply systems fail. The system includes a diesel-engine pump designed to supply water to steam generators from the demineralized water tanks (two tanks with 500 m3 capacity each).

In this case the residual heat from the reactor core is removed, using the following pathway:

- Primary circuit heat is transferred to the second circuit via natural circulation of the primary circuit coolant - SG steam is discharged into atmosphere using SDV-A or SG SV – SG the make-up is made by diesel-engine pump.

1.3.2.2. LAY OUT INFORMATION ON THE HEAT TRANSFER CHAINS: ROUTING OF REDUNDANT AND DIVERSE HEAT TRANSFER PIPING AND LOCATION OF THE MAIN EQUIPMENT. PHYSICAL PROTECTION OF EQUIPMENT FROM THE INTERNAL AND EXTERNAL THREATS

According to the design, the equipment and pipelines of routine cooling system and emergency steam generator make-up are located inside the turbine building at the following elevations:

- Make-up and emergency make-up electric pumps -1.8 m;
- CP, PC, PCC 0.0 m;
- SDV-A +14.7 m.

Feedwater and main steam pipelines are designed (with margin) for MCE having a ground acceleration of 0.35 g.

The water of the unessential service water system cooling system is used as a coolant. It is also possible to supply cooling water from ESWS to the PC and PCC.

The systems process flow chart is shown in Appendix A.

ECCS, HP and LP cooling systems equipment are located in the B-001/1 room (Unit 2 boron room) at elevation of -8.7 m. Boric acid tank is also installed in the B-001/2, at elevation of -7.9 /- 7.3 m. Backup demineralized makeup water tanks (BDMWT) are installed outdoors at 0.0 m elevation.

All system equipment (except DMWT) is designed (with a margin) to withstand the MCE.

Service water of the ESWS is used as a coolant. Ventilation systems (V-6 and P-6) are designed to remove heat produced by equipment in the B-001/1 room.

System flow chart is shown in Appendix A.

Diesel-engine pump for SG second make-up is located in the B-001/1 room (Unit 1 boron room) at elevation of -8.7 m.

System equipment and piping is designed to withstand MCE (with a margin).

1.3.2.3. POSSIBLE TIME CONSTRAINTS FOR AVAILABILITY OF DIFFERENT HEAT TRANSFER CHAINS, AND POSSIBILITIES TO EXTEND THE RESPECTIVE TIMES BY EXTERNAL MEASURES (E.G., RUNNING OUT OF A WATER STORAGE AND POSSIBILITIES TO REFILL THIS STORAGE)

When the routine cooling system is in operation, there is no time limit set for unit cooling. The system failure may be caused by blackout and/or seismic impact resulted in loss of the system integrity. In case there is a complete power blackout of the unit, but with the system integrity remained intact the reactor cooling is performed in accordance with the following pathway:

The fifth DG connected to SEC is started, a cable (from SEC) is connected to one of the ESWS pumps, a cable (from SEC) is connected to one of CP pumps, ESWS and CP pumps are started and routine cooling system is activated. All switching operations are to be performed by operator in accordance with the procedures. Cables from SEC to services are laid and phased. The maximum time of required for establishing connections does not exceed one hour. The reactor cooling time is limited by the diesel fuel tank capacity for DG in SEC system. Available diesel fuel stock in DG room is sufficient to operate during at least 50 hours.

In case of hot or semi-cold shutdown emergency core cooling the time depends on three factors: the inventory of the demineralized water, power supply and inventory of the diesel fuel kept in the tank to operate SG diesel-engine pump. When electrical power is available (through normal or emergency power source), the time to maintain the reactor in hot or semi-hot state is unlimited.

When only the reserve power supply is available (from DGS), the time for maintaining the reactor in hot condition is limited by the diesel fuel inventory at ANPP. The available amount of diesel fuel (300m^3 emergency reserve and 50m^3 in two DG compartments) ensures operation of four DGs at nominal power for at least 15 days.

Also, it should be noted that maintaining the reactor in hot condition for more than 24 hours could result in failure of the sealing of the MCP and leak of the primary circuit.

In case of plant blackout, without leaks of demineralized water, time for removal of residual heat from the core is limited to 5 days (1000m^3 of demineralized water of SG auxiliary diesel make-up system is used). Also it should be noted that the reserve of diesel fuel for the diesel pump ensures operation of the diesel for one day.

In case of seismic event, when only one emergency feedwater seismic pump (EFWSP) is in operation, practically there is no time limitation for maintaining reactor in hot or semi-hot condition, as far as:

- Operation of EFWSP doesn't require cooling water;
- Demineralized water tanks have make-up line from service water, and service water could be supplied with diesel pumps of the circulating water system.

A problem could occur in maintaining hot condition of the reactor due to the failure of the MCP's sealing. Calculation from OKB Gydropress (General designer of ANPP) Nr270.200.D5 of 1995, in the frame of the scenario of seismic event and black-out of the unit, shows that in case of maintaining the reactor in hot condition (primary circuit temperature above 260 °C), a failure of MCP's sealing is possible after 24 hours, when temperature of primary circuit is above 260°C. This is proved by tests carried-out in 1997 by NTTs ENERGONASOS TsKBM (manufacturer of MCP) on D500 mock-up.

Thus, the emergency operating procedures of the plant related to reactor transfer to safe condition, as a main mode, assume reactor transfer to semi-cold shutdown (<150°C) in case of seismic event and black-out of the unit.

For refuelling in case the procedure of core heat accumulation in the boric acid storage tank water is used, time for the procedure implementation is limited due to temperature increase in B-8/2, which could result in failure of boron pumps supplying the solution in refuelling pond. Time before water temperature increase in B-8/2 up to 80°C (from 60°C) is approximately 16 hours (without B-8 cooling).

1.3.2.4. AC POWER SOURCES AND BATTERIES THAT COULD PROVIDE THE NECESSARY POWER TO EACH CHAIN (E.G., FOR DRIVING OF PUMPS AND VALVES, FOR CONTROLLING THE SYSTEMS OPERATION)

Unit electric generators and the external grid are AC sources to supply power to the equipment designed for routine cooling. On the other hand there is a cable available to connect SEC with the pump; the cooling water can be supplied to PC and PCC from ESWS whose pumps are fed from Class II buses, and SDV-A are fed from the Class I bus. All pumps (EMP, ESWP, EFWP, ECP, MP except SFP-MP) required in different modes of emergency core cooling are fed from Class II buses. Control systems for these safety systems are fed from Class I buses. In addition there are cables led to the following pumps EMP, ESWP, EFWP, ECP and SFP-MP from SEC.

The detailed power supply system of ANPP is provided in § 1.3.5.

1.3.2.5. NEED AND METHOD OF COOLING EQUIPMENT THAT BELONG TO A CERTAIN HEAT TRANSFER CHAIN; SPECIAL EMPHASIS SHOULD BE GIVEN TO VERIFYING TRUE DIVERSITY OF ALTERNATIVE HEAT TRANSFER CHAINS (E.G., AIR COOLING, COOLING WITH WATER FROM SEPARATE SOURCES, POTENTIAL CONSTRAINTS FOR PROVIDING RESPECTIVE COOLANT)

Equipment for routine unit cooling system (PC, PCC, CP and EFWP) is cooled using unessential service water system. EMP and ECP units are cooled by essential services water system.

ESWP, EFWS, MP and SFP-MP pumps are cooled using air and by pumped water, so no cooling water from other sources is required.

Diesel-engine pump of SG second make-up system is cooled by air.

1.3.3. Heat transfer from spent fuel pools to the ultimate heat sink

Two identical cooling systems for spent fuel pools are designed to maintain the temperature in the pools. Each system (Units 1 and 2) includes two pumps (SFP CP) and heat exchangers (SFPHE) and one pump to fill the spent fuel pool (SFP-MP). In normal operation conditions the following pathway is used to cool down water in the SFP: SFP - SFP CP –SFP HE - SP.

ESWS water is used as coolant (the ultimate heat sink). No emergency cooling systems are foreseen for spent fuel pools in the ANPP design.

In case of emergency dealing with malfunction in SFP cooling system, the following pathways are provided for water cooling:

- B-8/2 – 2 SFP-MP (or 2 CPP-1, 2) – 2 SFP – B-8/2;
- B-8/1 – 1 SFP-MP (or 1 CPP-1, 2) – 2 SFP – B-8/2;
- B-8/2 – 2 SFP-MP (or 2 CPP-1, 2) – 1 SFP – B-8/1;
- B-8/1 – 1 SFP-MP (or 1 CPP-1, 2) – 1 SFP – B-8/1.

SFP cooling pathway for Units 1 and 2 is shown in Appendix A.

SFP cooling equipment of Units 1 and 2 is located in the reactor building:

- 1 SFP CP – 1.2, room V-110/1, elevation +2.7 m;
- 1 SFPHE – 1.2, room V-241/1, elevation +6.3 m;
- 1 SFP-MP, room B-001/1, elevation - 8.7 m;
- 2 SFP CP– 1.2, room V-110/2, elevation +2.7 m;
- 2 SFPHE – 1.2, room V-241/1, elevation +6.3 m;
- 2 SFP-MP, room B-001/2, elevation - 8.7 m.

All equipment and piping of SFP cooling system in Units 1 and 2 are designed to withstand MCE. The power supply to the pumps is organized as follows:

- 1 SFP CP – 1, 2 are fed from reliable power supply buses of Class II. Also 2 SFP CP-1/2 can be fed from SEC;
- 1, 2 CPP– 1, 2 are fed from the normal power supply;
- 1, 2 PFP are fed from reliable power supply bus of Class II. As an alternative, 1 SFP-MP can be fed from SEC.

In normal operating conditions the SFP cooling time is not limited. In case of incident when SFP cooling pathway (B-8 - pump - SP - B-8) is applied and external power source fails, the time of SFP cooling is limited by capacity of the diesel fuel emergency tank of the EDG that ensures operation at the rated load during about 50 hours. When this SFP cooling pathway is used the temperature in B-8 raised up to 80^{o}C during about 3.3 days in Unit 2 and about 19 days in Unit 1.

It should also be mentioned that the operation of the SFP cooling system is required for cooling of the storage pool during normal operation (from 60°C to 40°C):

- For Unit 2, less than 2 hours per day;
- For Unit 1, less than 2 hours per 5 days.

1.3.4. Heat transfer from the reactor containment to the ultimate heat sink

1.3.4.1. ALL EXISTING HEAT TRANSFER MEANS / CHAINS FROM THE CONTAINMENT TO THE PRIMARY HEAT SINK (E.G., SEA WATER) AND TO THE SECOND HEAT SINKS (E.G., ATMOSPHERE OR DISTRICT HEATING SYSTEM)

The main equipment of the reactor installation is installed in a confinement area. The designed ventilation system provides removal of the excessive heat from the area and from reactor cavity. The air is circulated in a closed loop. The system consists of five ventilation units and air coolers. Ventilation unit motors are fed from 0.4 kV bus of Class II. Unessential service water system is used as a coolant for air coolers. In case of emergency involving coolant leaks in primary circuit the confinement is cooled using a spray system. The spray system pump motors are fed from the Class II bus. ESWS water is used as spray unit cooling water. Emergency systems to cool the confinement cooling were not provided in ANPP design.

1.3.4.2. RESPECTIVE INFORMATION ON LAY OUT, PHYSICAL PROTECTION, TIME CONSTRAINTS OF USE, POWER SOURCES, AND COOLING OF EQUIPMENT AS EXPLAINED UNDER 1.3.2.

Ventilation units and air coolers of the confinement are located in the reactor building at elevation of -1.8 m.

Spray system equipment is installed in the boron room. In case power supply is lost both from internal and external power sources, the system of the recirculation ventilation system of the confinement cannot cool the confinement due to unavailability of the water supply from the unessential service water system.

Spray system operation from EDG may be limited by the capacity of diesel fuel tank (50 hours).

1.3.5. AC Power supply

1.3.5.1. OFF-SITE POWER SUPPLY

Armenian NPP is connected to the power grid of the Republic of Armenia (RA) which is connected to the power grid of Iran, using two 220 kV transmission lines with 400 MW transmission capacity each.

Armenian NPP is connected to the RA power grid using five 220 kV and six 110 kV transmission lines. These lines are used to connect NPP to five substations. These 220 and 110 kV lines are connected via bus bars of 220 and 110 kV switchyards. Bus bars of 220 and 110 kV switchyards are interconnected through a coupling autotransformer.

Switchyards power supply is provided using two operational transformers of 15.75/6 kV. Each transformer is 25 MVA. Two redundant 110/6 kV and 35/6 kV are provided. Each transformer is 32 MVA.

Two switchyards with coupling autotransformer provide reliable connections to the power grid.

Off-site 220 and 110 kV transmission lines are the backup power sources for the ANPP 220 kV and 110 kV switchyards. The power is supplied both from hydroelectric power stations and conventional plants. Power can be provide to the auxiliary 6/0.4 kV buses both through the normal circuit using the main transformers 3T/4T and tapped transformers 23T/24T, or through the backup circuit using start-up transformers 1TR and 2TR. ANPP auxiliary power supply diagram is shown in Appendix A.

In accordance with Guidelines for RA Power Grid Dispatching in case of the RA power network blackout ANPP power supply will be recovered from a standby hydroelectric power station (50 MW) using a direct 110 kV transmission line.

In case of RA power network blackout, the Guidelines provide for ANPP power supply recovery within 3-5 minutes regardless of availability of ANPP communication with the central dispatcher service of the RA power network.

ANPP power supply recovery from external sources is also possible using 110 kV and 220 kV transmission lines connected to the RA power network or to the network of the Islamic Republic of Iran.

During the entire period of ANPP operations there was only one case of loss of outside AC supply from external and internal sources (LOOP). This happened during the 1982 Unit 1 fire. Three hours after occurrence of the accident the power supply was restored to one primary circuit emergency cooling pump using a temporary cable from the diesel generator and thus primary circuit make-up system was powered. About 4 hours later ANPP power supply from the external sources was recovered.

The experience of power supply from diesel generators using a temporary cable gained during ANPP fire resulted in installation of the SEC system.

1.3.5.2. POWER DISTRIBUTION INSIDE THE PLANT

Diagrams of auxiliary power supply to ANPP safety related systems (SRS) are shown in Appendix A.

To supply power to auxiliary ANPP SRS the following was designed:

- 6 kV buses, 3RB-2 and 4RB-2;
- 0.4 kV buses, 25BNN and 26BNN, Class II;
- 0.4 kV buses, 28 NA and 29 NA, Class I.

Under normal conditions 6 kV buses (3, 4 RB-2) are fed from auxiliary transformers (aux.) 23T and 24T.
In case of blackout two parallel DGs are automatically started and connected to each 6 kV bus.

The 0.4 kV buses (25, 26 BNN) are fed from 6 kV buses.

The 0.4 kV buses (28, 29 BNN) are powered by 0.4 kV 25 BNN and 26 BNN buses. To ensure uninterruptible power supply to 28 NA and 29 NA buses from 25 BNN and 26 BNN, reversible motor generators are working on "recharge" mode. In case of LOOP, the reverse motor generators switch to battery mode and supply 0.4 kV AC to Class I consumers. Safety Systems power supply uses a two-train concept with physical and galvanic separation. Cables from transformers 23T, 24T and 1 TR and 2

TR routed in four cable ducts are divided into two circuits, at 3.6 m underground, are connected to CDD 6 kV switchgear at the 0.0 m elevation. Similar two-train concept is used to supply voltage to 0.4 kV buses. Then power is supplied to Safety Systems 6/0.4 kV consumers using dedicated supply cables.

Power cables in the tunnels and on trays are coated with a fireproof coating for cables. Every 30 m the cable trays (tunnels) are equipped with fire stops in order to prevent fire propagation along the cables.

Power cables from the EDG to CDD 6 kV switchgear are routed in individual underground trays with two channels per tray. These cables are also coated with fireproof coating and equipped with fire stops.

Power cables from SEC are routed in an individual underground tray coming to the cable tunnels and then via cable tunnels and cable shafts are connected to the consumers.

All power distribution cabinets are installed at the ground level.

All electrical equipment of emergency power supply systems of Class I and II providing power supply to Safety Systems are seismically resistant to 0.4 g ground acceleration.

Fire (inflammation) in cable tunnels and shaft is automatically extinguished using a foam fire-fighting system.

Cable tunnels are equipped with spouting chutes to remove the water from tunnels by using gravity directed flow.

1.3.5.3. MAIN ORDINARY ON-SITE SOURCE FOR BACK-UP POWER SUPPLY

Diesel generators are used at ANPP as an on-site emergency power supply system of Class II.

Class II emergency power supply system is designed to provide electric power for Class II essential consumers both in normal operation and in case of the loss of the outside power supply. The system consists of two independent trains. Each train includes:

- Two diesel generators of 1500 kW each;
- 6 kV bus (3 RB-2 and 4 RB-2);
- 0.4 kV buses (25 BNN and 26 BNN, 69 NO and 70 NO);
- Transformers 6/0.4 kV (65 T, 66 T, 69 T and 70 T);
- VARTA rechargeable battery of 450 A/hours.

Diesel generators are automatically started at power failure signal in 6 kV buses (3 RB-2 and 4 RB-2). Diesel generators start-up time and power supply to 6 kV buses is 25-30 s.

DG in each train (1DG-1, 1DG-2, Train I and 2DG-1, 2DG-2, Train II) is started only if the voltage failure is detected in its 6 kV section.

The diesel generator load sequencer (DGLS) automatically starts of respective equipment subject to the operational mode. Each train of diesel generator plant is equipped with tree diesel fuel tanks: two feed tanks (one per diesel generator) and one reserve fuel tank for both diesel generators.

Diesel fuel stock at diesel generator plant provides generators with full-load operation during at least 30 hours. In addition, 300 m^3 of emergency diesel stock is available at the ANPP site. Emergency diesel fuel stock will provide a full-load operation of both trains (4 diesel generators) for at least 15 days.

It should be noted that in any initiating event that does not involve primary circuit leak more than DU-32 and external power supply failure operation of only one DG is sufficient to maintain reactor under safe condition. Hence the time of performance of one train of emergency power supply of Class II at above initial conditions can be twice as long.

Class II Emergency Power Supply system is designed as Class I seismic resistance. It is qualified to withstand 0.4 g ground acceleration.

- Emergency diesel tanks (300 m3) and their piping is certified for Class II seismic stability and their failure cannot be ruled out in case of MCE;
- DG are cooled using the essential services water supply system;
- Diesel generators in each train are installed in different rooms (compartments) of EDG. Trains I and II of emergency power supply system are completely separated physically;
- Below zero the EDG compartments are equipped with gravity water drainage system.

1.3.5.4. DIVERSE PERMANENTLY INSTALLED ON-SITE SOURCES FOR BACK-UP POWER SUPPLY

The first emergency auxiliary power supply source for ANPP is 50 MW Arguel HPP. In case of failure of ANPP external power supplied by 220 kV and 10 kV transmission lines (e. g., in case of the RA power system disruption) Arguel HPP is started on power network dispatcher's order and ANPP is powered by a direct 110 kV transmission line (Bzhni). Dispatching service and ANPP operator actions are described in details in guidelines.

The second emergency auxiliary power supply source for ANPP is provided by an on-site diesel generator plant (refer to § 1.3.5.3).

ANPP design provides no special system to protect confinement from destruction after reactor core meltdown.

1.3.5.5. OTHER POWER SOURCES THAT ARE PLANNED AND KEPT IN PREPAREDNESS FOR USE AS LAST RESORT MEANS TO PREVENT A SERIOUS ACCIDENT DAMAGING REACTOR OR SPENT FUEL

Ultimate last resort power source designed to prevent severe accident is the second emergency cooling system (SEC).

SEC system is designed to supply power to essential consumers of the unit in case of beyond design basis severe accidents, when 6 kV and 0.4 kV buses voltage fails and power supply from working and emergency sources is not possible.

SEC system includes:

- Emergency transformer (1 TR);
- 1500 kW diesel generator (DG);
- Outdoor switch-gear (CDDN-6 kV);

- Outdoor transformer substation (KTPN-0.4 kV).

The system is started manually by the shift supervisor. The SEC system start-up procedure includes:

- Power cable disconnection from the consumer;
- SEC cable connection to the consumer. SEC power cables are laid to respective consumers, they are phased and fixed to the buildings walls, where respective consumers are located;
- Diesel generator start-up and supply to CDDN and KTPN 6/0.4 kV buses;
- Consumer's start-up in a one-by-one manner.

SEC system preparation and start-up time equals to 30-40 minutes (less than one hour).

SEC system cables are routed to the following consumers:

Two emergency boron make-up pumps, two ESWS water pumps, Fire Fighting Water Pump (FFWP), Fire Fighting Foam Pump (FFFP), emergency feedwater pumps (EFWP) in the turbine hall; primary circuit make-up pump; boron supply pumps (MP and SFP-MP); Safety Group II valves' electric cabinets required for cooling; spent fuel pool cooling pumps (SFP CP), reactor building lighting electric cabinets, boron unit drain pumps, NFMS (emergency power supply), drain tank pumps (DTP). During design of the plant, no requirement was specified about the seismic qualification of the SEC system.

Additional reliability analysis of SEC system performed after Fukushima NPP accident revealed that, when 6 kV buses fail and no electric power can be provided by normal or emergency power sources, the pumps of the essential service water system would not start resulting in SEC system failure. In case of failure (due to any reason) of cooling water supply to essential consumers the whole Class II emergency power supply system will also fail (due to the same reason).

To improve reliability of the emergency power supply from internal sources a second cooling water line was designed and built to DG-4 (DG-4 is connected to SEC) from diesel-engine pump station for circulation water supply. After DG-4 start-up it will be possible to start the pump of essential services water system and to prevent a failure of safety systems by the same reason: the lack of cooling water.

Respective changes have been introduced into operational manuals.

Existing operational manuals on personnel emergency response are focused on prevention of design accidents development into the beyond design basis and on mitigation of consequences caused by the beyond design basis accidents.

The initiating event such as full blackout may assume a series of scenarios and respective responses from the operating personnel subject to revealed symptoms on systems and equipment status. The respective actions of the operating personnel are governed by the following documents:

- Manual on beyond design basis accident management at ANPP;
- Guidelines for ANPP electrical equipment failures response;
- Guidelines for the RA power network failures response;

- Guidelines for ANPP auxiliary power supply recovery in case of the RA power network disruption;
- Off-spec operational modes of Class I EPS;
- Off-spec operational modes of Class II EPS;
- Alarms response forms.

Operating personnel receives training in ANPP Training Centre and on-the-job training under supervision of an experienced operator. After the on-the-job training, the operating personnel passes the exam in the Regulatory Authority (ANRA) or in front of the Plant Commission (subject to job position), where his knowledge of rules, standards and operational guidelines as well as his practical skills are verified. After successful exams the operating personnel is allowed to work as an apprentice guided by an experienced operator. After successful on-the-job practical work the apprentice is allowed to work as an operator.

The operating personnel attend training and advanced training courses in TC on an annual basis; periodically the exams are held to verify the knowledge of rules, standards and operation manuals. The emergency response drills are organized on the monthly basis for operating personnel. Various emergency response drills assess the personnel preparedness to cope with emergency situations.

1.3.6. Batteries for DC power supply

The system is designed to supply power to safety related systems (SRS) of Reliability Group I and to ensure the conditions for their functioning.

Class I EPSS consists of two independent trains that comply with safety systems design approach.

Each train provides 100% of the capacity required for the safety system train.

Each train of Class I EPS includes:

- Earthquake resistant VARTA rechargeable battery with a capacity of 1500 Ah;
- Two reversible engine-generators: one in operation and one in stand-by;
- Two automatic transfer switch (ALT) thyristor devices;
- 0.4 kV bus (28 BNN, 29 BNN);
- DC electric board (BSPT-1. 2);
- Rectifier unit to recharge battery.

Power sources of Class I DC devices are rechargeable batteries. The design capacity of Class I EPS ranges from 3 to 3,5 hours at the maximum load without need of battery recharge.

There is also a DC electrical board available with its own battery "VARTA" with a capacity of 600 A hours.

Batteries are recharged as follows:

- When the unit is under load: from operational unit generators powered from auxiliary transformers;
- When the unit is on stand-by: from external power grid through standby transformers;
- When the unit is without power: from emergency diesel generators of Class II.

After Fukushima NPP accident the duration of power supply was assessed for the batteries feeding Class I DC and AC consumers without recharge.

Calculations were made for Unit Direct Current Board (UDCB-1, 2). The duration of power supply from UDCB-1 to DC and AC consumers was assessed with and without supply from the generator shaft seal oil pump (GSOP).

The two following scenarios were evaluated:

1) All consumers of 28 NA and 29 NA buses which may fed from both buses are connected to one bus;
2) 28 NA and 29 NA buses loads are equally distributed;

Calculation results are shown in Table 1-12.

	Scenario 1	**Scenario 2**
Time UDCB-1 operates with GSOP	5.4 hours	7.5 hours
Time UDCB-1 operates without GSOP	6.0 hours	7.8 hours
Time UDCB-2 operates with GSOP	8.8 hours	12.0 hours

Table 1-12: Calculation results for UDCB time operation

Results of calculations:

- The minimum duration of Class I EPSS operations without battery recharge is 5.4 hours;
- When GSOP load is removed from UDCB-1, UDCB-1 operates 0.6 hour longer;
- If 28 NA and 29 NA buses loads are equally distributed, UDCB-1 operation can be extended up to 8 hours.

1.4. Significant differences between units

Two identical power units, WWER-440 (V-270) reactor type, are installed at the ANPP site. The main difference between the units is that reactor of Unit 1 is in permanent shutdown mode without fuel in the reactor core. The spent fuel pool of Unit 1 is used to store spent fuel from Unit 2.

1.5. Scope and main results of Probabilistic Safety Assessments

The following probabilistic safety analysis (PSA) has been carried out for ANPP Unit 2.

- Level I PSA of internal initiating events;
- Level I PSA of external impacts;
- internal fire and flooding PSA;
- Seismic PSA.

Item No	Initiating event	Reactor core damage frequency, per year (Contribution, %)
1	Transients	2.08E-05 (39.4%)
2	Primary Circuit LOCA, equivalent diameter up to 265 mm nominal diameter	8.73E-06 (16.5%)
3	Primary Circuit LOCA, equivalent diameter above the 265 mm nominal diameter	8.54E-06 (16.2%)
4	Leaks from primary circuit to second circuit	6.90E-06 (13.1%)
5	Second circuit ruptures	6.39E-06 (12.1%)
6	Power failure	1.41E-06 (2.7%)

Table 1-13: Key outcomes of Level I PSA of internal initiating events

Total core damage frequency due to internal initiating events is 5.28E-05 per year.

Outcomes of Level I PSA of external impacts

For external impacts assessment the list of potential events was defined based on the following documents:

1. Records of external natural and industrial impacts on nuclear and radioactive hazardous facilities. PNAE G-05-035-94, Moscow, 1995
2. Draft Safety «External Events (Excluding Earthquakes) in Relation to Nuclear Power Plant Design» IAEA, 50-SG-D5.
3. Code of Regulations on Nuclear Power Plants Safety: Sites Selection for NPP. IAEA, 1990.

After the screening of the external initiating events on basis of the geographic location of the Republic of Armenia and the NPP site itself, the following initiating events were selected for further analysis in terms of impact on the NPP:

- Dust storms;
- Strong wind and tornado;
- Plane crashes;
- Snow;
- Low temperature;
- Rainfall.

Item No	Initiating event	Reactor core damage frequency, per year (Contribution, %)
1	dust storms	1.8E-05
2	strong wind and whirlwind	2.0E-06
3	plane crashes	1.7E-07
4	Snow	no impact
5	low temperature	no impact
6	rainfall	no impact
7	electric power fluctuation	no impact

Table 1-14: Key results of PSA for external initiating events

Results of Internal Fire and Flooding PSA

Total Core Damage Frequency caused by fire in ANPP rooms is 1.85 x 10^{-5}(1/year). Key contributors to reactor core damage are as follows: fires in cable tunnels, where the majority of the cables are running to pumps supplying water to primary circuit and/or second circuit.

Major fire inside the room (A-013/2), where pressure lines from primary circuit emergency make-up pumps are located, also results in substantial impact. Fire may cause a medium leak from primary circuit eliminating any chance of leak isolation or water supply to primary circuit.

Flooding analysis reveals that the risk of ANPP internal flooding is negligible. The probability is less than 1E-08/year.

Basic Outcomes of Seismic PSA

Total seismic core damage frequency is estimated equal to 7.93E-05/year.

The key scenarios resulting in reactor core damage are:

- Earthquake-related external power source failure: 42%;
- Earthquake-related building failure: 19%;
- Earthquake-related primary circuit rupture: 15%;
- Earthquake-related second circuit rupture that cannot be isolated: 12.5%;
- Earthquake-related transients: 7.5%;
- Strong seismic event (an earthquake with maximum ground acceleration over 0.8g): 4%.

At ground acceleration of 0.8 g a lot of systems, components and structures of the plant will fail (including EDG building, DG diesel tanks and demineralized water tanks) with a probability close to 1. That is why the probability of reactor core damage related to an earthquake at ground acceleration over 0.8 g is assumed to be close to 1.

Note:

- PSA for internal initiating events and fire PSA were reviewed by end of 2012;
- PSA for external initiating events and seismic PSA were implemented on basis of the PSA's model of 2007 for internal initiating events. Both are currently under reviewing process.

Level II PSA

Work on the PSA level II began at the end of 2011 and is scheduled for completion in the second half of 2013.

The results of Level II PSA of similar WWER-440 Power Units show that in terms of accidents having substantial impact on large releases a special attention should be paid to leaks from primary circuit to second circuit.

Based on the experience gained for similar power units the following activities should be implemented at the ANPP:

- To perform analysis, develop and justify accident management for leaks occurring from the primary circuit to the second circuit in order to minimize the number of actuation cycles of the second circuit safety valves which have discharge to the atmosphere;
- To improve the operating personnel preparedness in accident response through emergency drills using the full scope simulator;
- To revise and improve accident response guidelines in order to ensure reliable localization of a leak inside a faulty steam generator.

Review of PSA data on the ANPP revealed several drawbacks:

- The probabilistic analysis data available is dated of January 2006;
- No PSA data is available for low load and shutdown reactor state;
- No data is available on PSA Level II;
- The available PSA data does not consider ANPP systems and component upgrades and modernizations since 2006;
- The large modernization projects of unit 2 require revising the PSA model and additional calculations.

Measures to be taken:

1. Revise the PSA basic model as of January 2012 plant status;
2. Perform additional calculations and review the available PSA data;
3. Develop PSA for low load and shutdown reactor state;
4. Develop PSA Level II.

Currently all above issues are being addressed. The deadline to complete these activities is set as end of 2013.

REFERENCES TO CHAPTER 1

1. Nuclear Power Plants General Safety Regulations
   OPB-88/97 NP-001-97, Moscow, 1997
2. Nuclear Safety Rules for Nuclear Power Plant Reactor Facilities, NP-082-07, Moscow, 2007
3. Nuclear Safety Rules, SFP-CP-99, and Basic Sanitary Rules OSP-72/87, Moscow, 1988
4. Safety Investigation Report: Power Unit 2, ANPP, CJSC Armatom, Yerevan, 2008.
5. Feasibility Study: Safety Investigation of Vibration-Resistant Profiled Assemblies of 3.82% Medium Enrichment, Unit 2, Armenian NPP, 270-TP-024, OCB Gidropress, Moscow, 2009.
6. Performance Analysis: Reactor Core Reactivity Control Disturbance; Unit 2, Armenian NPP (V-270) RNC Kurchatov Institute, Moscow, 1995.
7. Manual for Accidents Analysis in NPP having WWER Reactors, IAEA, 1995.
8. Safety Analysis and Substantiation of Refurbishing Algorithm to Protect Reactor Facility in Case Steam Lines are Ruptured. 270.200.D6, OCB Gidropress, 2000.
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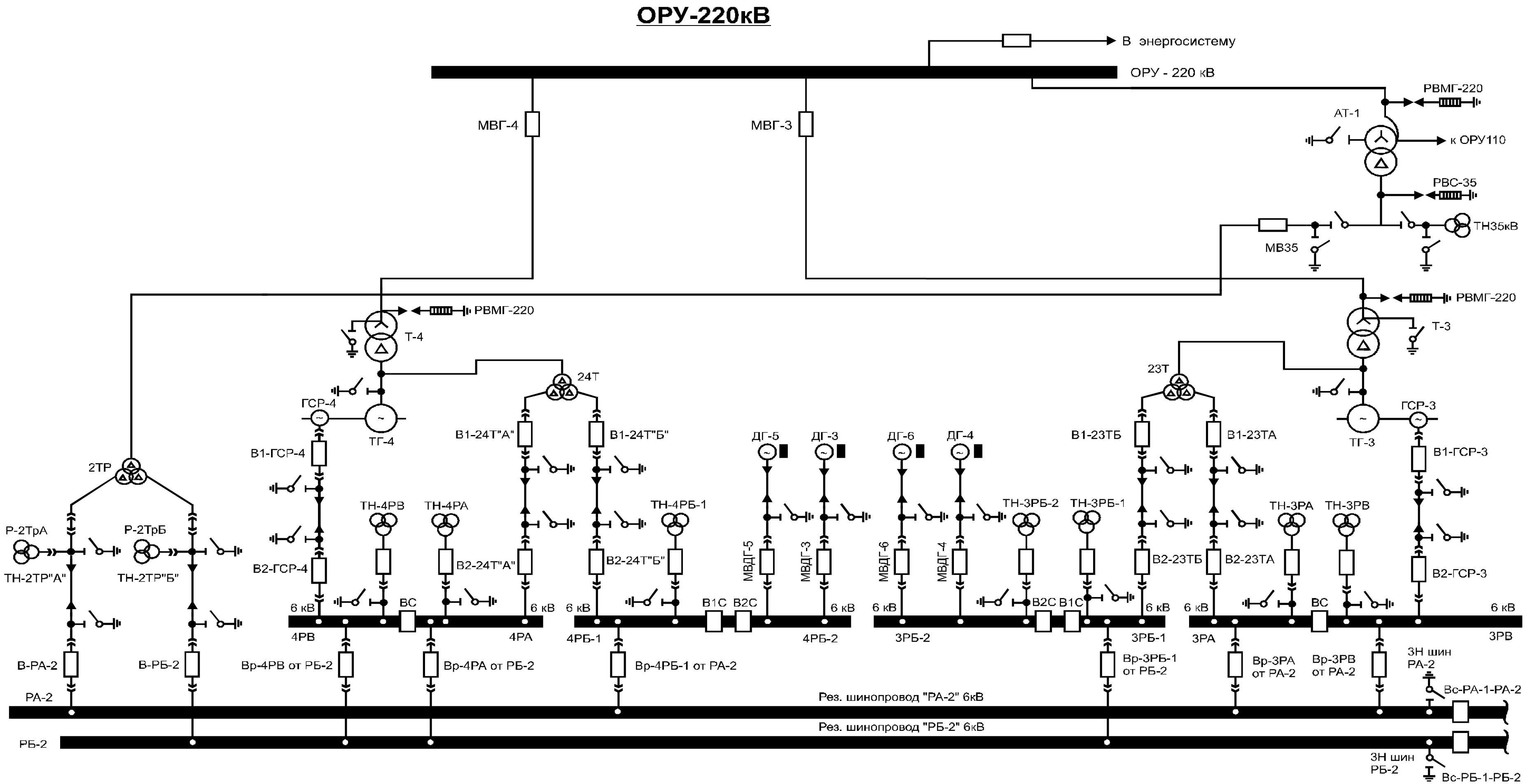
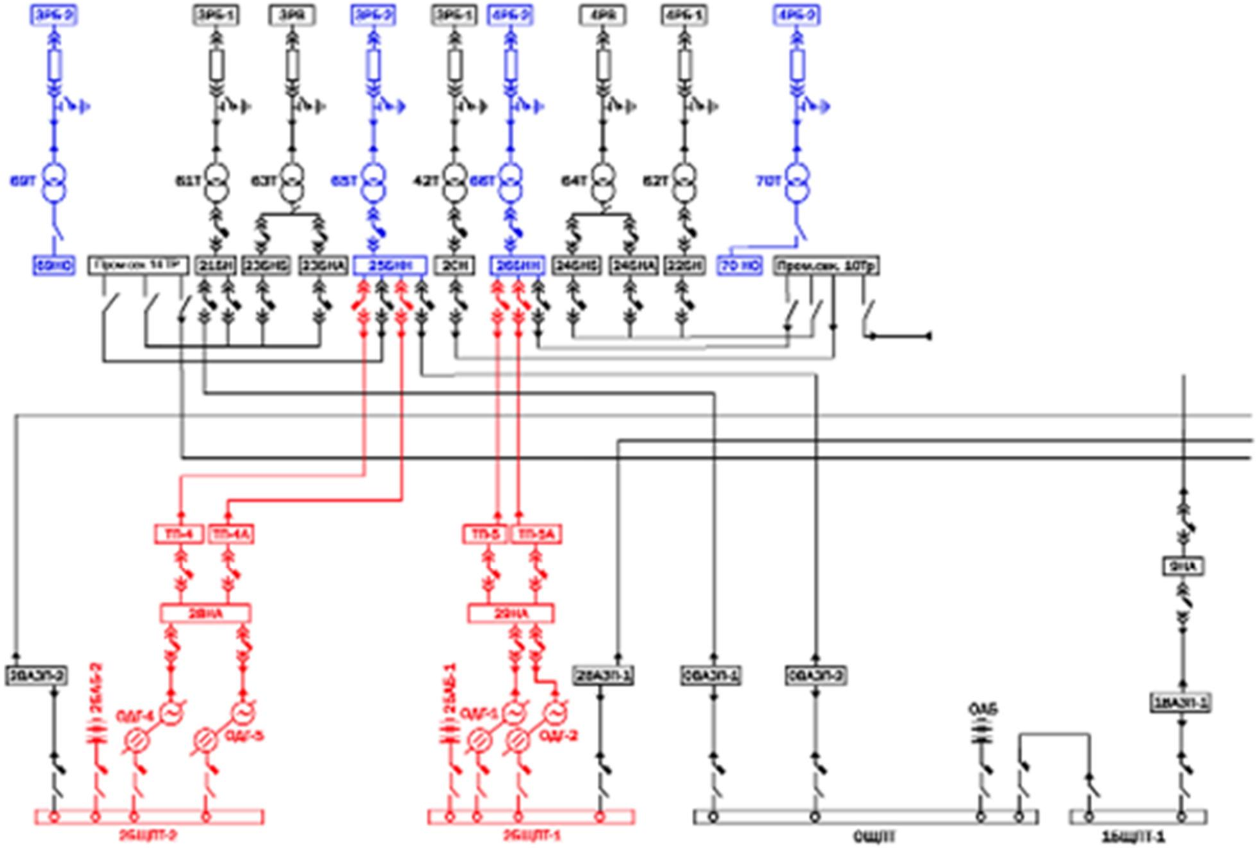
   NP-061-05. Moscow, 2005.
10. Safety Manual No 50-SG-D10. Nuclear Power Plants Fuel Handling and Storage Systems. IAEA, Vienna, 1985,
11. Medzamor NPP Analytical Safety Review. Storage NUHJVS. FRAMATOME. 2000.
12. Power Plant Loss of Outside Power Supply Case Study. UB.ETD.06.OYB-001, 2011.
13. Complete Loss of Internal and External Power Sources. Kiev, 2011.
14. Estimation of Time Reserve Available for Operator in Case SP Cooling is Malfunctioning. UB.ETD.06.OYB-001, 2011.
15. Process Procedures to Operate Power Unit, Unit 2, ANPP, Reactor WWER-400 (V-270).
16. ESWS Feasibility Study Including Environmental and Radiation Safety. Technical Report A-67217. NIIAEP, 2000.
17. Verification of Criteria of 72-Hour Period of Essential Services Cooling following RLE, ANPP, 2005.

APPENDIX A

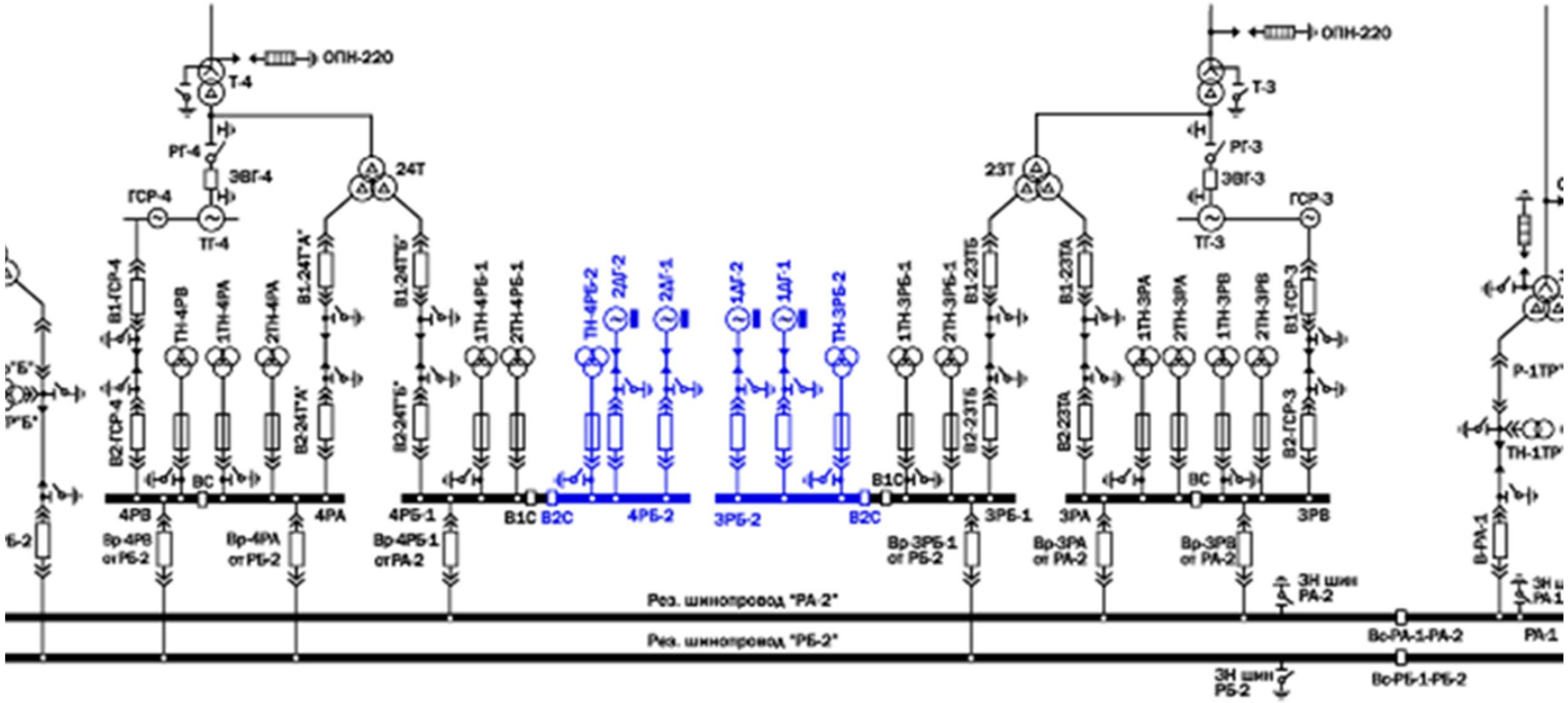
FLOW DIAGRAMS OF SYSTEMS IMPORTANT FOR ANPP SAFETY

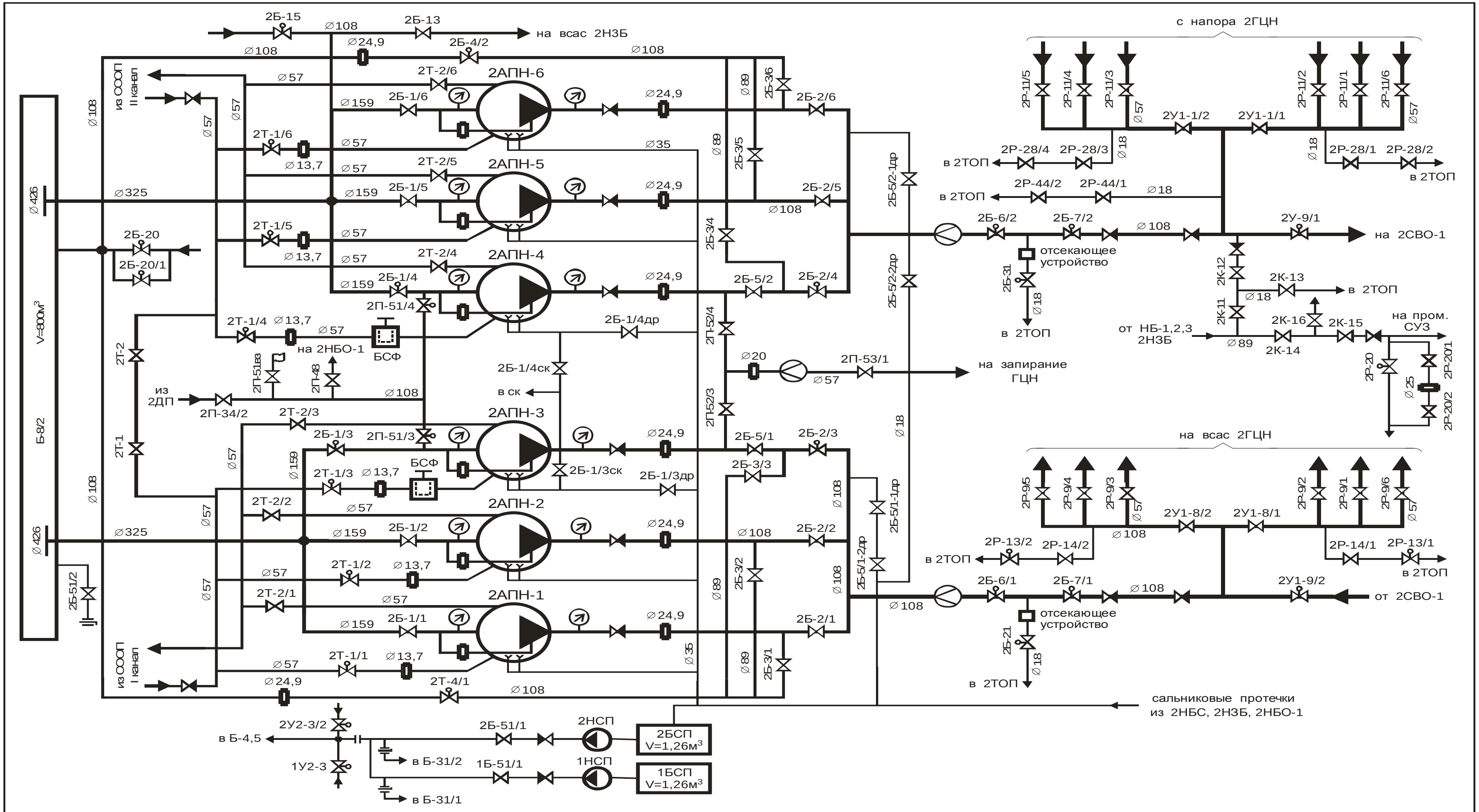
PRINCIPAL DIAGRAM OF 220 KV SWITCH YARD AND 6 KV POWER SUPPLY SYSTEM OF UNIT 2





6kV POWER SUPPLY OF UNIT 2





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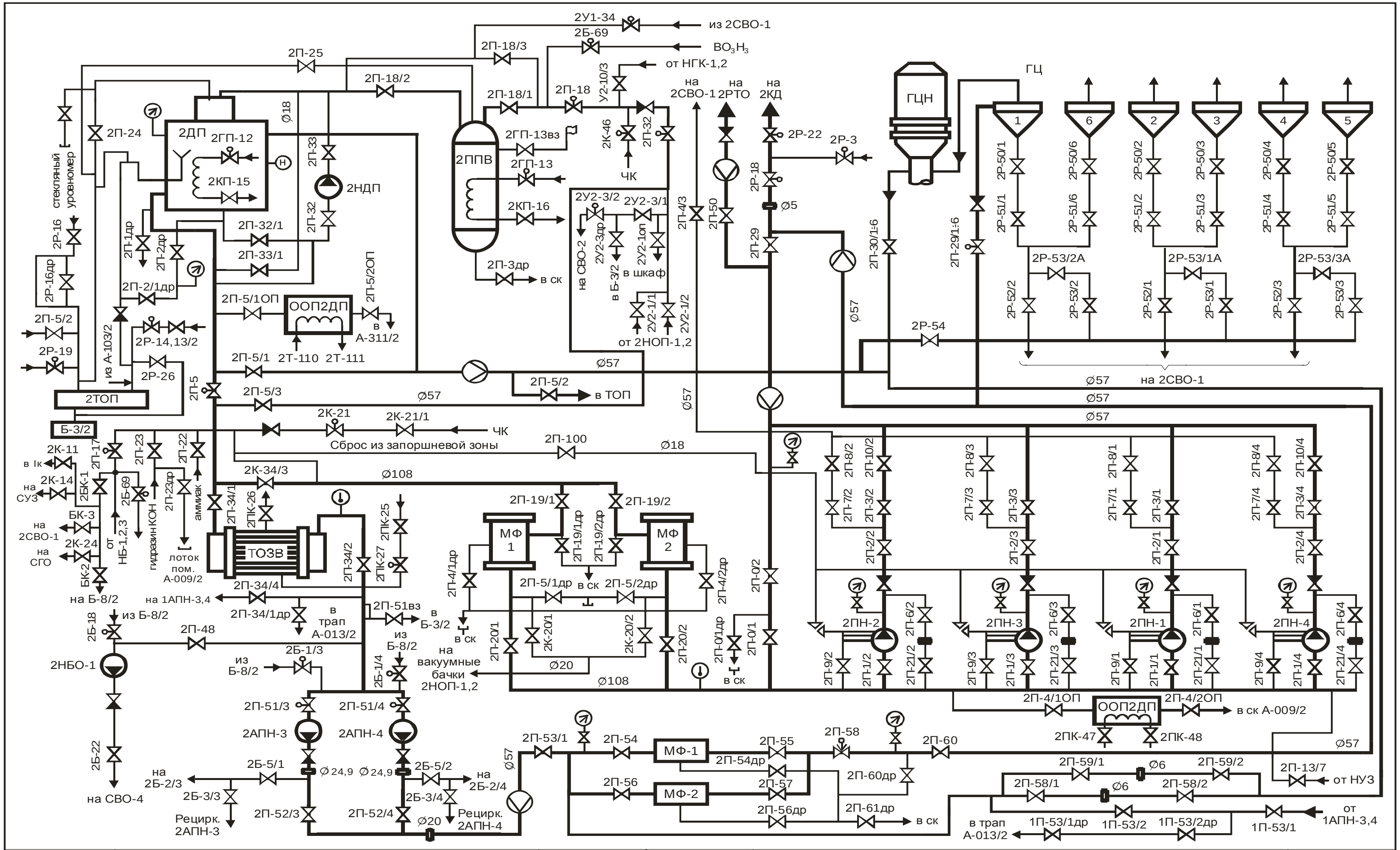

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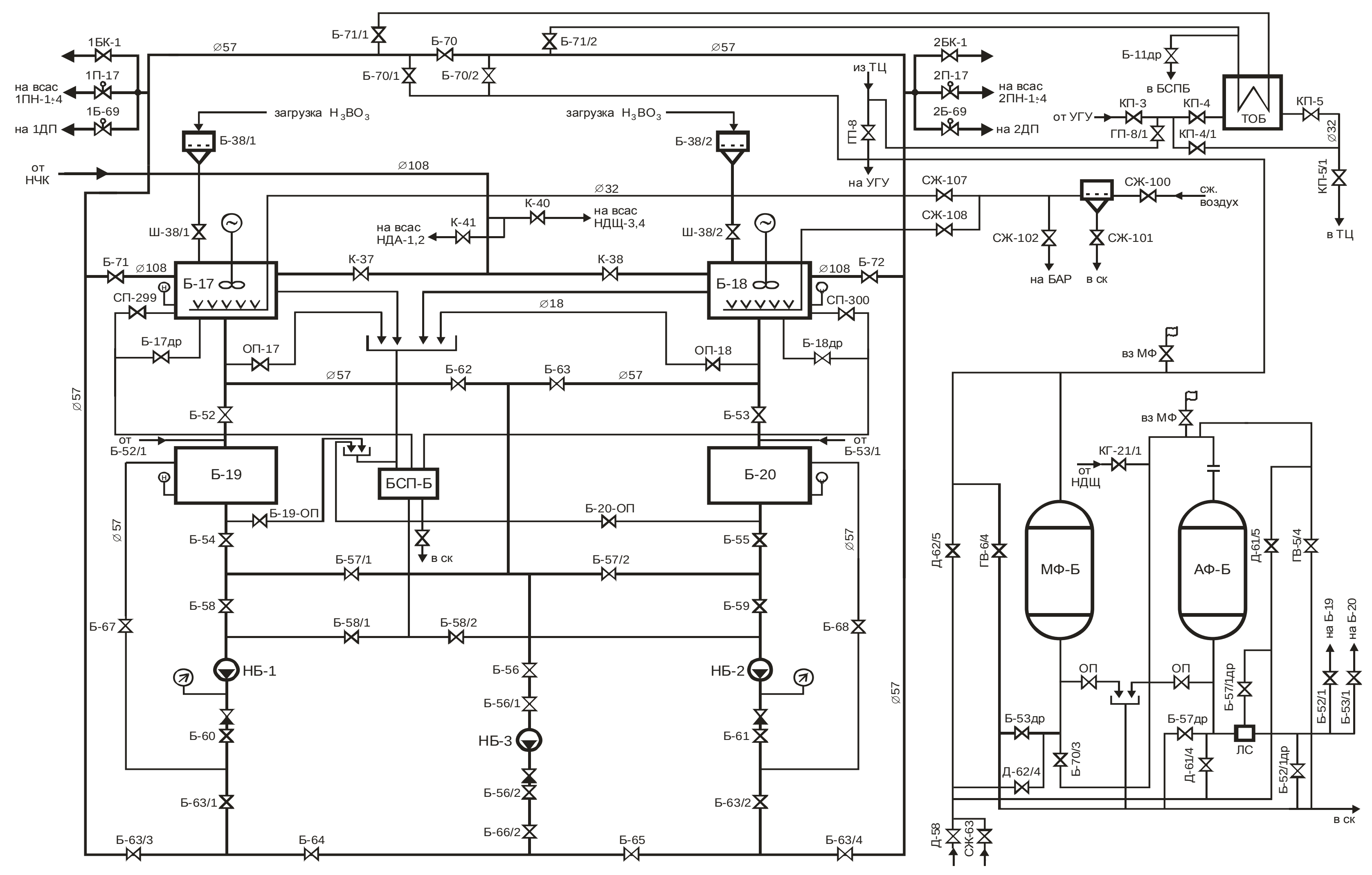

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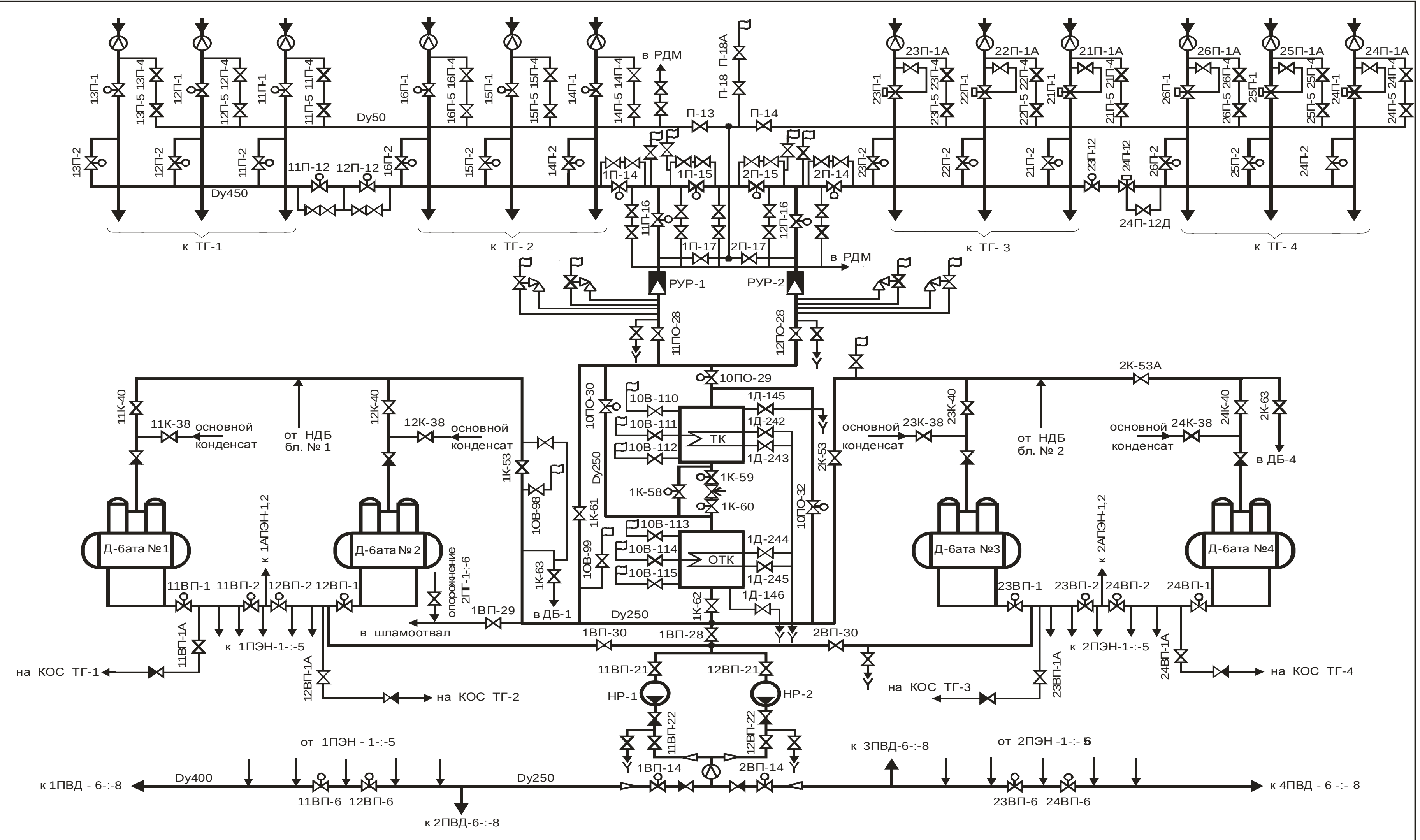

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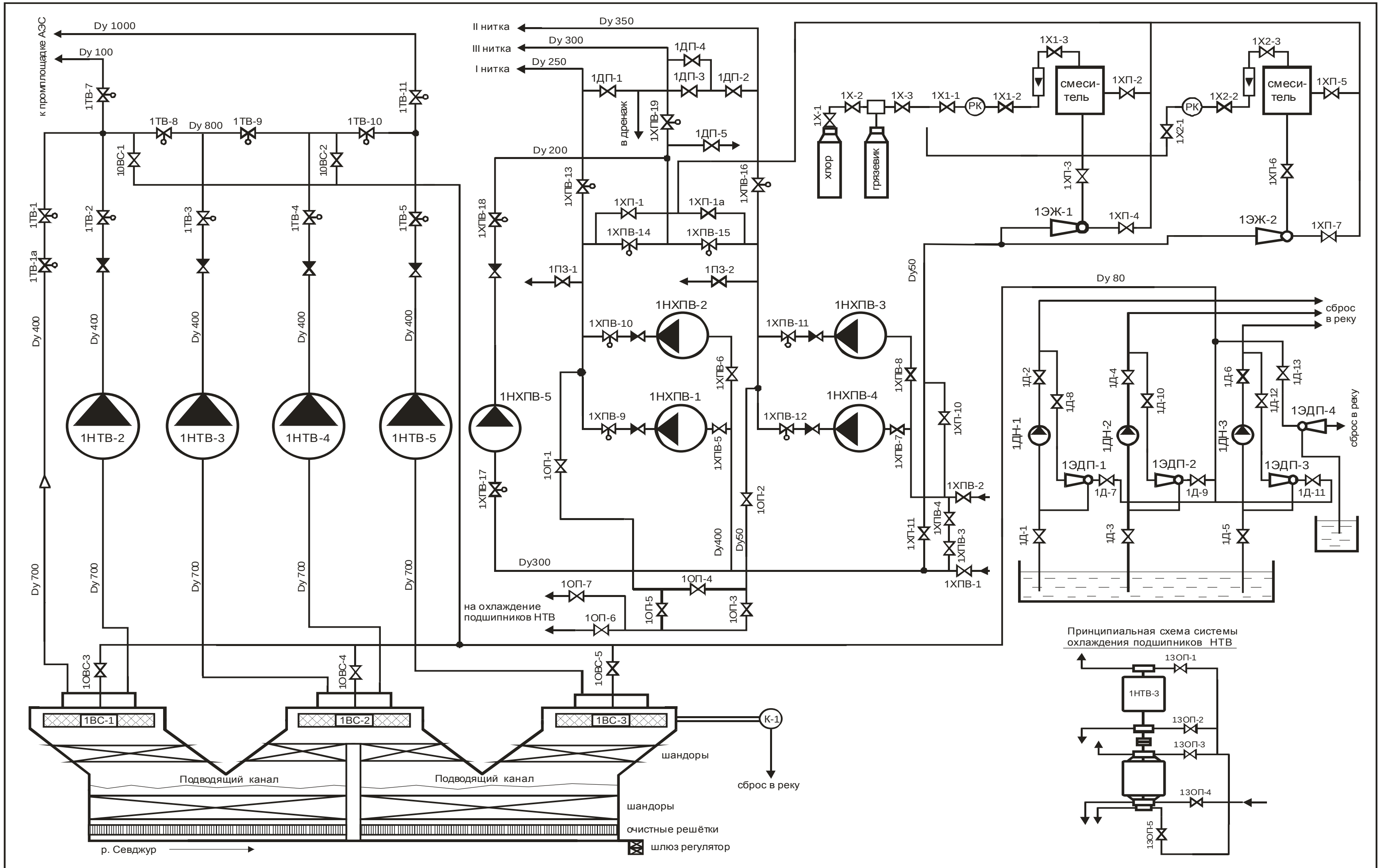

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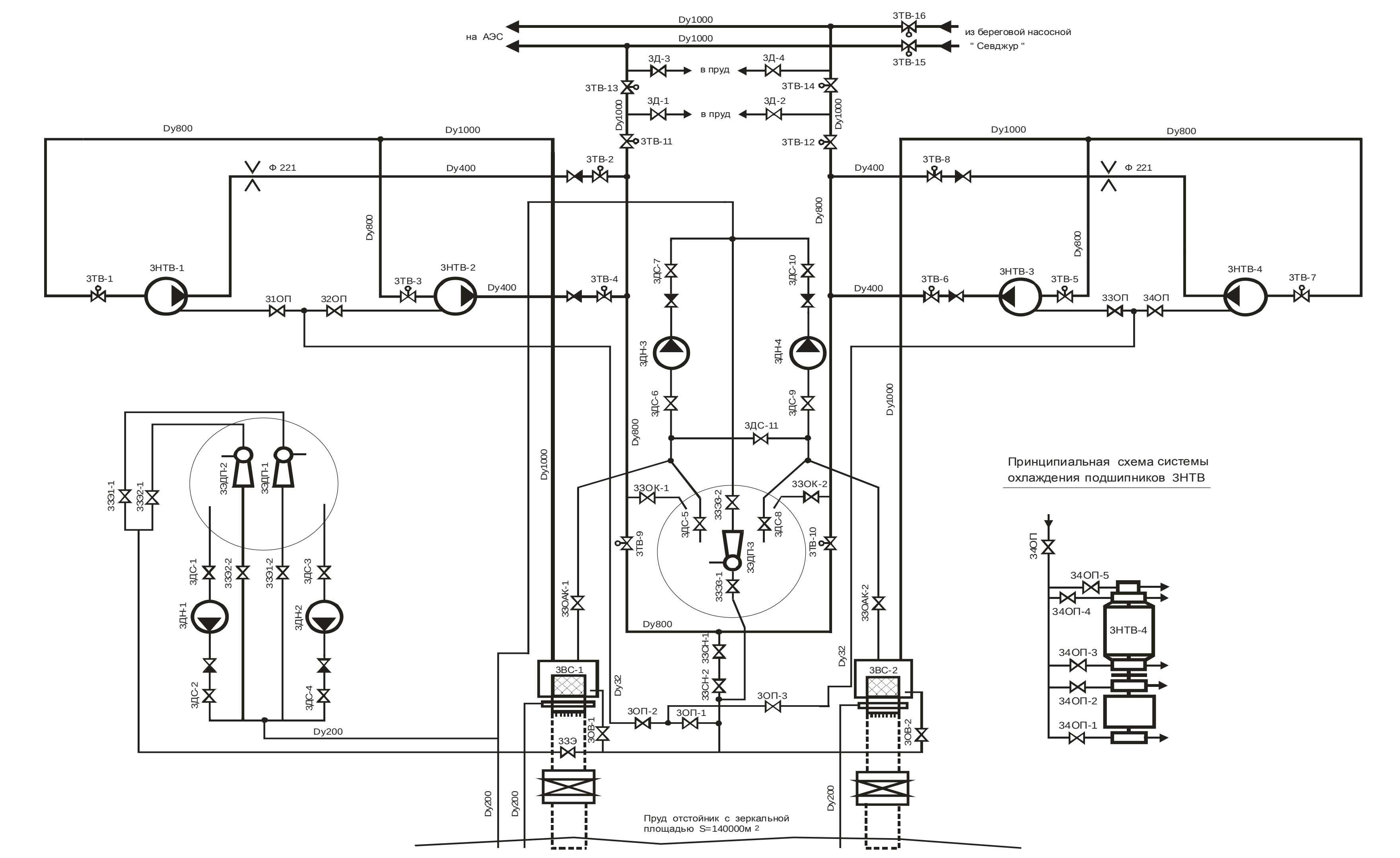

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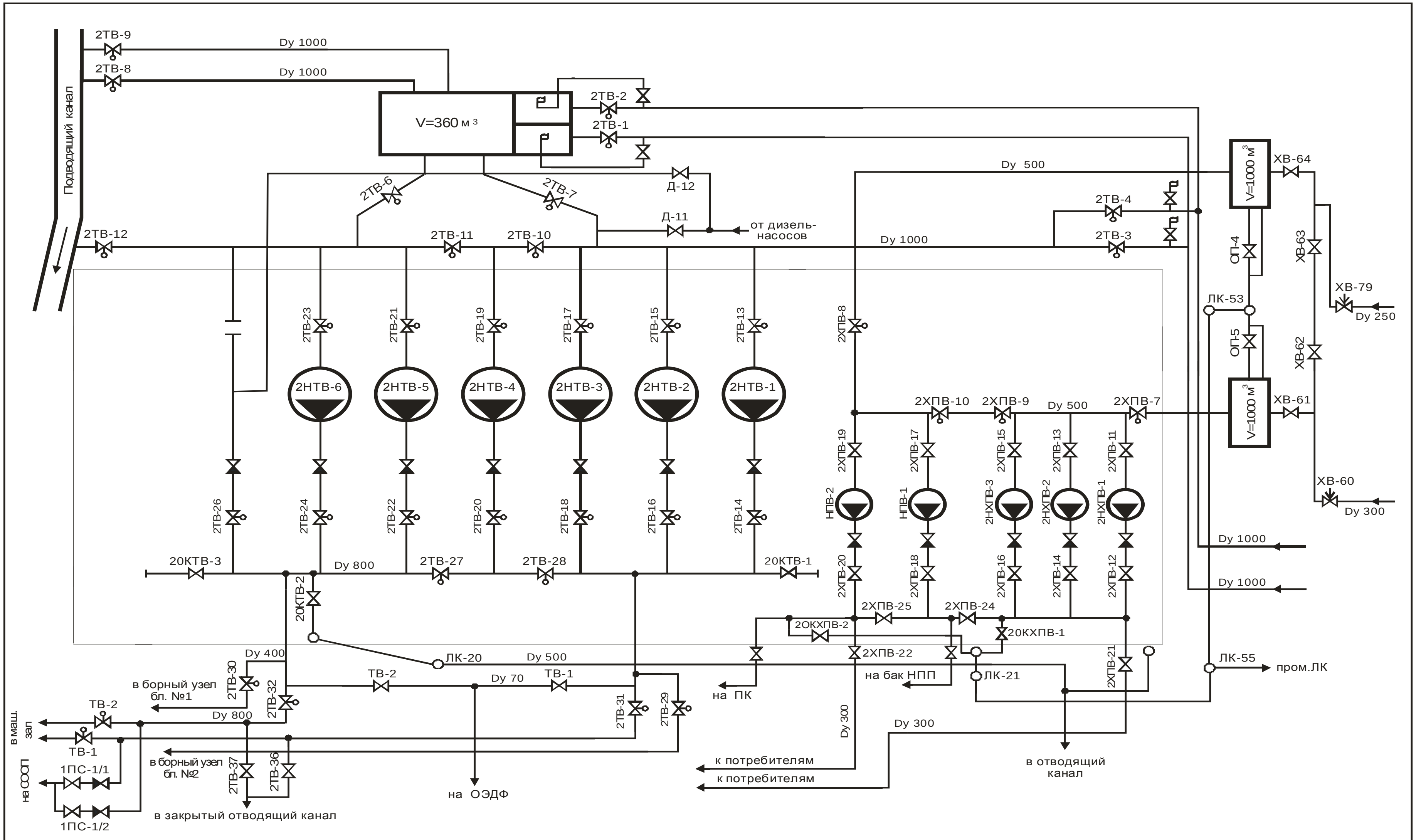

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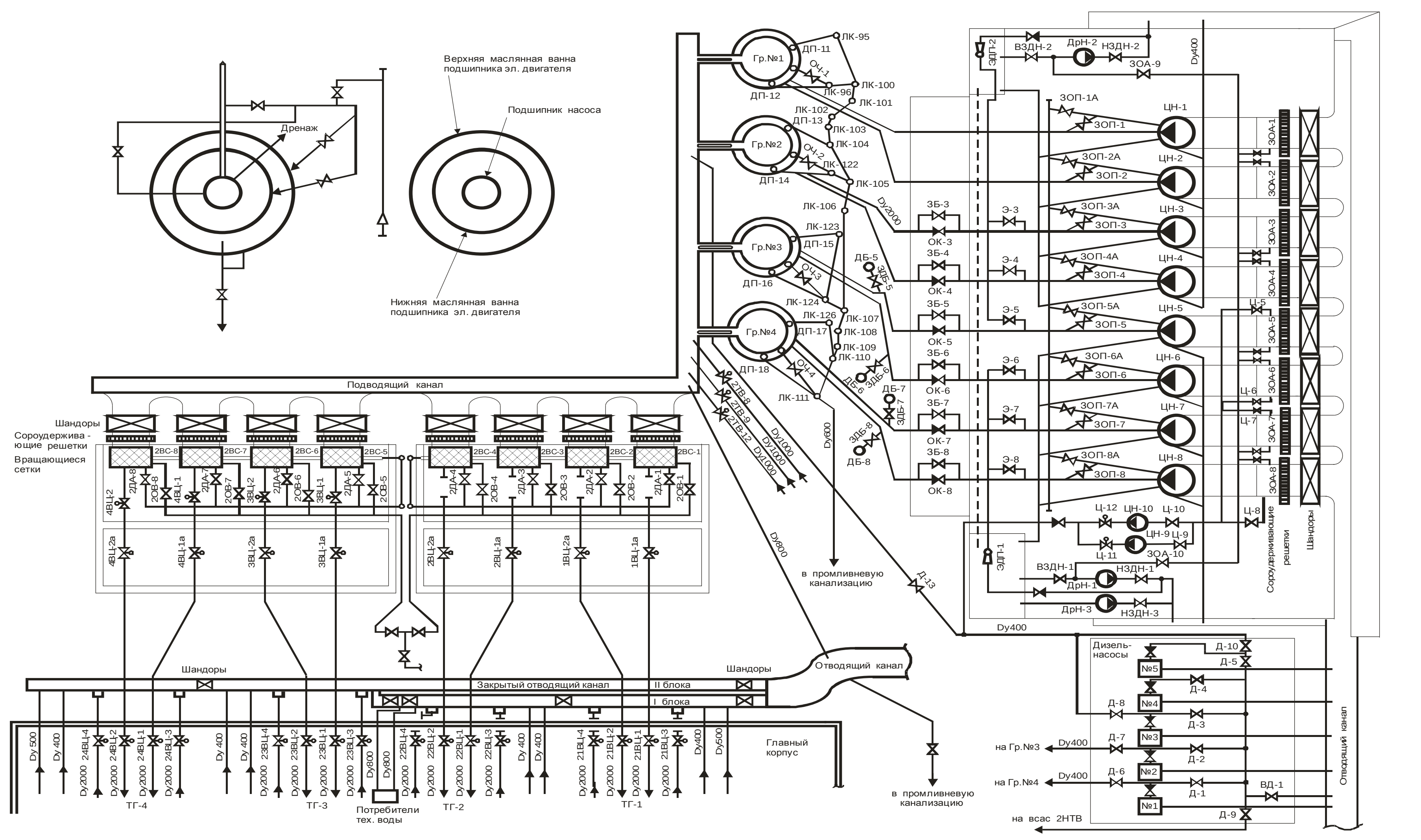

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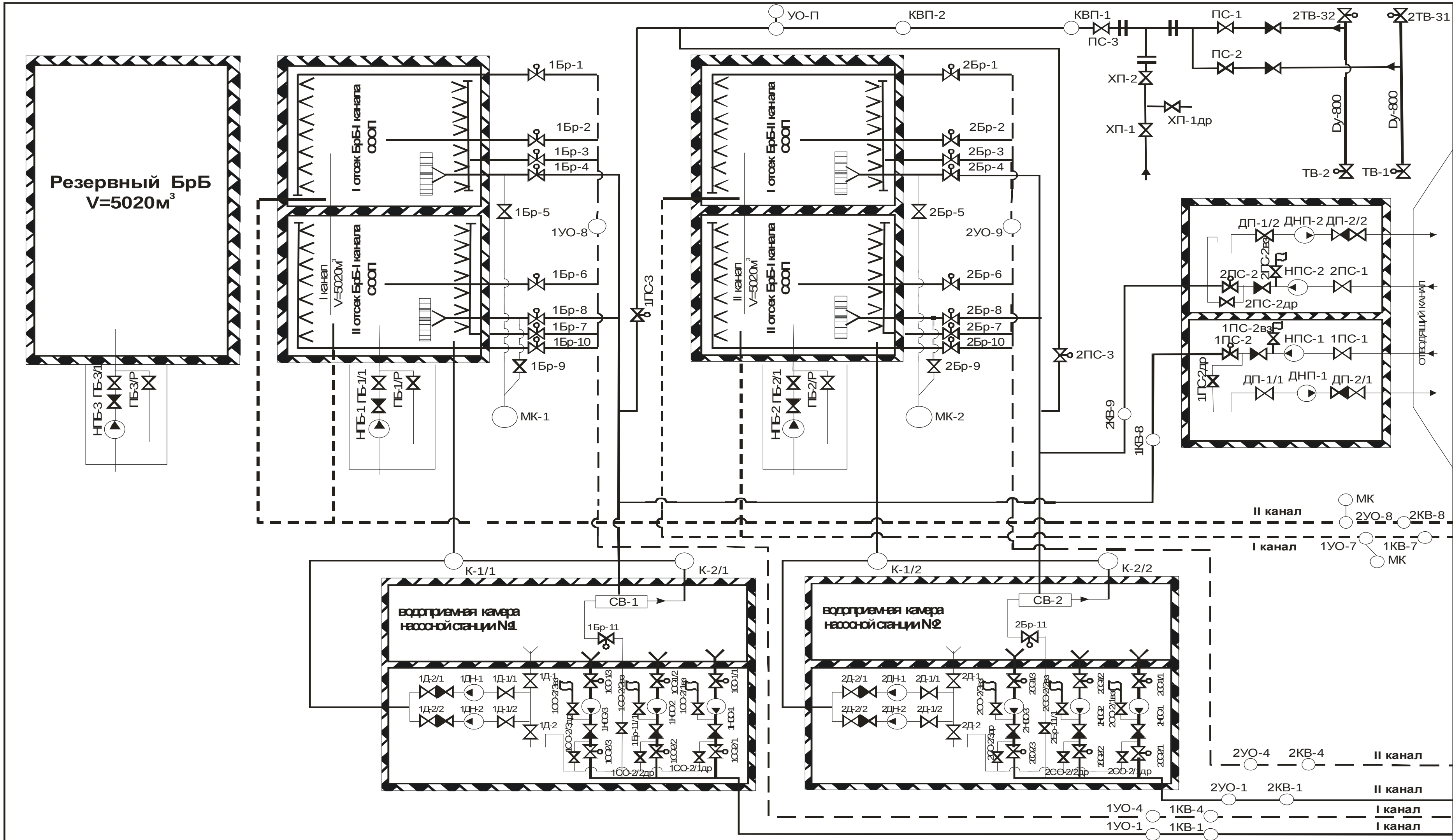

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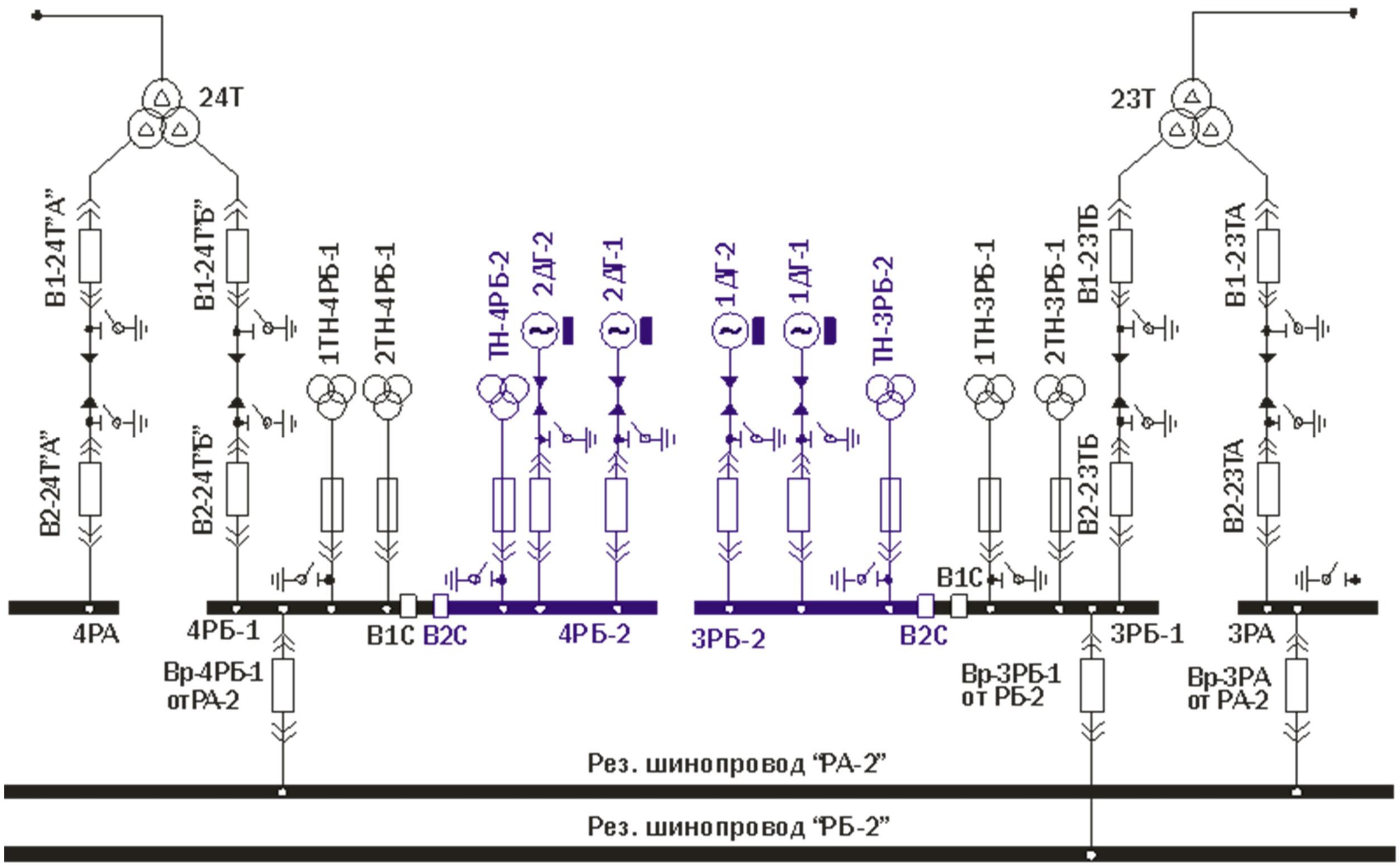
24T
23T
В1-24Т"А"
В1-24Т"Б"
В2-24Т"А"
В2-24Т"Б"
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ТН-4РБ-2
2ДГ-2
2ДГ-1
1ДГ-2
1ДГ-1
ТН-3РБ-2
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2ТН-3РБ-1
В1-23ТБ
В1-23ТА
В2-23ТБ
В2-23ТА
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4РА
4РБ-1
В1С В2С
4РБ-2
3РБ-2
В2С
3РБ-1
3РА
Вр-4РБ-1
отРА-2
Вр-3РБ-1
от РБ-2
Вр-3РА
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CLASS I 0,4kV BUS BARS, 28NA AND 29NA

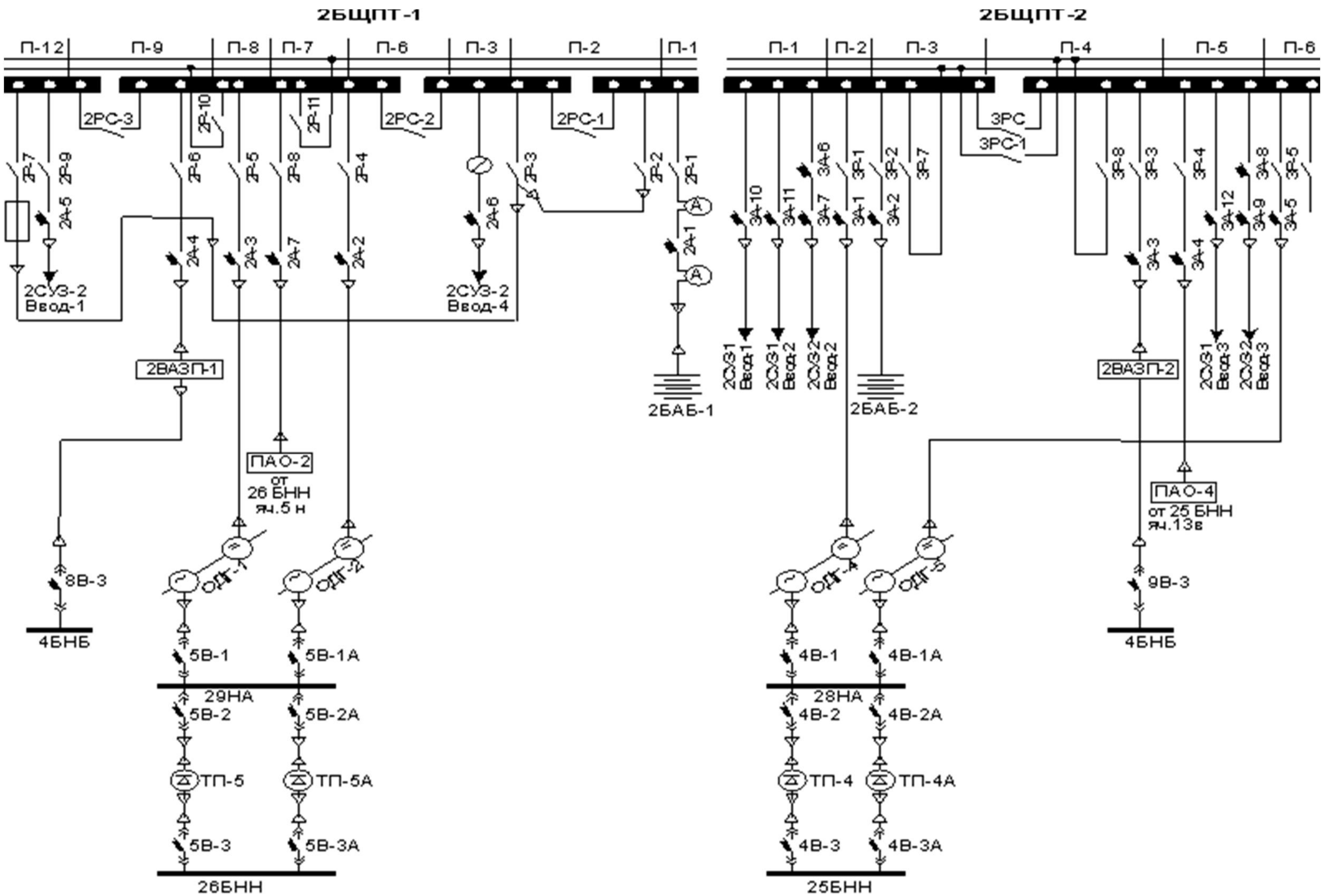
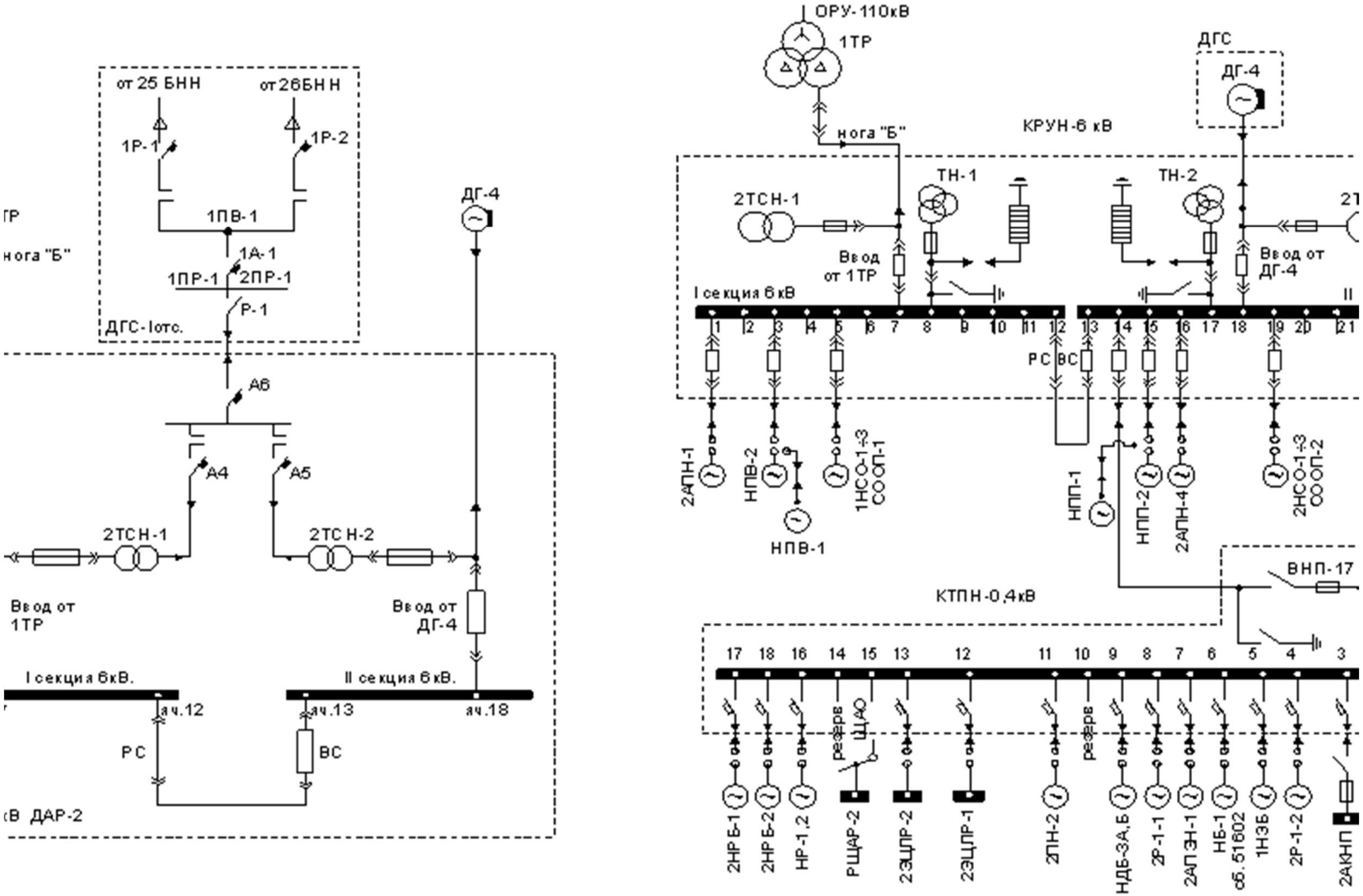
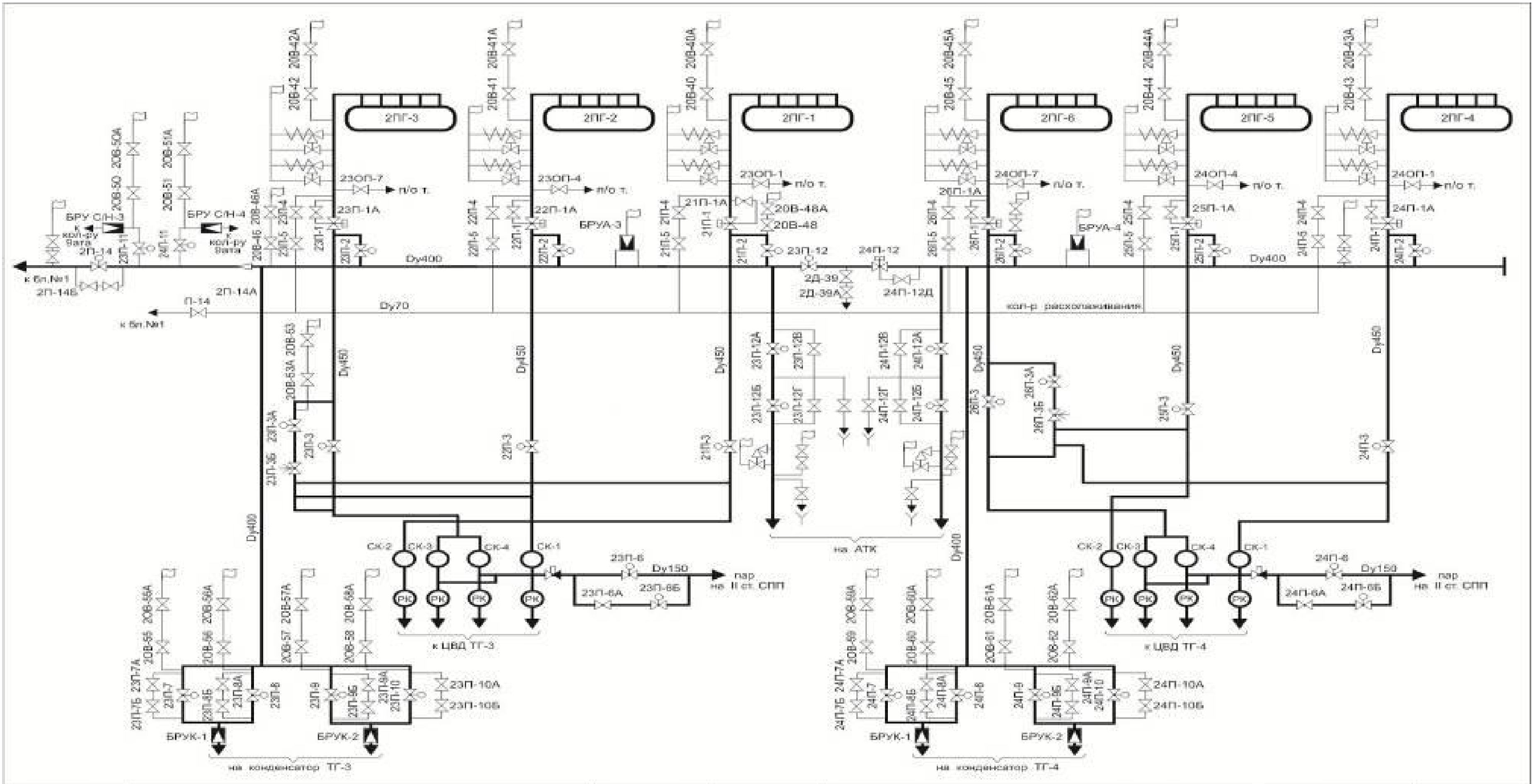


DIAGRAM OF ADDITIONAL EMERGENCY COOLING SYSTEM





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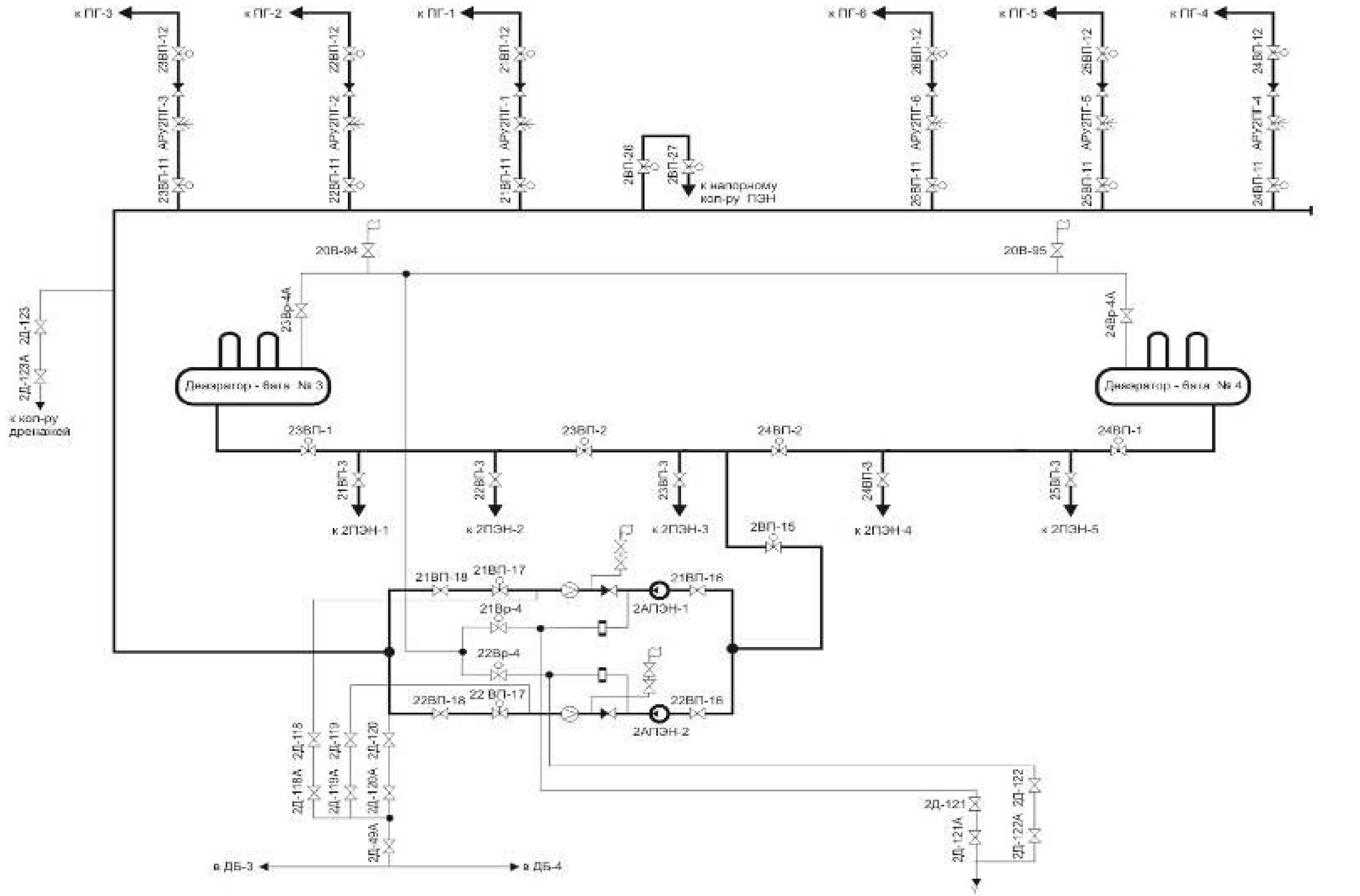

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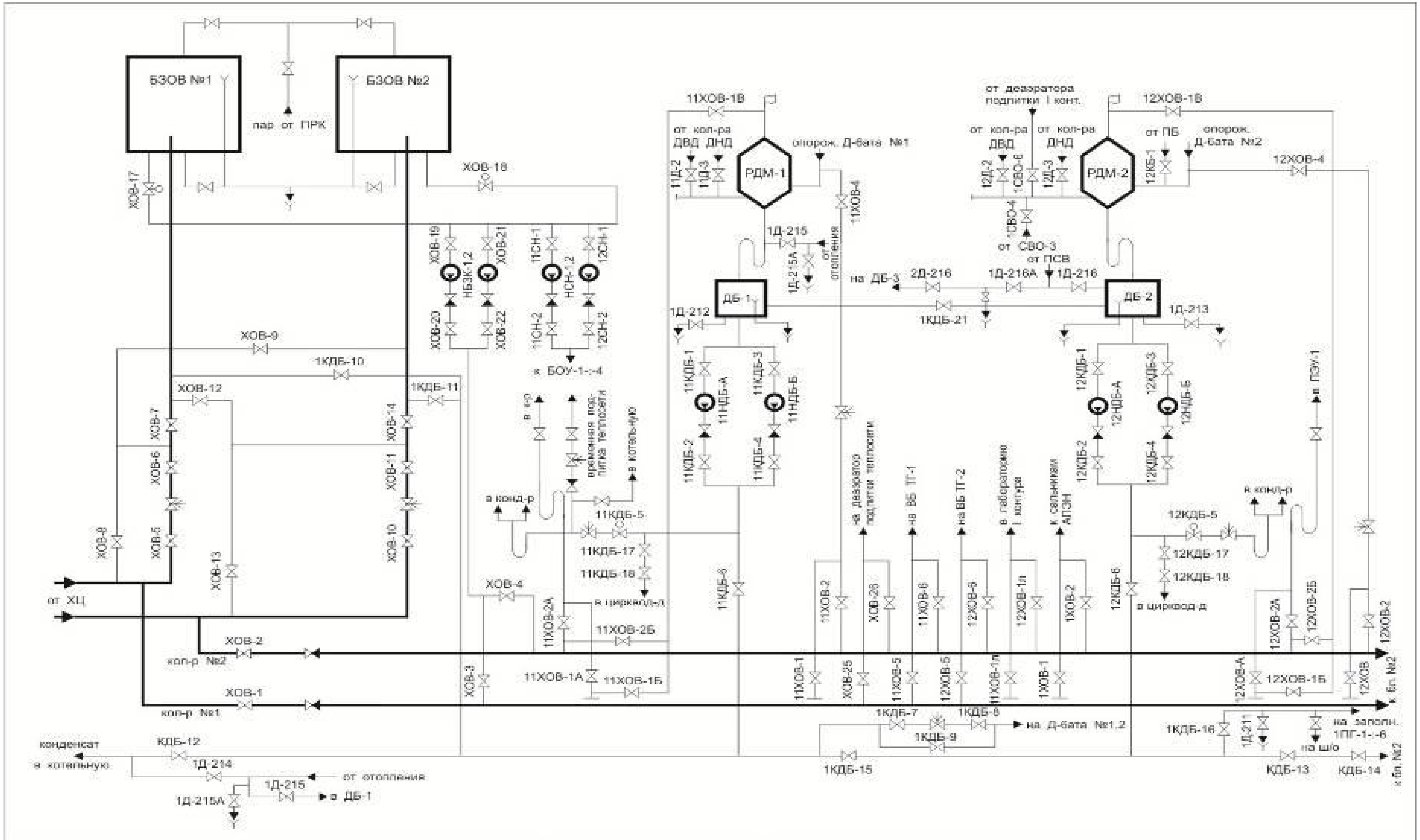

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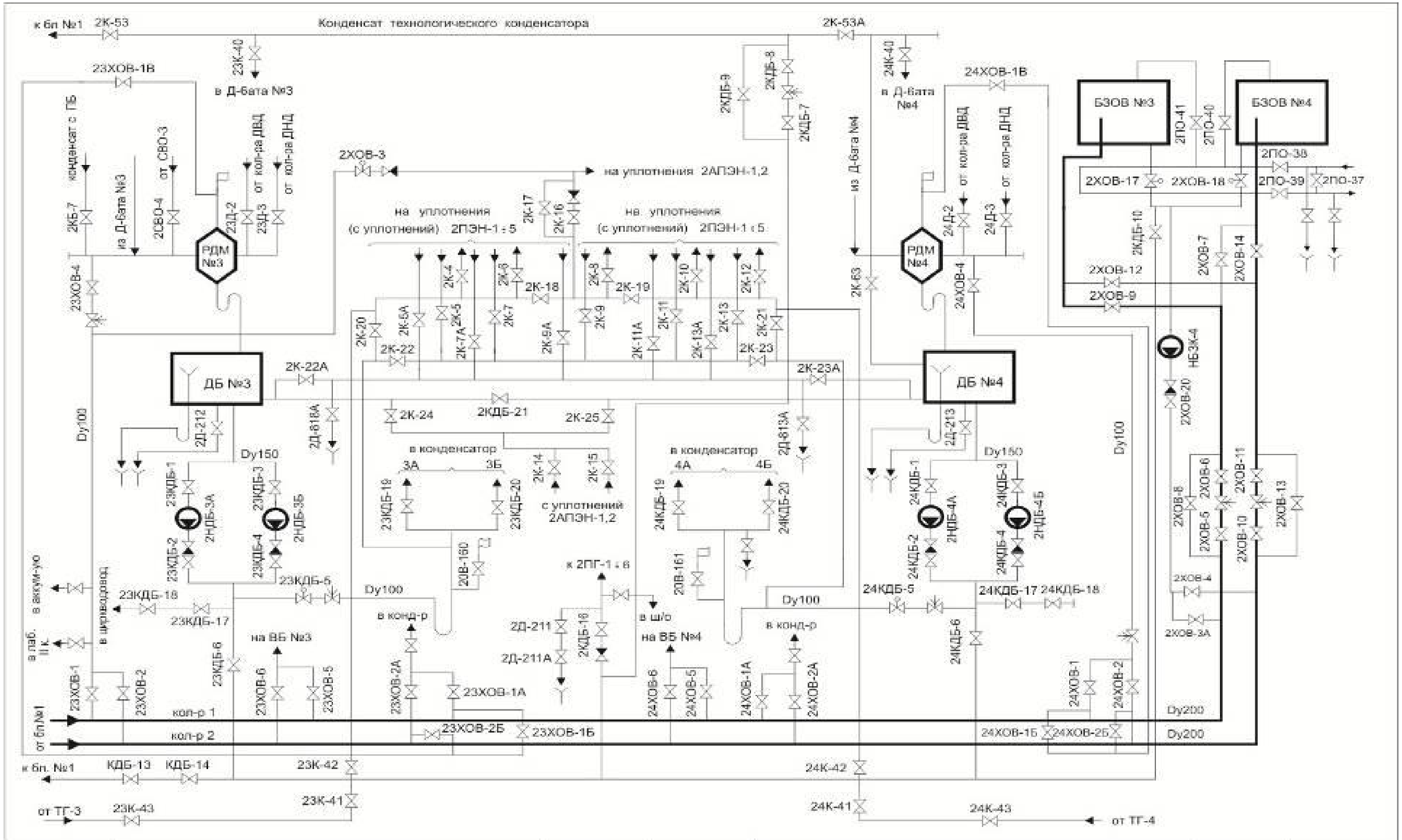

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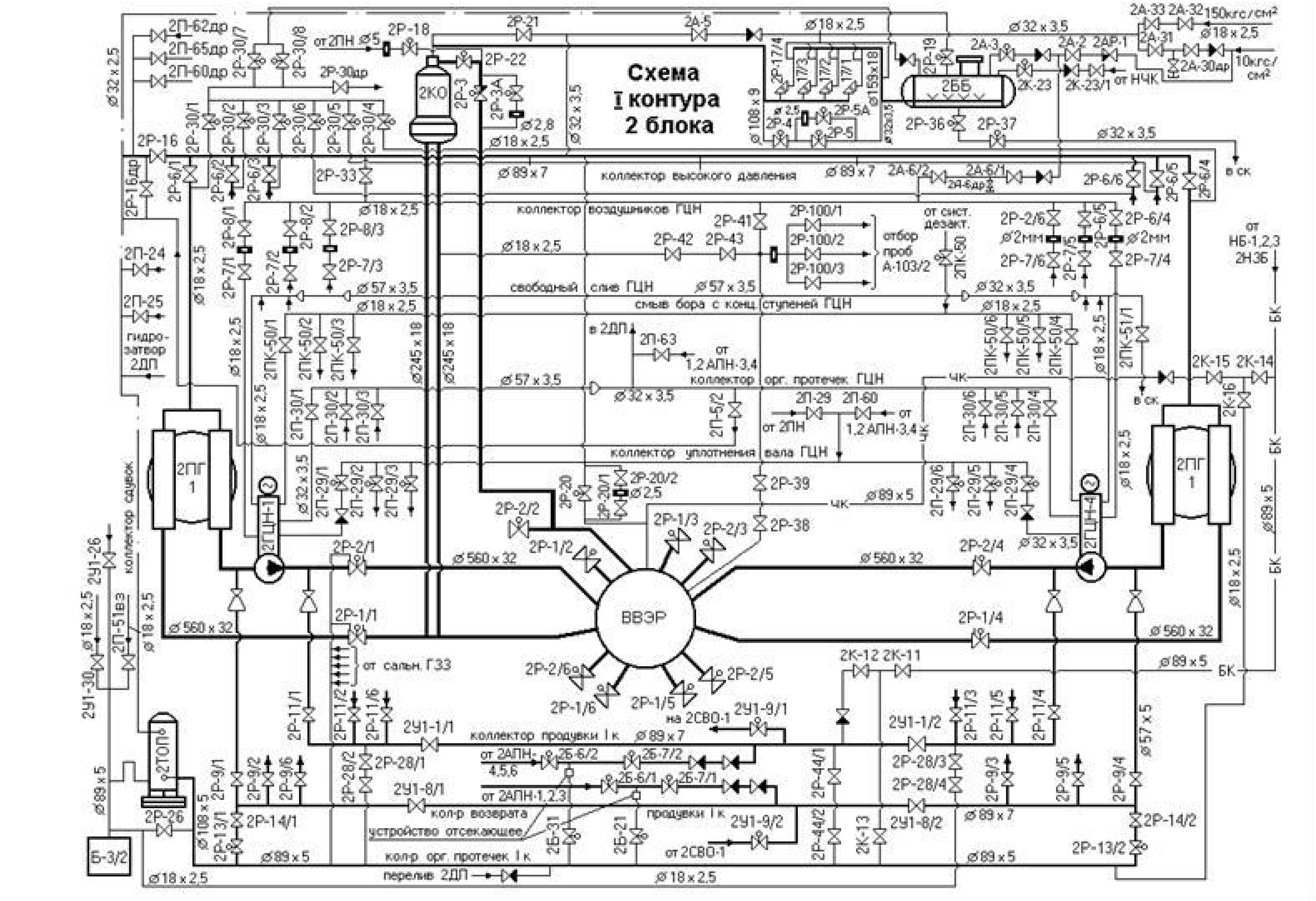
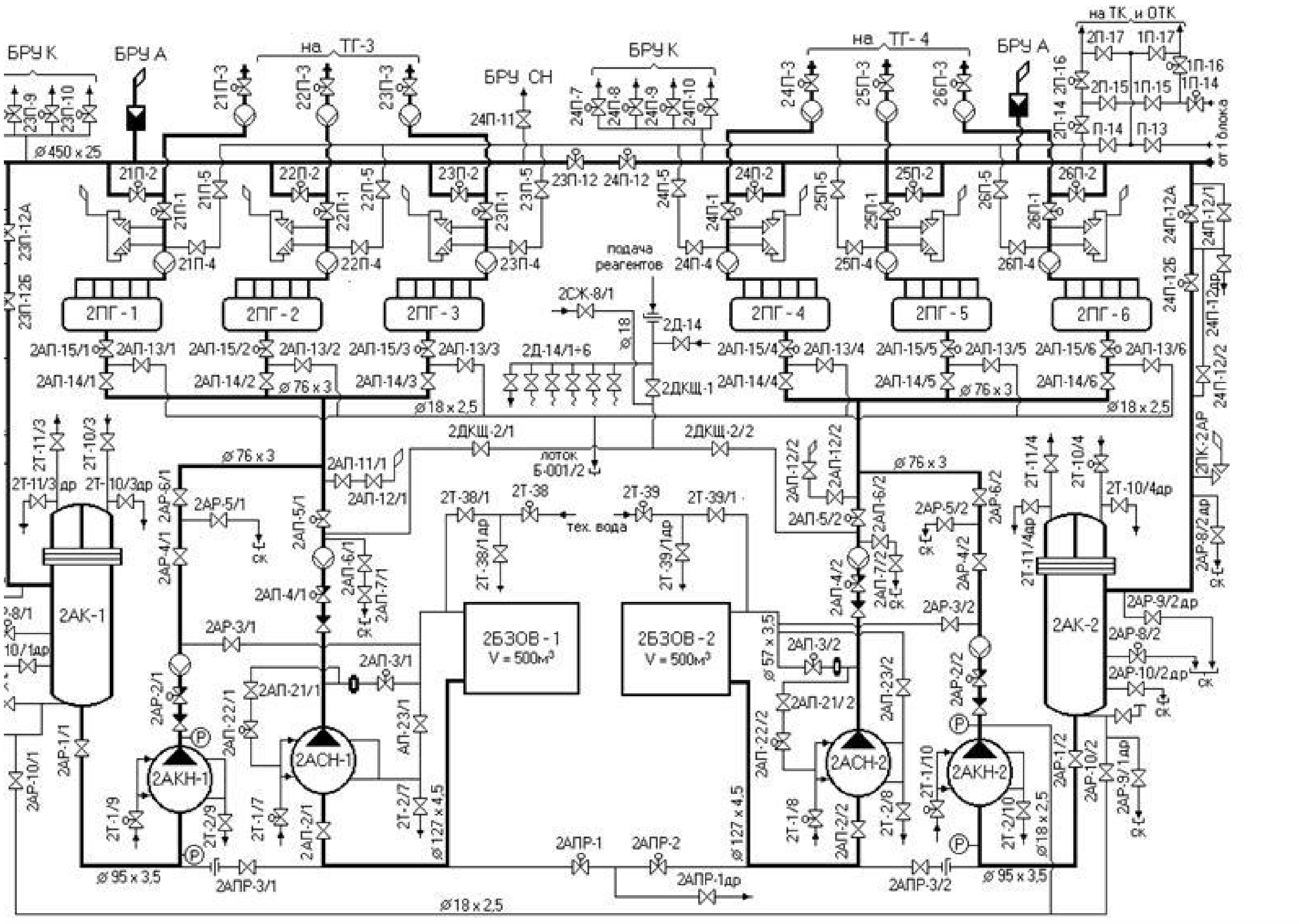
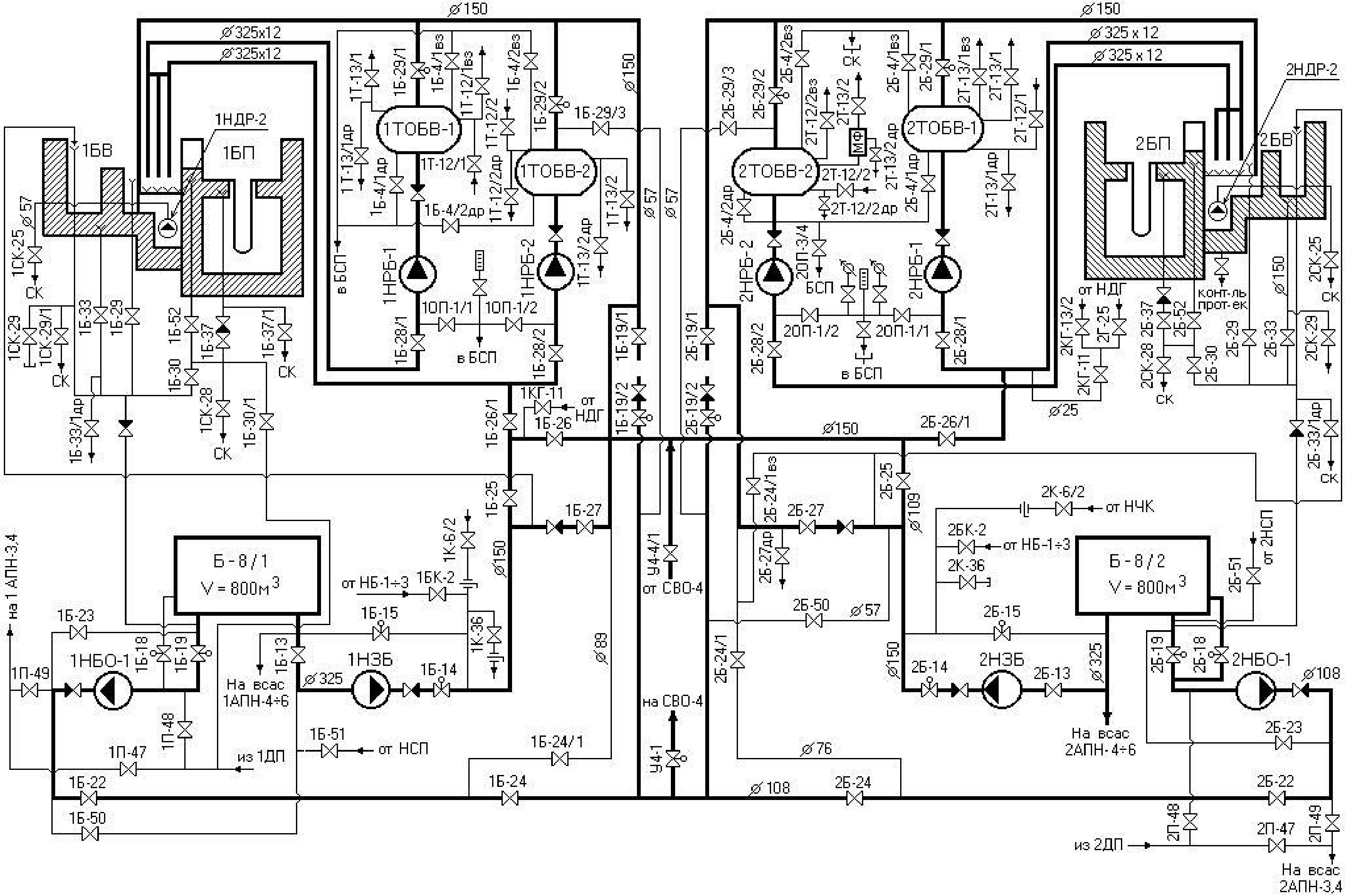
FLOW DIAGRAM OF UNIT 2 PRIMARY CIRCUIT



FLOW CHART OF UNIT 2 NSSS EMERGENCY COOLDOWN



FLOW CHART OF SFP 1&2 COOLDOWN



2. CHAPTER 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 Armenian NPP 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 = 5km) is covered with a thick (400m) 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 400m) 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/cm^2 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 - 2,6 $gram/cm^3$;
- Bearing capacity: 800 - 1000 kg/cm^2;
- V_p : 2500 - 3000 m/s;
- V_s : 1800 - 1960 m/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 - 2,1 $gram/cm^3$;
- Bearing capacity: 40 - 80 kg/cm^2;
- V_p : 600 - 900 m/s;
- V_s : 350 - 750 m/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 [1-3]. Table 2-1 shows the chronology of events and activities related to ANPP seismic design.

Nr	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 I construction	
4	1972	Categorization of SSC on three seismic categories	(2)
5	1975	Start of Unit II construction	
6	1974-1975	Seismic analyses and experimental studies for Seismic Category I equipment and piping for seismic demand 0,4g	(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 (~ 0.2g)	
9	1977	Commissioning of Unit I	
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 II	
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 II	
23	1995	Armenian government's decision to increase PGA level up to 0,35g	(11)
24	1996-1998	Analysis of Seismic Category I and II piping for Floor Response Spectra corresponded to PGA = 0,21g	(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)

Nr	Period	Activities	Notes
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 (DMWT-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, [46-48]) 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	

Table 2-1: Chronology of events and activities related to ANPP Seismic Design

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 (0.2g). Additionally amplification factor equal to 2,7 was installed for the assessment according to SNiP, [4-5].
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 - 0,4 g [6-8]. 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; 0,2 g - Category II, 0,1g - Category III [9, 10].

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) accelerograms scaled to PGA = 0,4g [11-13].
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 [14]. 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 [15].
5) During second phase of testing (1977-1978) 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, [16].
   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 [17] 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 10 000 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 [18], 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, [19]. 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 [20] 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, [21].
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 84th percentile.
12) Seismic Category I and II piping have been analysed in period 1996-1998 according to the modern Standards and requirements. Seismic input was defined in terms median shaped FRS with ZPGA = 0,21g, [22].
13) After restarting the ANPP Unit 2 it was decided to revaluate Plant's seismic design for Review Level Earthquake with use 84th 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), [23].
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, [24]. 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;
    - Localisation 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. As for initiating event for developing an SSEL a simultaneous loss of in-site and off-site power during 72 hours was considered.
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 [25].
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 [26]. 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 Mmax5,8÷8,0, and also dispersed seismicity zones with Mmax5,6÷7,6 in the main seismotectonic model (R=150km).

The nearest Sardarapat seismic zone is in about 13km south to the ANPP, and it is assessed as having Mmax =6,2. The ANPP site is located in M_{max} = 6,1 dispersed seismicity zone.

This analysis uses Ambraseys and ass 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 probabilities were 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) [27, 28]. 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 [29].

As could be seen from the Table 2-1, 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, [30]:

- 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 [31] was developed aimed at implementation of the first two items.

These activities (see Table 2-1, item 21) were resulted in Regional seismotectonic model (R=150km) identifying the areas of potential earthquake foci (PEF). Deterministic analysis of PEF area maximum magnitude (MMAX) was performed with following methods:

- Maximal historical earthquake (M_{max}^{hist});
- Segmentation;
- The overall length of the fault;

- The fractional length of the fault.

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

The catalogue 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 50th 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 50 m each were drilled near the ANPP main building. Special seismic works were performed. Selected drill samples were analysed 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 centre 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 [23] 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 84th 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 [33].

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 FRISK88M™ 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}-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% and 95% confidence and for mean assessment (see Fig. 2-1, Fig. 2-2, Table 2-2 and Table 2-3).

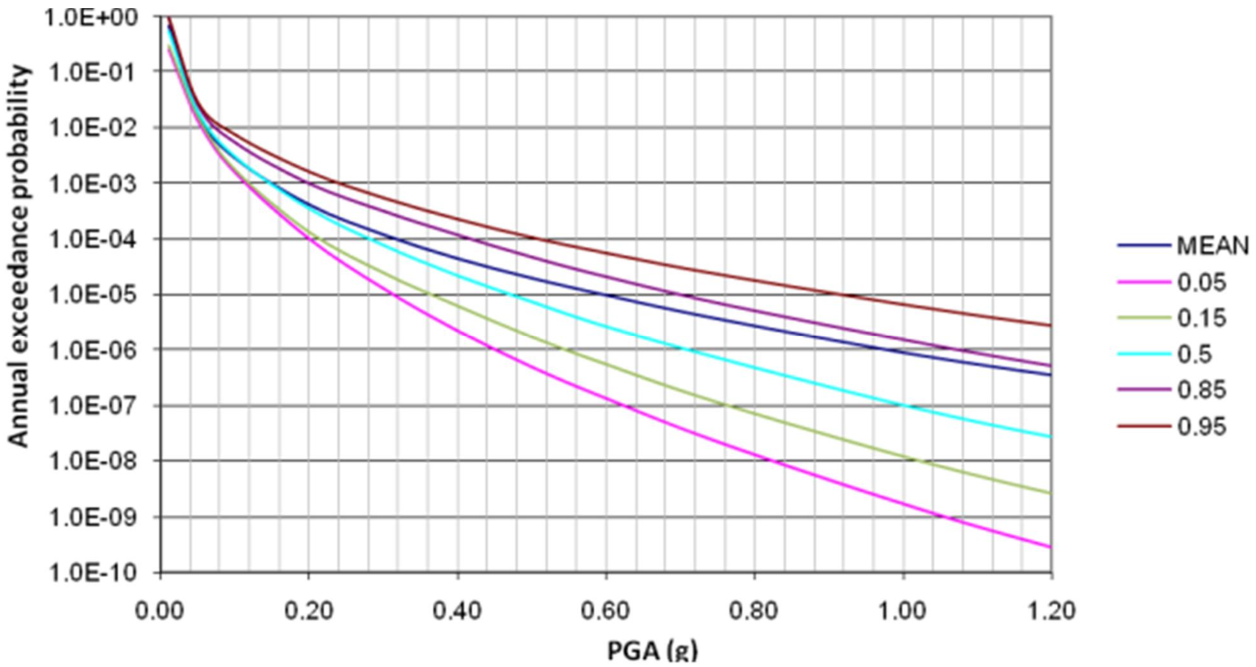


Fig. 2-1: Seismic Hazard Curves for ANPP site (horizontal direction)

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

Table 2-2: Digitization of seismic hazard curves

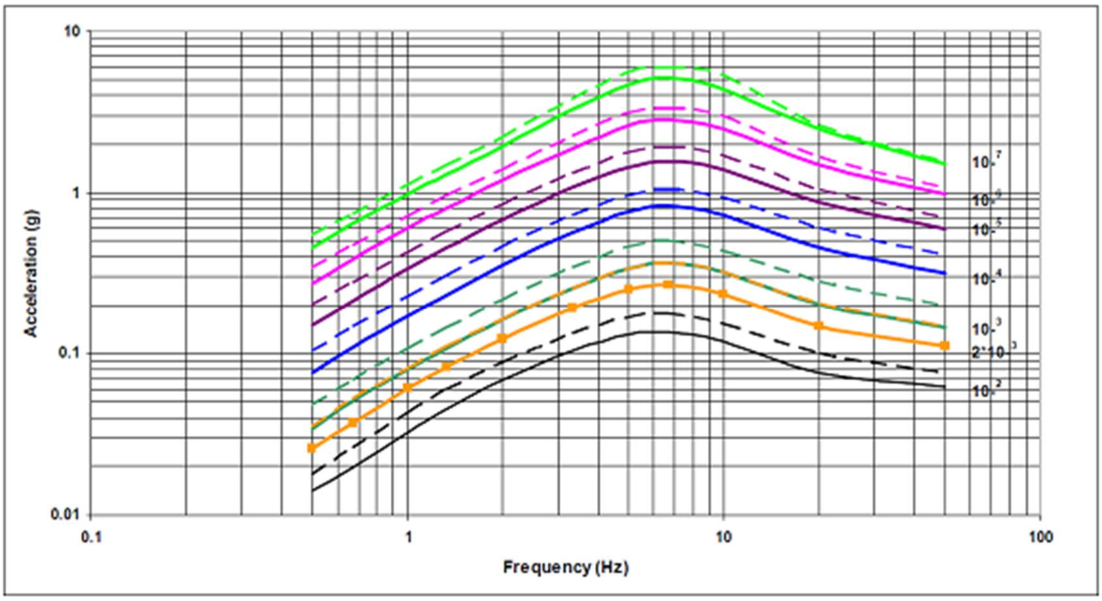


Fig. 2-2: Horizontal Uniform Hazard Response Spectra (UHRS) for ANPP site

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

Table 2-3: Digitization of Horizontal Uniform Hazard Response Spectra

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, archaeological 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 [34], being agreed by regulatory authority.

2.1.2. Provision 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 [35]:

- 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), main coolant pumps (MCP), main loop isolation valves (MLIV), 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 refuelling machine breaking, closure of valves located on seismic category I pipes of primary and second circuits and gate valves of cooling water system channels;
- Structural model of the crane was included in the Reactor Building analysis. During NPP operation, crane is fixed in parking position and even in the case of failure it would not produce significant damage.

Table 2-1 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. IDENTIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS (SSC) THAT ARE REQUIRED FOR ACHIEVING SAFE SHUTDOWN STATE AND ARE MOST ENDANGERED DURING AN EARTHQUAKE. EVALUATION OF THEIR ROBUSTNESS IN CONNECTION WITH DBE AND ASSESSMENT OF POTENTIAL SAFETY MARGIN

During the period from 2001 to 2002 a procedure for the Reactor Safe Shutdown under seismic impact [24] 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]

29 systems were included in SSEL:

1. Reactor;
2. Main coolant circuit;
3. Control and protection system (RCPS);
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. MCP 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 Service Water System (ESWS);
23. Second 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, [36]) 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, 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's safety valves, SG's 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) [37-39].

In 2004 a new system DAP was developed for SG makeup from chemically demineralized water tanks (DMWT-500) in case of the ANPP blackout. It involved installation of additional emergency diesel pump in the Boron Unit (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 [40-43].

Seismic qualification of ANPP Buildings and Structures was confirmed by analyses:

Analysis of the ANPP main building were performed and approved by regulatory authority in 2004 [44]. 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 [46-48] at RLE (0,35g), [45-47].

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 [48] 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 1g load (own weight) compliance, and according to the analysis [49] 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 ESWS 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 Unit 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 same requirements were applied to it as to the hermetic compartment (amplification factor 2,7 applied).

However, according to IAEA recommendation verification of the Boron Unit was also included in seismic qualification project ARM/9/022, and in the framework of that project the analysis was completed [50] concluding that the Boron Unit 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 [51].

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 [27, 28], allowing performance of the final seismic walkdown.

According to "ToR for detailed seismic walkdown of ANPP Unit 2" developed by IAEA [52] 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 [53], which was later provided to the IAEA experts. Dr. J.D. Stevenson performed Peer Review of this Project [54].

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 [55-62].

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 revaluation 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 [63-64], 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 [64].

Improvement of the seismic performance of piping systems located at el. +14,7m was included in TACIS project [65]. Resulting recommendations for upgrading of Steam Lines, Feedwater and Emergency Feedwater piping are currently implemented: it led to installation of the additional hydraulic snubbers, piping rigid struts and whip restraints, 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 Unit2". 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, Primary Circuit temperature sensors, piping hermetic penetrations for the Main Steam and Feedwater lines, as well as for number of piping systems (list of analyses is provided in Appendix B)

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, [66].

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 [67].

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 $50cm/s^2$) 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 refuelling 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.

At the present time this operational instruction is revised taking into account new seismic data and results of performed analyses.

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

2.1.2.3.1. Assessment of potential failures of heavy structures, pressure retaining devices, rotating equipment, or systems containing large amount of liquid that are not designed to withstand DBE and that might threaten heat transfer to ultimate heat sink by mechanical interaction or through internal flood.

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. Several instances of such interactions were identified during 2007 and following walkdowns: one example is a service water storage tank 40 m^3 (BTV-1) of the cooling system of responsible consumers (SOOP) located at elevation 21 m. Seismic Margin of this tank led to very low HCLPF = 0,1g. Corresponding recommendation for seismic restraining were developed, however ANPP later has decided to demount this equipment. Other sample is large bore pipe located in the Turbine Hall in the vicinity of Electrical Control Cabinets. Appropriate recommendations were issued and pipe was restrained.

Most of the safety-related equipment and piping located in the Boron Unity were seismically qualified for RLE level. However, even in case of the hypothetical rupture of DN400 piping from ESWS (room B-001/2 in Boron Unit which located approximately 700 m from ESWS), the volume of the spilled water would be equal to 88 m^3. Such volume could not affect operability of the emergency pumps (EMP, EFWSP, sprinkler pump, etc.), since a real threat for EMP is free volume of 120 m^3 and three drainage pumps located in this compartment provide a total draining capacity 115m^3/hour.

A potential source of flooding for the Main Building (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, [67]. 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 (-4.2 m). Assuming that the basement is completely flooded, the water through the existing gates may spread to the site, and cannot threaten the components of the emergency power supply. Additionally all significant compartments located at ground level are equipped with hermetic doors.

To prevent flooding of the RDGS building a long opening by 50 m length and 0,5 m height was made on the low level of the canal's wall (opposite CWPH) which provides water discharge outside the ANPP.

Despite the fact that SSEL boundaries on the +14,7m level were limited by Main Steam line Isolation 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 armoured 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 indirect effects (e.g. fire or explosion)

ANPP fire-fighting 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.

Fire-fighting 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

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), [23].

12 main tasks were defined according to the Technical Guide (see flowchart in Fig. 2-3). 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 with 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 levelling 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 levelling: 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, [68];
- 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.

Safety Related Systems are checked for the functionality in accordance with yearly schedule. During shutdown period such checking is performed according to separate specific schedule. All upgrading and modernization of the safety classified equipment are performed according to the guidelines "Comprehensive planning modifications" and "Procedure for modifications" and are strictly controlled in accordance with ANRA document "Technical modification of nuclear installations" (order number 1-U, January 18, 2010).

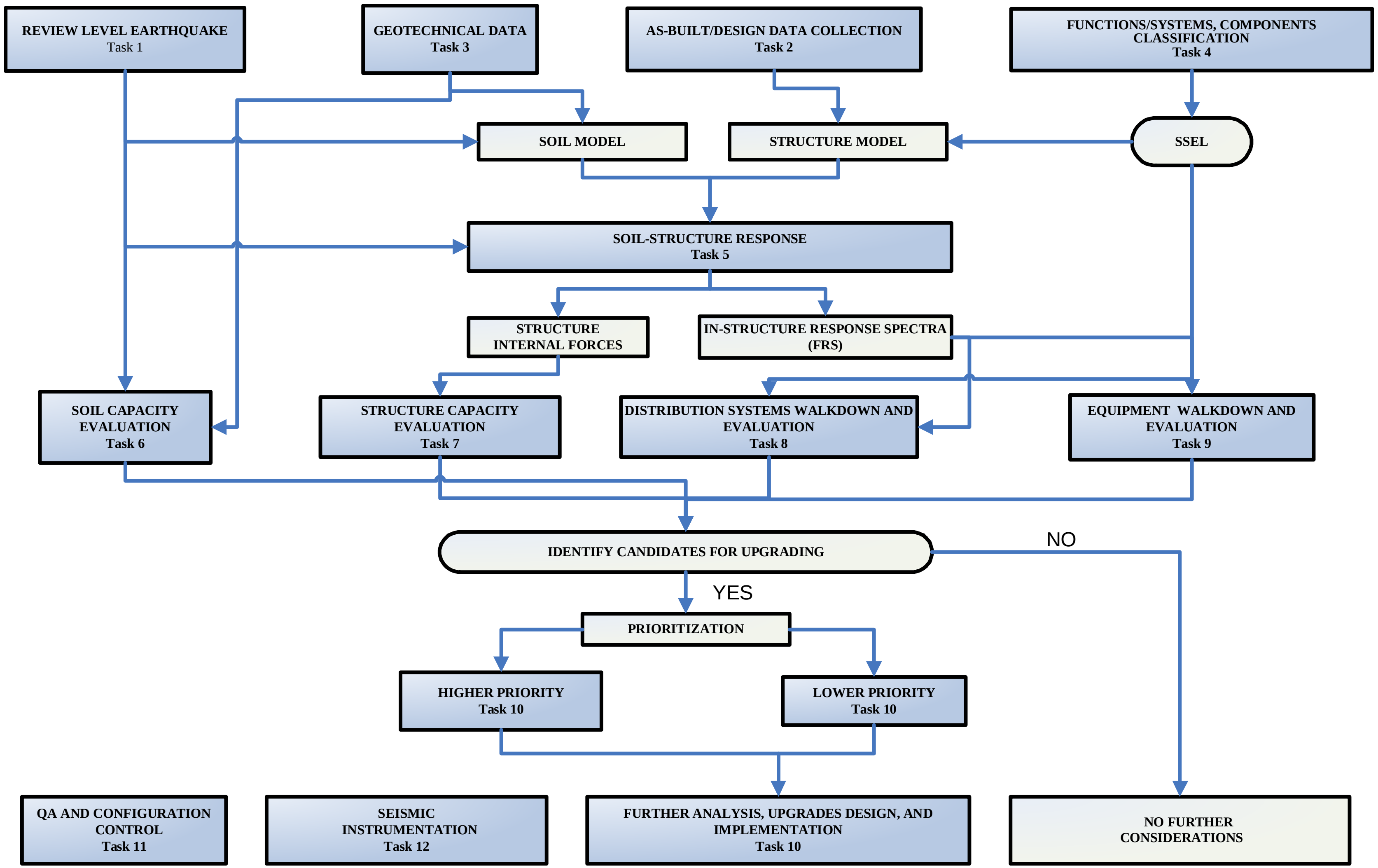


Fig. 2-3: ANPP Seismic Reevaluation Tasks

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 DMWT 1 and 2 tanks with the diesel pump installed in more seismically robust Boron Compartment (Unit 1).

DMWT 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 [69].

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), Fig. 2-1. 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 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 Fig. 2-4, 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 > A_{SL}, where A_{SL} = 0,35g - a screening level corresponding to the conservative GIP-WWER ([70], [71]) 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 A_{SL} = 0,5g, taking into account the following circumstances:

1. 0,5g screening level is applicable for the existing GIP procedures: GIP-DOE [72] and for the Seismic Margin procedure presented in the documents EPRI [73];
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 [23];
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 revaluation program.

Seismic Margin Assessment results are given in the summary Table 2-4. Appendix B presents a list of reports and documents with analyses and estimates made for assessment.

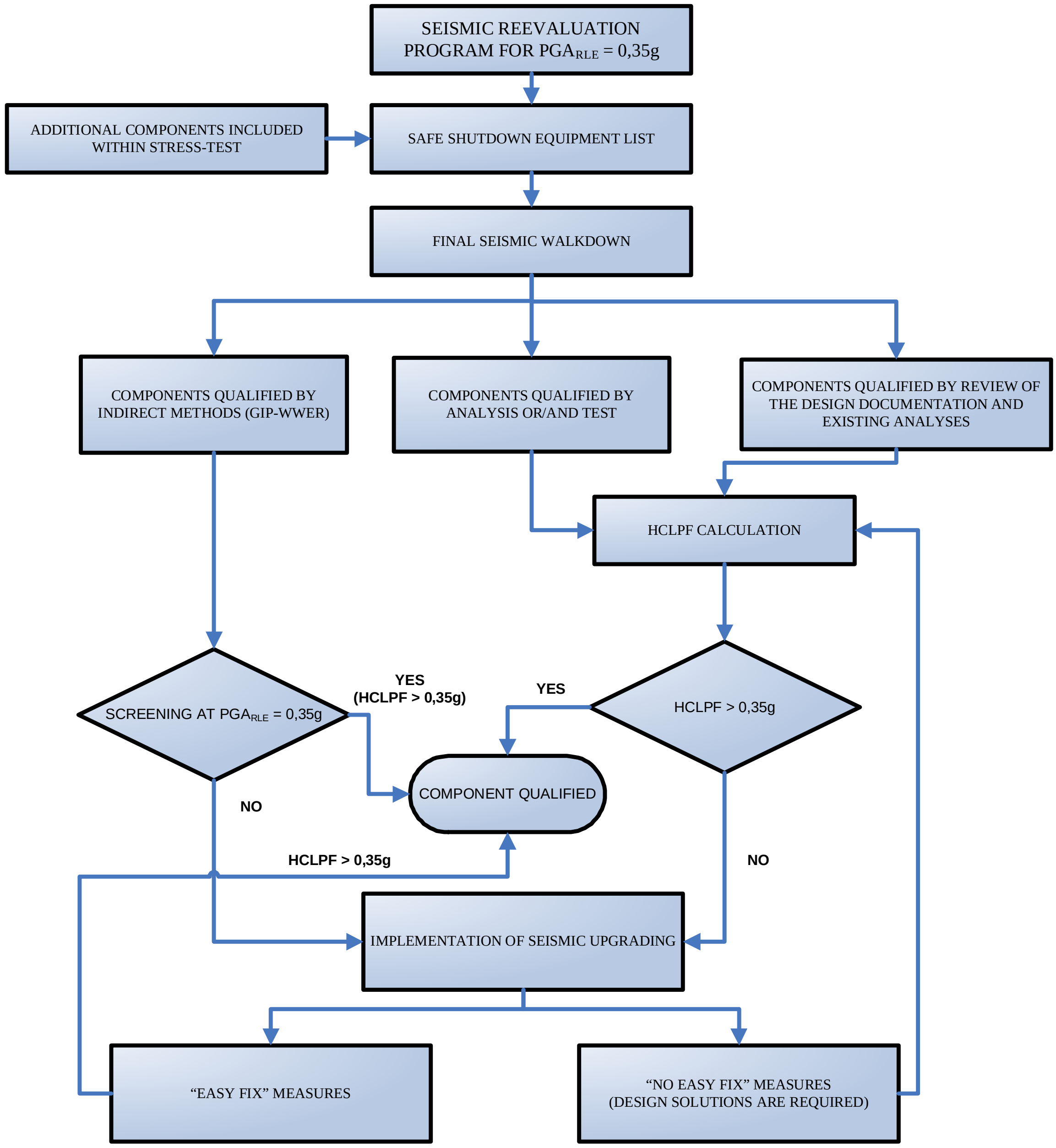


Fig. 2-4: ANPP Seismic revaluation flowchart

№	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Note
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 EP 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			0,50	(2)
4	Unit 2 main control room			0,50	(2)
5	Neutron Flux Monitoring System			0,50	(2)
6	Control and monitoring system			0,50	(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	0,60		
8	Primary circuit overpressure protection system			0,50	(2)

№	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Note
9	MCP system			0,50	(2)
10	Primary circuit auxiliary feedwater system	EFWP1 (Emergency Feedwater Pump)	0,58	0,58	
		EFWP2	0,71		
		EFWP3	0,62		
		EFWP4	0,98		
		EFWP5	0,94		
		EFWP6	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	0,50	
		SFP CP-1	0,5		
		SFP CP-2	0,56		
		SFP MP	0,54		
14	Industrial seismic protection system (SIAZ)		0,50		(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		

№	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Note
16	Main steam pipeline system	from SG1	0,40	0,39	
		from SG2	0,50		
		from SG3	0,50		
		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		
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		
		DMWT -1,2	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		
		1 ESW pump	0,77		
		2 ESW pumps	0,62		
		DMWT-3,4	0,55		(6)
21	Emergency gas removal system	Piping	0,78	0,78	
		Supports	1,6		

№	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Note
22	Essential Service Water System (ESWS)	ESWS piping in the Boron Compartment	0,46	0,46	
		ESWS piping (Pump Station)	0,64		
		ESWS buried pipes	0,55		(7)
23	Second 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	0,90		
		Return Diesel Generator	0,60		
		ESWS buried pipes	0,55		(7)
25	Diesel generator automatic load sequencer			0,50	(2)
26	Ventilation system	Frame of the pedestal	0,57	0,57	
		load-bearing frame	1,10		
		side panels	0,62		
		testing of active components	0,62		
27	Radiation monitoring system			0,50	
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		

№	System/Equipment	Element/Failure mode	HCLPF, g	HCLPF, g (minimal)	Note
		RDGS	0,44		(8)
		ECLS Pump Stations 1 and 2	0,621		
		Dry Spent Fuel Storage	1,18		
		Auxiliary Building	0,77		
		Ventilation Stack	2,30		(8)
		Intake Water Building N 2	0,46		(8)
29	Operative communication			0,50	(2)

Table 2-4: Summary Table of Seismic Margin Assessment for ANNP SSEL components at RLE with PGA = 0,35g

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, [55], the rationale is given in the report [67].

(4) HCLPF value is revised in comparison with data presented in report Rep08.ARM9014-89032S, (see Appendix B, item 11), the rationale is given in the report [67].

(5) Rep09.ARM9022-84188 (see Appendix B, item 36) 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, (see Appendix B, item 43), the rationale is given in the report [67].

(7) HCLPF value is revised in comparison with data presented in report Rep12.ARM9014-89032S, (see Appendix B, item 14), the rationale is given in the report [67].

(8) HCLPF assessment is given in the report [67].

The following conclusions could be drawn from the results presented in Table 2-4:

1. No cliff edge effects were identified for the items included in SSEL.

2. 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.

3. 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 [74] 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.

4. 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 2-4 follows, that the minimum HCLPF value for the systems provided sealing and integrity of the Confinement is 0,49 g. 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 0,98 g. 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 for the plant 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 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. Modification of the emergency gas removal system in compliance with the project “Gas removing system from under the Reactor's Lid, ANPP Unit2".

3. In order to provide an emergency power supply for period of 72 hours, an additional fuel tank with capacity of ~50 to 100 tons will be installed. The fuel tank shall have seismic design and meet the required safety standards.

4. Performing of seismic margin evaluation of the fire extinguishing system and issuing of recommendations (reinforcing, modification, etc.) for seismic

safety improvement up to DBE and implementation of the issued recommendations.

5. Perform analysis of the SRS damage in case of explosion of nitrogen recipients and hydrogen storage tanks.

6. Perform analyses for the consequences of the possible flooding of the following rooms and compartments: CDD-6/0,4kV room, intermediate relay panel–2 rooms, flooding of the MCP control room through the ceiling if the Turbine Hall (elevation of 14,0 m) is flooded. On the basis of these analyses the relevant protective measures will be developed.

7. Implementation of the program for seismic upgrading of I&C equipment and monitoring system.

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APPENDIX B
LIST OF WORKS PERFORMED FOR RLE SEISMIC REEVALUATION

№	Description of work	Contractor	Year	Note
1	Developing of the Safe Shutdown Equipment List (SSEL)	ANPP	2007	
Calculation of the floor response spectra:				
2	RLE Floor Response Spectra for the ANPP, Unit 2 Main Building, report A-101a-1	NRSC	2006	
3	RLE Floor Response Spectra for the ANPP, Unit 2 RDGS, report A-101a-2	NRSC	2006	
Civil structures seismic analysis:				
4	RLE Seismic Analysis for the Main Building Pedestal, ANPP, Unit 2, report NRSC A-101.03/02	NRSC	2004	
5	RLE Seismic Analysis of the ANPP, Unit 2 Civil Structures. Ventilation Stack, report NRSC A-205/1	NRSC	2007	
6	RLE Seismic Analysis of the ANPP, Unit 2 Civil Structures. Main Building, report NRSC A-205/2	NRSC	2007	
7	RLE Seismic Analysis of the ANPP, Unit 2 Civil Structures. RDGS, report NRSC A-205/3	NRSC	2007	
8	Final Seismic Walkdown	IAEA S&A (CZ) CVS ANPP	2007	
Analyses performed within project for Final Seismic Walkdown:				
9	Seismic Margin Assessment of "Cold" injection Pressurizer Piping (DN100). ANPP Unit 2. №02.ARM9014-89032S	CVS	2008	
10	Seismic Margin Assessment of Cable Trays, report №07.ARM9014-89032S.	CVS	2008	
11	Seismic Margin Assessment of Sprinkler Pipelines located in Boron Compartment, №08.ARM9014-89032S.	CVS	2008	
12	Seismic Margin Assessment of Sprinkler Pumps, №09.ARM9014-89032S	CVS	2008	
13	Seismic Margin Assessment of Emergency Seismic Pump Piping within Boron Unit (system of Primary Circuit High Pressure Emergency Cooling), №14.ARM9014-89032S	CVS	2008	

№	Description of work	Contractor	Year	Note
14	Seismic Margin Assessment of Buried pipes (Service water system from Pump Station N 1 up to Boron Compartment), №12.ARM9014-89032S	CVS	2008	
15	Elimination of seismic deficiencies found during seismic walkdown according to 2008 plan-schedule	ANPP	2008-2011	
Analyses and Tests performed within project ARM 9022:				
16	Seismic Margin Assessment of "Cold" injection Pressurizer Piping (DN100) Damper Support. № 20.1.ARM9022-84188	CVS	2011	
17	Seismic Margin Assessment of the Precast Concrete Panel (DG and Turbine Buildings) Unit 2 ANPP, №14.ARM9022-84188	CVS	2011	
18	RDGS Wall Panel Seismic Test (Protocol of 23 March 2012)	ANPP CVS	2011	
19	Seismic Margin Assessment of the Joint Column Line between the Turbine Hall and the Electrical Building UNIT 2 ANPP, №15.ARM9022-84188	CVS	2011	
20	Seismic Margin Assessment of the Boron Facility, Unit #2 ANPP, №16.ARM9022-84188	CVS	2011	
21	Seismic Margin Assessment of Pipelines of blowdown, blowdown return and primary circuit drainage, Unit 2 ANPP, №03.ARM9022-84188	CVS	2011	
22	Seismic Margin Assessment of the discharge piping from the Pressurizer to Barbotage Tank, Unit 2 ANPP, №04.ARM9022-84188	CVS	2011	
23	Seismic Margin Assessment of plant essential service water system. Unit 2 ANPP, №05.ARM9022-84188	CVS	2011	
24	Seismic Margin Assessment of Pipelines Spent Fuel Pool Cooling system, Unit 2 ANPP, №06.ARM9022-84188	CVS	2011	
25	Seismic Margin Assessment of Pipelines Spent Fuel Pool Cooling system Supports. Unit 2 ANPP, № 20.2. ARM9022-84188	CVS	2011	
26	Seismic Margin Assessment of SG auxiliary feedwater system. Unit 2 ANPP, №07.ARM9022-84188	CVS	2011	
27	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 1), №08.1.ARM9022-84188	CVS	2011	
28	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 2). №08.2.ARM9022-84188	CVS	2011	
29	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 3). №08.3.ARM9022-84188	CVS	2011	
30	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 4). №08.4.ARM9022-84188	CVS	2011	

№	Description of work	Contractor	Year	Note
31	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 5). №08.5.ARM9022-84188	CVS	2011	
32	Seismic Margin Assessment of Main Steam and Feedwater Lines (Loop # 6). №08.6.ARM9022-84188	CVS	2011	
33	Seismic Margin Assessment of Main Steam Lines Supports. Unit 2 ANPP, № 20.3. ARM9022-84188	CVS	2011	
34	Seismic Margin Assessment of Feedwater Lines Supports. Unit 2 ANPP, № 20.4. ARM9022-84188	CVS	2011	
35	Seismic Margin Assessment of the Piping Wall Hermetic Penetrations (Main Steam and Feedwater Lines), №08.7.ARM9022-84188	CVS	2011	
36	Seismic Margin Assessment of SG blowdown system. Unit 2 ANPP, №09.ARM9022-84188	CVS	2011	
37	Seismic Margin Assessment of SG blowdown system Supports. Unit 2 ANPP, №20.5.ARM9022-84188			
38	Seismic Margin Assessment of the Primary Circuit Emergency Makeup System, Unit 2 ANPP, №10.ARM9022-84188	CVS	2011	
39	Seismic Margin Assessment of the Primary Circuit Emergency Makeup System Supports. Unit 2 ANPP, №20.6.ARM9022-84188	CVS	2011	
40	Seismic Margin Assessment of Boron Solution Supply Piping, Unit 2 ANPP, №11.ARM9022-84188	CVS	2011	
41	Seismic Margin Assessment of the Control Rods and Movable Fuel Elements, Unit 2 ANPP, №12.ARM9022-84188	CVS	2011	
42	Seismic Margin Assessment of the primary circuit temperature sensors Unit 2 ANPP, №13.ARM9022-84188	CVS	2011	
43	Seismic Margin Assessment of Demineralized water reserve tank, №17 ARM 9022-84188	CVS	2011	
Development of design solutions within the project ARM 9022:				
44	Pipeline venting of the condenser. Unit 2 ANPP, ARM9022-84188.R00.00.2804NNA00	CVS	2011	
45	Demineralized water reserve tank. Unit 2 ANPP, ARM9022-84188.R17.00.REC1-00	CVS	2011	
46	"Cold" injection Pressurizer Piping Unit 2 ANPP. Damper Device, ARM902284188R00.00CIM53-00	CVS	2011	

№	Description of work	Contractor	Year	Note
47	Spent Fuel Pool Cooling Pipelines. Unit 2 ANPP, FP ARM902284188.R06.00.REC(00÷03)-FP	CVS	2011	
48	Feedwater Pipelines. Unit 2 ANPP, ARM9022 84188.R08.(00÷06).REC(00-02)-FW	CVS	2011	
49	Main Steam Pipelines. Unit 2 ANPP, MS ARM9022- 84188. R08.(00÷06).REC(00;03)-MS	CVS	2011	
50	Primary Circuit Emergency Makeup Pipelines. Unit 2 ANPP, ARM9022-84188.R10.00.REC(00÷003)-PCEM	CVS	2011	
51	SG blowdown pipelines. Unit 2 ANPP. ARM9022-84188.R09.00.SG(1÷6).REC(00÷04)-SGB	CVS	2011	
52	Implementation of the seismic improvements according to 2012 time schedule	ANPP	2012	

Changes recording list

Item No	Page numbers			Basis for making changes (in the copy of Customer)	Name and/or signature	Date	Validity date
	Changed	New	Cancelled				

3. CHAPTER 3: FLOODING

3.1. Design basis

3.1.1. Flooding against which the plant is designed

3.1.1.1. CHARACTERISTICS OF THE DESIGN BASIS FLOOD (DBF)

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 934,5 m 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 860 m above sea level;
- The Aknalich Lake is 5 km away from the site; the lake is located at 880 m above sea level.

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

The groundwater is about 86 to 95 m 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 excluded.

The existence of sloping of the plant site area can result in run-off due to precipitation on the upstream area (northward) of the site. For the protection of the site against mudflows on north-western side of the plant, where the sloping is relatively sharp, a mountain canal is created. The ESWS cooling ponds and pump stations are located on this side of the plant. This part of the site is protected by a dyke. The mountain canal is, at least, 0,8m below (from 1 to 3 m, on the most of the length) the top of the dyke. The site's fence concrete wall is located on top of the dyke. This wall can serve as additional protection against water flows in case, during extreme weather conditions, the flow rate is significantly higher than the design value and the level of the canal is increased up to the top of the dyke.

On the northern side of the plant site (near the location of open switchyard) the slope of the ground is negligible and there is no risk of intense mudflows formation. This statement is made on basis of the walk-down specially performed for the stress-tests self-assessment in order to identify potential flooding threats. The site's fence concrete wall can serve as additional protection against water flows in case, during extreme weather conditions, insignificant mudflows are nevertheless formed on the northern area of the plant site. Based on the above the conclusion is made that there is no risk for penetration of mudflows into the plant site.

Based on the description of the geographical location and relief 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 not considered as hazard and the Design Basis Flood was not postulated nor analyzed in the plant design.

The only hazard for ANPP that can be considered is the possibility of flooding of certain parts of the site due to heavy rainfall and not effective operation of water draining systems.

Concerning the hydrogeological conditions of ANPP site it should be noted that geology creates favorable conditions for infiltration of atmospheric precipitations.

Draining systems are being operated at the ANPP in order to ensure controlled drainage of surface waters. The first system is dedicated to the plant site and the second one to the site of spray ponds and pump stations of Essential Service Water System (ESWS).

The draining systems operate on basis of gravitational principle (on basis of level difference). The collectors and pipes of the system are located on the ANPP site and also outside of the site down to Sevjur River. The capacity of the collector is 1200 m3/h. Manholes are installed on the collectors and pipes at distance from 25 up to 50 m. These manholes are designed for cleanout and inspection of the collectors and pipes.

The format and content of the chapter somewhat differs from the ENSREG specifications and WENRA format due to the fact that the Design Basis Flood was not postulated nor analyzed for the plant site (due to favorable location of the site) and that measures were not foreseen to protect the plant site against external flooding. In §3.1.2.1 an analysis, taking into consideration very conservative assumptions, of the possibility of flooding of certain parts of the site is detailed.

3.1.1.2. METHODOLOGY USED TO EVALUATE THE DESIGN BASIS FLOOD

As already mentioned the Design Basis Flood was not considered for the ANPP site due to favorable location conditions.

3.1.1.3. CONCLUSION ON THE ADEQUACY OF PROTECTION AGAINST EXTERNAL FLOODING

As already mentioned the water basins existing near the ANPP site are more than 50 m lower than the absolute ground elevation of the plant. Consequently no protective measures against external flooding are foreseen.

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

3.1.2.1. IDENTIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS (SSC) THAT ARE REQUIRED FOR ACHIEVING AND MAINTAINING SAFE SHUTDOWN STATE AND ARE MOST ENDANGERED WHEN FLOOD IS INCREASING

A detailed analysis of possible flooding of different rooms of the plant is carried out with the objective to identify vulnerable systems and equipment. The analysis is performed with assumption of sequential failures at different levels and using (where necessary) engineering judgment in compliance with the ENSREG stress tests specifications.

An analysis of the operation of the draining system of the ANPP site during a rainfall with probability of 0.01 % (recurrence once every 10.000 years) was performed in the frame of the external event PSA [2]. One of the objectives of the analysis was to define if the failure of equipment is possible due to high water level on the site. The analysis results show that the existing draining system practically excludes the flooding of NPP rooms and that there is no hazard of failure of safety-related equipment.

The following parameters are considered in the PSA:

- Frequency of exceedance: once per 10.000 years;
- Intensity 286 l/sec/ha for a time period of 20 minutes (PSA results show that for all frequencies of exceedance the precipitation for 20 minutes interval makes about 50% of the 24-hours interval precipitation).

In Table 3-1 the 24-hours maximum precipitations for different frequencies of exceedance as well as the observed maximums are presented [2].

Station	**Observed maximum**	**Frequency of exceedance, years**							
		100 000	**10 000**	**1000**	**100**	**50**	**20**	**10**	**5**
Aragats	64	104	88	70	54	48	36	31	24
Armavir	42	81	68	54	41	38	29	26	19
Ashtarak	56	99	88	68	52	46	35	30	23
ANPP	44	99	85	66	49	44	32	27	20
Yerevan agro	40	77	62	52	41	37	29	25	21
Yerevan Zvartnots	34	72	62	50	39	35	27	24	20
Karakert	45	105	90	70	52	46	34	28	20
Talin	63	109	92	76	58	52	39	32	25
Echmiadzin	46	86	72	58	44	40	29	25	19

Table 3-1: 24-hours maximum precipitations (mm) for different frequencies of exceedance

In Table 3-2 the intensities of 20-min interval for different frequencies of exceedance are presented for the Armavir meteorological station which is the nearest station to the plant site [2].

Station	**Frequency of exceedance, years**							
	100 000	**10 000**	**1000**	**100**	**50**	**20**	**10**	**5**
Armavir	350,0	286,7	225,0	165,0	143,3	110,0	83,3	56,7

Table 3-2: Intensities of 20-min interval rainfall (l/sec/ha) for different frequencies of exceedance

Taking into account the conservative parameters of rainfall considered in PSA, i.e. a frequency of exceedance once per 10.000 years, in coherence with good practice adopted in IAEA member states for addressing the external events, the evaluation of rainfall parameters for more rare frequencies of exceedance was not needed in the frame of the stress tests self-assessment. In order to meet the stress tests objectives some conservative assumptions have been made with the result of water accumulation in several rooms of the plant. The possible places of water accumulation and possible paths of water penetration to the rooms were identified during walk-down.

The analysis is performed with an assumption that the intensity and duration of rainfalls are higher than the conservative values taken into account in external event PSA, with hypothesis of inadequate operation of the site's draining system. The assumption is made that due to such extreme conditions some accumulation of water takes place on the site and that the heavy rainfalls are combined with strong winds.

In the performed analysis the following rooms which house equipment of safety systems are considered:

- ESWS pump stations where the equipment of Essential Service Water System is installed;
- Diesel-generator plant where the equipment of II Category Emergency Power Supply System and Additional Cooling System is installed;
- Boron Unit Nr 2 where the equipment of following systems is installed:
    - Reactor Installation High Pressure Emergency Cooling System;
    - Primary Emergency Make-up System;
    - Spray System.
- Boron Unit Nr 1 where the SG make-up diesel pump is installed;
- Turbine hall (lower levels) where EFW pumps are installed and from where the water can overflow to cable tunnels and boron unit.

The possibility of flooding of cable tunnels, 6/0,4 kV switchgear set rooms and underground cable tunnels is also analyzed.

The equipment used for cooling of spent fuel pools is not liable to failure as result of rainfalls because the equipment is installed in rooms above the ground level:

- 1(2) FSPHE-1,2 – room B-214/1(2), level 6,3;
- 1(2) FSSFP CP-1,2 – room B-110/1(2), level 2,7.

In the analysis, loss of off-site power is conservatively assumed. This assumption is considered as conservative because the analysis didn't reveal any reason that the considered weather conditions could lead to failure of equipment ensuring power supply from the grid. Although the occurrence of strong wind could lead to the tearing off of the metal sheets of the cooling tower coating or of the turbine hall roof which could drop on power transmission lines and produce short circuit. Nevertheless the probability that all power transmission lines are lost and that there is no possibility to restore any of them is very low.

Essential Service Water System (ESWS)

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 part of the site where the pump stations are located is protected with dike on three sides; the open fourth side is located in the direction of the ground sloping. The site is also protected against mudflows from the neighboring rising grounds by the site's fence concrete wall (about 2 m high) which is located on the top of the dyke;
- Outside the dyke perimeter a canal is foreseen for disposal of the rain water and also a mountain canal for disposal of possible mud flows;
- Open conduits are foreseen around the cooling pools and pump stations for disposal of the rain water;
- The door to the pump station is hermetic and located about 10 cm above the surrounding area;
- The relief of the area surrounding the pump station excludes any water accumulation near the doors;
- The pump stations don't have any doors/penetrations that are located lower than the surrounding area level.

Taking into consideration the above-mentioned measures we can exclude important quantities of water entering into the pump stations from outside.

Open switchyard, 6/0,4 kV switchgear sets, underground cable tunnels

The open switchyard of the site is protected against possible insignificant mudflows from the neighboring rising grounds by the site's fence concrete wall (2 m high). The switchyard equipment is installed on the concrete foundations. The rainwater will be removed from the switchyard side thanks to the site's sloping and the infiltration. Even during the heavy rainfall the accumulation of the water on the switchyard site and occurrence of such a water level that can lead to equipment failure are excluded.

6/0,4 kV switchgear electric distributors are located in the reactor department rooms at the ground level, and the open cable connections are located about 15 to 20 cm above the floor. Even in case of water penetration the occurrence of such a water level that can lead to equipment failure is excluded. These rooms don't have any compartments that are located lower than the surrounding landscape level. Therefore there is no door/penetration that could be broken under the water column static pressure.

The underground cable tunnels are waterproof and are located about 2 m under the ground.

Diesel Generator Station (DGS)

The DGS has three separate compartments; each of them has its own basement area. In this area some equipment are installed for which their failure can lead to the DGs' failure. The level of basement room floor is -3.65m. Five DGs are installed in DGS (2 for each train of II category emergency power supply system and 1 for the SEC system).

Water can penetrate the basement area through the external non waterproof emergency door at the level of basement and also can drain from the main compartment when water penetrates there through the roof leakage or through damaged windows due to the strong wind. The emergency door is always closed (is under surveillance of "Regime" department). The door is 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. The ways of water inflow to the cavity are the following:

- Water can drain there when a water accumulation takes place on the building blind area;
- Rain water cans directly inflow the cavity when the rain is combined with corresponding direction wind.

With the water level rise in basement area, the following equipment (that can potentially lead to DGs failure) is under the threat of failure:

1. Located at about 45 cm from the floor the motor of the fan of DG generator air cooling system can fail;

2. Located at about 50 to 55cm from the floor some valves on the lines of cooling water can fail; at this level the control assemblies of valves are installed (for some valves the control assemblies are installed at about 1m above the floor level). These valves are being opened automatically at DG start-over;

3. The motors of "hot standby" water pump and "hot standby" oil pump which ensure the hot standby state of the DG are installed at about 70 to 75 cm from the floor. Herewith, after some hours (depends on the temperature conditions in DGS) the oil temperature can decrease below 30 ^{o}C and the DG start over will become impossible. The failure of these pumps during the operation of DG cannot influence the operability of the DG;

4. Located at about 1m from the floor the DG pre-starting pump can fail in consequence of which the DG start-over will become impossible. The failure of this pump during the operation of DG cannot influence the operability of the DG.

Therefore, during the operation of DGs the flooding of the basement areas of DGS can lead to DG failure in the case of:

- Generator air cooling failure;
- Unintended closing of valve on the cooling water line cause of the water penetration to the control assembly.

The DGS basement areas are equipped with conduits of "dirty" drainage system. The water that theoretically can penetrate the basement area during the heavy rainfalls will be removed by this 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 pumps (see Fig. 3-1). The consequence of assumed loss of off-site power is the loss of power supply of DFP pumps, as the buses from where the power is supplied cannot be powered from DGs. As result accumulation of water and increase of the level in the system will take place. An important increase of level will result in occurrence of water in the DGS basement areas as well as in the turbine hall (level -3.6m), as these areas are 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 (level -2.3 m). This corresponds to a level of 1.35 m above the floor of the basement.

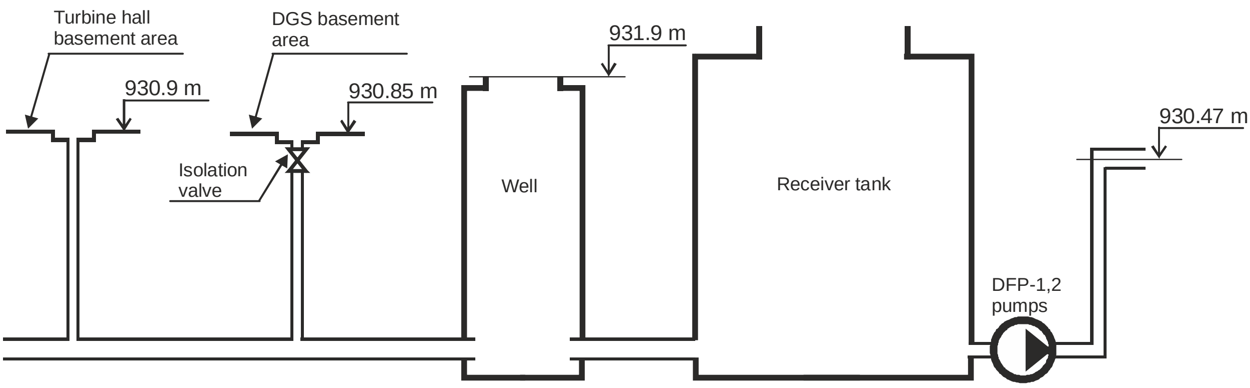


Figure 3-1: Elevation marks in drainage system

The walk-down on the DGS surrounding area didn't reveal any ways for penetration of important volumes of water to the DGS basement areas. The facts presented hereafter allow excluding the possibility of inflow of important volumes of water into the cavity where the basement area emergency door is:

- The area of the cavity is small (excluding the possibility of direct rainfall into the cavity or of runoff from the building walls and shed);
- The outlet canal of the condenser cooling system protects the DGS from the water flows coming from upper parts of the site (when there is an inflow of important volumes 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; the water will bypass the DGS (see Fig. 3-2);

- The DGS building is located far enough from the main roads, from which the rain water will flow down on the site (we can expect that the water will mainly flow from asphalted roads due to the fact that the resistance to the flow on these roads will be lower than on the parts covered with vegetation).

It should be noted that the non hermeticity of emergency doors and the leakage through it was taken into account in the analysis. Therefore there is no need to consider the design pressure that the door can support without leakage.

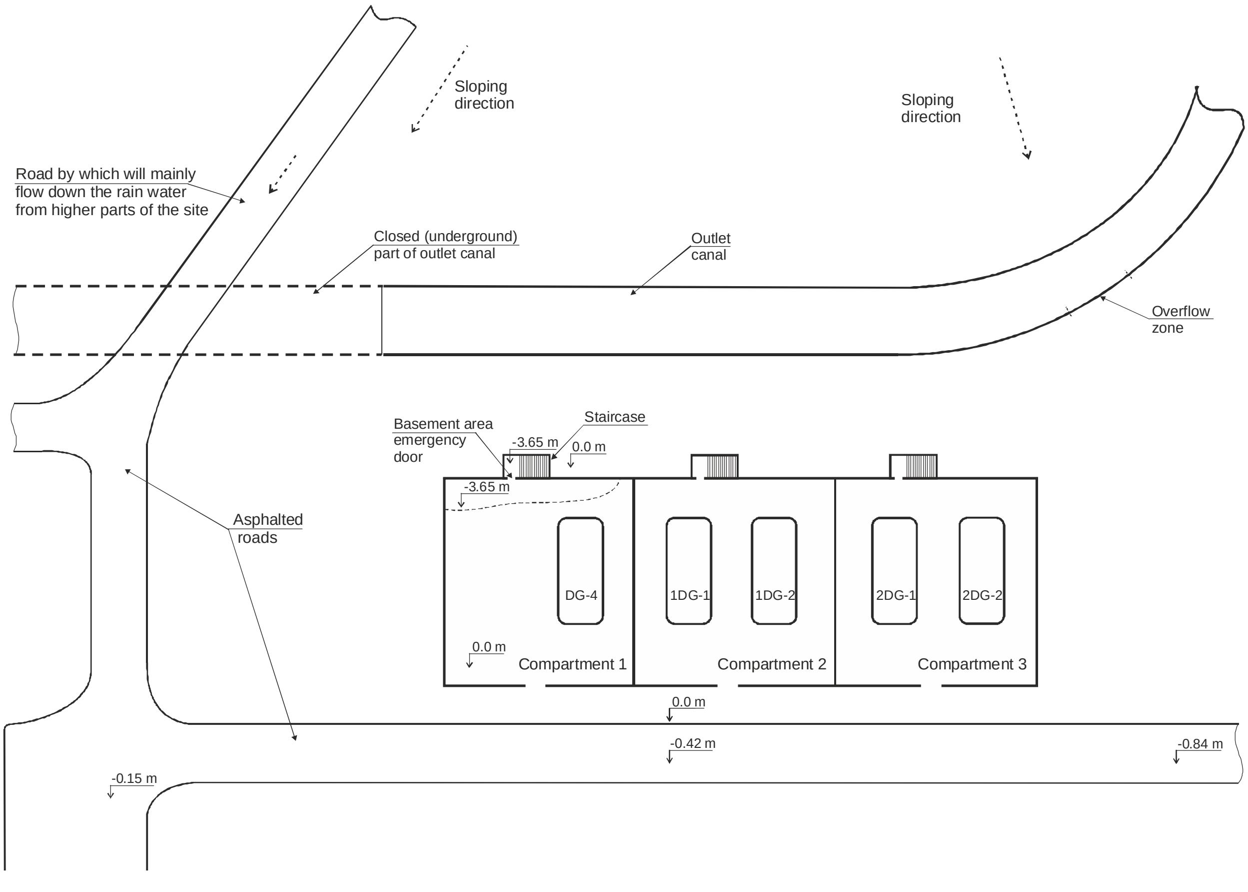


Figure 3-2: Diesel-generator plant location

The volume of the "dirty" drainage system pipes from turbine hall up to the well of the receiver tank is about 45 to 50 m3, and the working volume of the receiver tank is 20 m3. Thus, when about 60 to 65 m3 of water is accumulated in the system, the water will not be removed anymore from DGS and turbine hall basement areas. These areas are at about same level and are communicating vessels. Therefore the water level increase will be simultaneous in both areas.

The turbine hall is about 8.700 m2. Some part of this area (visually estimated at no more than 10-15 %) in basement area, is taken up by columns, supports of equipment and pipelines as wells as pipelines. If conservatively we assume this part equals to 25 % of this area, each centimeter of height equals to about 65 m3 of volume. It means that about 2.500 m3 of water is needed to reach the critical level of 45 cm inside the Turbine Hall.

Despite it is impossible to assess by calculations the flow rate of water that can penetrate into the rooms (and assess the speed of water level increase), it can be argued that the personnel will have enough time to undertake appropriate measures (e.g. set-up barriers on the ways of water flow, water pumping out using fire machines). But it should be noted that the current technical means in operation don't inform the operator about the occurrence of water level in the basement area of the DGS. This can result in delay in undertaking protective measures for preventing the flooding.

The only scenario when the flooding of the DGS basement area is possible is water inflow from outside with simultaneous failure of drainage system. In this scenario the water accumulation will take place mainly due to water inflow through non waterproof emergency door. In order to prevent a return flow, i.e. inflow from the drainage system, a cut-off valve is foreseen. The closing of the cut-off valve will prevent the flooding of the area if there is no water inflow from outside.

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. When the fan is not in operation the temperature of coils increases. Some alarms are actuated on controlled temperatures when the temperature increases. There is no protection for the cutting-off of the generator. In these conditions the generator can operate till the degradation of the coils.

The analysis performed allows drawing the conclusion that the possibility flooding of the DGS basement areas cannot be excluded in the considered scenarios and this will result in DGs failure. But such a scenario is possible only during very long rainfalls and in case it is impossible to undertake any protective measure.

Turbine Hall

The water can penetrate the turbine hall through:

- The doors at the ground floor in conditions of accumulation of water in surrounding area;
- The turbine hall roof; In case of strong wind the damage of the roof is not excluded (the wind can tear off poorly fixed metal sheets of roof);
- Damaged windows (in case of strong winds the damage of windows is not excluded).

The flooding of turbine hall (basement area, level -3,6m) can lead to:

- Overflow of water into the cable tunnels;
- Overflow of water to the boron unit of power units Nr 1, from the doors of emergency exit to the staircase on the level -3,6m (see Fig. 3-3);
- Failure of EFW pumps which are installed at the level “-1,8 m”;
- Possible failure of cables installed in cable ducts on line "B" at the level “-1, 8 m”.

The other important equipment for safety is above the ground level.

The water entering into the turbine hall will be drained to the basement area. Walk-down in the turbine hall reveals that during the draining of water from the ground floor, the water could not fall on equipment important for safety.

The basement area of the turbine hall is equipped with conduits of "dirty" drainage system. The operation of this system is analyzed in paragraph 3.1.2.1.3. According to the analysis the maximum level of water in the considered scenario can be 1,3 m. The cable ducts (line "B") are at 1,7 to 1,8 m from the floor, and the EFW pumps are installed at 1,8 m from the floor. We can therefore exclude the failure of this equipment in the considered scenarios.

The analysis performed allows concluding that the only risk of flooding of the turbine hall during heavy rainfalls is the possible water overflow to the boron units and cable tunnels.

It should be noted that the non hermeticity of doors located on the lower levels of the turbine hall and the leakage through them was taken into account in the analysis. Therefore there is no need to consider the design pressure that the door can support without leakage.

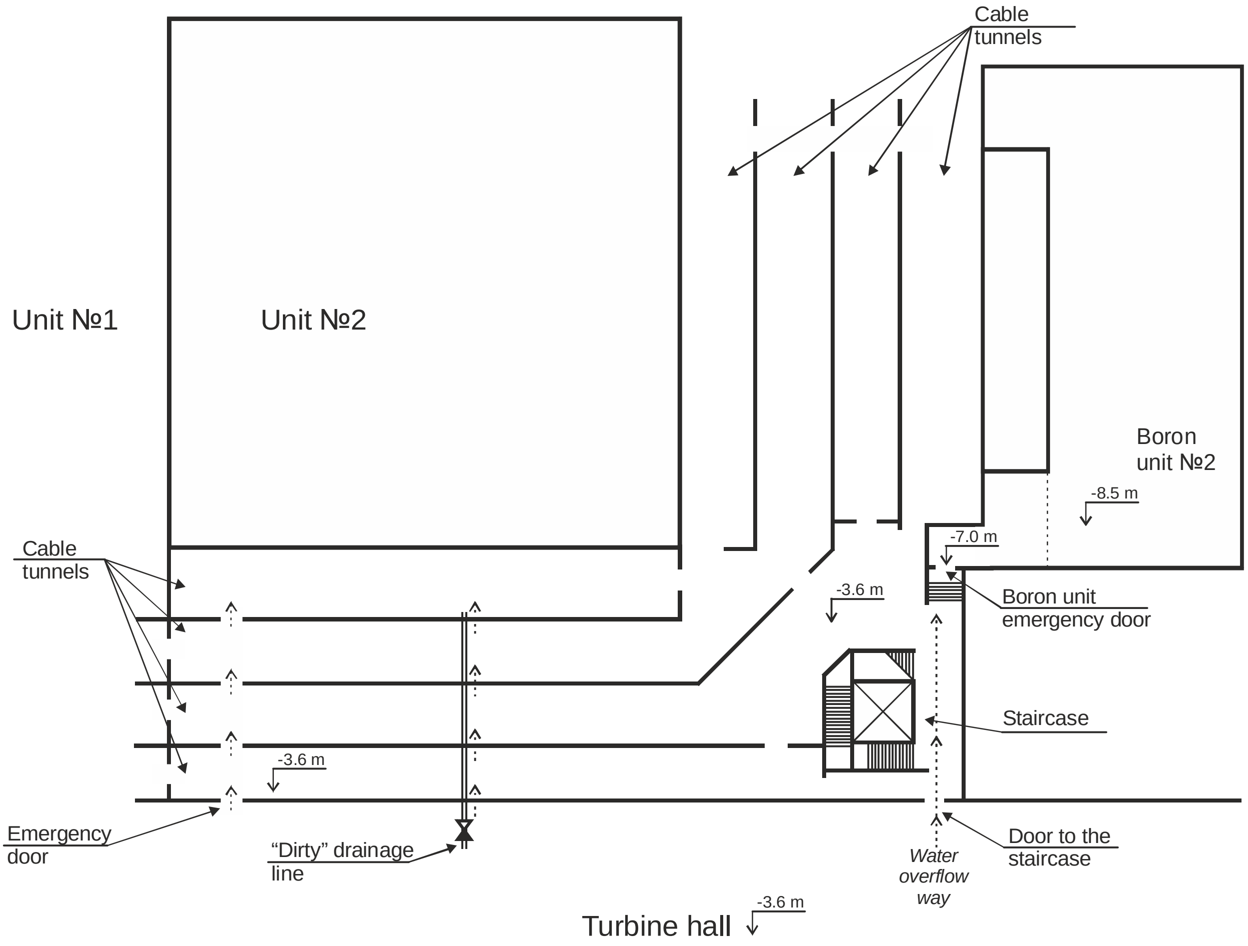


Figure 3-3: Rooms arrangement of the Unit 2 (for Unit 1 the arrangement is identical)

Boron unit rooms

The water can penetrate into the boron unit rooms through the non-waterproof emergency door, exiting to the staircase connected to the turbine hall by a non-waterproof door. The boron unit emergency door is about 3.5 m lower than the basement area. The door is opened 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 unit room under static pressure of the water, the flow rate will not be so important due to the very small opening surfaces.

The boron room of Unit 2 is equipped with two drain pumps to remove water from the boron unit sump. The capacity of each pump is 20 m^3/h. The pumps are energized from both I and II trains of II category emergency power supply system. The pumps are actuated automatically. In addition a third drain pump of submersible type with flow rate of 65 m^3/h is installed in the boron unit. The pump is energized from II category emergency power supply bus bar. Power supply from the SEC system is also foreseen. The probability of simultaneous failure of all 3 pumps is very low. The scenario with the failure of the 3 pumps will lead to the failure of all DGs (with condition of loss of offsite power).

The boron Unit 1 is also equipped with two drain pumps actuated automatically.

Considering that the drain pumps have redundancy, the flooding of boron units can be excluded in the considered scenarios. 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 performed assessments about 350 m^3 of water should penetrate to the boron unit room to lead to an equipment failure).

Cable tunnels

The water can flow from the basement area of the turbine hall to the cable tunnels through a non-waterproof emergency door (see Fig. 3.2). 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, but we cannot exclude the valve of being sticked in open position or the leak-free closing of this valve during its periodical actuation (the position of valve is not checked). Between these areas some pipe penetrations exist, but they are sealed and are at the level of 1.5 to 2 m.

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 cable trays are on different heights, some of them are very close to the floor. It is difficult to judge about the consequences of the flooding as the water cannot damage the cables' insulation, but if some cables had damaged insulation, they can fail.

I can be concluded that the flooding of the turbine hall will certainly lead to cable tunnels flooding. As a consequence some equipment could lose their power supply or some measurements could be lost.

The cable penetrations passing through the ceiling of the rooms or through higher parts of the walls are at such level that the water cannot reach them, in the considered scenarios.

Drainage of rain water from buildings' roof

One of the hazards related to heavy rainfalls is the possible accumulation of water on the roofs in case of inadequate drainage systems performance and then, possible collapse of the roof in case the structural capacity is not strong enough to support the maximum anticipated accumulation.

For assessment of hazard of roof failure due to snowpack's weight, the loads for which the roofs can withstand were assessed in the external event PSA:

- Reactor building: 1,28 kN/m^2, equal to 128 mm water column;
- Turbine hall: 1,8 kN/m^2, equal to 180 mm water column;
- DG station: 2,69 kN/m^2, equal to 269 mm water column;
- Service water pump station: 1,8 kN/m^2, equal to 180 mm water column.

In the external event PSA the maximum intensities of rainfall for different time periods were assessed. For 24-hours period the maximum value of integral precipitation for recurrence once every 100.000 years is 99 mm for the plant site area. Within the results got for the other meteorological stations the maximum for this parameter is equal to 109 mm. The observed maximum by the meteorological stations is 64 mm (see Table 3-1).

The intense rainfall can take place during a very limited period of time (representative case for the ANPP site is 20-min interval with about 50% of 24-hours period precipitation). Even conservatively assuming that the entire 24-hours precipitation is fallen in a short period of time and even neglecting the removal of water from the roofs by their drainage systems, the precipitation will not create issue of roof failure.

For the assessment of the drainage system capacity, the operation of the drainage system of the reactor building, which has the minimum strength margin, is considered. The system has 4 separate discharges with a design capacity of about 27-46 l/sec/ha. The system design capacity makes 42.8 l/sec. The roof surface is 1.07 ha. The analysis is performed with the following assumptions on the extreme weather conditions and conditions of system operation:

- Rainfall intensity: 350 l/sec/ha (corresponds to the frequency of exceedance of once per 100 000 years for 20-min interval, see Table 3-2);
- Duration of rainfall with considered mean intensity: 40 minutes instead of 20 minutes (+100%);
- Mean intensity of long-lasting precipitation after heavy rainfall corresponds to the maximum intensity of the "intense rain": 49 mm/ 12 hours, i.e. 11.4 l/sec/ha;
- One of the 4 discharges of the system (with highest capacity) is clogged – system capacity is 27.9 l/sec (-35%).

The accumulation of water during the analyzed time period of 40 minutes will produce about 83 mm of water. This volume of water will be fully drained from the roof in 15.7 hours considering the long-lasting intense rain.

Thus, it can be stated that even during the extreme rainfalls which are followed by long-lasting intense rain, there is no hazard for roof failure due to rainwater accumulation.

Maintaining of safety functions

The analysis shows that during continuous heavy rainfalls, when considering very conservative assumptions, the following rooms are under risk of flooding:

- Basement areas of diesel-generator plant;
- Basement area of turbine hall and cable tunnels.

As it was already mentioned these areas are connected by the "dirty" drainage system, thus their flooding will be simultaneous, only if the DGS is not isolated from the system by closing of isolation valve.

Within the systems needed to maintain the fundamental safety functions or supporting functions under the risk of failure cause of the flooding can be II Category Emergency Power Supply System and Additional Cooling-down System part of which is the diesel-generator plant. The main risk in case of flooding of the DGS basement area is the loss of DG generator air cooling which will result in DG failure.

In the scenario when all DGs fail the only way to ensure the decay heat removal from reactor core remains the system of SG additional feeding system which includes a diesel pump (SG DFWP) installed in boron unit of the power unit Nr 1. The system possesses considerable autonomy and doesn't depend on plant support systems.

The room of boron unit Nr 1 can be flooded only when the power supply on both bus bars of II Category power supply is lost, or in result of failure of both drain pumps. The SG DFWP is installed on a base. The component that will first fail in contact with water is the battery which is also installed on the base. When the diesel pump is in operation the failure of electric part which ensures the control of equipment state will not lead to the stop or failure of the pump. Considering that the SG DFWP doesn't have any redundancy we have to assume also its failure (for reasons not related with flooding). Such scenario can lead to fuel damage (such scenarios are considered in Chapters 5 and 6 of this report).

For SFP-1,2 the emergency power supply and cooling systems (ESWS) are the same as for the reactor installation of Unit 2.

In [4] is performed analysis of initiating event related with failure of SFP cooling. Based on calculations, the time available for operators before any damage in case of loss of SFP cooling is assessed. The calculations are performed with very conservative assumptions for 2 cases: (1) complete core discharge into the spent fuel; (2) after reactor core refueling.

The results show that in case of core full discharge the time margin before the moment the water level reaches the top of the assemblies is 26,7 hours and for fuel damage, 33,0 hours. In case (2) these values are respectively 70,1 and 104,6 hours.

Such a time margin should be enough to restore the power supply from any source. If it is not enough to restore power supply, this time margin allows to assure water make up to the spent fuel pools to avoid damage to the fuel assemblies in the SFP. Additional means to assure water make up for the SFP in case of loss of SFP cooling are defined in chapters 5 or 6.

3.1.2.2. MAIN DESIGN AND CONSTRUCTION PROVISIONS TO PREVENT FLOOD IMPACT TO THE PLANT

The main design provision to prevent the flooding of plant rooms is the availability of drainage systems of rooms as well as plant site drainage systems, whose function is the removal of rain and melted snow water from ANPP area.

3.1.2.3. MAIN OPERATING PROVISIONS TO PREVENT FLOOD IMPACT TO THE PLANT

The protection of plant site against flooding is ensured by adequate operation of the site's drainage systems. To ensure effective operation of the systems, regular inspections are performed in order to exclude possible clogging of conduits and occurrence of barriers on the water flow path.

3.1.2.4. SITUATION OUTSIDE THE PLANT, INCLUDING PREVENTING OR DELAYING ACCESS OF PERSONNEL AND EQUIPMENT TO THE SITE

Two roads lead to the plant site: one from the side of highway from Yerevan city, and the second one from the side of Metsamor town. The first one, along its length, is sufficiently above the surrounding landscape for not being subject to flooding. The second one has a section which is on a relatively low level and is liable to water accumulation.

The accessibility through first road excludes the possibility of prevention of access to the plant site.

3.1.3. Plant compliance with its current licensing basis

3.1.3.1. LICENSEE'S PROCESSES TO ENSURE THAT PLANT SYSTEMS, STRUCTURES, AND COMPONENTS THAT ARE NEEDED FOR ACHIEVING AND MAINTAINING THE SAFE SHUTDOWN STATE, AS WELL AS SYSTEMS AND STRUCTURES DESIGNED FOR FLOOD PROTECTION REMAIN IN FAULTLESS CONDITION

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 (these issues are covered in [5]). No systems or structures are foreseen for flood protection.

3.1.3.2. LICENSEE'S PROCESSES TO ENSURE THAT MOBILE EQUIPMENT AND SUPPLIES THAT ARE PLANNED FOR USE IN CONNECTION WITH FLOODING ARE IN CONTINUOUS PREPAREDNESS TO BE USED

The flood is not considered as threat for the plant, and no mobile equipment is foreseen for use related to flooding.

3.1.3.3. POTENTIAL DEVIATIONS FROM LICENSING BASIS AND ACTIONS TO ADDRESS THOSE DEVIATIONS

In the frame of the justification of ANPP's safety, the issues related with flooding were not considered (due to above-mentioned reasons), thus the issue of deviations from licensing basis is excluded from the analysis.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

As the external flooding of the plant site is excluded due to its favorable location, the estimation of safety margin for water level increase is not applicable for ANPP.

The main weak points of the power plant from the point of view of flooding of rooms of the power plant during rainfalls are the turbine hall gates that have large openings (the walk-down have shown that the relief of area near the turbine hall door on the Unit 2 is favorable for water accumulation and there are large openings at the bottom of the door), non-waterproof doors of DGS basement area as well as the emergency doors between turbine hall basement area, cable tunnels and staircase.

The analysis has shown that from the point of view of maintaining the safety functions the main risk is the flooding of DGS basement area. "Cliff edge effect" is considered as the water level increase up to such a level that the DG air cooling system fan motor will fail.

With the purpose of preventing such an event, the following additional protective measures can be considered:

- Availability of means for building barriers on the way of water flows;
- Water pumping out of DGS basement area using fire machines.

Conclusion

Thereby, the flooding of ANPP site due to external natural phenomena is not possible. Only the accumulation of water in certain parts of the site is possible, related to important water flows and the non-adequate operation of drainage system.

In the performed analysis conservative assumptions were made at different levels. Theoretically the flooding of basement areas of the DGS and the turbine hall is not excluded. But considering that this flooding is only possible during very long heavy rainfalls and in case of very conservative assumptions, the probability of such a scenario is really low.

Though it is impossible to calculate the water volume that could penetrate the areas of the turbine hall and the DGS, it can be affirmed that hours, even days, of rainfall would be needed to flood these zones, with 2.500 m^3 of water.

In external event PSA [2] a conclusion is made that the value of precipitation for 20-minutes period makes about 50% of 24-hours period precipitation, thus about 50% of the precipitation fall during the first 20 minutes. When analyzing the data given in [3] (see Table 3-3) related to the intensity of the maximum of precipitation for different time periods, it can be concluded that the duration of heavy rainfalls cannot be more than 2-3 hours.

	Minutes			Hours		
	5	**10**	**20**	**1**	**12**	**24**
Precipitation intensity, mm/min	2,1	1,9	1,1	0,4	0,04	0,02
Integral precipitation, mm	10,5	19	22	24	28,8	28,8

Table 3-3: Precipitation maximum intensity for different time periods (mm/min)

With regard to the low process of water level increase in the basement of the DGS, it should be noted that a very long period of time will pass until all DG will fail due to flooding of the basement. It can be stated the before loss of power system of safety systems, the plant staff will have enough time to bring the reactor to "cold shut down" state. Considering also that, at that time, the decay heat will be relatively low; a long period of time will be available to perform remedial actions before reaching the conditions for which the core damage can occur. This same can be argued for spent fuel pools.

It should also be noted, from return of experience of ANPP operation, that even during the strongest rainfalls penetration of important volumes of water into the plant rooms never occurred, and no deficiencies were observed in drainage system operation.

Based on the analysis it can be stated that the maximum intensity of rainfall that the plant can withstand without any consequences corresponds to the rainfall parameters with frequency of exceedance less than once per 10 000 years. Such level of safety against external events is consistent with the criteria adopted in international guidance.

Based on this reasoning it can be concluded that the probability of serious consequences during heavy rainfall is rather low. However, the analysis shows that the plant has some weak points and the implementation of measures to increase the robustness is desirable with the purpose to prevent the inflow of rain water to the turbine hall and basement areas of the DGS compartments as well as to equip the plant and its staff with the appropriate hardware and procedures for mitigation of situation (see § 3.2.2).

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

As result of the stress tests self-assessment, the following measures are being planned in order to increase the robustness during the rainfalls:

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 both in DGS operator room and outside the DGS.

3. Seal the penetrations of the turbine hall located at the level of -3.6m.

4. Add operator’s actions in the DGS operation manual, in abnormal modes, in case of water inflow in the basement area.

5. Foresee mobile pumps for water pumping out of the DGS.

6. Develop and implement measures with the purpose of building barriers on the way of water flow to the turbine hall gates.

REFERENCES TO CHAPTER 3

1. ANPP: extension considering the heat supply of Yerevan city. Chapter 9. Construction and operation safety technical justification. Volume 1.
2. Level 1 PSA. External events. Volume 2. Essential events analysis. NRS STC, April 2004.
3. Level 1 PSA. External events. Volume 1. Screening of external events. NRS STC, April 2004.
4. Assessment of main time margins available for operator in case of storage pond cooling failure. UB.ETD.06.OYaB-001, 2011.
5. ANPP Unit 2 Safety Analysis Report.
6. ANPP site layout plan.

4. CHAPTER 4: EXTREME WEATHER CONDITIONS

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

Brief Description of the climate conditions at the ANPP site:

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/m².

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.

Winter in the Ararat valley is moderately cold. The coldest period is the end of January. In the winter months the region is subject, from time to time, to strong cooling of the runoff because of the cold air from the mountains and the stagnation of the water in the plain.

- The minimum temperatures reach about -27/-30°C;
- The snow depth is typically about 10 cm. 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.

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.

Data from meteorological stations are given in Table 4-1.

Stations	Height above sea level (m)	Distance from the nuclear power plant (km)	Work period (number of years)
Aragats	1254	40	65
Armavir	861	6	73
Ashtarak	1090	22	26
ANPP	945		31
Yerevan - Agro	942	20	52
Yerevan - Zvartnots	858	20	35
Karakert	1085	28	41
Talin	1582	30	60
Echmiadzin	853	15	82

Table 4-1: Data from meteorological stations

The following data have been selected for the ANPP site as a result of the analysis of available meteorological data.

Precipitation - average value

Average annual snow depth	120 mm
Average annual rainfall	298 mm
Average number of days of hail per year	2.1 days

Precipitation – maximum value

Maximum snow depth	240 mm
Maximum annual rainfall	340 mm
Maximum number of days of hail per year	2.4 days

Temperature and atmospheric pressure

Average temperature of the hottest month	26.2C
Average temperature of the coldest month	- 1.7°C

The maximum temperature recorded	+ 40.8°C
The minimum temperature recorded	- 30°C
Average value of atmospheric pressure	681 mmHg
Humidity	
Average humidity	56 %
Humidity of the coldest month	74 %
Humidity of the hottest month	44 %
Wind	
Average wind speed	1.1 m/s
Average wind speed during the coldest month of the year	3.0 m/s
Average wind speed during the hottest month of the year	2.1 m/s
The maximum recorded wind speed	24 m/s
The maximum recorded gust speed	32 m/s
Prevailing wind direction	
Cold period (11-3)	NE (31), E(20), W(14)
Warm period (4-10)	NE (34), E (21), SE(15)
In a year	NE (33), E (20), SE(13)
Repeatability of calms, %	
Cold period	64
Warm period	39
Year	50
Dust storms	
Average of days of dust storms	9.1 days
The greatest number of days of dust storms	21 days
Average duration	1.5 hour
Maximum duration	6.42 hours

4.1.1.1. VERIFICATION OF WEATHER CONDITIONS THAT WERE USED AS DESIGN BASIS FOR VARIOUS PLANT SYSTEMS, STRUCTURES AND COMPONENTS: MAXIMUM TEMPERATURE, MINIMUM TEMPERATURE, VARIOUS TYPE OF STORMS, HEAVY RAINFALL, HIGH WINDS, ETC.

The design of the nuclear power plant was carried out on the basis of construction norms and rules (Construction Norms and Regulations II A11-62, II Construction Norms and Regulations - 6-74).

According to these documents the design basis events were defined for the following extreme loads:

- Wind loading;
- Snow loading;
- High temperature;
- Low temperature;
- Hurricanes, tornadoes.

The geographical arrangement of Armenia excludes such extreme influences as cyclones, hurricanes, tornado, tsunami etc. Therefore for the ANPP hurricanes and tornadoes were not considered.

The following parameters were taken into account in the design basis of the plant:

- Wind loading: 45 kg/m^2 (0,445N/m^2), at a wind of 27 m/s;
- Snow loading: 80kg/m^2 (0,8kN/m^2);
- High temperature: +42 °C;
- Low temperature: - 40 °C.

4.1.1.2. POSTULATION OF PROPER SPECIFICATIONS FOR EXTREME WEATHER CONDITIONS IF NOT INCLUDED IN THE ORIGINAL DESIGN BASIS.

International practice, regulations and procedures require consideration of the following events at the NPP site:

- Air temperature;
- Wind;
- Precipitation;
- Extreme snow falls;
- Avalanche snow;
- Flood;
- Hurricane, tornado;
- Dust storms;
- Mudflow;
- Ice;
- Landslides and washouts;
- Lightning stroke;
- Tsunami;
- The ice phenomena on waterways;
- Mode of a coastal zone of the seas (surges, storm disturbance);
- Seiches;
- Inflow and big waves;
- Tropical cyclone (typhoon) and others.

The analysis of the above natural events for the region of the ANPP was executed in 2004 in the frame of PSA level 1 - «External events».

The selection of external initiating events that require detailed analysis is made taking into account the location of the ANPP site and plant design.

In order to draw up the final list of external initiating events, applicable to ANPP, all possible events, on the basis of the above-mentioned events, have been screened out according to the following criteria:

- Event not applicable to ANPP territory ;
- The frequency of occurrence of the event is less than 10^{-6} per year ;
- Events that cannot initiate a sequence of events leading to damage to the core (the influence on NPP).

On the criteria of inapplicability the following phenomena are eliminated:

- Tropical cyclone (typhoon);
- Tsunami;
- Inflow and big waves;
- The ice phenomena on waterways;
- Seiches ;
- Mode of a coastal zone of the seas (storm disturbance);
- Landslides and washouts;
- Avalanche snow.

These phenomena are impossible around ANPP site.

- The site is located far from large water pools;
- The phenomenon "avalanche" is impossible due to lack of forming factors (rather high massif with an unstable snow covering etc.).

Questions related to floods and site flooding as a consequence of rains and heavy rain in the area of the plant are considered in chapter 3 "Flooding".
Based on frequency of occurrence the following phenomena are eliminated:

- Tornado;
- Water tornado;
- High temperature of reservoirs;
- Ice/hoarfrost ;
- Meteorites;
- Volcanism (emission/activity), see § 2.1.1.2 and 2.1.1.3.

Based on the influence on the plant, the following phenomena are eliminated:

- Mudflow (see §3.1.1.1);
- Snow break;
- Extreme sunshine;
- Hail;
- Drought;
- Fog;
- Dampness;
- Biological phenomena (alga's, bacteria, fish, leaves);
- Natural fire.

The events to be considered are the following:

- Strong wind and tornado;
- Dust storms;
- Heavy rain;
- Air temperature;
- Snow (snow loading);
- Lightning.

These events are characterized by the following impacts on the plant:

Process, phenomenon and factor	**Potential impacts to the plant**
Air temperature	Temperature loads of buildings, constructions, networks and so forth.
Wind	Wind pressure. Flying objects.
Tornado	Wind pressure on buildings and constructions. Loadings from pressure difference between the periphery and the center of rotation of the funnel. Loadings from the flying objects which have been carried away by a tornado. Water carrying out from technological reservoirs coolers
Dust storms	Failure of ventilation systems and supply of air. Damage of rotating mechanisms
Snow (extreme snowfalls)	Drift by snow of access roads, platforms, communication lines, etc. Snow loads of a roof of buildings and constructions.
Ice	Extra loads on construction covered by ice. Break of cables, lines of transfers, deflection and roof collapse.
Lightning	Impact of an electric discharge on buildings, constructions, networks, equipment

Table 4-2: Impact of applicable weather conditions on the plant

From comparison of the received list of events and design assumptions it is visible that dust storms and lightning have not been considered within the ANPP design.

For the further analysis the following constructions and systems related to safety are defined:

- Main building;
- Turbine hall;
- CCWS (ponds and pump);
- Diesel Generator;
- Ventilation stack.

These structures are shown in red on Fig 4-1.

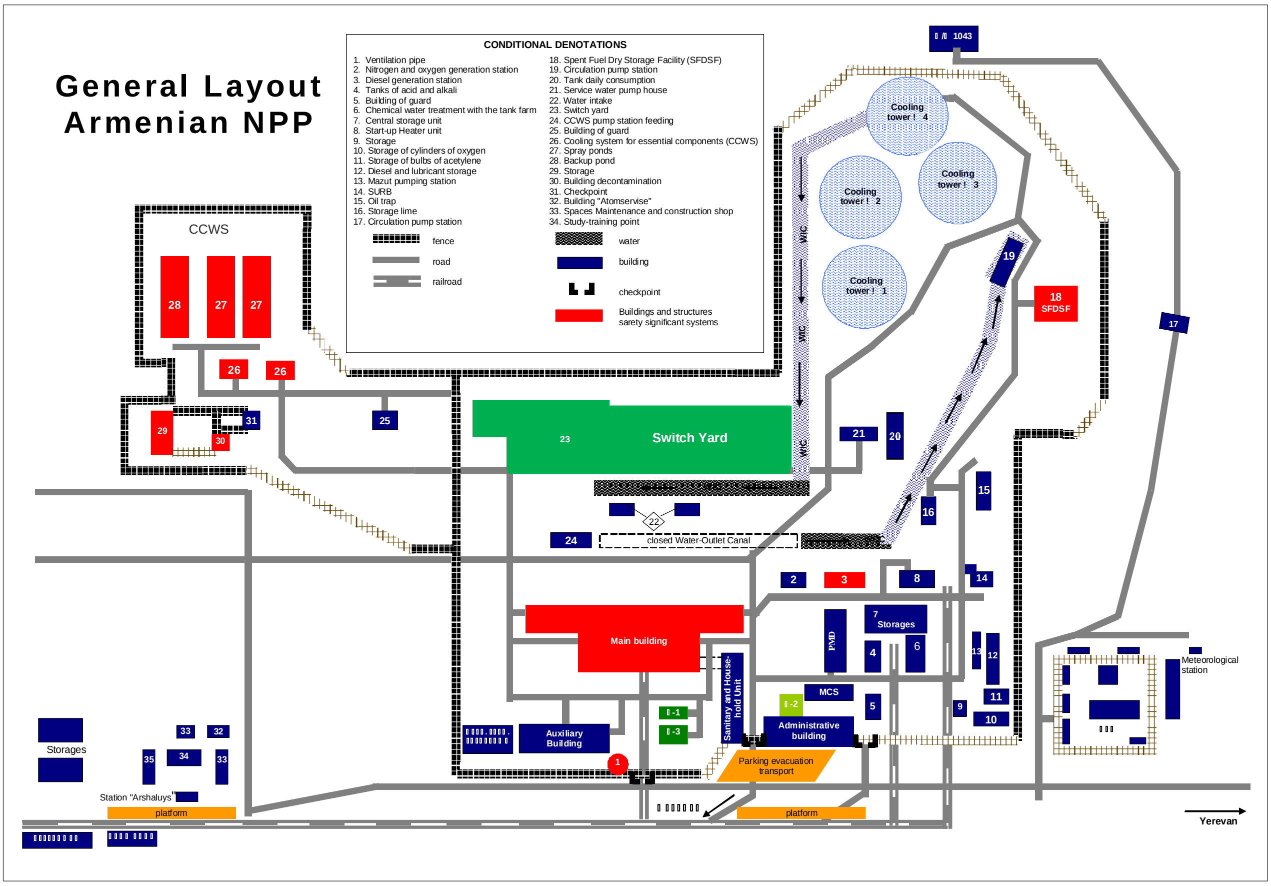


Figure 4-1: ANPP General Layout

4.1.1.3. ASSESSMENT OF THE EXPECTED FREQUENCY OF THE ORIGINALLY POSTULATED OR THE REDEFINED DESIGN BASIS CONDITIONS

The analysis of the expected frequency of adverse weather conditions on the conditions laid down in the plant design, is made in the report of PSA level 1, "External events".

After obtaining calculation values of meteorological characteristics, the methodology of SNiP (Norm and Rules in Construction - Tome "Climatology") and the handbook applied for building design were used.

The calculated values of all characteristics for following year frequencies are given: 0.001% (100000 years), 0.01% (10000 years), 0.1% (1000 years), 1% (100 years), 2% (50 years), 10% (10 years), 20% (5 years).

The calculation of extreme temperature frequencies is performed both by empirical and analytical methods. The empirical curves are constructed on the basis of normal probability distribution law using frequency of the members of the statistical series which is calculated from the formula:

$$P = m/(n+1) \times 100\ \%$$

Where, m = order,

n = number of series members.

When calculating maximum wind speed with different frequency, the yearly maximum speeds are used. On basis of the initial data, integral distribution curves are constructed and the maximum speeds with given frequencies are extracted. Besides of empirical curves, analytical approach of extreme calculations for given frequencies is also used based on the 1st limit distribution of Humbel:

$$F(v_{max})=\exp[-\exp(y)]$$

Into which the parameter of velocity (v), the variance of the series of maximum wind speeds and the auxiliary value "y" of input frequency are introduced. The values obtained from the probability curve and calculated from this formula are almost identical.

The calculations of snow loads are performed on basis of the data from filed snow surveys. Based on the thickness of the snow layer and its density, the water reserve on the last day of each decade is determined. This reserve, expressed in mm of the water column is equivalent to the mass of snow layer in kg/m^2. For each winter, the maximum value of the reserve is selected and an empirical of normal probability was developed. The probability is calculated by using the formula:

$$P = (m-0.5)/n$$

Where, m = order,

n = number of series members.

Results of the analysis:

1. Snow load on the roof of buildings

The calculated maximum snow loads for the ANPP site are presented in Table 4-3.

Return Period (years)	**100000**	**10000**	**1000**	**100**	**50**	**20**	**10**	**5**
Snow load (kg/m2)	180	173	156	122	108	84	64	45

Table 4-3: Calculated snow load for the ANPP site

Extreme snow loading can lead to a collapse of roof of buildings housing equipment ensuring safety of the plant.

The considered buildings are:

- The reactor building;
- The turbine hall;
- The CCWS pump station;
- The DG building.

These buildings are designed to sustain a snow load of 80 kg/m^2.

For the reactor building, the DG building and the turbine hall, calculations have defined critical load of snow and structures for destruction of the roof. Failure criterion reaches 95 % of the elasticity margin.

Calculations have shown that the value of snow loading leading to destruction is:

Reactor Building	290kg/m^2
DG building	269 kg/m^2
Turbine hall	180 kg/m^2
The CCWS pump station	more than 200 kg/m^2

Comparing the received values of snow loading leading to destruction of roofs of buildings, with extreme snow loadings of 0.001 % of security (expected in 100000 years) it is possible to note that extreme snow loadings do not represent a threat for safety of ANPP.

2. Wind loading

The calculated maximum values of wind speed and wind gust speed at the site of the ANPP are given in Table 4-4.

Return Period (years)	**100000**	**10000**	**1000**	**100**	**50**	**20**	**10**	**5**
Speed of a wind for 10 minute interval (m/s)	61	52	43	33	30	24	21	16
Wind gust speed (m/s)	88	74	60	47	42	37	33	28

Table 4-4: Calculated wind speed values for the ANPP site

Negative consequences on safety of the power unit, because of extreme wind, can be caused by falling of the ventilation stack on the reactor building.

Falling of the ventilation stack on the reactor building can bring to:

- Collapse of a part of a roof on components of regulation of RCPS that can lead to the impossibility of operation of the emergency protection of the reactor;
- The total destruction of the roof of the reactor building with damage to the reactor primary circuit and systems;
- Damage the spent fuel in the fuel pool.

In PSA, the wind speed which leads to the falling of the ventilating stack is calculated. Calculation shows that probable loading which can lead to a collapse, is 213 kg/m^2 which is equivalent to a wind speed of 59 m/s at height of 10 meters.

Considering wind data at the site, frequency of occurrence of ventilation stack fall on the reactor building is determined to be $2x10^{-6}$ in a year.

3. Dust storms

In summer months rather long dust storms are possible. In a combination with a strong wind they can also cause failure of diesel generators due to the ingress of particles into the engine cylinder. Diesel generators are installed inside building at the ANPP. Protection against a dust, provided by the building has been defined as sufficient. However, strong winds can break tightness of rooms of DG building (damage of windows) allowing penetration of dust into the rooms.

Calculations have shown a probability of this event of $1{,}66 \times 10^{-4}$.

4. Low temperatures

Low temperatures at the site of the ANPP are presented in Table 4-5.

Return Period (years)	**100000**	**10000**	**1000**	**100**	**50**	**20**	**10**	**5**
Temperature (°C)	-39	-36	-32	-27	-25	-23	-21	-18

Table 4-5: Low temperature at the ANPP site

Probable negative consequences related to very low temperatures can be:

- Loss of operability of DG because of blocking (freezing) of supply of fuel in DG;
- Pipelines freezing from DMWT to EFWSP and DFWP;
- Water freezing in spray pond of CCWS.

In the first case, the consequence of the event is the loss of all diesel generators which are used as back-up AC power supply.

In the second case there is a water supply loss from Demineralized Water Storage Tanks to both Emergency Feedwater pumps and Diesel Feedwater Pump which provide long-term water supply to SG's if main feedwater and auxiliary feedwater pumps are lost.

In the third case, a consequence of an event can be the loss of operability of ESWS.

In order to prevent freezing of pipelines from DMWT to EFWSP and DFWP, the periodic connection (1 time every 22 hours) of a pump for DMWT recycling has been foreseen in the design of the plant.

In PSA1 the possibilities of pipeline freezing has been analyzed, leading to the implementation of design measures in order to prevent freezing.

Results of the analysis have shown that at a temperature lower than -35 °C, the water in the pipes from the DMWT froze between consecutive recirculation. On the basis of this conclusion the technical solution of heating of the DMWT pipelines has been developed and implemented by ANPP (when temperature is below 8°C) in order to exclude the possibility of freezing of the above-mentioned pipelines.

Data on the diesel fuel used in DG specify that at temperatures below -35°C the fuel starts to stiffen. Supply system of diesel fuel is located in a room of the DG building. The room is heated by radiators, working on plant auxiliary steam system. Moreover the room may be heated using a stand-alone boiler. The room temperature must be maintained above 5°C, and this requirement is always satisfied. The freezing of the diesel fuel, preventing fuel supply, is therefore excluded.

Extremely low temperatures do not influence the operability of cooling water system for the following reasons:

- Water pipes of the CCWS are laid out at a depth of more than 2 meters below the ground which is deeper than the depth of soil freezing;
- Preservation of water heat in spray pond, avoiding heat loss during winter period, by bypassing the spray nozzle; Hot water coming from heat exchangers is injected directly in the spray pond;
- In case of emergency, combined system of cooling is foreseen: half of the spray pond works without spraying and the other half with spraying.

Reliable operation of cables and batteries in case of extreme low temperatures of atmospheric air are insured by:

- Installation of all batteries inside the buildings;
- Cabling of power supply and control cables between the buildings on depth of 3m which ensures exclusion of low temperatures;
- The cables are designed for operation at low temperatures up to -50^{o}C.

5. High temperatures

Possible negative consequence of extreme high temperature can be the failure to ensure the heat transfer to the ultimate heat sink.

Service water with temperature between +5°C to +35°C should be provided to the ESWS consumers to ensure normal work.

According to design calculations, one ESWS division can ensure a temperature of 33°C in the spray ponds under the following conditions:

- Temperature of external air 42 °C;
- Humidity up to 50 %;
- Wind speed of 2,1 m/s.

Within 72 hours without spray pond make-up.

6. Lightning

In Armenia there is no monitoring and data recording system about lightnings. For this reason, carrying out the quantitative analysis of danger of a lightning on ANPP is not obviously possible.

The assessment is based on engineering judgment.

Operating experience, from the Armenian nuclear power plant and from other power units, shows that the most negative impact from a lightning stroke consists in disconnection the power unit from the external network, with subsequent transfer of the power unit to a safe condition. On basis of this experience we consider that lightning strokes do not represent a threat for reliability of safety systems of the plant.

Besides in recent years, many new devices of protection against lightning's were installed. The work has been carried out on the basis of regulatory documents "instructions for lightning protection of buildings and structures of RD 34.21.122-87".

The reactor building and auxiliary building of ANPP are protected from a lightning stroke by means of the grounding of metal building components. On the ventilation stack lightning conductors are installed.

7. Tornado

The annual probability of a tornado on the territory of Armenia is about 1x10-7.

- Estimated class of intensity of a probable tornado: 1.07;
- Maximum horizontal speed of rotation of a wall of a tornado: 43 m/s;
- Pressure difference between the periphery and the funnel centre of the tornado: 22hPa.

According to meteorological service of Armenia, such phenomena were not observed in a current of the last 100 years and not mentioned in the history of Armenia.

Proceeding from the aforesaid, the analysis of influence of a tornado on ANPP is not executed.

4.1.1.4. CONSIDERATION OF POTENTIAL COMBINATION OF WEATHER CONDITIONS

From the point of view of negative impact on safety of the plant two events of possible combinations of adverse weather conditions should be considered:

1. Strong wind + dust storm + loss of external power supply.
2. Low temperature + loss of the ultimate heat sink.

1. Strong wind + dust storm + loss of external power supply

Strong winds are the potential reason of loss of external power supply, and also violation of tightness of compartments of DGS. In DGS backup sources of power supply (5 diesel engine generators) are stored.

Dust storms, with loss of the tightness of the rooms hosting the diesel generators, can cause failure of diesel generators due to the ingress of particles into the engine cylinder. Loss of external power supply leads to a CSBO of the plant with failure of the SEC systems. As it has been presented the probability of occurrence of such event is $1{,}66\text{x}10^{-4}$. However considered very conservative for the following reasons:

- To maintain the unit in safe condition, only one diesel engine generator is needed;

- The maximum duration of a strong wind is up to 1,5 hours, and a dust storm till 6 hours;
- Failure of all five DG, for the specified period (at DG work serially) is unlikely.

2. Low temperature + loss of a ultimate heat sink

In the case of loss of the ultimate heat sink and a complete loss off-site power, the heat removal from the reactor core can be provided at least for 120 hours using second side feed and bleed procedure: supply of feedwater from Demineralized Water Storage Tanks to SG's.
As shown here above, freezing of water in spray ponds, demineralized water storage tanks as well as freezing of diesel fuel is not possible.

Thus, we can conclude that the combination of low temperature and loss of the ultimate heat sink does not represent a threat for the plant.

4.1.1.5. CONCLUSION ON THE ADEQUACY OF PROTECTION AGAINST EXTREME WEATHER CONDITIONS

1. The ANPP design did not consider all possible natural phenomena requested by the current standards (e.g. tornado, dust storms).
2. The design basis of the plant ensures its operability under the extreme weather conditions of wind loading, snow loading, high temperature and low temperature.
3. The condition of dust storm combined with strong wind, not taken into account in the design of the plant, can create a threat.
4. There is no analysis of the impact of a tornado on the plant.Evaluation of safety margins.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

The parameters of weather conditions covered by the design basis and their recorded extreme values at the plant location are presented in Table 4-6.

Parameter name	Design Basis	Extreme recorded value
Wind pressure	45 kg/m^2	35 kg / m^2
Snow loading	80 кg/m^2	55кg/m^2
High temperature	+42^{0}C	+41^{0}C
Low temperature	-40^{0}C	-30^{0}C

Table 4-6: Weather conditions covered by design basis and recorded extreme values

For extreme temperatures, wind speed and snow load specific extreme values were taken into account in the plant design.

From the data of Table 6 we can conclude that design basis for the parameters of the concerned weather conditions is adequate, in comparison with the registered extreme values, and provide appropriate resistance of the plant to weather conditions.

Evaluated extreme values of weather conditions are presented in Table 4-7 for the following return periods: 10, 20, 50, 100, 1000, 10000 and 100000 years.

Parametre	Return Period (years)						
	100000	10000	1000	100	50	20	10
Snow loading, kg/m^2	180	173	156	122	108	84	64
speed of a wind for 10 minute interval, m/s	61	52	43	33	30	24	21
Low temperature °C	-39	-36	-32	-27	-25	-23	-21
High temperature °C	+43	+42	+41	+40	+39	+38	+38

Table 4-7: Extreme values of weather conditions

Taking into account:

- The results PSA level 1 «External events»;
- The comparison between the values of parameters of weather conditions and the assumptions in the design of the plant project, and
- The probability of extreme values of these parameters.

It can be stated that the design features relevant to weather conditions are fully adequate for possible extreme events with return period of 10000 years and there are enough margins.

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

The following measures are being planned to increase the robustness of the plant against extreme weather conditions:

- To develop and implement technical solution in order to avoid damage of window of the DG building in case of extreme wind conditions;
- To reconsider PSA level 1 taking into account the following:
  - To present a more detailed description and justification of extreme natural phenomena screening out;
  - To present a more justified choice of combinations of different natural factors;
  - To check the calculations of meteorological characteristics for different frequencies of exceedance;
  - To perform the analysis of possible impact of a tornado on the safety of the plant.
- To study the need of filtering devices on air inlets of diesel generators.

REFERENCES TO CHAPTER 4

1. Basic security provisions adopted in the project I and II nuclear reactor (Atomteploelektroproekt, Gorky, Russia 1986);
2. The safety CCWS including environmental and radiation safety (Armenergoproekt, Gorky, Russia 2013);
3. Level 1 PSA. External events. Volume 2. Essential events analysis. NRS STC, April 2004;
4. Level 1 PSA. External events. Volume 1. Screening of external events. NRS STC, April 2004;
5. Meteorological and hydrological hazards in site evaluation for nuclear installations (IAEA safety standards NoSSG-18);
6. NR-064-05 "The account of the impacts of natural and man-made objects on the use of nuclear energy." Moscow, Federal Service for Environmental, Technological and Nuclear Supervision, 2005

5. CHAPTER 5: LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

5.1. Loss of electrical power

5.1.1. Loss of off-site power

5.1.1.1. DESIGN PROVISIONS TAKING INTO ACCOUNT THIS SITUATION: BACK-UP POWER SOURCES PROVIDED, CAPACITY AND PREPAREDNESS TO TAKE THEM IN OPERATION

In case of total loss of AC power supply of the NPP from off-site sources, the plant design provides two-way emergency power supply system (EPSS). Design of the emergency power supply system of safety and safety related systems ensures the following safety functions in any operational mode of the unit and both Unit 1&2 SFP:

- Reactivity monitoring and control;
- Cooling of the fuel in the reactor core;
- Cooling of the fuel in Unit 1&2 SFP ;
- Confinement of radioactivity.

<u>Reactor in the “hot shutdown” mode and at any level of power</u>

The specified safety functions are ensured by:

- Control and Protection System;
- System of the primary circuit overpressure protection (PZ PORV);
- Radiation control system;
- Spray system;
- Main steam pipelines, SG SV, MSL, SDV-A;
- Emergency feedwater system (EFWS);
- High pressure emergency feedwater seismic system (unit 2 EFWSP-1&2);
- Emergency power supply system (EPSS);
- Primary circuit make-up system (unit 2 MP-1-4);
- Primary circuit emergency make-up system;
- Essential Service Water System (ESWS).

<u>Reactor installation in “cold” shutdown mode (including refuelling, reactor is opened or closed)</u>

The specified safety functions are ensured by:

- Primary and second technological parameters control system;
- Normal reactor cooling systems (PC, PCC, CP-1,2). Service water to PC, PCC is supplied from ESWS;

- Low pressure emergency feedwater system (unit 2 Emergency Condensate Pump-1,2 ; unit 2 Emergency Condenser-1,2). Service water to unit 2 Emergency Condenser-1,2 is supplied from ESWS;
- Primary circuit emergency make-up is ensured by unit 2 EMP-1-6 or by unit 2 SFP make-up pump;
- Steam generators are fed by unit 2 ESFW pumps-1,2 (from DMWT-3,4) or drainage tank's pumps or unit 2 EFWP-1,2 (from the de-aerator of 6 atm.);
- Emergency power supply system:

Cooling of Unit 1&2 SFP in any mode

- Cooling systems of unit 2 SFP (unit 2 SFP cooling pumps-1,2);
- Service water to unit 2 SFP heat exchangers is supplied from ESWS;
- Make-up of unit 1&2 SFP is provided from B-8/1&2 by unit 1&2 SFP make-up pumps. Pumps are connected to diesel buses;
- Emergency power supply system.

In case of loss of all sources of off-site AC power, two-train emergency power supply system (EPSS) provides power supply to all above-mentioned elements of safety systems which ensures the safety functions of the reactor and of the Unit 1&2 SFP. The operating personnel can switch on manually the part of consumers of 3RB-2, 4RB-2, 25BNN, 26BNN bus-bars (e.g. unit 2 SFP cooling pumps, CP, SFP make-up pump, etc), after completion of the DGLS programs, according to the DGLS operating procedures of the load sequencer.

Calculations confirm that safety functions can be ensured in case of start-up of at least one DG in any train operation.

All safety and safety related systems are designed as two-train safety systems. Start-up of one DG in any train ensures power supply to 3RB-2 (6 kV), 25BNN and 28NA (0,4 kV) (or 4RB-2, 26BNN, 29NA) bus-bars. Power supply by at least one train of EPSS ensures operation of at least one train of the above-mentioned safety systems, which provides a level of safe operation of the reactor and Unit 1&2 SFP in any mode of the operation of the unit (any level of power, hot shutdown, cold shutdown, refuelling).

Together with the power grid of the republic of Armenia (RA) personnel, special instructions for the operating personnel have been developed and implemented in case of collapse of the power grid. The objective of these instructions is to restore AC power supply of auxiliaries of ANPP from off-site sources of power. The list of consumers which have to be turned on after the connection of DG to its bus-bars is given in Appendix C.

In the "cold" shutdown and refuelling, 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) for make-up and cool-down of unit 2 SFP (unit 2 SFP cooling pumps, unit 2 SFP make-up pumps, ESWS pumps), for primary circuit make-up (unit 2 SFP make-up pumps, EMP), for feeding of SG (ESFW pumps, ECP, drainage tank pump, EFWP), for the organization of the primary circuit cool-down in the water-

water mode (CP, ECP, EFWSP, DTP).The operating personnel acts according to the DGLS operation procedure.

Design calculations performed by the designer take into account the operability of the various number of DG in any train of EPSS. For emergency power supply of ANPP auxiliaries, the following is foreseen:

- 3RB-2 and 4RB-2 6 kV bus-bars;
- 25BNN and 26BNN 0.4 kV bus-bars EPSS category II;
- 28NA and 29NA 0.4 kV bus-bars EPSS category I.

Two DG automatically start-up and connect in parallel to each 6kV bus bars. The power of each DG is 1500 kW.

- 1DG-1 and 1DG-2 are connected to 3RB-2;
- 2DG-1 and 2DG-2 are connected to 4RB-2.

Reliable power supply 0,4kV bus-bars of II category are powered from diesel 6kV bus-bars.

<u>0,4kV bus-bars of I category of reliability 28NA and 29NA (power supply circuit diagram is given in Appendix A of Chapter 1).</u>

When the power is available on 25BNN and 26BNN 0,4kV bus bars, the 28NA and 29NA bus bars are powered from these bus bars. To ensure un-interrupted power supply of 28NA and 29NA bus bars batteries of 2UDCB-1 and 2UDCB-2 and reversible motor generators (RMG) are foreseen. In case of AC power supply to 28NA and 29NA from diesel bus-bars, reversible motor generators (RMG) operate in charging mode. The 28NA and 29NA 0.4 kV bus bars are physically and electrically isolated. Each bus-bar is equipped with two reversible motor generator (RMG) (one operating, and the other in standby) and respective DC panel. Operating personnel operate according to the following documents:

- Beyond design basis accidents management guidelines;
- Reactor accidents mitigation procedures;
- Turbine island accidents mitigation procedures;
- Electrical equipment accidents mitigation procedures.

5.1.1.2. AUTONOMY OF THE ON-SITE POWER SOURCES AND PROVISIONS TAKEN TO PROLONG THE TIME OF ON-SITE AC POWER SUPPLY. DESIGN VALUE FOR THE AUTONOMY (FUEL AND OIL TANK CAPACITY) AND POSSIBILITIES TO EXTEND THE TIME OF USE OR TO PROVIDE MAKE-UP.

Two 6kV 3RB-2 and 4RB-2 bus-bars are designed for the emergency AC power supply of ANPP. In the case of loss power on 6 kV bus bars, 2 DGs are automatically started and connected to each bus.

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 compartment is furnished with compressed air systems and a direct current board, which ensure start-up of diesels ion case of loss of NPP AC power supply.

All elements that provide the start-up of DG are powered from the DC panel of the respective train.

DG of each train (1DG-1, 1DG-2, I train and 2DG-1, 2DG-2, II train) are started-up during the loss of power only of its 6 kV bus-bar.

Currently a new system of DG speed control and new excitation and voltage control system is installed at each DG.

Each train of two DGs has its own I&C system (DG SAU).

All above-mentioned systems ensure reliable and parallel operation of DG.

Load sequencer system provides a step-by step connection of consumers (for a smooth loading of the DG) to its power supply bus-bars. Load sequencer system automatically disconnects part of the consumers of its train if, for any reason, one of operating DG fails, in order to ensure normal operation for the remaining DG. Load sequencer scheme automatically connects only part of the consumers to the 3RB-2 (or 4RB-2) 6 kV bus bar if, for any reason, only one DG was connected to the bus-bar.

In case of loss of NPP AC power supply all pumps of the Essential Service Water System (ESWS) are switched off, then after the connection to its bus bar, load sequencer automatically switch on two pumps in each train of the ESWS (one pump in 5 seconds after the connection of DG, and the second pump in 15 seconds).

According to the calculations, reactor safety is ensured if at least of one DG in any train is turned on.

Load sequencer system allows switching on consumers of safety related systems (SRS), but pumps are switched on only if technological factors to switch them on are present. The list of consumers which are switched when DG is connected to its bus bars is given in Appendix C.

Each DG compartment 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 26 hours.

There is one oil storage tank with a capacity of 3m^3 in each DG compartment. This capacity is sufficient for the simultaneous operation of two DGs with a full load for at least 5 days. For SEC DG-4 the capacity is sufficient for at least 10 days. Whenever necessary, feeding of oil storage tanks can be done from oil trucks.

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

In this mode the following equipment remains operable:

- SG auxiliary feedwater system (diesel pump from DMWTs 1&2);
- SEC DG (see section 5.1.2.1);
- Primary power supplies of (28NA, 29NA) 0.4 kV bus-bar from batteries (see section 5.1.3.1).

5.1.2.1. DESIGN PROVISIONS TAKING INTO ACCOUNT THIS SITUATION: DIVERSE PERMANENTLY INSTALLED AC POWER SOURCES AND/OR MEANS TO TIMELY PROVIDE OTHER DIVERSE AC POWER SOURCES, CAPACITY AND PREPAREDNESS TO TAKE THEM IN OPERATION

In case of loss of AC power of NPP auxiliaries at any power mode of the unit, and failures of primary and second trains of emergency power supply, there is completely autonomous and independent additional emergency cooling-down system of the reactor (SEC DG see Figure 3).

- SEC system is manually put into operation by the operating personnel in accordance with the operating procedures of the system;
- SEC DG is start-up (DG is located in the first DG compartment) and is connected to the SEC of 6kV bus-bars;
- Preparation and putting into operation of the autonomous SEC system could be done within one hour in accordance with the operating procedures of the given system;
- Consumers of SEC DG (see Chapter 1, Appendix A) are chosen in order to ensure the safety functions of the Unit 1 &2 SFP and the reactor in any mode of operation of the unit, in accordance with section 5.1.1.1.

<u>SEC DG powers (see Chapter 1, Appendix A):</u>

- SFP cooling pumps (unit 2 SFP cooling pumps-1,2);
- One pump in each ESWS train;
- Train of the primary circuit emergency make-up;
- Second circuit cooling pump of the reactor in the water-water cooling mode;
- The SFP make-up pump (unit 1 SFP make-up pump) (for make-up of unit 1 SFP from B-8/1);
- Fire-fighting water pump (FFWP);
- Fire-fighting foam pump (FFFP);
- High pressure primary circuit make-up pump;
- Drainage tank pump;
- Pump of the recirculation system of containment cooling (2R-1);
- Emergency Feedwater pump (EFWP);
- Lighting cabinet;
- Boron make-up pump;
- Drainage pump (DP) of B-001/2 compartment.

5.1.2.2. BATTERY CAPACITY, DURATION AND POSSIBILITIES TO RECHARGE BATTERIES, AUTONOMY OF THE PERMANENTLY INSTALLED DIVERSE BACK-UP POWER SOURCES AND PROVISIONS TAKEN TO PROLONG THE TIME OF ON-SITE AC POWER SUPPLY. DESIGN VALUE FOR THE AUTONOMY (FUEL AND OIL TANK CAPACITY) AND POSSIBILITIES TO EXTEND THE TIME OF USE OR TO PROVIDE SUPPLY

Reliable power supply of I category consumers is ensured by two autonomous 0.4 kV bus-bars (28NA and 29NA) (Circuit diagram of the reliable power supply of I category 0.4 kV bus-bar is given in Chapter 1, Appendix A). In the case of power availability at 25BNN & 26BNN 0.4 kV bus-bars 28NA and 29NA are powered by these bus-bars through thyristor breakers TP-1 or TP-2 (one operating and the other in stand-by). During loss of power supply of 25BNN (26 BNN) bus-bars thyristor breakers are automatically closed.

Each of 28NA (29NA) bus-bar has two RMG (reversible motor-generator), one operating and the second in standby. Switching of the standby RMG on is performed by the operating personnel in accordance with operating procedures of system.

When 28HA (29HA) bus-bars are powered from 25BNN (26BNN) bus-bars RMG operate in the batteries charging mode. Each of 28HA (29HA) 0,4 kV bas-bar has the direct current board. After closing of the thyristor breakers RMG is automatically transferred into an inverter mode (without interruption of power supply) and provides alternating current to 28HA (29HA) 0,4kV bus-bar.

Capacity of each battery is sufficient to provide power supply to 28HA (29HA) bus-bars for at least 7 hours, see Chapter 1.3.6 which lists durations depending on various conditions. This time can be prolonged by switching off of part of consumers (not important instrumentations, panels of the power supply of MCR-2, etc.). Period of batteries operation is not enough to ensure the safety functions in accordance with paragraph 5.1.1.1.

The direct current board and compressed air system of I DGS compartment ensure autonomous start-up of DG-4 in the absence the off-site power supply sources and failures of reserve sources (failure of all diesel-generators).

DG-4 is started without Essential Service Water System (ESWS) pumps. After start and connection of SEC DG 6kB bus-bars, operating personnel immediately switch on one pump in each train of the ESWS. The reserve of diesel fuel (approx. 25м3) in the I DG compartment is sufficient for SEC DG operation within approx. 50 hours with rated power (1500kW).

In case of failure of all ESWS pumps the operating personnel has possibility to supply service water by the diesel pumps from the outlet channel of circulating water supply system.

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

In this mode, the following equipment remains operable:

- SG auxiliary feedwater system (the diesel pump from DMWT 1&2);
- Power supply of primary (28NA, 29NA) 0,4kV bus bars from rechargeable batteries (section 5.1.3.1).

5.1.3.1. BATTERY CAPACITY, DURATION AND POSSIBILITIES TO REDUCE CONSUMPTION (LOAD SHEDDING) AND TO RECHARGE BATTERIES IN THIS SITUATION

I category safety consumers AC power supply is ensured by 28HA and 29HA 0,4 kV bus-bars (see Figure 2).

Power supply to I category direct current consumers is ensured by physically and galvanically separated rechargeable batteries. Capacity of each rechargeable battery is sufficient for RMG operation during 7 hours. This time can be considerably extended by switching off of part of consumers (not important instrumentations, panels of the power supply of MCR-2, etc.).

5.1.3.2. ACTIONS FORESEEN TO ARRANGE EXCEPTIONAL AC POWER SUPPLY FROM TRANSPORTABLE OR DEDICATED OFF-SITE SOURCE

For the moment ANPP doesn't have mobile diesel-generators which could be used in case of total loss of AC power and external AC power failure (both the main and stand-by).

SG feedwater supply is ensured by the operating personnel using the diesel pump installed in the B-001/(Unit 1) area. The SG auxiliary feedwater system is completely independent (operability doesn't 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 900 m^3). Discharge line of diesel pump is connected to the discharge line of ESFW pumps. The reserve of the diesel fuel is sufficient for 24 hours of operation.

Diesel pump parameters:

- Rated pressure: 53,6 kgf/cm^2;
- Rated flow rate: 65,9 m^3/hour.

Additional calculations must be performed to validate the adequacy of the diesel pump performance to feed the steam generators in various reactor power modes (cold shutdown, refuelling, closed reactor, open reactor).

A special instruction exists at Armenian NPP to restore the AC power supply of auxiliaries of NPP with the AC from the outdoor switchgear SY-110/220kV during the blackout (disruption) of the Republic of Armenia power supply system of the Republic of Armenia. Provision of power supply to any 110/220 kV grid is controlled by RA power supply management services.

5.1.3.3. COMPETENCE OF SHIFT STAFF TO MAKE NECESSARY ELECTRICAL CONNECTIONS AND TIME NEEDED FOR THOSE ACTIONS. TIME NEEDED BY EXPERTS TO MAKE THE NECESSARY CONNECTIONS

The operational personnel of the ANPP receives emergency response training for the initiating events, associated with the NPP auxiliaries' blackout of NPP and collapse (disruption) of the power supply system of Armenia.

Operating personnel is trained to (with tests of knowledge and via specific emergency drills) put into operation SEC and diesel SG feedwater pump systems.

Various trainings with various shifts of operating personnel show that the time to transfer SEC system into operational mode does not exceed 1 hour.

The diesel pump switches into operation within 15 to 20 min.

According to time-schedule approved by CE, the DG is tested every month under load (1DG-1, 2 and 2DG-1, 2, DG SEC).

Every year after annual outage of the unit or after unit shutdown for more than 3 days a test of emergency power supply system is carried out in compliance with the program approved by CE.

5.1.3.4. 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)

Open reactor, with natural circulation via two loops and normal reactor level

T_{mean}=55˚C, the reactor is shut down 5 days ago.

- Start of coolant boiling in the reactor core: 4 hours;
- Time of decrease of the coolant level in the reactor till hot leg nozzles: 5,5 hours;
- Beginning of fuel rods damage: 18,5 hours.

Closed reactor, T_{mean}= 150˚C P_{pc} 30 kgf/cm^2

- Complete evaporation of SG: 10,5 hours;
- Boiling of coolant in the reactor core: 16 hours;
- Damage of fuel rods: 19,5 hours.

Reactor at any power mode and in the "hot shutdown" mode

- Time for the complete evaporation of the steam generators water inventory (at nominal level, feeding is completely unavailable) is about 8 hours;
- Primary circuit make-up is performed with SFP MP via reserve loop of primary circuit cooldown. It is specified in the operating procedure of primary circuit;
- Make-up of SFP is performed with the same SFP MP (simultaneous make-up is possible) via the SFP cooldown system piping (SFP CP). This mode is considered in the operating procedure of SFP cooldown system.

	Full unloading	Fuel is located at the bottom tier only
Time of boron solution heating in Unit 2 SFP by 1°C	5,3 min	34,8 min
Time of boron solution reach 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
Start of fuel damage 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)	82,1m^3/ hour	4,8m^3/ hour
Water flow rate to maintain constant level in unit 2 SFP	6,1m^3/ hour	0,36m^3/ hour

Table 5-1: Calculation results for the SPF 2

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.

Note: The specified values are preliminary and are subject to change after carrying out of detailed calculations with the analyses of scenarios, related to the conditions of the Unit 1 & 2 SFP (open reactor, closed reactor, refuelling).

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

- The amount of diesel fuel in DG tanks does not assure the operation of DG at full capacity for 72 hours (Unit 1 DG-1&2, Unit 2 DG 1&2 and SEC DG);
- The diesel fuel storage does not assure the diesel feedwater pump operation for at least 72 hours;
- The capacity of rechargeable batteries is enough for providing the AC power of 0,4 kV reliable power supply of the I category bus-bars (28NA and 29NA) for 7 to 8 hours only; This is not enough for assuring the safety functions for at least 72 hours;
- In case of loss of all off-site AC power supply sources simultaneously with loss of all on-site reserve sources of AC power supply, it is necessary to implement the additional measures listed in section 5.1.5;

- The existing procedures do not fully reflect power modes and operating personnel actions with regard to the reactor and units 1&2 SFP.

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

To increase the level of the safe operation of NPP in the case of simultaneous loss of normal and stand-by AC power supply sources it is proposed:

- To review symptom-based operating procedures of the reactor (in all operating modes of the unit) and units 1&2 SFP, and, if necessary to develop new procedures and programs for training of the personnel under various accident scenarios and common failures of safety systems;
- To perform analyses of the units 1&2 SFP cooling and to introduce additional measures based on the performed analyses. The issues of concern are, for example, the timeline calculation for the recent fuel in case of loss of cooling, or the need to provide power supply/load sequencer for unit 1 SFP cooling pumps and/or unit 1 SFP make-up pumps in case of loss of external power supply to ANPP;
- To develop additional measures to extend the operating time of reversible motor generators (RMG) in an inventor mode that will lead to increase the time to provide SRS I&C AC power supply;
- To implement autonomous alternative power supply sources. For this measure, it is necessary to perform analysis with the objective to identify parameters (output, etc.), required qualification, configuration and allocation (including the mobility option or the possibility to use fire brigade pumps) for those power supply sources in order to assure the fulfilment of the necessary safety functions.

  For example, it could be possible to use:
  - DG 2.0MW, 6kV; 0.4kV; 220 VAC;
  - 2 DGs 0.4kV; 500kW.

- To install an additional tank of diesel fuel on the DGS (for each bus bar) in order to run the train I and II of emergency power supply during 72 hours at full load, independently of the fuel inventory in the oil-fuel facilities of the plant;
- To replace all reversible motor generators (RMG) with modern inverters;
- To implement the scheme for battery charging from SEC and/or the portable diesel generator;
- To implement autonomous alternative means for make-up of the primary circuit, Unit 1&2 SFP, SGs 1-6 of the Unit 2 and other required systems (essential service water, DMWT 1-4). For this measure, it is necessary to perform analysis with the objective to identify parameters (output, etc.), required qualification, configuration and allocation (including the possibility of mobility or the possibility to use fire brigade pumps) for those make-up means in order to assure the fulfilment of the necessary safety functions.

  For example, it could be possible to use:
  - A diesel pump 5-10kgf/cm^2, 600-1000m^3/h;
  - A medium pressure diesel pump 70-90kgf/cm^2, 100-250m^3/h;
  - Two low pressure diesel pumps 10-15kgf/cm^2, 150-500m^3/h.

- To implement schemes (system) of water supply to primary circuit, Unit 1&2 SFPs, SGs 1-6, DWSTs 1-4 from alternative water sources;

- 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, refuelling, closed reactor, open reactor);
- To implement analysis of circuit diagram for consumers power supply from SEC. Develop and implement activities aimed at minimizing personnel manual actions to activate the SEC system.

5.2. Loss of the ultimate heat sink

5.2.1. Design provisions to prevent the loss of the primary ultimate heat sink, such as alternative inlets or systems to protect main water inlet from blocking

Following systems are designed to remove heat from the reactor core and from Unit 1&2 SFP:

- System of circulation water supply ("Ponds" and “Sevdjur” pumping stations, cooling towers, inlet and outlet channels);
- Unessential Service Water System;
- Essential Service Water System (ESWS);
- System of normal reactor cooling down through the second circuit (PC, PCC, CP-1,2);
- SG auxiliary feedwater system (diesel pump, DMWTs 1&2);
- High pressure reactor installation emergency cooling system (unit 2 EFWSP 1&2, DMWTs 3&4);
- Low pressure reactor emergency cooling system (unit 2 Emergency Condensate Pump-1,2 and unit 2 Emergency Condenser-1,2);
- Unit 1&2 SFP cooling system (SFP heat exchangers, SFP cooling pumps);
- Emergency SG feedwater system (unit 2 EFWP-1,2 , unit 2 de-aerators D-6 atm.);
- Main steam pipelines (including SG SV, SDV-A, SDV-C).

5.2.2. Loss of the primary ultimate heat sink (e.g. loss of access to cooling water from the river or loss of the main cooling system)

Simultaneous failure of the "Pond" and "Sevdjur" pumping stations when the unit is in any power mode:

- In case of failure of pumping stations and decrease of the level in the channels down to 2,2 m, the operating personnel should transfer the unit into a "cold" shutdown mode according to procedure requirements;
- In case of failure of the cooling system of all cooling towers, operating personnel shutdown the reactor, on basis of the parameters of the second circuit and if there is no cooling, they transfer the unit into “cold” shutdown mode;
- At any power mode the water volume in the inlet channel is 77.240 m^3 and 36.000 m^3 in the outlet channel;

- According to the calculations, the available inventory of service water is completely sufficient to bring the reactor in a "cold" shutdown mode and to maintain it in this mode for at least 72 hours.

5.2.2.1. AVAILABILITY OF AN ALTERNATE HEAT SINK

The alternative heat sink for reactor cooling via second circuit is the ESWS . In any operating mode of the reactor (including open reactor, closed reactor, refuelling) in case of failure of "Pond" and "Sevdjur" pumping stations or failure of circulating water supply, the following systems remain operational:

- Essential Service Water System (ESWS);
- The basic safety features of all safety systems are ensured by the design.

In case of loss of Unessential Service Water Cooling System, the operating personnel manually transfer process condenser and process condenser cooler to ESWS.

ESWS spray ponds can be refilled by using output channel make-up system (pumps are powered by diesel bus-bars).

5.2.2.2. POSSIBLE TIME CONSTRAINTS FOR AVAILABILITY OF ALTERNATE HEAT SINK AND POSSIBILITIES TO INCREASE THE AVAILABLE TIME

Conservative calculations indicate that the existing minimum inventory of the service water in spray ponds of the ESWS, in case of loss of water supply to them, is sufficient to cool down the unit during 72 hours. The back-up supply pipeline of the ESWS as well as the feeding by the diesel pumps from the water inventory of the output channel ensures ESWS operation during 15 days.

Currently ANPP doesn’t fully use the possibility of alternative heat sink (e.g. an inventory of the service water in circulating channels, water supply in steam generators and DMWT-1-4 from alternative sources).

All available calculations are given in section 5.2.3.2.

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

In case of loss of the primary ultimate heat sink and alternate heat sink the following systems remains available:

- SG auxiliary feedwater system (diesel pump from DMWT-1,2);
- Main steam lines (including SG SV, MSIV 1-7, SDV-A);
- Normal primary circuit make-up system;
- Emergency SG feedwater system (unit 2 EFWP 1&2 from de-aerator 6 atm.);
- High pressure reactor installation emergency cooling system (DMWTs 3&4, unit 2 EFWSP 1&2);
- Units 1&2 SFP make-up pumps (unit 1 SFP make-up pump and unit 2 SFP make-up pump).

5.2.3.1. EXTERNAL ACTIONS FORESEEN TO PREVENT FUEL DEGRADATION

Reactor in any power mode and in “hot shutdown” mode

- Operating personnel use one of the systems listed in section 5.2.3 in order to maintain parameters of primary circuit, with steam relief to the atmosphere through SG SV or SDV-A;
- Normal primary circuit make-up pumps are used (2MP,1-4) to feed the primary circuit ;
- "Feed & Bleed" is used to maintain parameters of Unit 1&2 SFP, using Unit 1&2 SFP make-up pumps from B-8/1&2.

Reactor in a "cold" shutdown mode (including refuelling)

- The Unit 2 SFP make-up pump from B-8/2 is used for the primary circuit make-up (the reactor is opened);
- Any systems listed in section 5.2.3 are used for steam generator feeding.

5.2.3.2. TIME AVAILABLE TO RECOVER ONE OF THE LOST HEAT SINKS OR TO INITIATE EXTERNAL ACTIONS AND TO RESTORE CORE COOLING BEFORE FUEL DAMAGE: CONSIDERATION OF VARIOUS EXAMPLES OF TIME DELAY (TIMELINE ANALYSIS) FROM REACTOR SHUTDOWN TO LOSS OF NORMAL REACTOR CORE COOLING CONDITION (E.G., START OF WATER LOSS FROM THE PRIMARY CIRCUIT).

Use of the unit 1&2 SFP «Feed & Bleed" system from B-8/1&2 with Unit 1&2 SFP make-up pumps:

- Water temperature in Unit 1& 2 SFP is 60˚C; Time during which the temperature in the unit 2 SFP reaches about 80˚C: ~3,3 days; For the unit 1 SFP the time to reach this temperature is about 19 days;
- The results of calculations for the Unit 2 SFP without use of the "Feed & Bleed" system are listed in the table of section 5.1.3.4.

The reactor is opened; Natural circulation through the two loops, the level in reactor = nominal level.

- The results of calculations without use of the "Feed & Bleed" system of the primary circuit are listed in the section 5.1.3.4.

The reactor is closed (T_{mean}=150˚C P_{pc}=30 kgf/cm^2)

- Time for full evaporation of SG, with available diesel pump is about 5 days.

Reactor at any power level and in “hot shutdown” mode

- With operational/available EFWSP, diesel FW pump (DMWT 1-4), time for full drying of SG is about 16 days.

Results of calculations in case of failure of AC power supply and SEC DG are given in the section 5.1.3.4.

Note: The specified values are preliminary and are subject to change after detailed calculations for beyond design basis accident analysis of scenarios.

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

Current procedures do not fully prescribe operating personnel actions in the 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.

5.2.5. Measures envisaged to increase robustness of the plant in case of loss of ultimate heat sink

It is necessary to,

- Perform additional calculations and analysis of different scenarios related to the reactor mode (opened reactor, closed reactor, refuelling);
- Develop additional procedures on the basis of new calculations and analysis;
- Organize training of personnel on the use of the new procedures;
- Develop and implement additional measures to use a large reserve 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 back-up 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: 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 remains operable:

- SG auxiliary feedwater system (diesel pump from the DMWT-1,2) (see 5.1.3.2);
- Reliable I category power supply (28NA and 29NA) from rechargeable batteries (see 5.1.1.1);
- Main steam pipelines (including SG SV, SDV-A, MSIV 1-7).

5.3.1. Time of autonomy of the site before loss of normal reactor core cooling condition (e.g., start of water loss from the primary circuit)

All the available calculations are given in the section 5.1.3.4.

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:

- Operational personnel puts into operation the SEC DG within 1 hour (see 5.1.2).

In case of loss of ultimate heat sink with simultaneous loss of off-site power and on-site stand-by and back-up power sources:

- The SG auxiliary feedwater system is used (diesel pump from DMWT 1&2).

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 purchase autonomous alternative power supply sources. In order to fulfil the required safety functions, it is necessary to perform preliminary analysis in order to identify parameters (output, etc.), required qualification, configuration and allocation (including mobility option) for those power supply sources;

  For example, it could be possible to use:
  - Diesel Generators of 2.0MW, 6kV, 0.4kV, 220 VAC;
  - 2 Diesel Generators of 0.4kV, 500kW.

- To purchase autonomous alternative means to supply water to the primary circuit, the Unit 1&2 SFP, the SGs 1-6 of the Unit 2 and other systems (essential service water, DMWTs 1-4). In order to fulfil the required safety functions, it is necessary to perform preliminary analysis in order to identify parameters (output, etc.) required qualification, configuration and allocation (including mobility option or the possibility to use fire brigade pumps) for those means;

  For example, it could be possible to utilize:
  - 1 diesel pump of 5-10kgf/cm^2, 600-1000m^3/h;
  - 1 medium pressure diesel pump of 70-90kgf/cm^2, 100-250m^3/h;
  - 2 low power diesel pumps of 10-15kgf/cm^2, 150-500m^3/h.

- To implement system of water supply to primary circuit, Unit 1&2 SFPs, SGs 1-6, and DMWTs 1-4 from alternative sources;
- To develop and implement measures to recharge the batteries;
- To develop additional symptom based operating procedures for the plant personnel and to carry out training programs based on results of additional calculations and analyses.

APPENDIX C

LOAD SEQUENCER

Приложение 1.

АВТОМАТИКА СТУПЕНЧАТОГО НАГРУЖЕНИЯ
I КАНАЛ

РЕЖИМ №1. ВКЛЮЧИЛИСЬ В РАБОТУ ДВА ДГ

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	1НСО-1	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2АПЭН-2	25БНН	Безусловно
		2МНС-3Б	25БНН	Выбранный ключом ПБ в положение “Работа”
II	15	1НСО-2	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		1НСО-3	3РБ-2	При не включении 1НСО-1 или 1НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
III	25	2В-2А	25БНН	Безусловно
		2МНС-4А	25БНН	Выбранный ключом ПБ в положение “Работа”
		2ПН-2	25БНН	Безусловно
		2ПН-4	25БНН	При не включении 2ПН-2 (через 2 сек.)
IV	35	2НПК-2	25БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
V	180	69Т	3РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке
При ↓Н в 2КД на 2560мм от ном. или ↓Р в Iконтуре 95кгс/см^2		2АПН-1 2АПН-2 2АПН-3	3РБ-2	Выбранный ключом 1ПВ и нахождения своего ПБ в положение “Работа”
Через 10 сек.		2АПН-1 2АПН-2 2АПН-3	3РБ-2	Не выбранный ключом 1ПВ и нахождения своего ПБ в положение “Работа”
При ↑ Р в боксе ПГ до 0,2кгс/см^2		2НБС-1	25БНН	При нахождения своего ПБ в положение “Работа” или “Деблокировка”
Через 10 сек.		2НБС-2	25БНН	При нахождения своего ПБ в положение “Работа” или “Деблокировка”

На 5-ой секунде АСП дается разрешение на включение 2АПН-1,2,3 и 2НБС-1,2. 2АПН-1÷3 и 2НБС-1,2 включаются независимо от ступеней АСП при формировании технологического параметра на их включение.

Приложение 1.

АВТОМАТИКА СТУПЕНЧАТОГО НАГРУЖЕНИЯ
II КАНАЛ

РЕЖИМ №1. ВКЛЮЧИЛИСЬ В РАБОТУ ДВА ДГ

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	2НСО-1	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2АПЭН-1	26БНН	Безусловно
		2МНС-4Б	26БНН	Выбранный ключом ПБ в положение “Работа”
II	15	2НСО-2	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2НСО-3	4РБ-2	При не включении 2НСО-1 или 2НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
III	25	2В-2Б	26БНН	Безусловно
		2ПН-1	26БНН	Безусловно
		2МНС-3А	26БНН	Выбранный ключом ПБ в положение “Работа”
		2ПН-3	26БНН	При не включении 2ПН-1 (через 2 сек.)
IV	35	2НПК-1	26БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
		2НПК-3	26БНН	При не включении 2НПК-1 (через 2 сек.)
V	180	70Т	4РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке
При ↓Н в 2КД на 2560мм от ном. или ↓Р в Iконтуре 95кгс/см^2		2АПН-4 2АПН-5 2АПН-6	4РБ-2	Выбранный ключом 2ПВ и нахождения своего ПБ в положение “Работа”
Через 10 сек.		2АПН-4 2АПН-5 2АПН-6	4РБ-2	Не выбранный ключом 2ПВ и нахождения своего ПБ в положение “Работа”
При ↑ Р в боксе ПГ до 0,2кгс/см^2		2НБС-3	26БНН	Безусловно

На 5-ой секунде АСП дается разрешение на включение 2АПН-4,5,6 и 2НБС-3. 2АПН-4,5,6 и 2НБС-3 включаются независимо от ступеней АСП при формировании технологического параметра на их включение.

Приложение 1

РЕЖИМ №2. ОДИН ИЗ ДВУХ РАБОТАЮЩИХ ДГ ОТКЛЮЧИЛСЯ – I КАНАЛ

При наличии фактора течи из Iконтура отключаются следующие потребители:

2АПН-1 2АПН-2 2АПН-3	3РБ-2	Не выбранный ключом 1ПВ и нахождения своего ПБ в положение “Работа”
2НБС-2	25БНН	Если 2НБС-1 в работе
2АПЭН-2	25БНН	Безусловно
2МНС-3Б 2МНС-4А	25БНН	Безусловно
2В-2А	25БНН	Безусловно
2ПН-2(4)	25БНН	Безусловно

РЕЖИМ №2. ОДИН ИЗ ДВУХ РАБОТАЮЩИХ ДГ ОТКЛЮЧИЛСЯ – II КАНАЛ

При наличии фактора течи из I контура отключаются следующие потребители:

2АПН-4 2АПН-5 2АПН-6	4РБ-2	Не выбранный ключом 2ПВ и нахождения своего ПБ в положение “Работа”
2АПЭН-1	26БНН	Безусловно
2МНС-3А 2МНС-4Б	26БНН	Безусловно
2В-2Б	26БНН	Безусловно
2ПН-1(3)	26БНН	Безусловно

Примечания:

1. Если эти потребители до отключения одного ДГ еще не включились в работу по АСП, то их дальнейшее включение по любому каналу не происходит.
2. При отсутствии фактора "Течь из I контура" и отключении одного ДГ в любом канале отключение механизмов не происходит.

Приложение 1

РЕЖИМ №3. ВКЛЮЧИЛСЯ ТОЛЬКО ОДИН ДГ
(I ПРОГРАММА-ОБЕСТОЧЕНИЕ БЕЗ ТЕЧИ)

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	2НСО-1	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2АПЭН-1	26БНН	Безусловно
		2МНС-4Б	26БНН	Выбранный ключом ПБ в положение “Работа”
II	15	2НСО-2	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2НСО-3	4РБ-2	При не включении 2НСО-1 или 2НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
III	25	2В-2Б	26БНН	Безусловно
		2ПН-1	26БНН	Безусловно
		2МНС-3А	26БНН	Выбранный ключом ПБ в положение “Работа”
		2ПН-3	26БНН	При не включении 2ПН-1 (через 2 сек.)
IV	35	2НПК-1	26БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
		2НПК-3	26БНН	При не включении 2НПК-1 (через 2 сек.)
V	180	70Т	4РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке

Примечание: При возникновения фактора "Течь из I контура" производится перевод с I программы на II программу.

РЕЖИМ №3. ВКЛЮЧИЛСЯ ТОЛЬКО ОДИН ДГ (II ПРОГРАММА-ОБЕСТОЧЕНИЕ С ТЕЧЬЮ)

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	2АПН-4 2АПН-5 2АПН-6	4РБ-2	Выбранный ключом 2ПВ и нахождения своего ПБ в положение “Работа”
		2НБС-3	26БНН	Безусловно
		2АПН-4 2АПН-5 2АПН-6	4РБ-2	При не включении 2АПН выбранный ключом 2ПВ и нахождения своего ПБ в положение “Работа” (через 2 сек.)
II	15	2НСО-1	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
III	25	2НСО-2	4РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2НСО-3	4РБ-2	При не включении 2НСО-1 или 2НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
IV	35	2НПК-1	26БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
		2НПК-3	26БНН	При не включении 2НПК-1 (через 2 сек.)
V	180	70Т	4РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке

Приложение 1

РЕЖИМ №3. ВКЛЮЧИЛСЯ ТОЛЬКО ОДИН ДГ (I ПРОГРАММА – ОБЕСТОЧЕНИЕ БЕЗ ТЕЧИ)

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	1НСО-1	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		2АПЭН-2	25БНН	Безусловно
		2МНС-3Б	25БНН	Выбранный ключом ПБ в положение “Работа”
II	15	1НСО-2	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		1НСО-3	3РБ-2	При не включении 1НСО-1 или 1НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
III	25	2В-2А	25БНН	Безусловно
		2МНС-4А	25БНН	Выбранный ключом ПБ в положение “Работа”
		2ПН-2	25БНН	Безусловно
		2ПН-4	25БНН	При не включении 2ПН-2 (через 2 сек.)
IV	35	2НПК-2	25БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
V	180	69Т	3РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке

ПРИМЕЧАНИЕ: При возникновения фактора “Течь из I контура” производится переход с I программы на II программу.

Приложение 1

РЕЖИМ №3. ВКЛЮЧИЛСЯ ТОЛЬКО ОДИН ДГ (II ПРОГРАММА –

ОБЕСТОЧЕНИЕ С ТЕЧЬЮ)

ступени	Выдержка времени (сек)	Наименование оборудования	Источник питания	Условия включения
I	5	2АПН-1 2АПН-2 2АПН-3	3РБ-2	Выбранный ключом 1ПВ и нахождения своего ПБ в положение “Работа”
		2НБС-1	25БНН	При нахождения своего ПБ в положение “Работа” или “Деблокировка”
		2АПН-1 2АПН-2 2АПН-3	3РБ-2	При не включении 2АПН выбранный ключом 1ПВ и нахождения своего ПБ в положение “Работа” (через 2 сек.)
		2НБС-2	25БНН	При не включении 2НБС-1 и нахождения своего ПБ в положение “Работа” или “Деблокировка” (через 2 сек.)
II	15	1НСО-1	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
III	25	1НСО-2	3РБ-2	При нахождении своего ПБ в положение “Работа” или “Резерв”
		1НСО-3	3РБ-2	При не включении 1НСО-1 или 1НСО-2 и нахождения своего ПБ в положение “Работа” или “Резерв” (через 2 сек.)
IV	35	2НПК-2	25БНН	Безусловно
		2П-5/3	2504БНН	Безусловно
		2П-5/4	2604БНН	Безусловно
		2В-6А	2604БНН	Безусловно
		2В-6Б	2504БНН	Безусловно
		2АН ГЦН-1	2505БНН	Безусловно
		2АН ГЦН-2	2506БНН	Безусловно
		2АН ГЦН-3	2504БНН	Безусловно
		2АН ГЦН-4	2505БНН	Безусловно
		2АН ГЦН-5	2506БНН	Безусловно
		2АН ГЦН-6	2604БНН	Безусловно
V	180	69Т	3РБ-2	Безусловно
		2НДР-3/1	2614БНН	По блокировке
		2НДР-3/2	2613БНН	По блокировке

6. CHAPTER 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. Organisation 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. Organisation 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:

- 2 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;
- 2 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 Nr 2. The shelter Nr 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 Nr 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 MLIV/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 6-1 below.

Unit	Plant state	Number
ANPP unit 2	Full power, hot stand-by and intermediate state	4 operators in MCRand 4 machinists on-site for local inspections
	Cold shutdown	3-4 operators in MCR and 4-5 machinists on-site for local inspections

Table 6-1: Composition of the shift crew

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 ORGANISATION 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 § 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.

Fig. 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 Emergency Situations of the RA (involving State Reserve Agency) (MES);
- 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 Territorial Management;
- 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 Diaspore;
- 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-Н 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 the ANPP cooperates with 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”.

id.	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

Table 6-2: Involved forces of the medical response

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.

MES 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 RA MES 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 MES.

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.

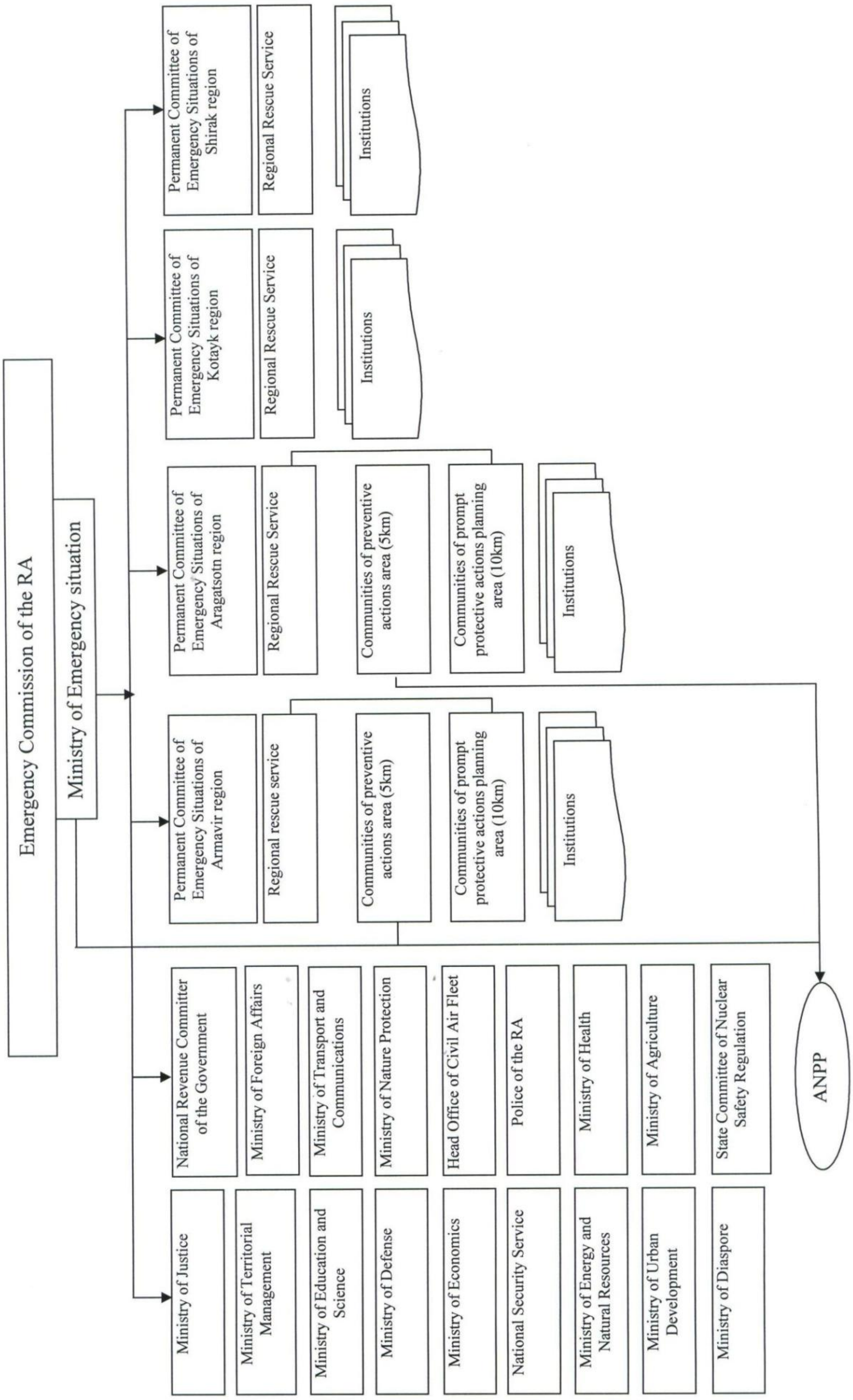


Fig. 6-1: Structure of Population Protection Planning System

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

a) 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.

b) 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

a) 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 Unit2 (reg. № 01.2.2);
- Guideline on beyond the design basis accidents management at ANPP;
- Procedure on mitigation of accidents in the electric part of ANPP, УБ.ЭТД.12.ЭЦ-026 (reg. № 845);
- 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, УБ.ЭТД.12.ЭЦ-015.

b) 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 (unborated) 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 unit1 of ANPP (which is in permanent shutdown without fuel in the reactor) can be considered, especially the water from the Borated Water Storage Tank (BWST). Except of this, further non-conventional means are currently under development for ANPP unit 2 (see p. 5.1.1.5).

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 30 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 4 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

The current status is described. 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. 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)

Radioactive releases at ANPP are monitored with facilities like RCS-03 and System 8004-01 which measure activity of the following components:

- LLA (long- lived beta-active aerosols);
- IC (gamma-active couples of molecular iodine -131);
- BG (beta-active gases);
- NRG (noble radioactive gases).

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.

In 5km zone in ANPP vicinity there is gamma radiation intensity monitoring system installed. But so far the data are not communicated to MCC, correction is planned.

In the occurrence of a nuclear accident, the management of the situation should be primarily focused on:

- The containment internal pressure reduction by efficient spraying to prevent opening of pressure relief and reduce releases;
- The aerosols and iodine in-containment inventories which could deposit thanks to use of containment sprays;
- The amount of water inside the borated water storage tanks and the spent fuel pools;
- The radioactive releases that might occur to the environment are limited by:
- The radioactivity permitted in the primary circuit under normal operating conditions;
- The maximum leak rates permit ted for the primary and second circuits;
- The containment natural leak rate to the environment.

Reactor hall and containment designs

The containment 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. The containment is located below the reactor hall.

In case of radioactive release towards the containment, the containment structures and the spray system will guarantee that most of the radioactivity remains inside the containment. As long as this radioactivity stays inside the containment, one can rely on the radioactive decay to limit the releases to the environment. Indeed, in function of the isotope, the in-containment source term can decrease by 5 to 25 % within solely half a day.

The internal containment pressure is measured by a series of classified equipment, backed-up with some additional ones.

The spray system is used to limit the pressure increase inside the containment and reduce the leak rate.

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 unit2 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.

Containment isolation

Each containment penetration is designed with a serial double leak-tight isolation system that can be remotely controlled from the main control room. Those not required for accident mitigation are automatically shut down if an abnormal situation is detected requesting the isolation of the containment. In the occurrence of a failure of the containment isolation, some mitigation actions are described in the procedures to recover the situation.

Reduction of airborne contamination inside the containment

Using the containment spray system helps to retain radioactive aerosols and some of the iodine within the liquid phase. Moreover, potassium metaborate ($K[B(OH)_4].xH_2O$) 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.

Internal containment leakage monitoring

The leak-tightness of the containment is measured periodically by pressure tests performed at reduced pressure compare to the design pressure, due to the existence of the pressure relief system operating below the design pressure.

Other provisions

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 m^3 from which 1850 m^3 is currently free.

Release calculation and assessment programs

Some provisional studies have been performed with the RASCAL 4 code to assess the impact on radioactive releases on the environment [2]. However, other assessments are also foreseen to be used to support the management of potential releases in the environment. The objective is to support the development of PSA level-2 and Severe Accident Management Guidance's for ANPP-2 (see § 6.2.3).

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 – Ministry of Emergency Situations, 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 § 6.1.1.4).

6.1.3.2. LOSS OF COMMUNICATION FACILITIES / SYSTEMS

See § 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 § 6.2.5.

a) Main Control Room and Back-up Control Panel

This point is dealt with in § 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:

1. MCC – create an airlock in the crisis centre;
2. MCC - purchase additional dose metering instrumentation and equipment;
3. Creating a new back-up crisis centre BCC of ANPP.

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

1. Installation of communication and notification devices;
2. Installation of computers, office equipment, special software for situational analysis, hearings and meetings;
3. 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 § 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 § 6.1.3.1

6.1.3.6. FEASIBILITY AND EFFECTIVENESS OF ACCIDENT MANAGEMENT MEASURES UNDER THE CONDITIONS OF EXTERNA HAZARDS (EARTHQUAKES, FLOODS)

See previous chapters.

6.1.3.7. UNAVAILABILITY OF POWER SUPPLY

See § 6.3.1.6

6.1.3.8. POTENTIAL FAILURE OF INSTRUMENTATION

The 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.

This point is also dealt with in § 6.2.4, where a list of instrumentation needed for SAMG is included.

6.1.3.9. POTENTIAL EFFECTS FROM THE OTHER NEIGHBOURING 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 organisational 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.

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 [1] 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 procedure B-270 “270.200.Д50 КБГП 2001”. 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 [1] 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 [1] 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 (see the report [4]). 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 120 hours (5 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 [5], [13]. 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 [5]. 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 [13]. 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. In [5], this penetration was reached at 120942 s (33:35 hours). The other scenarios [13] 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 is included in SAMG and it was analyzed in [13].

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 [1]. 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 [2]. The need for comprehensive operating procedures is also indicated in IAEA-TECDOC-640 ([3], specific for VVER-440/230 but used as a basis in [2]).

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 [2]. 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, [4]) 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 [5]. 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 ANPP2.**

The development of Severe Accident Management Guidance (SAMG) has been recommended by IAEA mission in 2009 [2]. In the framework of the Comprehensive Modernization Program for ANPP-2 [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 [2].

Presently, SAMG at full power are under development. They are based on generic WOG SAMG philosophy [2]. 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 ANPP2 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 2014.

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 [7] 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 ANPP2. Revision of post-Fukushima generic WOG SAMG is awaited at the end of January 2013. **All new means, features as a follow-up of the stress tests or resulting of the modernization program of ANPP2 will be included in SAMG.**

Regarding the technical tools to manage a loss of core cooling, it is recognized in [3] 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 ANPP2 [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[1]. 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;
- Electrical supply $U_{3,4}$ for unit 2.

It has been recognized during the different IAEA missions that I&C of ANPP2 were very old fashioned and might not be reliable [2]. In the framework of the different related safety upgrading programs for ANPP2, 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 program [2].

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 [3] is adequate for accident management before core damage: the plant-specific PAMS are qualified for accident monitoring.

In addition, this upgrade program for instrumentation allowsANPP2to adopts the generic WOGSAMG approach [4] 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 in reference [8] that “concerning the instrumentation needs and capabilities, the experience has shown that this strategy to be workable for the following reasons:

1. 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.
2. 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).
3. 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 ANPP2WOGSAMG 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:

1. Core temperature;
2. Pressure in the primary circuit;
3. Water level in the steam generators;
4. Pressure in the containment;
5. Water level in the containment sump;
6. Hydrogen concentration in the containment atmosphere;
7. 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:

1. Core thermocouples are located at the exit of selected assemblies. They can be supplemented by cold/hot legs thermocouples;
2. RCS pressure is measured at several places in the RPV, primary circuit loops and pressurizer;
3. Water level is measured in all 6 steam generators;
4. Containment pressure is measured at several places in the containment rooms;
5. 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;
6. **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;

7. 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 [2]. 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 [2]. 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 CO_2 exceeding permitted level [2].

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 is planned in 2013. 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 unit #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 in [2]. 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 § 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 [1] 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 § 6.2.1). Practically, the use of the guideline [1] 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 may 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 [1] 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 4:30 h between the start of high temperature Zr oxidation and vessel failure in the SBO scenario [2]). 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 2. This situation is inherent to the specific design of such a plant for which:

- The containment[1] volume is relatively small ([1] 12330 m^3, [4] 13785 m^3);
- The total mass of Zr in core is relatively large, 19268 kg [4].

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):

1. 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.
2. Station Black-Out (SBO) with partial inertization by steam.
3. Other scenarios without primary system rupture like transients or a small break LOCA with sprays operating.

The case 1 was studied in detail in [4] 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 [4]. A typical solution recommended in [4] was to manage sprays to avoid any risk of hydrogen deflagration.

[1] 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.

The case 2 is described in reference [2]. These calculations show partial containment inerted by steam and partial leak of hydrogen and oxygen to the environment which suppressed the hydrogen risk. Because of the high natural leak rate of the ANPP 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 the activity release rate, in-containment pressure should be reduced by sprays. However, as already stated in case 1, this action could lead to an increase of the hydrogen explosion risk (also highlighted in [5]).

The case 3 can lead to high risk of fast hydrogen deflagration or transition to detonation.

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 2 and analyzed both in frame of the ongoing PSA level 2 study and within the ongoing project aiming at developing a set of SAMG, specific for ANPP 2 (see § 6.2.3).
- Providing hydrogen concentration measurement and installing PARs are listed in [4] 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

ANPP2 containment presents no risk of over-pressurization mainly because it is equipped with automatic and passive pressure relief to the atmosphere before reaching its design pressure. Moreover it has a high natural leak rate. In spite of it, pressurization of the containment should be prevented as much as possible by other means (sprays) to reduce the activity release to the atmosphere.

6.3.1.4. PART IV: PREVENTION OF RE-CRITICALITY

The VVER-440 reactor of the ANPP 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 800 m3 located below the containment. The minimum boric acid concentration in this tank is 12 g/kg [3] and the temperature 50 °C, for which parameters a subcritical core is assured [3];
- 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 (BMMT) at ANPP 2 in case of a severe accident. It has to be noticed that the cavity is surrounded by soil and BMMT would not lead to direct activity release to the atmosphere. A sound strategy should be developed and discussed for ANPP2 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 BMMT 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 BMMT) is suitable for smaller power reactors like ANPP 2 and it is foreseen in the list of planned plant upgrades in [4]. 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 m3). It also requires a heat removal strategy to prevent cavity water from boiling off (sprays for containment heat removal can be used).

ANPP2 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 recriticality 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 base mat 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 § 5.3.3, 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): existing spray system using AC power (no additional requirement) 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 § 6.3.4

6.3.1.8. CONCLUSION ON THE ADEQUACY OF SEVERE ACCIDENT MANAGEMENT SYSTEMS FOR MAINTAINING THE CONFINEMENT CAPABILITIES

See § 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 § 6.3

6.3.2. 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. In [2], a severe accident originated from a complete station black-out was analyzed and no opening of the valves was visible even without sprays. The early source of hydrogen from Zr-steam reaction was small, about 150 kg and it was not ignited because large part of it escaped through the natural leak.

The second strong cliff-edge effect is fast hydrogen deflagration or its transition to detonation that may fail the containment. According to simple calculation in [3], this might happen already during the Zr-steam reaction about 1 hour after the onset of core damage [2]. Developing SAMG for hydrogen together with technical solution – installing hydrogen concentration measurement and igniters – are proposed in § 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 because of oxygen-lean atmosphere. A simple calculation in [3] has shown that burning or recombining 350 kg of hydrogen - the amount corresponding to about 40% of oxidation of Zr in the core - leads to the complete consumption of the oxygen present in the containment. Using PARs (e.g. of small capacity) is proposed in § 6.3.1 to avoid hydrogen deflagration after 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 base mat or radial penetration by debris. The consequences of the cavity base mat 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 § 6.3.1 to eliminate this risk.

The base mat 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 § 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.

6.3.3. 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 [1] 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 § 6.2.3. The adequacy of WOG SAMG is also recognized for protecting the containment integrity.

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 [2] starting at the design pressure;

- Containment pressure relief valves to keep the integrity of the containment during large LOCA.

Concerning the first item, the plant is aware of the situation and measures to reduce this leak to about 100%/day are already proposed [3]. 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 [2]. 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 § 6.3.1).

Other recommendations have already been made in § 6.3.1 and 6.3.2. These should be analysed 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 § 6.2.3).

6.3.4. 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 § 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.

6.3.5. 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 § 6.2.5.

6.3.6. 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 realisation of some measures to mitigate the hydrogen risk inside the containment – installing igniters and Passive Autocatalytic Recombiners (PARs) as proposed in § 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 [2] 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.

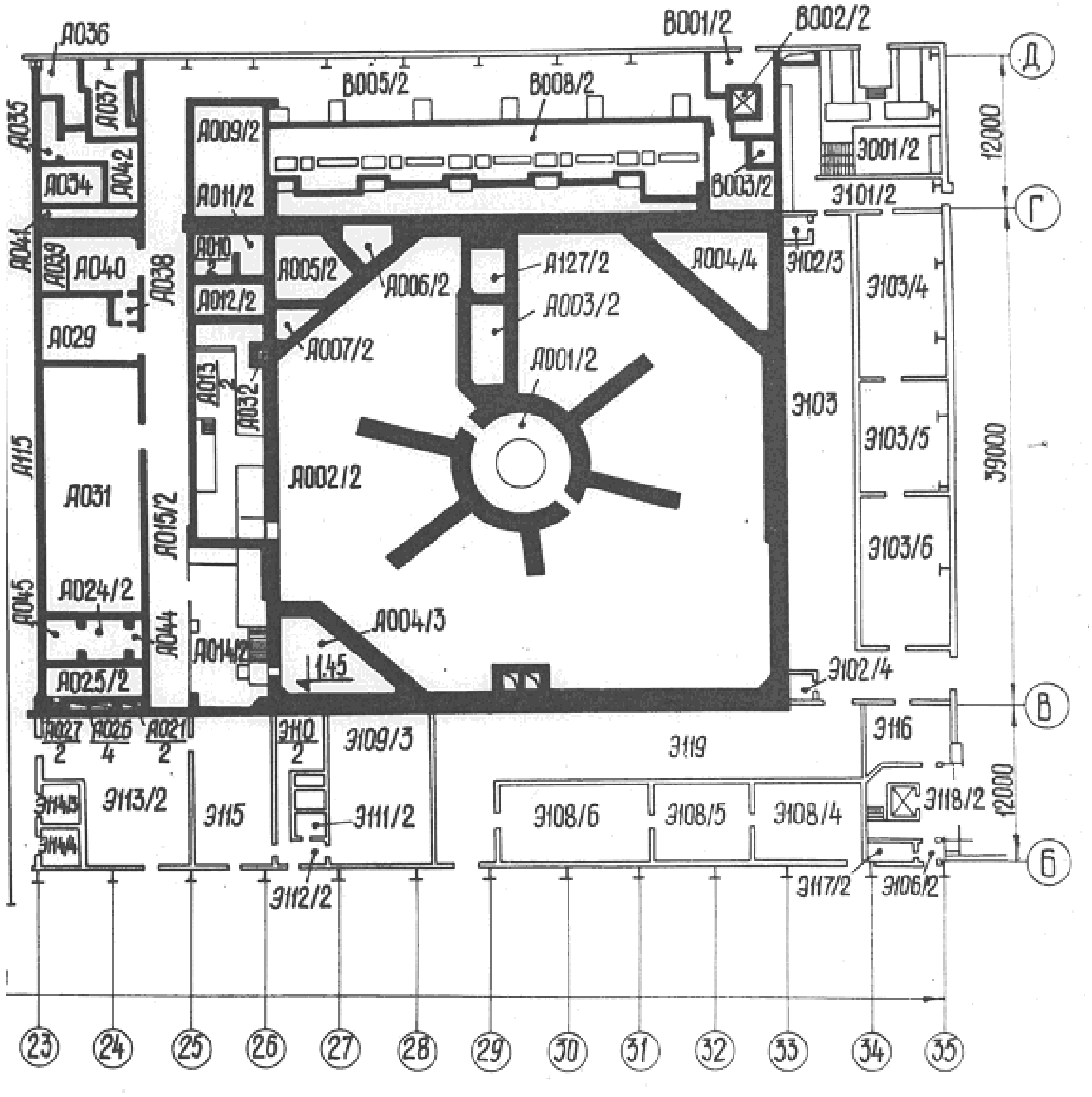
Much of the hydrogen coming in the containment during the early and the late phases of the severe accident will be released outside. Hydrogen is released mainly directly towards the environment via the containment pressure relief valves, which therefore prevails too much accumulation of hydrogen that can enhance leakages to the neighbouring 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 100000 m^3, so the accumulation would require a large quantity of hydrogen [2]. 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 § 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.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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 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 0.02 m^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 as recommended in [1].

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 together with a little carbon monoxide. Although the inflammable gas concentration and oxygen concentration were high enough to start hydrogen combustion, no burns were identified in the hermetic rooms during the whole scenario (33 hours). 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 m^3), 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 5.1.5 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 2. The implementation of SAMG is under way (see § 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 § 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 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 § 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 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 [17], 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 § 6.3.1.4.

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 a future revision of ANPP2 SAMG.

There are sufficient time margins to restore the SFP cooling (see § 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.

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7. CHAPTER 7: ACTION PLAN

The actions for improvement or further analysis identified in the frame of the stress test self-assessment are summarized hereafter.

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. Modification of the emergency gas removal system in compliance with the project “Gas removing system from under the Reactor's Lid, ANPP Unit2".
3. Performing of seismic margin evaluation of the fire extinguishing system and issuing of recommendations (reinforcing, modification, etc.) for seismic safety improvement up to DBE and implementation of the issued recommendations.
4. Perform analysis of the SRS damage in case of explosion of nitrogen recipients and hydrogen storage tanks.
5. Perform analyses for the consequences of the possible flooding of the following rooms and compartments: CDD-6/0,4kV room, intermediate relay panel–2 rooms, flooding of the MCP control room through the ceiling if the Turbine Hall (elevation of 14,0 m) is flooded. On the basis of these analyses the relevant protective measures will be developed.
6. Implementation of the programs for seismic upgrading of I&C equipment and monitoring system.

Flooding

7. 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.
8. Equip DGS with alarms indicating occurrence of water level in basement area with output of light signals both in DGS operator room and outside the DGS.
9. Seal the penetrations of the turbine hall located at the level of -3.6m.
10. Add operator’s actions in the DGS operation manual, in abnormal modes, in case of water inflow in the basement area.
11. Foresee mobile pumps for water pumping out of the DGS.
12. Develop and implement measures with the purpose of building barriers on the way of water flow to the turbine hall gates.

Extreme Weather Conditions

13. To develop and implement technical solution in order to avoid damage of window of the DG building in case of extreme wind conditions.
14. To reconsider PSA level 1 taking into account the following:
    a. To present a more detailed description and justification of extreme natural phenomena screening out;

b. To present a more justified choice of combinations of different natural factors;
c. To check the calculations of meteorological characteristics for different frequencies of exceedance;
d. To perform the analysis of possible impact of a tornado on the safety of the plant.

15. To study the need of filtering devices on air inlets of diesel generators.

Loss of Electrical Power and Loss of Ultimate Heat Sink

16. Review of the existing symptom-based operating procedures of the reactor (in all operating modes of the unit) and units 1&2 SFP, and, if necessary to develop new procedures and programs for training of the personnel under various accident scenarios and common failures of safety systems.

17. Perform analyses of the units 1&2 SFP cooling and introduce additional measures based on the performed analyses. The issues of concern are, for example, the timeline calculation for the recent fuel in case of loss of cooling, or the need to provide power supply/load sequencer for unit 1 SFP cooling pumps and/or unit 1 SFP make-up pumps in case of loss of external power supply to ANPP.

18. Develop additional measures to extend the operating time of reversible motor generators (RMG) in an invertor mode that will lead to increase the time to provide SRS I&C AC power supply.

19. Installation of an additional tank of diesel fuel on the DGS (for each bus bar) in order to run the train I and II of emergency power supply during 72 hours at full load, independently of the fuel inventory in the oil-fuel facilities of the plant. The fuel tank shall have seismic design and meet the required safety standards.

20. Replacement of all reversible motor generators (RMG) with modern inverters. This will result in upgrading I category loads power supply logic reliability and the decrease of battery loads.

21. Implementation of the scheme for battery charging from SEC and/or the portable diesel generator.

22. Perform analysis of basic diagram for consumers power supply from SEC. Develop and implement activities aimed at minimizing personnel manual actions to activate the SEC system.

23. Implement logics (systems) of water supply to primary circuit, Unit 1&2 SFP’s, SGs 1-6, DWSTs 1-4 from alternative water sources.

24. 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, refuelling, closed reactor, open reactor).

25. Perform additional calculations and analysis of different scenarios related to the reactor mode (opened reactor, closed reactor, refuelling). Development of additional procedures on the basis of new calculations and analysis. Organization of personnel training on the use of the new procedures.

26. Develop and implement additional measures to use a large reserve of service water in the inlet and outlet channels, as an alternative heat sink.

27. Purchasing of autonomous alternative power supply sources. In order to fulfil the required safety functions, it is necessary to perform preliminary analysis prior to purchasing in order to identify parameters (output, etc.), required qualification, configuration and allocation (including mobility option) for those power supply sources.

    For example, it could be possible to use:
    - DG 2.0MW, 6kV; 0.4kV; 220 VAC;
    - 2 DGs 0.4kV; 500kW.

28. Purchasing of autonomous alternative means for:
    - Boron solution supply to primary circuit;
    - SG make-up from alternative water sources;
    - SFP 1&2 cooling from alternative water sources;
    - Filling of spray ponds of ESWS and DWST 1-4 from alternative water sources or direct cooling water supply to essential loads from alternative water sources.

    It is necessary to perform preliminary analysis, prior to purchasing, with the objective to identify parameters (output, etc.), required qualification, configuration and allocation (including the possibility of mobility or the possibility to use fire brigade pumps) for those make-up means in order to assure the fulfilment of the necessary safety functions.

    For example, it could be possible to use:
    - A diesel pump 5-10kgf/cm2, 600-1000m3/h;
    - A medium pressure diesel pump 70-90kgf/cm2, 100-250m3/h;
    - Two low pressure diesel pumps 10-15kgf/cm2, 150-500m3/h.

Severe Accident Management

29. Implement additional analysis of MCP shaft sealing behavior after 24 hours at primary circuit temperature above 260°C.

30. Development of a complete set of emergency operating procedures (EOP).

31. Development of a full set of severe accidents management guidelines.

32. Perform modernization of Core Cooling Emergency System.

33. Implementation of measurements of hydrogen concentration in containment.

34. Installation of afterburners and passive autocatalytic hydrogen recombines.

35. Development and implementation activities aimed at maintaining melting fuel inside RPV via external cooling of the reactor vessel.

36. Ensure improvement of containment tightness.

37. Implement modernization of the spray system.

38. Perform a detailed analysis of possibility of hydrogen accumulation in rooms outside the containment.