REPUBLIC OF TURKEY

EUROPEAN "STRESS TESTS" FOR NUCLEAR POWER PLANTS

NATIONAL REPORT OF TURKEY

REVISION 2

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TURKISH ATOMIC ENERGY AUTHORITY

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ABBREVIATIONS

ACNS : Advisory Committee on Nuclear Safety
ACS : Automated Control System
AEC : Atomic Energy Commission
ALT : Automatic Load Transfer
AOO : Anticipated Operational Occurrences
APC : Akkuyu Project Company (Akkuyu Nuclear JSC)
ARMS : Automated Radiation Monitoring System
ASS : Automatic Step-By-Step Startup
AT : Auxiliary Transformer
BDBA : Beyond Design Basis Accident
BDBAMG : Beyond-Design-Basis Accident Management Guide
BOP : Balance of Plant
BRU-A : Fast-Acting Relief Valve for Steam Discharge into Atmosphere
BRU-K : Fast-Acting Relief Valve for Steam Discharge into Condenser
CCF : Common-Cause Failure
CMC : Core Melt Catcher
CPS : Control and Protection System
CPS CR : Control and Protection System Control Rod
CSF : Critical Safety Functions
DBA : Design Basis Accident
DBE : Design Basis Earthquake
DCB : Direct Current Board
DCH : Direct Containment Heating
DG : Diesel Generator
DSHA : Deterministic Seismic Hazard Assessment
DTHA : Deterministic Tsunami Hazard Assessment
EAL : Emergency Action Level
ECCS : Emergency Core Cooling System
ECR : Emergency Control Room
EPSS : Emergency Power Supply System
ECS : Emergency Cooldown System
EOP : Emergency Operating Procedure
EP : Emergency Protection
EPCS : Emergency and Planned Cooldown System
EPS : Emergency Power Supply
ESFAS : Engineered Safety Feature Actuation System

EUR : European Utility Requirements
FA : Fuel Assembly
FDF : Fuel Damage Frequency
FE : Fuel Element
FFSF : Fresh Fuel Storage Facility
GIS : Gas Insulated Switchgear
HA : Hydro Accumulator
HCLPF : High Confidential of Low Probability of Failure
IAEA : International Atomic Energy Agency
IRG : Inert Radioactive Gases
JSC : Joint Stock Company
LDL : Low Detonation Level
LFL : Lower Flammability Level
MCP : Main Coolant Pipeline
MCR : Main Control Room
MCS : Monitoring and Control System
MDGS : Main Diesel-Generator Station
MENR : Ministry of Energy and Natural Sources of Turkey
MER : Maximum Emergency Release
MS : Meteorological Station
MSIV : Main Steam Isolation Valve
MSL : Mean Sea Level
NF : Nuclear Facility
NO RPSS : Normal Operation Reliable Power Supply System
NOS : Normal Operation Power Supply System
NPP : Nuclear Power Plant
NPT : Non-Proliferation Treaty
OBE : Operating Basis Earthquake
OJSC : Open Joint-Stock Company
OHES : Offshore Hydraulic Engineering Structures
PAR : Passive Autocatalytic Recombiners
PCTA : Peak Coastal Tsunami Amplitude
PECR : NPP Protected Emergency Command Room
PHRS : Passive Heat Removal System
PMF : Probable Maximum Flood
PMP : Probable Maximum Precipitation
PMRF : Probable Maximum River Flood
PMSS : Probable Maximum Storm Surge

PMT : Probable Maximum Tsunami
PORV : Pilot-Operated Relief Valve
PRZ : Pressurizer
PSA : Probabilistic Safety Assessment
PSAR : Preliminary Safety Analysis Report
PSHA : Probabilistic Seismic Hazard Assessment
PTHA : Probabilistic Tsunami Hazard Analysis
PTL : Power Transmission Lines
QMS : Quality Management System
RCP : Reactor Coolant Pump
RF : Russian Federation
RI : Reactor Internals
RP : Reactor Plant
RV : Relief Valve
SAMG : Severe Accident Management Guideline
SB : Storage Battery
SDGS : Standby Diesel-Generator Station
SF : Safety Functions
SFA : Spent Fuel Assembly
SFP : Spent Fuel Pool
SG : Steam Generator
SHA : Seismic Hazard Assessment
SNF : Spent Nuclear Fuel
SNFS : Spent Nuclear Fuel Storage
SPDS : Safety Parameters Display System
SPR : Site Parameters Report
SS : Safety System
SSE : Safe Shutdown Earthquake
SSR : Specific Safety Requirements of IAEA
TAEK : Turkish Atomic Energy Authority
TEH : Tubular Electric Heater
TPS : Transport Packaging Set (cask)
UPS : Uninterruptible Power Supply
URAP : National Radiological Emergency Plan
TEK : Turkish Electricity Authority
VVER : Russian design Pressurized Water Reactor

INTRODUCTION

This report is prepared in accordance with the European Nuclear Safety Regulators Group (ENSREG) requirements for performing comprehensive risk and safety assessment in the light of the Fukushima accident.

Turkey has currently no nuclear power plants in operation. Negotiations to build a NPP at Akkuyu site in Turkey started with the Russian Federation in February 2010 and concluded with an Intergovernmental Agreement based on a Build-Own-Operate model. The Agreement was signed on May 12, 2010. Relying on the agreement, “Akkuyu Nuclear Power Plant Electricity Generation Joint-Stock Company (Akkuyu Project Company (APC), soon changed his title to Akkuyu Nuclear JSC), responsible for the construction, operation and decommissioning of 4 units Water-Water Energetic Reactor, VVER, of each to produce 1200 MW power, was established. The nuclear regulatory body of Turkey, Turkish Atomic Energy Authority (TAEK), recognized APC as the owner (hereafter referred to as Applicant) on February 7, 2011.

The Akkuyu Site on the Mediterranean coast was granted a site license for building a Nuclear Power Plant (NPP) in 1976. In 2011, this site was allocated to Applicant as specified in the Intergovernmental Agreement. Applicant started site investigations in Akkuyu for updating the site characteristics and parameters according to the national procedures laid out in the Decree on Licensing of Nuclear Installations [1]. Upon completion of updating the information on the characteristics and parameters of the site, Site Parameters Report is presented by the Applicant to TAEK. Site Parameters Report also includes the results of detailed site investigations performed at the NPP site and the precise values of the project parameters. On February 9, 2017 project parameters are approved by Turkish Atomic Energy Authority in accordance with the relevant articles of the Decree

On March 2, 2017 Applicant applied for construction license of Akkuyu NPP Unit 1. As the results of review and assessment of the application, limited work permit was given to Applicant for Akkuyu NPP Unit 1 at the 146th meeting of Atomic Energy Commission on October 19, 2017.

With the limiting work permit, the Applicant is allowed to proceed with the installation of structural foundations of reactor and environmental safety related buildings and facilities and construction of other structures, systems and components in accordance with the Decree.

On April 2, 2018, construction license is granted for the Akkuyu NPP Unit 1 based on the application of the Applicant by the decision number 148/2 of the Atomic Energy Commisssion on March 30, 2018, in accordance with the Law on Turkish Atomic Energy Authority and related regulations.

According to the Decee on Licensing of Nuclear Installations, operation and construction licenses for nuclear facilities can be issued based on general and specific conditions. So, there are also general and specific conditions as integral part of the construction license of Akkuyu NPP Unit 1. The license conditions are mainly related to the detaileddesign of the plant issues to be finalized during the operation license phase. The license conditions are being fulfilled by the Applicant and foreseen to be fully fulfilled before operation license. Therefore some of the topics mentioned in this report will be detailed before operation phase of the plant.

On June 22, 2018 Applicant applied for construction license of Akkuyu NPP Unit 2. As the results of review and assessment of the application, limited work permit was given to Applicant for Akkuyu NPP Unit 2 at the 149th meeting of Atomic Energy Commission on November 30, 2018.

The Stress Tests National Report of Turkey, which included evaluations of the regulatory body and comments of the relevant bodies of Turkey (Ministry of Energy and Natural Resources of Turkey and Ministry of Foreign Affairs). The finalized report was sent to the European Commission on June, 2012.

On October 29, 2013 the "Stress Tests for Nuclear Power Plants in EU Neighboring Countries: Experience and Follow-up" meeting was held in Luxembourg. At that meeting, TAEK and Applicant delegations presented their findings about the National Report of Turkey and requested the Report to be published at the ENSREG website. Also it was stated that Akkuyu Project was in an early stage to have the peer review and in the developing phases a peer review would be considered. After the meeting, Stress Tests National Report of Turkey was published in ENSREG website among the neighboring countries.

After Applicant's application for construction license of Akkuyu NPP Unit 1, studies for updating the Stress Tests National Report of Turkey was started.

Revision 2 of the Stress Tests National Report of Turkey is prepared based on the Applicant's report [2] and Turkish Atomic Energy Authority's findings. The report is based on input data from the Site Parameters Report, Preliminary Safety Analysis Report and Probabilistic Safety Analysis Report of Akkuyu NPP Unit 1.

This report contains mainly seven topical areas in conformity with the ENSREG Guidance for the content and format of National Reports, and additionally an introductory chapter on Turkey's legislative and regulatory framework. The first chapter includes general information about the site and the plant design. In Chapters 2, 3 and 4, site and design features related to external events including earthquakes, floods and extreme weather conditions in the Akkuyu site are presented respectively. The information on the scenarios involving loss of electrical power and loss of ultimate heat sink are provided in Chapter 5. Chapter 6 focuses on evaluation of actions considered in preliminary design documents and reference plant design to prevent severe fuel damage in the reactor core and spent fuel pool and also the plant response and the effectiveness of the preventive measures to be implemented in severe accident management strategies.

LEGISLATIVE AND REGULATORY FRAMEWORK IN TURKEY

General Information on Legal Framework

Turkish regulatory structure is composed of laws, international treaties, decree laws, regulations, guides and standards. The hierarchical pyramid of Turkish legislation structure is presented in Figure 1.

Turkey’s legislative and regulatory framework ensures that nuclear materials and facilities are utilized and nuclear activities are performed with proper consideration for health, safety, security and protection of people and the environment. In this respect, Turkey signed and/or approved international agreements and conventions, a list of which is given in Annex I.

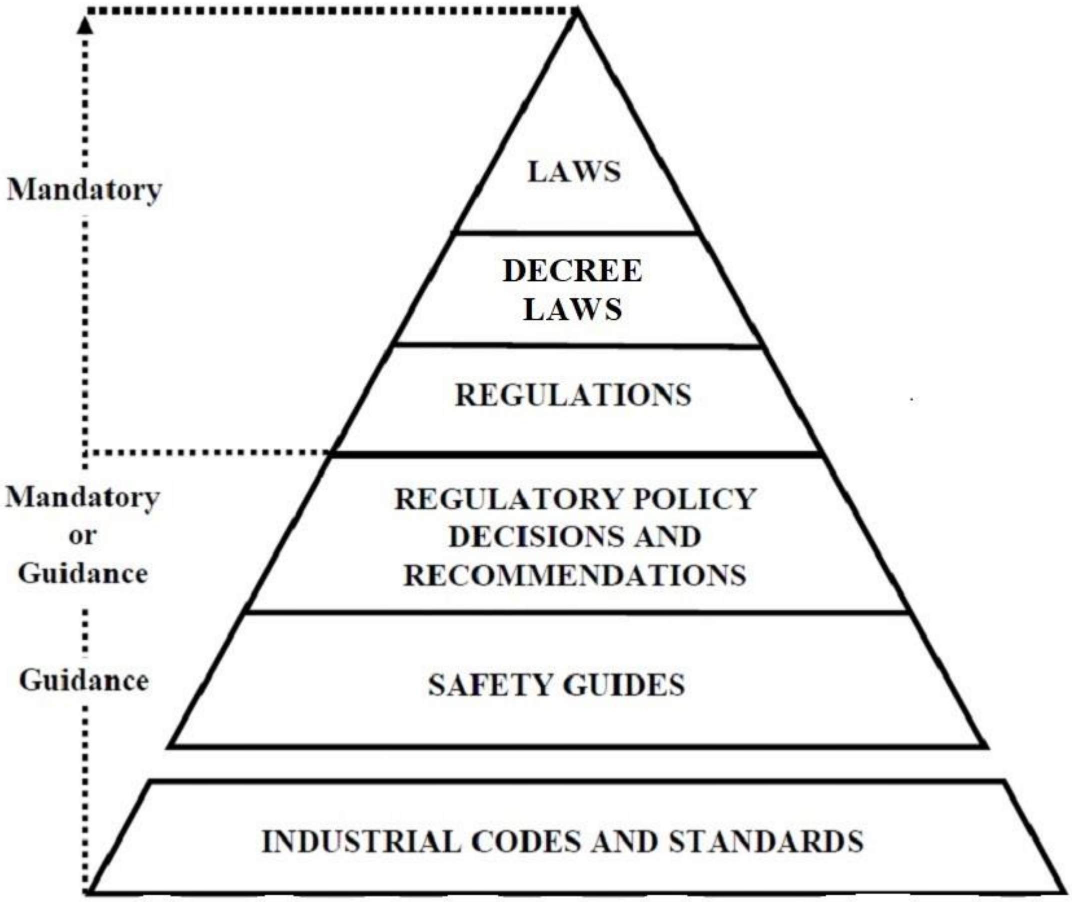

Figure 1 Hierarchy of legislation in Turkey

Altough it is not shown in Figure 1, according to Article 90 of Constitution of the Republic of Turkey, international agreements duly put into effect have the force of law. Therefore, they are also a part of the Turkish legislation. In this regard:

- Turkey is party to the Convention on Nuclear Safety.
- As a non-nuclear weapon state party to the NPT, Turkey has established a system of accounting for and control of nuclear materials based on the Safeguards Agreement and the Additional Protocol with the IAEA. Turkey has received an ISSAS mission of IAEA in June 2010, who reviewed this system with respect to the requirements of the Safeguards Agreement and the Additional Protocol.
- Turkey is also party to the Convention on Physical Protection of Nuclear Material and Nuclear Facilities implemented its requirements in national regulations. Current regulations are under revision to introduce latest changes to these systems.

Regulatory Infrastructure in Turkey (Before July of 2018)

The main Turkish legal instrument on nuclear installations is the "Law on Turkish Atomic Energy Authority", which establishes the Turkish Atomic Energy Authority as a regulator for nuclear and radiation facilities and activities on safety, security and safeguards, and as a research and development organization in nuclear technology and radiation applications, etc. TAEK reports to the Ministry of Energy and Natural Resources in accordance with the decision of the Cabinet of Ministers. On the other hand, the decision of TAEK on licensing of nuclear installations is not subject to approval of Ministry of Energy and Natural Resources.

"The Decree on Licensing of Nuclear Installations" is the second high level instrument regarding nuclear safety, security and radiation protection in nuclear installations. Rules and procedures related to the licensing of nuclear installations are laid out in the "Decree on Licensing of Nuclear Installations", entered into force in 1983. The decree defines permits and licenses to be obtained, requirements for applications to these permits and licenses, including lists of documents to be submitted, review and assessment procedures, the responsible entities within TAEK for each authorization, approval mechanisms for modifications during construction and operation; and authorizes TAEK for inspecting the installations throughout their lifetime and enforcing penalties such as limiting, suspending and revoking the licenses.Another important regulatory document is the "Directive on Principles of Licensing of Nuclear Power Plants", which lays out the principles for establishing a licensing basis for NPPs. Principles state that the issues insufficiently addressed by existing Turkish regulations on nuclear safety shall be covered by requiring compliance with the IAEA Safety Fundamentals, the Safety Requirements. If the provisions contained herein are insufficient, regulations of the vendor or designer country and the, particularly, safety fundamentals and safety requirements shall applied. For remaining issues, third party country laws, regulations, standards, or IAEA the Safety Guides are referenced. The directive also requires the Applicant to submit the regulatory body a reference plant of the proposed design for facilitating the licensing process.

Further details on safety principles are addressed in regulations. There are currently 23 regulations directly or indirectly addressing safety of nuclear power plants. The list of the laws, decrees, regulations and guides that are relevant to the nuclear power plants is given in Annex II.

Rules and procedures for accounting for and control of nuclear materials are described in the "Regulation on Accounting for and Control of Nuclear Materials", which satisfy the requirements of the Safeguards Agreement and Additonal Protocol with the IAEA. This regulation is under revision for ensuring compliance with the additional protocol. The national aspects of Convention on Physical Protection of Nuclear Material have been implemented in the "Regulation on Physical Protection Measures of Special Nuclear Materials". This regulation is under revision for ensuring compliance with INFCIRC 225/Rev. 5.

There are several regulations associated with nuclear safety. Suitability of NPP sites is addressed in the "Regulation on Nuclear Power Plant Sites". Basic requirements on design of an NPP are laid out in the "Regulation on Design Principles for Safety of Nuclear Power Plants" and on construction, commissioning, operation and decommissioning of an NPP in the "Regulation on Specific Principles for Safety of Nuclear Power Plants".

Nuclear and radiological emergencies are covered in the "National Regulation on Nuclear and Radiological Emergencies". This regulation only covers the roles and responsibilities of governmental authorities in case of a radiation emergency. Requirements on emergency preparedness and response are addressed by IAEA Safety Requirement GS-R-2.

Regulations that cover radiation protection, operating personnel qualification and licensing, clearance and release of sites from regulatory control and radioactive waste management in nuclear installations has been issued in recent years.

Legal framework is not only limited with the TAEK regulations. There is also the "Environmental Law" regarding environmental impact of these facilities; the "Penal Law", which also defines nuclear and radiological crimes and penalties; and the "Law on Electricity Market" regarding electricity production licenses.

Regulatory framework (Before July 2018)

As for regulatory framework; other than nuclear regulatory body, there are several other government bodies such as Ministry of Health, Ministry of Transport, Maritime Affairs and Communications, etc. which indirectly regulated an NPP as an industrial facility. TAEK is a subsidiary institution of Ministry of Energy and Natural Resources.

In Turkey, nuclear installations are licensed by TAEK regarding nuclear safety, security and radiation protection issues. Licensing procedure is initiated by the application of the owner to be recognized as such. Licensing process for a NPP comprises three main stages in succession: Site License, Construction License and Operating License. There are several permits functioning as hold points during the licensing process, such as limited work permit, commissioning permit, permit to bring fuel to site, fuel loading and test operations permit for operating license, etc. For each authorization, documents required for review and assessment of TAEK are defined in the Decree. The Decree also requires the owner to apply for authorization of TAEK for every modification that may have an impact on the safety of nuclear installation.

It is explicitly declared in the Decree on Licensing of Nuclear Installations that nuclear installations cannot be operated without a valid license. The Penal Law defines operating a nuclear installation without a valid license as a crime, punishable by imprisonment.

In addition, NPPs should obtain an affirmative decision on environmental impact assessment according to the “Regulation on Environmental Impact Assessment” from the Ministry of Environment and Urbanization as a prerequisite to the site license and an electricity production license from the Energy Market Regulatory Authority.

Regulatory inspection and enforcement activities cover all areas throughout the lifetime of a nuclear installation. Inspection of TAEK does not relieve the authorized person/organization of its responsibility for ensuring nuclear safety. The main philosophy for the regulatory inspection is “Trust and Verify”. This is achieved by planning the overall approach in scope and content of the inspection to be conducted, not only limited to the authorized organization but also to include its contractor and supplier chains. TAEK conducts inspections to satisfy itself that the authorized organization is in compliance with the conditions set out in the authorization and applicable regulations, based on the “Regulation on Nuclear Safety Inspections and Enforcement”. Enforcement actions may be taken, as deemed necessary, by TAEK in the event of deviations from, or non-compliance with, conditions and requirements.

Regulatory inspection includes a range of planned and reactive inspections over the lifetime of a nuclear installation and inspections of other relevant parts of the operator’s organization and contractors/suppliers to ensure compliance with regulatory requirements. The methods of inspection include examination and evaluation of all records and documentation, and surveillance, monitoring, auditing and interviewing of personnel and management, as well as performing of actual tests and measurements in all phases of the installation. In addition to TAEK staff, outside local or foreign services may be procured for specific inspection tasks for the purpose of pre-evaluation and obtaining data where necessary.

In case of non compliance with license conditions, security requirements and legislation, The Decree on Licensing of Nuclear Installations authorizes TAEK to grant, decline, limit the scope, suspend and revoke the licenses. TAEK may put a formal request to the Prime Minister to close down a nuclear installation.

In case of regulation violations, TAEK takes into account importance, urgency and seriousness of the violations in regard to nuclear safety for the imposed enforcement. All decisions and actions by TAEK may be challenged by any interested party through the legal system of Turkey.

Generally, current legislative and regulatory framework of Turkey satisfy IAEA requirements regarding nuclear safety and security.

Transition Period of Nuclear Legislative and Regulatory Framework (After July 2018)

The organizational structures of state instutitons are in a transition period at the moment. As mentioned above, TAEK is a subsidiary institution of Ministry of Energy and Natural Resources and before July 2018, TAEK performed regulatory tasks. In order to achive independence in regulatory decision-making, new legal and regulatory framework is introduced. Below is the summary of nuclear regulatory and legal framework established with newly introduced laws and decree. Also studies on drafting new regulations for supporting that framework has started.

In this context, Turkey has changed its nuclear regulatory and legal framework with two decree-laws and one presidential decree in July 2018. These are:

- Decree Law No. 702: Regulation of the Law on the Organization of the Nuclear Regulation Authority and its duties in some laws,
- Decree Law No. 703: Regulation of the Law on the Organization of TAEK,
- Presidential Decree No. 4 on the organization of authorities and institutions linked and related to Ministries (Articles 785-792).

According to these legal documents nuclear regulatory framework shaped as below:

"A new and independent regulatory authority is established under the name of Nuclear Regulatory Authority (NDK)."

This new framework aims to separate the regulatory and research and development responsibilities of TAEK. According to Decree Law No. 702; NDK is an independent regulatory authority which has a public legal entity, administrative and financial autonomy. NDK consists of the Nuclear Regulatory Board (NRB) and the Presidency. The decision-making body of NDK is NRB.

NDK regulates nuclear installations subjected to authorization, oversight of the authorized person, inspection process of the plant, prime responsibility of the operator etc.

In order to support and coordinate the nuclear energy, ionizing radiation and accelerator technology to make scientific and technical work for the benefit of the country used for peaceful purposes, Turkey Atomic Energy Agency (TAEK) is restructured.

TAEK has become an R&D instution and a body responsible from safe disposal of radioactive wastes. It also has roles and responsibilities in the area of radioactive waste management as a promotive and operative party. Another main function of TAEK is to provide training and development of human resources related to the field of duty.

Although NDK is established, TAEK has yet to finish its transformation. Decree on Licensing of Nuclear Installations and all regulations of TAEK related to regulation of nuclear safety, security and safeguards must revised according to new authorization processes. Until new regulations are made old ones are remain valid and applicable. Also the President of NDK and NRB have not appointed yet. Until these appointments are made, the Department of Nuclear Safety of TAEK and other departments and organs continue to work on the licensing of nuclear facilities.

1. GENERAL DATA ABOUT THE SITE AND NUCLEAR POWER PLANT

General information is presented under this chapter depending on information presented in Preliminary Safety Analysis Report of Akkuyu NPP Unit 1 [4]. Since Akkuyu NPP Unit 1 is under construction and some construction license conditions are not yet met, some of the information presented here will be finalized during the operation license phase in the Final Safety Analysis Report.

1.1. Brief description of the site characteristics

Akkuyu NPP site is located in the south of Turkey in Akkuyu bay on the Mediterranean Sea shore (in Mersin province) in the area with a radius at least 3 km. Geographical coordinates of the site center are 36°08′ N and 33°32′ E. The NPP site area falls within the Mediterranean region.

The licensee is Akkuyu Nuclear JSC, which is the project company registered in Turkey that will build, own and operate the nuclear power plant. Four NPP power units of AES-2006 design with VVER-1200 (B-509) type reactors shall be constructed on Akkuyu site in accordance with the Intergovernmental Agreement.

The site allocated for the NPP construction is a fenced area located in surrounding of hills up to 270 m high that are a natural boundary of the site area (Figure 2).

Total area of the NPP site is 1,023 ha allocated for the NPP construction area, dike dam, western and eastern berths, civil assembly yard and access roads. A complex of power distribution structures shall be constructed north of the construction site on a separately fenced terrace.

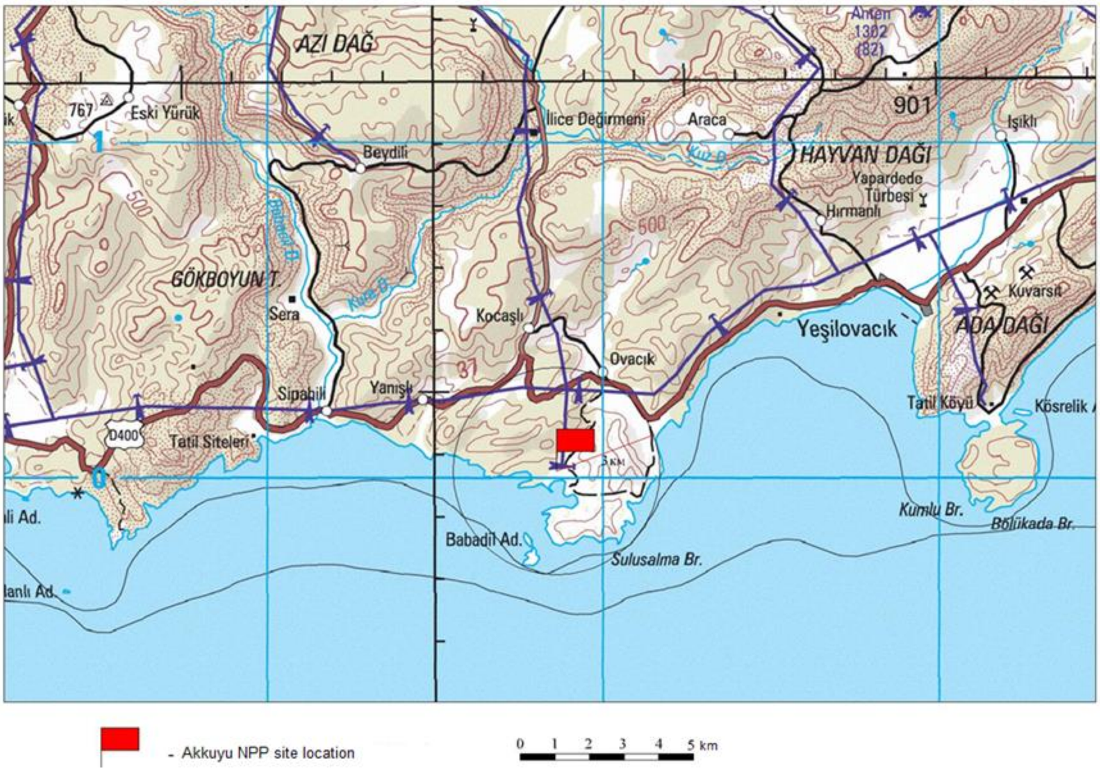

Figure 2 Akkuyu NPP site location

The total population in the emergency planning zone is 1,290 people (Gülnar/Ovacik district) (PSAR, Chapter 1) [4].

All four Akkuyu NPP power units are situated in the southwest part of the project site, which is a bare land plot excavated in the previous years to prepare soil foundation.

According to the design, the NPP grade elevation will be +10.5 m, and the GIS (power distribution structures) grade elevation will be +19.5 m above sea level (PSAR, Chapter 1) [4]. Maximum heights of the NPP structures are +110.5 m (vent stacks of NPP units for gas-aerosol releases), +50.5 m and +48.4 m (vent stacks of auxiliary reactor buildings) above sea level [5].

A set of engineered measures, including construction of a hillside interception ditch, shall be taken to protect the NPP site against landslides, mudflows, surface water flow, etc.

NPP power units are oriented by reactor buildings towards the north, turbine buildings are oriented to the south towards the sea.

NPP units shall be spaced 215 m to allow accommodation of utilities and transport lines between units, and commissioning of the units by start-up facilities.

Diesel generator stations of emergency power supply system (11,12UBN) are situated on opposite sides of each NPP power unit. Standby diesel generator station buildings of the emergency power supply system are spaced out in the general layout to prevent their simultaneous failure in the event of an aircraft crash. Each power unit is also provided with one normal operation main diesel generator station building (13UBN). The design also includes an alternative diesel generator set (ADGS).

Offshore hydraulic engineering structures are situated on the side of turbine buildings of each NPP power unit designed for drawing cooling sea water and discharging heated water into the sea.

Figure 3 shows the layout of Akkuyu NPP buildings and structures.

Each power unit comprises the following main buildings: reactor building (1-4) with transport portal, auxiliary reactor building (5-8), turbine building (9-12) with adjacent main demineralizer building and normal operation power supply building.

Common-plant buildings and structures are situated in the north-eastern part of the site from the side of the reactor building of Unit 1: radioactive waste storage and treatment building (17), fresh fuel storage (13), spent fuel storage (14), etc.

Engineered safety features of Akkuyu NPP include active and passive systems designed to ensure safe reactor shutdown, nuclear fuel residual heat removal, mitigation of consequences during anticipated operational occurrences and design basis accidents, to limit radioactive releases into the environment, and to prevent or mitigate beyond design basis accidents. The safety system design is based on a single failure criterion and principles of redundancy, diversity, independence and physical separation.

The source of service water supply for Akkuyu NPP is Mediterranean seawater with single circulation. Seawater from the Akkuyu bay flows straight-through heat exchangers cooling the systems of each NPP unit. After that, heated water is discharged under residual pressure through discharge channels into the Mediterranean Sea, which is the primary ultimate heat sink.

Power will be supplied into the Turkish grid from Akkuyu NPP via eight transmission lines (380 kV), including five long transmission lines (more than 70 km) for the main connection to 380 kV power grid (through Seydişehir, Konya, Mersin, Ermenek and Antalya substations), and two short transmission lines with a length of about 6 km to connect to the local 154 kV distribution network (through 380/154 kV autotransformers at Akkuyu-1 and Akkuyu-2 substations) (PSAR, Chapter 1, Chapter 8) [4]. Details of this topic is presented in Chapter 5 of this report.

Auxiliary buildings and structures, which are common for four power units, are situated from the side of power unit 1 in the eastern part of the NPP site.

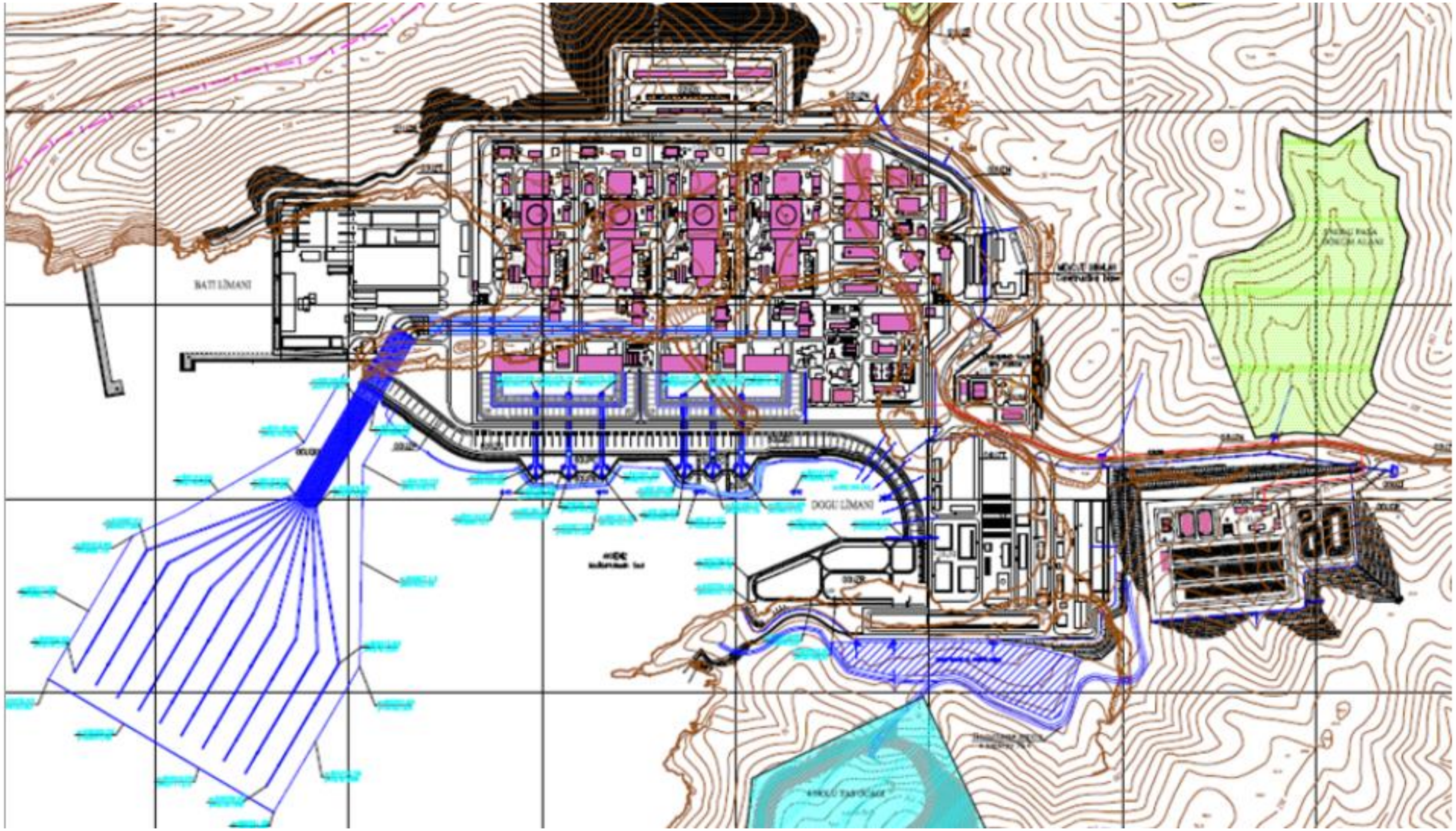

Figure 3 Akkuyu NPP site general layout

1.1.1. Main characteristics of the units

Akkuyu NPP design includes four power units comprising a VVER-1200/B-509 reactor plant and ARABELLE™ slow-speed steam turbine.

Estimated rated electric power of each Akkuyu NPP unit (with reactor thermal power of 3,300 MW and cooling water temperature of 22^0C) will be not less than 1,200 MW (gross).

Key specifications of Akkuyu NPP unit [4] are given in Table 1.

Mediterranean Sea water is the primary ultimate heat sink of Akkuyu NPP. Cooling sea water enters through water intake structures, which are common for all systems.

All Akkuyu NPP units employ straight-flow system of service (cooling) water supply with single circulation of Mediterranean Sea water as the ultimate heat sink. With the cooling water temperature of 25^0C, the cooling water flow to the turbine condensers of each NPP unit is about 254,000 m^3/h.

Service water supply on the Akkuyu NPP site includes the following systems:

- main cooling water system (PA),
- conventional cooling water system (PC),
- secured cooling water system (PE).

PA system is designed to remove steam condensation heat from turbine condensers.

PC system is designed to remove heat from the conventional cooling water system of the turbine building (UMA), chiller building (UQR), normal operation standby diesel generator station (UBN), compressor building (UTF).

PE system is designed to remove heat to the ultimate heat sink from safety-related equipment of systems located in UJA, UKC, UBN, UBN buildings in all unit operation conditions, including accidents.

The secondary ultimate heat sink is atmospheric air, which is used for the operation of passive heat removal systems (PHRS) and ventilation systems.

The thermal pattern of Akkuyu NPP units is double-circuit. The primary circuit is radioactive. This circuit comprises a reactor, four main circulating loops, pressurizer, and auxiliary equipment. Each circulation loop comprises horizontal steam generator, reactor coolant pump set and main coolant pipeline. Primary circuit equipment is housed inside double containment.

Table 1 Key specifications of Akkuyu NPP Units

No.	Parameter	Value
1	Reactor thermal power (rated), MW	3,300

No.	Parameter	Value
2	Active electrical power, not less than, MW	1,200
3	Number of RP main circulation loops	4
4	Parameters of primary and secondary circuits:	
	- primary coolant pressure at reactor outlet, MPa (abs.)	16.2
	- primary coolant temperature at reactor inlet/outlet, °C	297.2 / 328.8
	- reactor coolant flow rate, m^3/h	87,460
	- steam pressure at steam generator outlet, MPa (abs.)	7.0
	- steam flow rate from each steam generator, t/h	1,652
	- feed water temperature, °C	225
5	Maximum U-235 enrichment of fresh fuel, %	4.95
6	Spent fuel burn-up (average / maximum in unloaded fuel assemblies), MW·d/kgU	49.4 / 54.3
7	Refueling interval, months	18
8	Operation time of fuel in the reactor core (with steady-state fuel cycle), years	3 – 4.5
9	Number of FAs in the reactor core	163
10	Number of reactor control rod assemblies	94
11	Service life of RP equipment, years	60
12	Installed capacity utilization factor	0.93
13	Volume of ECCS hydraulic accumulators, m^3: - first stage - second stage - third stage	 200 960 720
14	Capacity of passive heat removal system, MW	80
15	Reactor containment structures: - height (inner/outer), m - inner diameter (inner/outer), m - upper part wall thickness (inner/outer), m	 61.7 / 65.4 44 / 50.8 1.2 / 1.5

The secondary circuit is non-radioactive and consists of the steam generating part of steam generators, main steam lines, turbine, auxiliary equipment and related deaeration, preheating and SG feedwater systems.

Each NPP power unit comprises the following main equipment:

- VVER nuclear reactor with a rated thermal power of 3,300 MW under a primary coolant pressure of 16.2 MPa (water with boric acid is the coolant and moderator in the reactor, and low enriched uranium dioxide is used as fuel in the reactor core) with nuclear reactor pit equipment,
- four horizontal steam generators,
- four reactor coolant pump sets,
- reactor coolant pipeline,
- pressurizer system,
- ARABELLE™ low-speed turbine with rotation speed up to 1,500 rpm.

The primary (inner) containment is made of pre-stressed reinforced concrete, and the secondary (outer) containment is made of cast-in-situ reinforced concrete. The primary containment has a core catcher at the bottom under the reactor designed for severe accident management. The double containment includes:

- inner containment of pre-stressed reinforced concrete designed to withstand environmental accidents in the containment,
- outer containment of non-prestressed reinforced concrete protecting against external natural and human-induced hazards and limiting the annulus space serving to capture radioactive leaks through the inner containment in accidents.

Spent fuel in the reactor building is stored in the spent fuel pool (SFP). The capacity of the spent fuel pool allows storing spent nuclear fuel on racks accumulated over ten years of electricity generation and an additional emergency full core unloading.

Spent fuel is stored in the SFP under the protective water layer with boric acid concentration of at least 17 g/kg. Rack geometry and boron-containing materials maintain subcriticality above 5% (neutron multiplication factor below 0.95) during spent fuel storage and management.

The design of the spent fuel pool represents rectangular reinforced concrete structure with double metal cladding, which retains active fission products generated during different NPP operation modes, and also reduces the ionizing radiation during fuel storage. The water temperature in the pool does not exceed 60^0C during scheduled refueling and emergency core unloading.

Residual heat is transferred from fuel assemblies to the component cooling water system and further to the ultimate heat sink. The design capacity of SFP racks for the 18-month fuel cycle is 601 cells for FAs (438 + 163 = 601 FAs, of which: 438 FAs – refueling of 72…73 FAs every 18 months accumulated over 10 years and 163 FAs – emergency core unloading). SFP also contains 20 cells for hermetic casing designed to store leaking SFAs (PSAR Chapter 9) [4].

The fuel pool is housed within the reactor building between reactor coolant loops. It is connected to the top of the reactor cavity by a refueling channel designed to transport one fuel assembly.

The top elevation of the pool (+ 26.3 m) is governed by the reactor design and protective water level (about 3 meters) above the active part of the spent fuel assembly when it is transported through the refueling channel.

The fuel pool consists of one compartment designed to store SFA, and a cask compartment - SFA cask loading and FFA jacket unloading area.

Fresh nuclear fuel shall be stored and prepared for loading into the reactor in a stand-alone fresh fuel storage facility (FFSF) with a capacity of 381 fresh FAs, including:

- 201 FAs in three racks, 67 FAs each,
- 180 FAs in packages, 2 FAs each.

The fresh fuel storage facility is classified as seismic category I and safety class 1.

The safety of FFSF is ensured by the design and thickness of walls and ceilings, which are designed for SSE. The FFSF building is designed above the flood-free elevation, in the absence of adjacent rooms, from which water or another moderator can enter the storage facility.

Akkuyu NPP design includes dry spent fuel storage (SNFS) to be located on the site.

SNFS is a common-plant building representing a stand-alone building with its own outside entrance road.

The spent nuclear fuel storage facility is classified as seismic category I and class 2N component in terms of its impact on safety.

The storage capacity is 60 casks. Spent fuel arrives at the SNFS in casks. The capacity of each cask is 18 FAs. Total residual heat from SFA in one cask is less than 40 kW. Cask cooling in SNFS is through natural convection, ruling out the increase of fuel cladding temperature above the designed levels (PSAR Chapter 9) [4].

Nuclear and radiation safety during SNF storage in casks is mainly achieved by properties of civil structures of the facility and cask features that do not require forced cooling. Therefore, the loss of off-site power in the SNFS is not considered, because SNF is cooled in casks by natural convection of the outside air, which does not require a power supply.

The NPP site has a common-plant temporary storage facility as part of 00UKS building to store and treat solid and solidified radioactive waste.

Capacity of the storage facility allows storage of solid and solidified low- and medium-level wastes for 10 years of the NPP operation, and high-level waste for 60 years. The general layout allows for expansion of the storage facility for the entire service life of NPP of 60 years.

Safety systems implemented in the Akkuyu NPP design with VVER-1200/B-509 reactor are based on active and passive actuation principles.

The active part of safety systems includes:

- reactor protection and control system (CPS),
- emergency and planned primary circuit and fuel pool cooldown system (EPCS) (active part of ECCS),
- spray system,
- emergency boron injection system,
- SG emergency cooldown system (SG ECS),
- BRU-A system,
- main steam line isolation system (MSIV),
- emergency gas removal system,
- essential-service component cooling system,
- secured cooling water system,
- ventilation and air conditioning systems in the reactor building, safety system and support system rooms,
- MCR/ECR air conditioning and life support systems,
- annulus ventilation and filtration system,
- emergency power supply system (EPS), including emergency diesel generators (SDGS) and batteries, and availability for connecting an alternative diesel generator.

The passive part of safety systems includes:

- passive heat removal system (PHRS),
- stage I hydraulic accumulators (ECCS passive part),
- stage II and III hydraulic accumulators,
- primary circuit overpressure protection system (PRZ PORV),
- secondary circuit overpressure protection system (SG PORV),

- containment hydrogen concentration monitoring and emergency removal system,
- hermetic enclosure system (double containment) with isolation valves.

In addition, the Akkuyu NPP design includes a ex-vessel core melt catching and cooling system (core melt catcher) to manage a severe accident.

Figure 4 shows the process flow diagram of NPP power unit with VVER-1200/B-509 reactor.

Safety systems are arranged in such a way that minimum required portion of pipelines, valves and equipment is located within the containment and does not require repair or maintenance during the unit power operation; the major portion of pipelines, valves and equipment is located outside the containment. Equipment accommodated outside the containment can be accessed, serviced and repaired even during power operation of the reactor.

In addition to the systems directly involved in power generation, the process flow diagram shows safety systems designed to prevent design-basis accidents or mitigate their consequences.

The power unit design includes a number of normal operation systems making up a unified complex and ensuring NPP operation in different conditions. Some of these systems are shown in the process flow diagram of the power unit.

NPP electrical systems consist of power generation and grid distribution systems and auxiliary power supply system.

The generator of each Akkuyu NPP unit is connected to 380 kV gas-insulated switchgear (GIS) through generator circuit-breaker as generator-transformer set with 24/380 kV main step-up transformer to distribute power to 380 kV grid. Auxiliary power for NPP unit start-up/shutdown is supplied from off-site sources. The generator-transformer set shall power two primary auxiliary transformers with a capacity of 2×80 MV·A and a voltage of 24/10.5-10.5 kV. Two standby auxiliary transformers with a capacity of 2×80 MV·A and a voltage of 380/10.5-10.5 kV are installed to backup auxiliary power supply of NPP units. Two auxiliary transformers with a capacity of 63 MV·A and a voltage of 380/10.5-10.5 kV are provided for auxiliary power supply.

Auxiliary power supply systems have primary, standby and emergency power supply sources and 10 kV and 0.4 kV AC and 220 V DC switchgears, and also 110 V DC switchgear for CPS CR drive power supply.

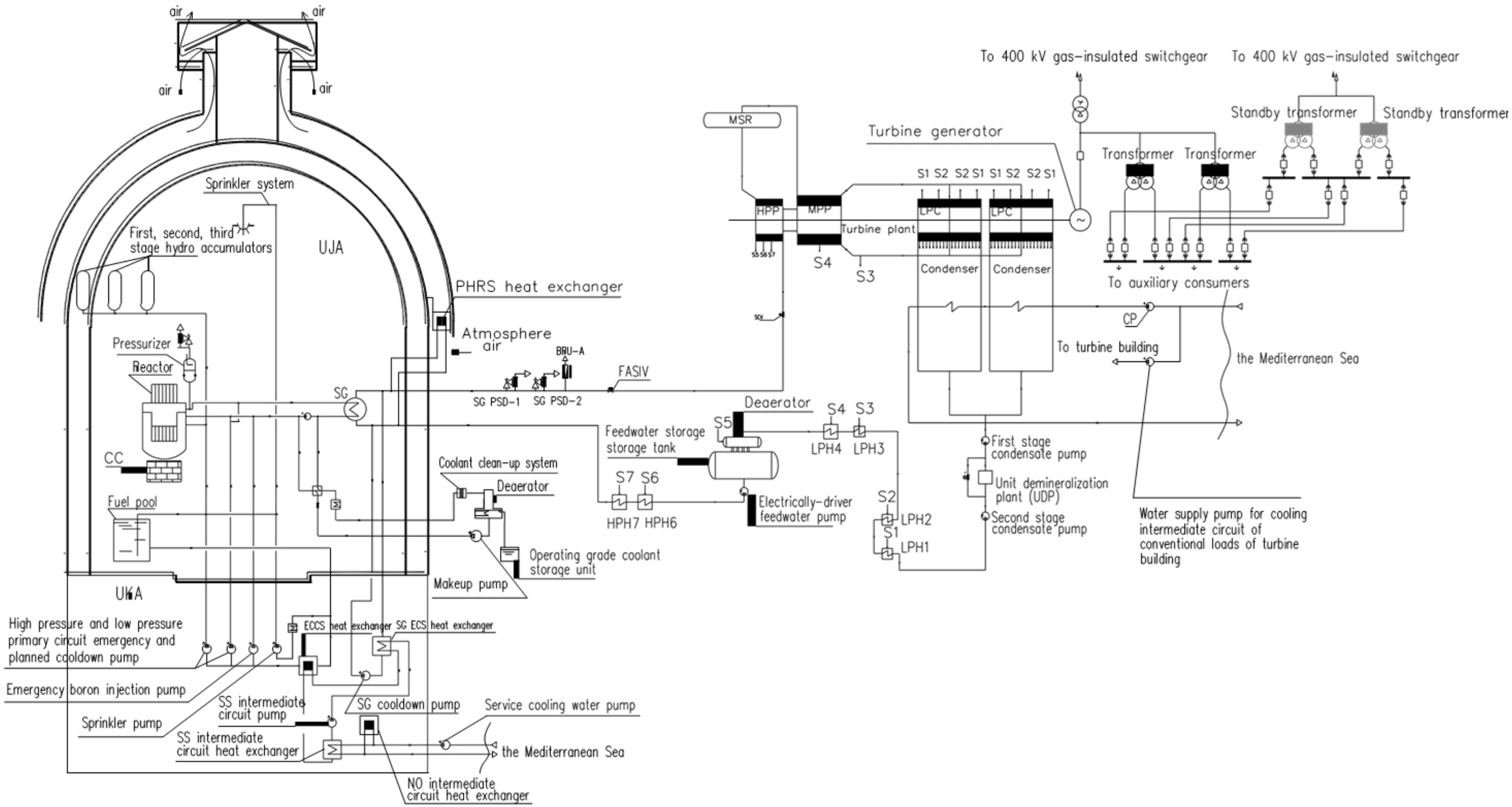

Figure 4 Process flow diagram of NPP power unit with VVER-1200/B-509 reactor

Each Akkuyu NPP unit shall have the following auxiliary power supply systems:

- normal operation power supply system (NOS),
- normal operation reliable power supply system (NO RPS) with one 10 kV main diesel generator station (MDGS),
- emergency power supply system (EPS) with two 10 kV standby diesel generating stations (SDGS).

Akkuyu NPP unit NO power supply system (NOS) powers non-safety systems, including BOP power supply system. The following sections and switchgears are used in the NOS design:

- 10 kV AC for connection of group 2 electric motors with a power over 200 kW and 10/0.4 kV transformers,
- 0.4 kV AC for connection of group 2 electric motors with a power below 200 kW, lighting and other systems,
- 110 V and 220 V DC with batteries to power group 1 DC and AC systems (through converters) requiring uninterrupted 0.4 and 0.22 kV power supply.

The following redundancy is provided to improve the reliability of NO RPS, which powers consumers of normal operation systems important for safety (safety class 3):

- automatic load transfer (ALT) from standby auxiliary transformers (backup power supply sources) when primary power sources are lost on each NOS 10 kV section connected to NO RPS 10 kV section,
- ALT closes jumper switches to power one NO RPS 10 kV section from another section when the former one is de-energized. When two NO RPS 10 kV sections are de-energized simultaneously (loss of off-site power), main diesel generator station (MDGS) is connected to both sections and automatic step-by-step start (ASS) of all group 2 equipment,
- power supply for group 1 equipment requiring uninterrupted 0.4 and 0.22 kV power supply is provided from the relevant NO RPS battery (for 2 hours of discharge).

EPS loads are NPP safety system equipment and devices (safety class 2) that require power supply in all operating modes of the NPP, including emergency shutdown of the reactor plant when off-site power sources are lost. EPS provides independent power to consumers of two trains of safety systems and enables connecting a 0.4 kV alternative air-cooled diesel generator set (ADGS), which powers some equipment required to manage beyond-design-basis accidents. The following redundancy is designed to make EPS more reliable:

- two independent EPS trains according to the principle of redundancy of process safety system equipment,
- each standby diesel generating station (SDGS) and all switchgears of each EPS train are physically separated and electrically independent from each other and their consumers,
- when NOS 10 kV section is de-energized (loss of off-site power) in each train, a standby diesel generating station (SDGS) is started independently and connected to the de-energized section with automatic step-by-step start-up (ASS) of all group 2 consumers of the train allowing for the maximum permissible power supply interruption for the safety systems,
- power supply for group 1 equipment requiring uninterrupted 0.4 and 0.22 kV power supply is provided from two batteries in each EPS train (for 2 and 72 hours of discharge).

AC consumers receive power from inverters connected to the DC board (from 2-hour discharge batteries) in each EPS train. Accident and post-accident monitoring system, valves for connecting HA-2 to the fuel pool and HA-3 to the reactor continue to receive power from batteries designed for 72 hours of discharge. During beyond design basis accident (total loss of power supply from all AC sources and/or loss of ultimate heat sink) after 72 hours, a 0.4 kV standby air-cooled diesel generator is utilized as power supply source in EPS.

1.1.2. Description of the systems for conduction of main safety functions

According to the defense-in-depth principle, Akkuyu NPP design includes safety systems and special engineered safety features to perform the following basic safety functions and sub-functions:

<u>Reactivity Control and Sub-Criticality Maintenance</u>

- reactor scram
- reactor subcriticality maintenance

<u>Heat Removal from the Reactor Core and Fuel Pool</u>

- heat removal from the reactor through primary circuit
- maintaining coolant inventory in the primary circuit
- heat removal from the reactor through secondary circuit
- limiting primary pressure
- limiting secondary pressure

- Heat removal from the fuel pool

Confinement of Radioactive Materials within Established Boundaries

- limiting radioactive releases into the environment from the containment
- limiting radioactive releases from steam generators

Supporting Functions

- emergency power supply
- component cooling and air cooling and ventilation indoors

Active and passive safety systems in the Akkuyu NPP design, which perform main safety functions, are listed in Table 2.

Table 2 List of safety systems that perform safety functions

Safety functions	**Safety systems and additional engineered safety features**		**Brief description**
	Active	**Passive**	
Reactivity Control and Reactor Subcriticality Maintenance			
Reactor scram	JND10-20 emergency boron injection system consisting of two trains (2×100%) in case of ATWS.	Reactor control and protection system (94 CPS control rods). The function is performed if one of the most effective control rod is stuck.	Insertion of absorbing rods into the reactor core. CPS CRs drop into the core (during blackout). If the CPS CRs fail, JND10-20 injects boron solution into the primary circuit to transfer the reactor to a subcritical state.
Reactor subcriticality maintenance	Primary circuit and spent fuel pool emergency and planned cooldown system (EPCS) JNA consisting of two trains (2×100%). JND10-20 emergency boron injection system consisting of two trains (2×100%)	JNG50 emergency core cooling system HA-1 (passive part) consisting of four trains (4×33%). Stage II and III hydraulic accumulators (ECCS passive part). JNG10 consisting of four trains (4×33%).	In case of LOCA, the systems inject boron solution into the primary circuit for emergency cooling of the core and maintaining the reactor subcritical.

Safety functions	Safety systems and additional engineered safety features		Brief description
	Active	**Passive**	
Heat Removal from the Reactor Core and Fuel Pool			
Heat removal from the reactor through primary circuit	Primary circuit and spent fuel pool emergency and planned cooldown system (EPCS) JNA consisting of two trains (2×100%).		In case of large-break LOCA, high and low pressure safety injection pumps are started. In case of small-break LOCA, high pressure safety injection pumps (2×100%) are started.
Maintaining coolant inventory in the primary circuit	Primary circuit and spent fuel pool emergency and planned cooldown system (EPCS) JNA consisting of two trains (2×100%). Water inventory in the fuel pool (1×100%) and in the containment sump.	JNG50 emergency core cooling system HA-1 (passive part) consisting of four trains (4×33%).	Coolant loss compensation and core cooling during design and beyond design basis LOCA. In case of large-break LOCA, high and low pressure safety injection pumps are started. In case of small-break LOCA, high pressure safety injection pumps (2×100%) are started.
		JNG10 second stage hydro accumulators HA-2 (ECCS passive part) consisting of four trains (4×33%).	Coolant loss compensation and core cooling during design basis and beyond design basis accidents with loss of coolant during 24 hours until the system is actuated.

Safety functions	Safety systems and additional engineered safety features		Brief description
	Active	Passive	
		JNG10 third stage hydro accumulators HA-3 (ECCS passive part) consisting of four trains (4×33%).	Coolant loss compensation and core cooling within 72 hours during beyond design basis LOCA when HA-2 supply is finished.
Heat removal from the reactor through secondary circuit	JNB10 SG emergency cooldown system (ECS) consisting of two trains (2×100%).	JNB50 passive heat removal system consisting of four trains (4×33%).	Residual heat removal and primary circuit cooldown during design and beyond design basis accidents.
Limiting primary pressure	JND10-20 emergency boron injection system consisting of two trains (2×100%). KTP emergency gas removal system.	Primary circuit overpressure protection system consisting of three PRZ PORV (3×100%) - one control, two working PORVs.	Primary circuit overpressure protection (PRZ PORV). Primary-to-secondary LOCA depressurization (JND10-20). Primary circuit depressurization to 1 MPa in case of severe accidents (KTP system with PRZ PORV).
Limiting secondary pressure	BRU-A (4×100%) to maintain pressure.	Secondary circuit overpressure protection system consisting of two SG PORV (2×100%) - one control and one working PORV for each SG.	Secondary circuit pressure limitation (BRU-A) and overpressure protection system (SG PORV).

Safety functions	Safety systems and additional engineered safety features		Brief description
	Active	Passive	
Heat removal from the fuel pool	Primary circuit and spent fuel pool emergency and planned cooldown system (EPCS) JNA consisting of two trains (2×100%). JMN spray system consisting of two trains (2×100%).	JNG10 second and third stage hydro accumulators HA-2, HA-3 (ECCS passive part) consisting of four trains (4×33%).	JNA and JMN systems cool or make-up the fuel pool. HA-2, HA-3 shall make-up the fuel pool during boiling (with the failure of active systems).
Confinement of Radioactive Materials Within Established Boundaries			
Limiting radioactive releases into the environment from the containment	JMN spray system consisting of two trains (2×100%).	Containment structures. A system of isolating devices on containment penetrations.	Radioactivity confinement in the inner containment. Pressure reduction in the inner containment during design and beyond design basis LOCA (JMN). Function of iodine binding in the inner containment atmosphere (JMN).
		JMU-JMT containment hydrogen concentration monitoring and emergency removal system, including passive catalytic hydrogen recombiners.	Prevention of explosive mixtures to generate in the inner containment by monitoring and maintaining the volumetric concentration of hydrogen during design basis, beyond design basis and severe accidents.

Safety functions	Safety systems and additional engineered safety features		Brief description
	Active	Passive	
	KLB22 annulus ventilation and filtration system (2×100%).		Collection and filtration of leaks through the inner containment to reduce the release of radioactive substances into the environment.
		JKM core melt catcher (CMC) (1×100%).	Retention, cooling and maintaining subcriticality of the core melt (after failure of the reactor vessel) to provide the containment integrity.
Limiting radioactive releases from steam generators	MSIV and shut-off gate-valve on SG steam lines (2×100%). Two shut-off gate-valves on SG feedwater lines (2×100%).		Isolation of failed SG from the environment and storage of the coolant inventory in the second SG circuit.
Supporting Functions			
Emergency power supply	Emergency power supply system (EPS) with diesel generators (SDGS) consisting of two trains (2×100%) and availability for connecting an alternative diesel generator.	Emergency batteries consisting of two EPS trains (2×100%).	Emergency power supply of safety system equipment.
Equipment and air cooling indoors	KAA10-20 essential service component cooling system (2×100%). PE secured cooling water system (2×100%). Special service water supply equipment (KAA25 alternative component cooling circuit pump and PEC10 mobile pumping unit).		Cooling of safety system equipment.

Safety functions	Safety systems and additional engineered safety features		Brief description
	Active	Passive	
	Indoor ventilation system of reactor building, safety systems and MCR/ECR.		Indoor ventilation system of reactor building and safety systems. MCR/ECR air conditioning and life support.

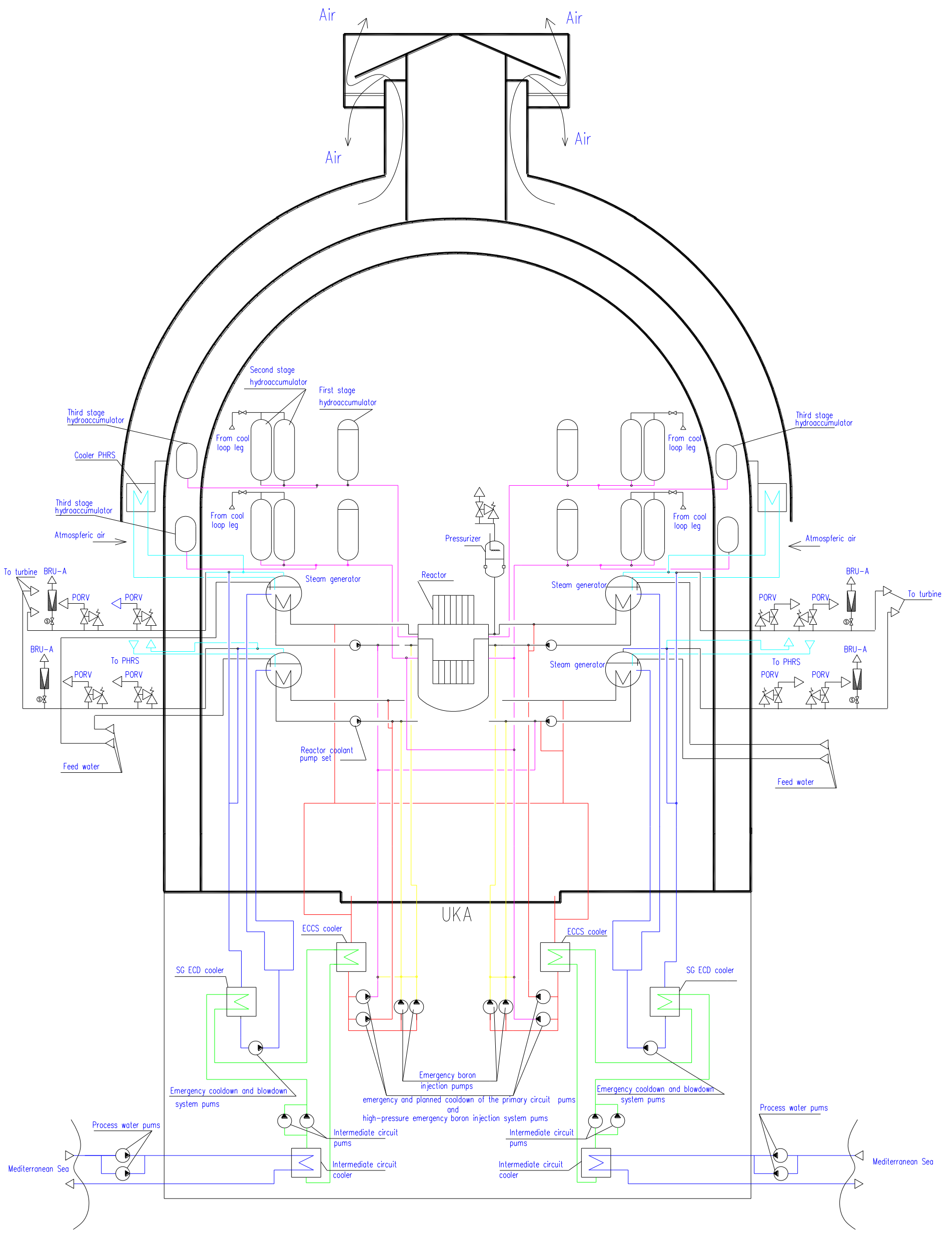

Figure 5 Diagram of primary reactor core cooling systems

The diagram of primary systems cooling the reactor core during accidents is shown in Figure 5.

Description of Safety Systems

Reactor Control and Protection System

The reactor control and protection system (CPS) includes reactor scram system with control rods (absorbing rods and their drives), absorbing/control rods and their drives, and safety control systems that generate reactor scram signals. The reactor scram system is triggered either by control SS or when CPS CR drives are de-energized causing absorbing rods to fall into the core under their own weight to the lower end position.

Emergency and Planned Primary Circuit Cool-Down and Spent Fuel Pool Cooling System

Emergency and planned primary circuit cool-down and spent fuel pool cooling system (JNA) is designed to perform the following functions:

- reactor cooldown to 70 ºC following reactor trip when heat removal via SG secondary circuit becomes inefficient (P_{1c} < 2.1 MPa, T_{1c} < 150 ^{0}C) in all operation modes of the unit (scheduled, maintenance, emergency and refueling shutdown),
- residual heat removal from the reactor core to the component cooling system in all NO modes of the power unit,
- residual heat removal from the fuel pool in all operation modes of the power unit (as a standby system),
- coolant inventory maintenance in case of 'large-break loss of coolant',
- primary circuit emergency makeup during small-break LOCA (DN25-80).

The system consists of two identical and fully independent, physically separated trains. Each train consists of two lines with one LP safety injection pump, one HP safety injection pump and one emergency cooldown heat exchanger. In all modes, heat is removed to the primary ultimate heat sink (Mediterranean Sea) as follows: JNA – essential service component cooling system - secured cooling water system - Mediterranean Sea.

In the event of coolant loss, the system's pumps start automatically to inject into the primary circuit using boric solution stored in the spent fuel storage tank (SFP) above the minimum level required to cool the SFP. When the SFP level drops to the minimum level, the system`s pumps are switched to the sump for recirculation through the containment.

Emergency Boron Injection System

Emergency boron injection system (JND10-20) is designed to perform the following functions:

- injection of boron solution into the pressurizer during primary-to-secondary LOCA to reduce pressure,
- maintaining the core subcritical in case of failure of control and protection system (CPS) of the reactor.

The system consists of two component-identical and fully independent trains. System train consists of two legs sparing each other. Each train leg has 50% capacity and injects boron solution into its 'cold' part of the RCC loop or into the pressurizer through a header common for both trains. Suction pipelines of emergency boron injection system pumps are connected to DN500 pipelines of relevant trains of the emergency and planned primary circuit and fuel pool cooldown system downstream of the heat-exchanger. Water volume above the fuel pool level required for fuel storage serves as boron solution (with concentration of 17-25 g/dm^3) supply.

Emergency Steam Generator Cooldown System

Emergency SG cooldown system (JNB10) is designed to:

- remove residual heat from the reactor core and cool down the reactor in accidents associated with the loss of off-site power or loss of normal heat removal through the SG secondary circuit, including leaks of SG steam lines and feedwater lines,
- remove residual heat from the reactor core and cool down the reactor in accidents associated with the primary circuit depressurization, including the reactor coolant line break and primary-to-secondary loss of coolant.

The emergency SG cooldown system consists of two trains, each of them connected to two steam generators. The system trains are physically separated and fully independent. Each train of the system consists of pipelines, emergency cooldown heat exchanger (process condenser cooled by water of KAA10-20 component cooling system), two emergency cooldown pumps and condensate return pipelines in two steam generators. Emergency cooldown of each SG is made through a closed circuit.

BRU-A System

BRU-A (fast-acting relief valve for steam discharge into atmosphere) is designed to dump excessive steam to avoid actuation of SG PORVs under load shedding and loss of off-site power, as well as to dump steam into the atmosphere during SG pressure maintenance.

Each steam line from SG is equipped with a fast-acting steam dump valve to atmosphere (BRU-A) and shut-off valve upstream of it.

Containment Spray System

The spray system (JMN) is designed to perform the following functions:

- limit pressure by spraying and recirculation after accidents to maintain the pressure in the containment within design limits during design-basis accidents,
- remove fission products from containment, thereby reducing the total amount of fission products in the air to prevent their release to environment,
- remove residual heat from the fuel pool in all operation modes of the power unit (as the main system during normal operation), as well as heat removal (together with JNA system) during scheduled core unloading.

The system consists of two identical and fully independent trains. One train of the system is capable to fulfill the function of temperature and pressure reduction inside the containment. In case of LOCA, JMN system is actuated to inject into the containment. Cooling or make-up of the fuel pool during accidents is provided by one train of JNA system (standby function) or one train of JMN system.

Emergency Gas Removal System

The emergency gas removal system (KTP) is designed to remove uncondensed gases escaping from coolant in the upper points of the primary circuit equipment to prevent explosive concentrations and explosions of hydrogen, and also to prevent loss of natural circulation of the reactor coolant in primary circuit. Such points are SG headers, reactor top unit, pressurizer. The system consists of pipelines and two shut-off valves installed in parallel in each line from reactor top unit, from the top of pressurizer, from steam generator headers and discharge lines to relief tank.

During reactor power operation the system does not function and is on standby. Shut-off valves are closed and periodically checked. The system may be used for air removal during primary circuit filling with water.

In case of loss of coolant, to improve primary circuit cooldown, the operator can open valves on the system lines connecting the reactor and SG air vents with the PRZ steam space and, if required, vent gas-steam mixture into the pressure relief tank by opening valves on the line connecting PRZ and the pressure relief tank.

During severe core meltdown accidents, the operator opens system shut-off valves together with PRZ PORVs to vent steam-gas mixture from reactor pressure vessel, PRZ and SG headers to reduce the primary circuit pressure to 1 MPa.

Emergency Power Supply System

Each of two EPS trains consists of diesel-generators (SDGS), batteries, uninterruptible power supply units (UPS), transformers, 10 kV and 0.4 kV switchgears, 220 V and 110 V DC boards.

Main Steam Line Isolation System

The system is designed for fast and reliable isolation of steam generators (using MSIV and shut-off valves on steam pipelines) in accidents:

- to prevent the reactor uncontrolled cool-down, and to enable the operation of the emergency heat removal system in case of steam line or feed water line breaks,
- to prevent radioactivity release into the environment in case of primary-to-secondary LOCA.

The system is also designed to prevent moisture reflux in the turbine in case of SG level rise due to feedwater flow controller malfunction and to prevent the loss of SG secondary coolant due to abnormal feedwater flow rate.

Component Cooling System

The component cooling system (KAA10-20), including KAA25AP001 alternative component cooling pump, is designed to remove residual heat from reactor components and safety-related systems to secured cooling water system (PE) in all operation modes of the unit, including emergencies. Besides, this system serves as a barrier preventing radioactivity releases to environment.

Secured Cooling Water System

The system (PE), including PEC10AP001 mobile pumping unit, is designed to remove heat from KAA10-20 to the primary ultimate heat sink (sea water) in all operation modes of the power unit, including emergencies.

Ventilation and Air Conditioning Support Systems

Ventilation, cooling and air conditioning systems are designed to:

- cool air in safety system rooms within the prescribed limits during operation of process systems,
- maintain standard air temperature in penetrations and ventilation chambers,
- air condition and life support the MCR/ECR,
- remove heat and maintain normal environment conditions in MCS rooms and auxiliary switchgear rooms.

Fire-Fighting Systems

Automatic fire-fighting systems are also designed for fire protection of cable floors under MCR and ECR. The rooms accommodate cables of different safety system trains, which are not separated from each other with fire barriers or safe clearances. This makes it impossible to contain fire in cable floors within one train. As a result, automatic fire-fighting systems are designed as supporting safety systems: double-train design, 100% each, thus fulfilling the single-failure principle. The systems are designed to function under extreme external hazards (SSE, hurricane, flooding etc.) as well as during design-basis accidents.

Primary Circuit Overpressure Protection

Primary circuit overpressure protection is achieved by the actuation of pressurizer relief valves (PRZ PORV) upon reaching of two pressure setpoints on discharge lines from PRZ to relief tank.

The number of relief valves to be installed is governed by the principle of N+1 (N is a number of relief valves controlling the flow rate required for depressurization) ensuring that the primary circuit pressure will not exceed operating pressure by more than 15%, including failure of one relief valve. The capacity of each pressurizer relief valve is at least 50 kg/sec (saturated steam).

Under low temperature (T_{1c}<100^0C), primary circuit is protected against cold overpressure by pressurizer relief valves that open automatically when the primary circuit temperature drops below 100^0C and primary circuit pressure above 3.7 MPa.

Each train of the residual heat removal and cooldown system is equipped with two relief valves. The relief valves of the residual heat removal and cooldown system maintain the primary circuit pressure below 2.2 MPa in combination with safety interlocks of associated systems and equipment.

Secondary Circuit Overpressure Protection

Secondary circuit overpressure protection is designed to prevent overpressure in steam generators and main steam lines above the specified value, i.e. not exceeding 15 % of steam generator operating pressure.

The system consists of two pilot-operated relief valves (SG PORV) to be installed in each main steam line downstream of the SG and actuated when two pressure set-points are reached. It comprises main and pilot valves. The main valve is controlled by work medium from the pilot valve. Usually the pilot valve is actuated by electromagnetic actuator in response to signals from pressure sensors. In case of power loss, the pilot valve is actuated as direct-action spring type safety valve.

Passive Emergency Core Cooling System

JNG50-80 passive emergency core cooling system (stage 1 hydro accumulators HA-1) is designed to inject boric acid solution with concentration of not less than 17 g/dm^3 into the reactor core for cooling during loss of coolant accidents, when the pressure in the reactor coolant system drops below 5.9 MPa.

The system consists of four independent trains with capacity of 4×33%. One hydro accumulator HA-1 is installed in each of the trains. Each hydro accumulator is connected to the reactor by a separate line with two check valves: two hydro accumulators are connected to the reactor inlet chamber with two other hydro accumulators connected to the reactor outlet chamber.

Each hydro accumulator with the total volume of $60m^3$ is filled with boric acid solution (H3BO3) with a concentration of 17 g/dm^3, volume of 50 m^3, temperature of 60^0C. The pressure of 5.9 MPa inside the hydro accumulators is maintained by nitrogen blanket. In order to prevent nitrogen release into the main cooling pipeline during injection, each pipe line has two fast-acting valves installed in series that are automatically closed at critically low level in the relevant hydro accumulator.

Stage 2 and 3 Hydro Accumulators

Stage 2 and 3 hydro accumulators JNG10 (HA-2 and HA-3) are designed to perform the following functions:

- boron solution supply from second stage hydro accumulators (HA-2) to maintain the coolant level in the core in case of LOCA, when the primary pressure drops below 1.5 MPa,
- boron solution supply from third stage accumulators (HA-3) to maintain coolant level in the reactor core during beyond design basis accidents with loss of coolant and the failure of active safety systems after the depletion of boron solution supply in the second stage accumulators (after 24 hours),
- boron solution supply to makeup the spent fuel pool from HA-2 during beyond design basis accidents with complete loss of all AC power supply and/or heat removal to the ultimate heat sink,
- storage of boric acid inventory to fill fuel pool compartments when the unit is shut down for refueling.

Total coolant inventory in HA-2 hydro accumulators is 960 m^3, which is the volume required in case of dependent failure of one train. Water inventory of second stage PHRS accumulators is sufficient to remove residual heat during 24 hours.

HA-2 hydro accumulators are actuated in passive manner ensuring residual heat removal in the event of loss-of-coolant accidents when the primary pressure drops below 1.5 MPa.

In case of loss-of-coolant accidents, JNG10 system supplies boric acid solution from two trains into the reactor inlet chamber and from the remaining two trains to the reactor outlet chamber through the relevant pipelines of first stage accumulators (HA-1).

Total coolant inventory in HA-3 hydro accumulators is 720 m^3, which is the volume required in case of dependent failure of one train. Water inventory of second stage PHRS accumulators is sufficient to remove residual heat for at least 72 hours from the accident initiation.

During loss-of-coolant accidents, as the primary circuit pressure reduces, HA-1 hydro accumulators first start injecting (at pressure below 5.9 MPa) and then HA-2 hydro accumulators start injecting (at pressure below 1.5 MPa) to flood the core under hydrostatic pressure. In case of beyond-design-basis accident with loss of all AC power sources, HA-2 hydro accumulators supply borated water to the reactor to remove residual heat in saturated conditions. Water stored inside the hydro accumulators allows core cooling for 24 hours. Once the water supply is depleted in HA-2, injection starts from HA-3 hydro accumulators (from 24 to 72 hours from the accident initiation).

<u>Passive Heat Removal System</u>

SG passive heat removal system (JNB50) is a protective safety system based on the passive action principle that removes residual heat from the reactor core through secondary circuit.

The system functions during design basis accidents and beyond design basis accidents requiring heat removal from the reactor plant. The system performs passive core heat removal from SG secondary circuit to the atmosphere as alternative ultimate heat sink. In case of loss of coolant accidents, the system removes residual heat simultaneously with safety injection into the primary circuit from HA-2 (at least for 24 hours) and HA-3 hydro accumulators (at least for 72 hours).

PHRS is able to provide natural circulation even with coolant in SG manifolds due to the condensation of steam generated in the primary circuit.

The system consists of four identical and fully independent trains based on natural air circulation. Each train comprises two heat exchanging modules, pipelines of steam condensing circuit with valves, supply and exhaust air ducts, air flaps and control devices.

SG steam is fed through a pipeline to the PHRS heat exchanger where it condensates by atmospheric air. Due to natural draft, cooling atmospheric air from outside the containment passes via baskets and enters common annular header. Air reaches heat exchanging modules through separate air ducts. Cooling air picks heat from steam in heat exchangers and flows out through air ducts, which are joined in a common header.

The system is automatically actuated due to de-energizing of electromagnets holding air locks in the closed position in case of EPS 0.4 kV section de-energizing, loss of coolant in any of hot legs with decrease of departure from boiling to 8 °C or a failure of relevant SG ECS train (JNB10).

Containment Hydrogen Monitoring and Emergency Removal System

The containment hydrogen monitoring and emergency removal system (JMU-JMT) ensuring hydrogen explosion safety in the containment employs passive catalytic hydrogen recombiners located in the area of probable hydrogen accumulation; thus, the system is able to perform its assigned function under any conditions of the containment atmosphere.

JMU-JMT system is designed for operation during LOCA. During normal operation, the emergency containment hydrogen removal system does not function and is on standby.

In case of design basis accidents and beyond design basis accidents, the containment hydrogen monitoring and emergency removal system prevents generation of explosive hydrogen concentrations above the design limits that lead to hydrogen burning.

Above listed safety systems and their components perform their functions under all external hazards covered in the design.

Autonomy

A provision is made in the design for sufficient inventory of process media, diesel fuel, stored energy, etc. required to maintain the power units in the safe state autonomously.

Water inventory in HA-2 and HA-3 allows safety functions to be performed for at least 72 hours, thereby cooling fuel in the containment and/or fuel pool. During beyond-vessel stage of severe accident, the water supply from HA-2 and HA-3 coming through the leak to the emergency sump removes heat from core melt in the corium retention and cooling device (catcher).

In case of AOOs associated with the loss of off-site power or in case of accidents occurring during NPP blackout, safety systems that are required to bring and maintain NPP units in a safe state shall be powered from diesel generators of the emergency power supply system. Diesel fuel inventory allows diesel generators to operate for at least 72 hours with the possibility of replenishing this inventory either from a centralized diesel fuel warehouse located on the NPP site and designed for the operation of diesel generators for another four days, or delivered from outside. Oil inventory in the diesel generator feeder tank is sufficient for the diesel generator to operate for 15 days. In the future, oil shall be delivered by truck tanks.

Therefore, off-site power supply, delivery of process media, fuels and lubricants, etc. or actuation of additional (alternative) BDBA management features are not required for at least 72 hours from the accident initiation.

Operation time and efficiency of the passive safety systems allow safety functions to be performed for at least 72 hours, including total blackout.

Besides, the dedicated equipment is designed to manage beyond design basis accidents beyond 72 hours after the accident initiation, including extreme external impacts.

During beyond design basis accident (associated with coolant loss and a complete loss of all AC power supply and/or loss of design heat removal to the ultimate heat sink) beyond 72 hours, BDBA management is provided by:

- connection of additional process equipment and mobile equipment in train 1 of SS,
- partial use of some components of safety systems specifically designed to manage DBA,
- power supply from alternative air-cooled DG included in the design.

The list of equipment used to manage BDBA:

- alternative air-cooled diesel generator,
- emergency boron injection pumps of 10JND system train 1 (2 pcs),
- spray pump of 10JMN system train 1,
- special service water supply equipment (KAA25 alternative component cooling pump and PEC10 mobile pumping unit),

- motor-operated valves of 10JNA train 1 (in suction pipelines from emergency containment sump and spent fuel pool), 10JMN, 10JND, 10KAA, 10JNG10 (on the pipelines connecting to 10JMN system), isolating valves of 10SCC system,
- accident and post-accident monitoring systems,
- ventilation systems for cooling electrical equipment and MCS equipment.

1.2. Significant differences between units

All four proposed Akkuyu NPP units are based on the same design, and there are no significant differences concerning nuclear safety.

1.3. Use of PSA as part of the safety assessment

Probabilistic safety analysis is included in the package of documents that substantiate nuclear and radiation safety and submitted for the NPP construction license application.

PSA approach for external initiating events is based on a sequential screening analysis, which allows us to concentrate efforts only on those initiating events that are potentially significant in terms of severe damage of the core and fuel in the fuel pool. The approach is based on gradual deepening and detailing of the analysis and consistent conservatism decrease as the significance of initiating events is identified.

Level 1 (PSA-1) and level 2 (PSA-2) probabilistic safety analysis was made at the design stage to assess Akkuyu NPP safety.

PSA was developed in accordance with recommendations of Russian regulations and taking into account IAEA recommendations, PSA experience and operating experience for NPPs with VVER reactors.

Fuel in the reactor core and fuel pool, including situations when all the core fuel is unloaded to the fuel pool, was considered as sources of radioactivity. The development considered internal initiating events as well as initiating events caused by on-site and off-site hazards typical for the NPP site, in all operational states of the unit.

Fuel damage frequency analysis for reactor core and fuel in the fuel pool was used to evaluate the vulnerability of NPP unit to initiating events and develop recommendations to improve the safety of NPP unit taking into account the results of uncertainty and sensitivity analysis.

At the same time, the following total fuel damage frequencies were estimated for all types of internal and external events: $5.15 \cdot 10^{-6}$ 1/year for the reactor core and $1.41 \cdot 10^{-6}$ 1/year for the fuel pool (PSAR Chapter 1) [5].

Level 2 PSA is developed to estimate the total frequency (probability) of maximum emergency release (MER) exceeding the established limits (for severe accidents with fuel melting in the reactor core and fuel pool).

The results of Level 2 are preliminary at the current stage of Akkuyu NPP project. This is due to conservative assumptions being used in Level 2 due to unavailability of detailed design information. This approach to the PSA gave over-conservative results regarding Akkuyu NPP safety. However, realistic approach is impossible at this stage of the design. But at the same time, the results obtained allow to determine dominant accident sequences that make the greatest contribution to the MER frequency and analyze their main causes.

2. EARTHQUAKES

Turkey lies along the Eastern Mediterranean sector of the seismically active and tectonically complex Alpine-Himalayan orogenic belt. The active tectonics of Turkey is the consequence of the convergence between the African, Arabian plates in the south and the Eurasian plate in the North which requires special attention to seismic issues in NPP projects. Turkish regulatory requirements address this by including additional specific requirements for seismicity of the site and the nuclear power plant. Akkuyu Nuclear Power Plant Site is selected because of the low seismic activities of the region. Site is located in one of the most favorable seismic zones of Turkey. Seismic zones are presented in Figure 6.

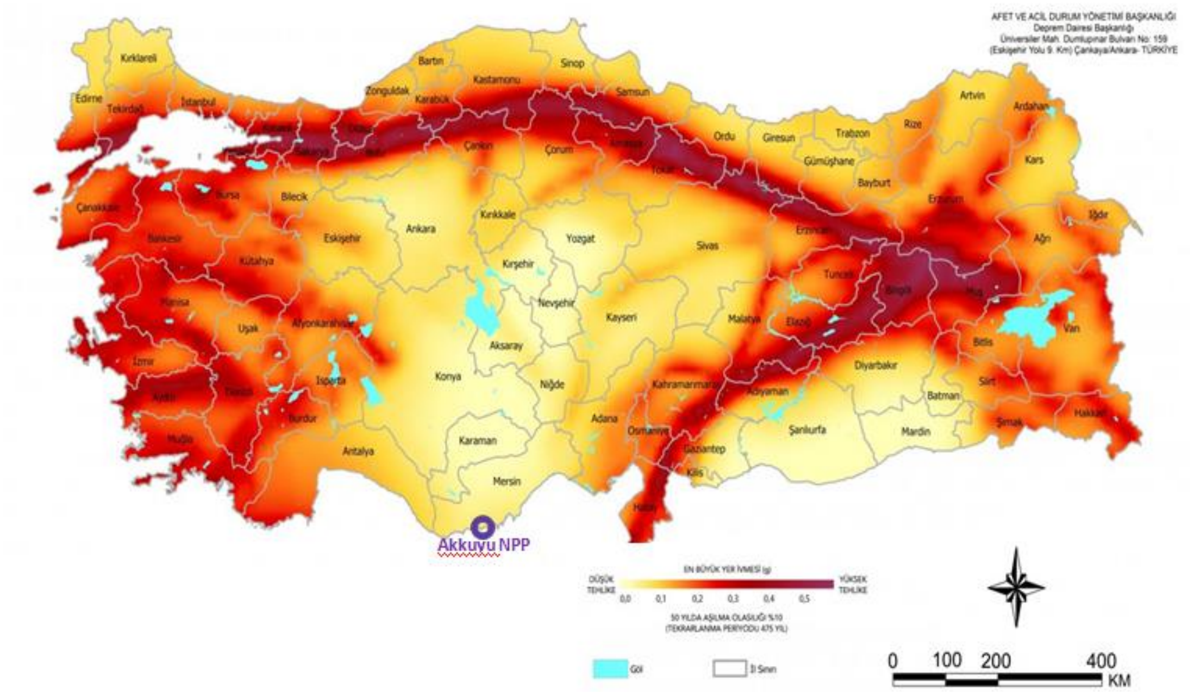

Figure 6 Seismic zones in Turkey

A plenty of seismic surveys and calculations have been carried out at Akkuyu site in the past since 1976. Early seismic studies were performed under the auspices of Turkish Electricity Authority. Seismological studies were performed by the Geology Department of Mining Research and Exploration Institute, Engineering Seismological Research Institute of Middle East Technical University (METU), Geophysics Department of Istanbul Technical University (ITU), and also by a specially organized Engineering-Consulting Consortium ENG (Emch-Berger, Basler und Hoffman).

Within the scope of the Akkuyu NPP project, these studies have been reviewed and updated at the stage of engineering survey in 2011, the historical seismicity data for the area were obtained by the following four research groups: ENVY/BU KOERI (Turkey);

WorleyParsons Nuclear Services JSC (USA); Paul C. Rizzo Associates, Inc. (USA); Institute of the Earth Physics of the Russian Academy of Sciences – IEP RAS (Russian Federation).

Historical data has been compiled and a new earthquake catalogue has been formed. Since new technical capabilities became available since the time of seismic site surveys. A sufficient number of seismic stations for various purposes were established to become possible to extend and update seismological database. A system of near-region instrumental observations (up to 40-50 km) was set up by KOERI (Kandilli Observatory and Earthquake Research Institute). The local seismic network consists of nine strong-motion and six weak-motion seismic stations.

2.1. Design basis

2.1.1. Earthquake against which the plant is designed

The seismic hazard assessment made in the Site Parameters Report of Akkuyu NPP site is based on several independently developed earthquake source zoning models. Probabilistic and deterministic seismic hazard analysis (PSHA and DSHA) was made for each zoning model.

The seismic design of Akkuyu NPP buildings and structures is developed in accordance with the requirements of NP-031-01 ‘Seismic Design Standards for Nuclear Power Plants’ included in the Licensing Basis [3].

Seismic analysis was based on conservative approaches in accordance with the requirements of NP-031-01 [10], MP 1.5.2.05.999.0027-2011 ‘Seismic Design Standards for Nuclear Power Plants. Guidelines’ [11] and MR 1.5.2.05.999.0025-2011 ‘Seismic Analysis and Design of Nuclear Power Plants. Guidelines’ [12], which are also included in the Licensing Basis [3].

According to Russian requirements for new NPP designs, seismic peak ground acceleration (PGA) for the safe shutdown earthquake (SSE) shall be assumed at least 0.10 g regardless of the site seismicity. Seismic accelerations for operating basis earthquakes (OBE) shall be assumed at least 0.05g. According to Turkish regulations, S1 design-basis peak ground acceleration shall be half the minimum S2 design-basis acceleration. The minimum acceptable value for S2 shall be 0.15g. These figures once again confirm the strictness of the Turkish national requirements for seismic safety.

Design parameters of SSE (10.000 years return period) and OBE level were determined by probabilistic and deterministic approaches. Akkuyu NPP structures, sysytems and components are designed based on the following peak ground accelerations:

- horizontal acceleration is 0.388 g during SSE
- horizontal acceleration is 0.194 g during OBE
- vertical acceleration is 0.295 g during SSE

- vertical acceleration is 0.147 g during OBE

Assessments made for Akkuyu NPP site are used to establish response spectra and peak ground accelerations for SSE and OBE with an equal probability of not exceeding the spectral amplitudes. Underlying 30-meter soil strata with the average shear wave propagation velocity Vs_{30} of 1,138 m/s is assumed as the datum for the entire site, to which the initial response spectra are referenced. The exact depth mark (ordinance datum) of this surface is calculated for each seismic section at the specified site stations.

The background seismicity events in seismic source models are postulated to occur on, or outside a circle with radius at least 5 km around the site. Such and even larger distances are proved to be safe on the basis of detailed site vicinity investigations (fault displacement hazard analysis) with respect to potential for surface or near surface tectonic deformations in the site vicinity including the offshore area.

In light of Fukushima NPP accident in 2011, seismic category I equipment is additionally tested for earthquakes with an intensity 40% higher than the SSE. 1.4 SSE seismic analysis is made using realistic approaches. In the event of such impact, the NPP is transferred to safe state, the release of radioactive substances into the environment is prevented, but the NPP may be lost for further commercial operation.

The analysis showed that the methods and approaches to assessment of design basis seismic impacts and beyond design basis impacts were defined for Akkuyu NPP in accordance with Licensing Basis [3].

2.1.2. Provisions to protect the plants against the design basis earthquake

The seismic resistance of reactor building containment components has been enhanced (as compared to the reference power unit of Novovoronezh 2 NPP) in accordance with the design requirements to improve the earthquake stability of Akkuyu NPP.

Building materials and prestressing elements used for the containment structure in the Akkuyu NPP design have higher strength properties (comparing to the reference power unit of Novovoronezh 2 NPP). At the same time the containment lifetime is increased from 60 to 100 years with a corresponding increase in the number of leak-tightness tests. Physical separation principle is employed in the design of systems and components as one of the measures to protect NPP against earthquakes.

Layout solutions are developed for Akkuyu NPP so as to physically separate systems of various seismic categories to rule out the effect of lower category equipment on higher category equipment under seismic impacts.

If equipment and pipelines of different seismic categories are accommodated in the same room, they are spaced, or equipment and pipelines of lower seismic category are additionally detached to achieve structural integrity and stability.

All the equipment, the failure of which may affect the operation of safety-important equipment, has either seismic category I or physically separated from safety-related equipment in the NPP design. Thus, protection against secondary effects of earthquake is achieved. This means that safety-related equipment will not fail during an earthquake up to SSE level inclusive.

One of the following approaches is used to analyze the interface of seismic category I components with lower category II and III components:

- checking the robustness (operability) of higher seismic category component under loads caused by the failure of lower category component,
- lower seismic category components are designed for all external loads and impacts to be covered in the design of an adjoining higher category component. Contact interaction of structures not belonging to seismic category I with category I structures as well as category I structures is excluded by aseismic joints and layout solutions. The adequacy of aseismic joint width is verified by calculations.

These measures allow preventing damage to safety-related components from indirect seismic impacts.

Akkuyu NPP PSAR [4] demonstrates that the plant is resistant to seismic impacts of OBE and SSE levels. The main safety functions are performed, safety systems, structures and components important for safety remain functional. Moreover, the NPP is designed to accommodate seismic loads combined with design-basis accidents.

2.1.3. Compliance of the plants with its current licensing basis

Akkuyu NPP is designed in accordance with regulations included in the Akkuyu NPP Licensing Basis [3]. An analysis of Licensing Basis of Akkuyu NPP Design for Compliance with IAEA and Turkish Regulatory Requirements was done as part of the licensing process. It demonstrated that the application of Russian regulations ensures acceptable compliance with the national requirements and IAEA safety standards, including seismic safety standards.

IAEA standards use two levels SL-1 and SL-2 (SSG-9 [9]) to evaluate seismic impacts. SL-2 level is derived to provide the most stringent safety requirements of the plant's design while SL-1 is usually associated with operational requirements only.

Similarly, Turkish Regulation has defined [7] two levels: S1 for operational basis earthquake (OBE) and S2 for safe shutdown earthquake (SSE).

Turkish Regulation on NPP Sites [7] gives a definition of the design levels S1 and S2:

'S1: Maximum earthquake ground motion level which reasonably can be expected to be experienced at the site area once during the operating life of the plant and carried on normal operation,

S2: Earthquake ground motion level that corresponds directly safety limits and maximum earthquake potential that can affect the site,

Levels S1 and S2 are determined based on seismotectonic considerations, seismicity and knowledge of the characteristics of site area geology and soil materials. The maximum earthquake potential in the seismotectonic province of the site should be assumed to occur at the site. Maximum earthquake potential provinces adjacent to the province of the site should be assumed to occur at the locations on the province boundaries nearest to the site. An appropriate attenuation function should be used to determine the ground motion intensity which these earthquakes would cause at the site'.

Turkish Regulation requires S1 to be determined as minimum half of S2. The minimum acceptable level for S2 is 0.15g. For design basis earthquake there are OBE and SSE seismic hazard levels defined in Russian codes and standards, which correspond to S1 and S2 (SL1 and SL2).

In accordance with these requirements, a sufficient number of seismic investigations of the region, area and site were done during Akkuyu NPP site and construction licensing.

According to Decree on Licensing of Nuclear Installations [1]:

'The applicant, in order to obtain a site license from the Authority has to submit a site report comprising information about the evaluation of the site with respect to natural phenomena such as earthquakes, floods and storms, and also their secondary effects'.

There are plenty of site survey reports concerning these issues.

The background seismicity events in seismic source models are postulated to occur on, or outside a circle with radius at least 5 km around the site. Such and even larger distances are proved to be safe on the basis of detailed site vicinity investigations (fault displacement hazard analysis) with respect to potential for surface or near surface tectonic deformations in the site vicinity including the offshore area.

Generalized three-component accelerograms were produced for SSE and OBE levels compatible with the initial response spectra that meet the requirements of Russian regulations NP-031-01 [10], NP-006-98 [16], and do not contradict the requirements of the Turkish regulation [7] and IAEA recommendations (SSG-9) [9].

The seismic hazard assessment on the basis of Probabilistic Seismic Hazard analysis (PSHA) and Deterministic Seismic Hazard Analysis (DSHA) which have been made for each zoning models have been evaluated. The final results of the three independent PSHA and DSHA studies differed from each other by 10% or less.

In conclusion it shall be noted that this stress test is developed assuming that the actions for maintaining the availability of equipment that transfer the reactor into safe state after earthquake will be developed within a set of operating procedures. Besides, the procedures for internal supervision over compliance with the Turkish safety requirements will be developed and put in force during the NPP commissioning according to the requirements of the project licensing basis.

2.2. Evaluation of safety margins

IAEA Safety Guide SSS-9 [9] on seismic hazards emphasizes on using both deterministic and probabilistic analyses when evaluating seismic hazard and ground motion for a nuclear installations:

"*Section 5.1- The ground motion hazard should preferably be evaluated by using both probabilistic and deterministic methods of seismic hazard analysis. When both deterministic and probabilistic results are obtained, deterministic assessments can be used as a check against probabilistic assessments in terms of the reasonableness of the results, particularly when small annual rates of exceedance are considered. The probabilistic results allow deterministic values to be evaluated within a probabilistic framework so that the annual rate of exceedance of each spectral ordinate of the deterministic response spectrum is known".*

SSG-9 Section 7.1- The assessment of seismic hazard by deterministic methods should include:

"For each seismogenic structure, the maximum potential magnitude should be assumed to occur at the point of the structure closest to the site area of the nuclear power plant, with account taken of the physical dimensions of the seismic source. When the site is within the boundaries of a seismogenic structure, the maximum potential magnitude should be assumed to occur beneath the site. In this case, special care should be taken to demonstrate that the seismogenic structure is not capable."

However, IAEA guides on both DSHA and PSHA do provide any specific recommendations how the results of these assessments shall be dealt with. This is related to the fact that some countries have their regulatory regimes based on deterministic approach while the other countries use more risk-informed approach.

The main difference between PSHA and DSHA is that PSHA uses seismic activity, while DSHA uses only maximum earthquake magnitudes. For sites with expected very low activity, DSHA produces higher results than probabilistic hazard values associated with the annual probability of exceedance of 10^{-4}. It is supposed to be the inverse for sites associated with high activity (for instance, subduction belt zones and active fault zones).

PSHA and DSHA produced similar results for Akkuyu NPP seismic assessment. Such results were not unexpected since all necessary requirements for surveys and calculations were met and considering the seismotectonic environment of Akkuyu NPP [5].

The adequacy of design provisions was assessed on the basis of standard safety margins. Safety margin is the ratio between design value and maximum permissible value. Available safety margins for safety important equipment were determined by conservative approach in accordance with equipment requirements.

The assessment was done by comparing the bearing capacity of structures and components against the maximum design seismic impact. Safety margin may be determined as a ratio of the actual bearing capacity of structural members under probable impact to the ground response with the safe shutdown earthquake covered by the design of structures, systems and components.

Seismic impacts are presented as three-component accelerograms defined on the free surface of the site. Seismic loads on the NPP buildings, structures and components are determined using detailed three dimensional finite element models and considering the soil-structure interaction.

Seismic category I buildings and structures are designed for SSE seismic impact assuming damping and effective stiffness values compatible with those defined in ASCE 4-98 [14] and ASCE 43-05 [15] documents.

Probabilistic seismic hazard is defined by the probability of exceedance of various strong ground motions at the site within a specified period of time.

Identification and parameterization of seismic source zoning models (ESO zones) include epistemic and aleatory uncertainties. Epistemic uncertainties associated with failures, maximum magnitude and magnitude recurrence as well as aleatory uncertainties associated with the hypocentral depth are addressed in the logic tree of each model.

Probabilistic assessment of seismic impacts using damage method is used to determine the seismic vulnerability of NPP components during ground motions.

Two seismic hazard levels are considered in PSHA. Peak horizontal ground acceleration (PGA) is 0.3875 at S2 hazard level. Peak horizontal ground acceleration (PGA) is 0.3875g (S2) for the safe shutdown earthquake (SSE) and 0.194g (S1) for operating basis earthquake (OBE).

The following types of failures of NPP components are considered [6]:

- functional failures related to elasticity (spurious actuation of relays and breakers, seizures of drives, elastic instability of vessel walls, excessive bend of fan blades, excessive mutual displacement of supports located on a building structure),
- brittle failures (anchor bolt and pin break-away and shear, welded seam break),
- failures induced by limit states of elasticity (plastic moment in pipeline, casing sections, plastic deformations of cable trays and racks).

PSA considers seismic impacts only for those NPP system structures and components, failures of which under seismic impacts may lead to the damage of equipment of systems that transfer the NPP into safe state.

Thresholds (safety margins), at which with 0.05 conditional probability of failure, structure or element strength may be lost with 95% confidence probability HCLPF (High Confidence of Low Probability of Failure) (PSA Chapter 16) [17], were calculated to assess the resistance of safety system structures and components to seismic impact. This value is compared with design SSE PGA increased by 40%, i.e. with value equal to 1.4× 0.388=0.54g. If the design threshold is not exceeded, 0.54g ensures operability of the systems retaining the NPP in the safe state and preventing radioactive releases beyond the containment. If this level of seismic impact is further increased, one should expect significant plastic deformation of inner containment of the reactor building with the sharp increase in the probability of radioactive releases to environment.

The structural strength of seismic category I buildings and structures, equipment, process and other lines was separately tested for beyond-design basis seismic impact exceeding SSE by 40%. In the event of seismic impact of 1.4 SSE level, the NPP is transferred to safe state, radioactive releases are prevented. The possibility of further commercial operation may be lost.

The assessment results demonstrate that this value does not exceed the robustness threshold (with assumed HCLPF) for the systems, structures and equipment that ensure the safe shutdown of the RP. Main systems, structures and equipment have sufficient margins to withstand 1.4 SSE loads. The inner containment remains tight, and reinforced concrete structures of the containment retain their strength. The release of radioactive materials beyond the containment as a result of 1.4 SSE seismic impact is impossible.

Table 3 contains the design thresholds of seismic impact causing the loss of robustness of safety important buildings, structures and components of Akkuyu NPP and the loss of ability to perform the assigned safety functions (PSA Chapter 16) [17].

Table 3 Seismic impact thresholds for seismic category 1 structures and components

Loss of robustness	HCLPF, g	Safety function failure
Certain bearing reinforced concrete structural elements of UJA building	0.58	Restriction of releases to environment and reactor building equipment enclosures
Loss of containment tightness as a result of cracking	0.68	Restriction of releases to environment
Bearing reinforced concrete structures of SDGS building 11UBN	0.72	Safety systems power supply
Bearing reinforced concrete structures of SDGS building 11UBN	0.71	Safety systems power supply
Bearing reinforced concrete structures of secured pump house 11UQC, 12UQC	0.74	Heat removal from the core
Reactor vessel	3.58	Heat removal from the core
Reactor vessel support structures	2.93	Heat removal from the core
Reactor internals	2.49	Maintaining subcriticality and reactivity control
SG support structures	1.11	Heat removal from the core
MCP components	2.05	Heat removal from the core

Loss of robustness	HCLPF, g	Safety function failure
PRZ support structures	1.50	Heat removal from the core
PHRS support structures	1.79	Heat removal from the core
Hanger-support system of pipelines and steam lines	0.63	Heat removal from the core
Heat-exchanger fastenings to the foundation	0.61	Heat removal from the core
Fastenings of electric control cabinets, inverters, rectifiers, switchgear, transformers	0.52	Safety systems power supply
Battery fastenings to racks	0.42	Safety systems power supply
Fastenings of cable runs	0.41	Safety systems power supply
Fastenings of cable runs	0.41	Safety systems power supply

2.2.1. Range of earthquake leading to severe fuel damage

According to the guidelines to the scope of stress-test report, the earthquake level leading to accident with severe damage of nuclear fuel (NF) shall be determined in the safety margin analysis. For this purpose the following shall be assessed: "....weak points and cliff edge effects: estimation of PGA above which loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable."

According to NP-001-97 in Licensing Basis [18], the average total core damage frequency analyzed in the design shall not exceed 10^{-5} per NPP power unit per year.

IAEA Safety Guide SSG-9 describes the basic concept of the Probabilistic Seismic Hazard Analysis (PSHA):

'6.2. The smallest annual rate of exceedance of interest will depend on the eventual use of the probabilistic seismic hazard analysis (i.e. whether for design purposes or for input to a seismic probabilistic safety assessment) and should be indicated in the project plan'.

A disaggregation procedure was developed to examine the spatial and magnitude dependence of PSHA results. Its aim is to determine the magnitudes and distances that contribute to the design exceedance frequency during a specified recurrence period and during a structural period considered in the design. The hazard for the specified recurrence period and during a specified ground motion period is broken down into selected magnitude and distance bins. The relative contribution to the total hazard of each bin is calculated by dividing the bin exceedance frequency by the total exceedance frequency of all bins. The results are displayed on a histogram showing the contribution (in percent) to the calculated hazard.

To visualize the probabilistic seismic hazard analysis approach, it shows the probability of events as a function of the peak ground acceleration. Thus, the considered significant range of intensity impacts is divided into 8 intervals with a 0.05g step.

The above intervals include design basis levels of OBE and SSE (S1, S2) and beyond design basis levels with annual recurrence of up to $5{\cdot}10^{-6}$ [17]. The seismic hazard mean curve discretization over intervals is shown in peak ground acceleration PGA, g in Figure 7.

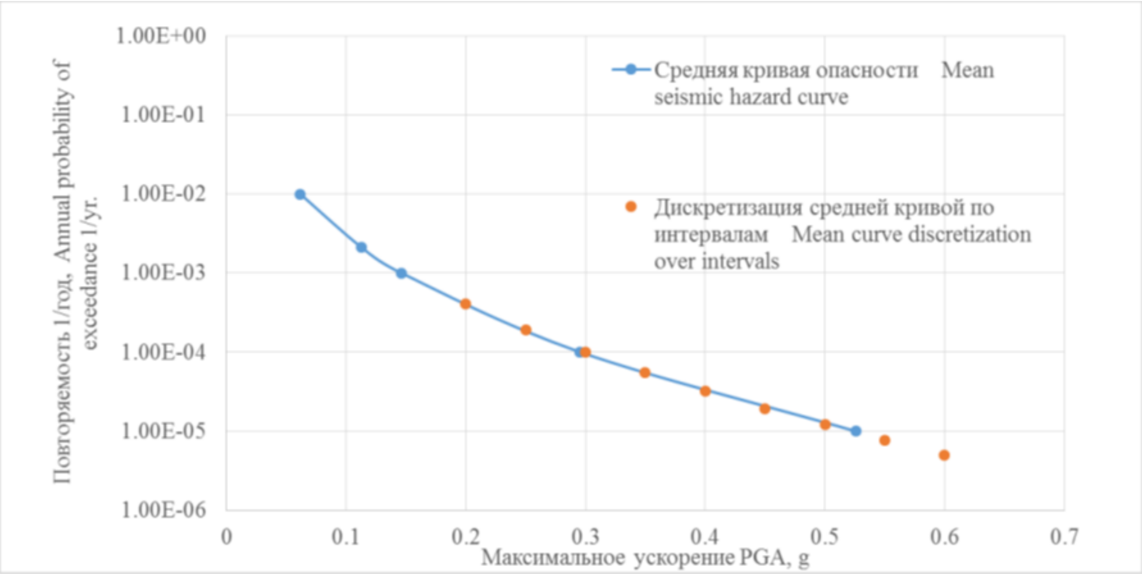

Figure 7 Seismic hazard curve

The integral point estimation based on simulation results and calculations gives the annual recurrence of $5.59{\cdot}10^{-6}$. At the same time, the following FDFs (Fuel Damage Frequencies) were calculated for various radioactive sources:

- $4.64{\cdot}10^{-6}$ per year for the core,
- $1.84{\cdot}10^{-6}$ per year for the fuel pool (PSA Chapter 16) [17].

Figure 8 shows the distribution of fuel damage frequency (FDF) estimation results over different intervals of seismic impact.

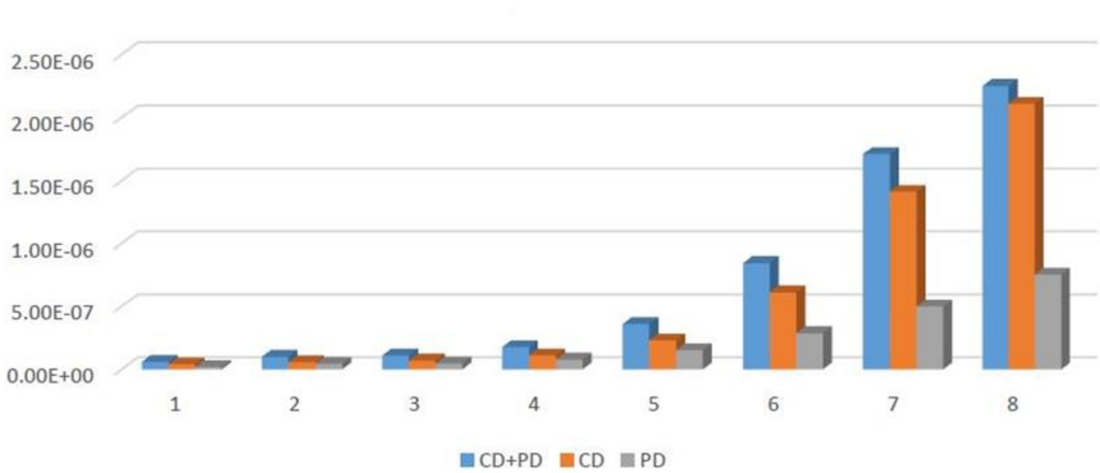

Figure 8 Fuel damage frequency over intervals

2.2.2. Range of earthquake leading to loss of containment integrity

The purpose of the containment is to limit the release of radioactivity to external environment in case of a design basis accident, limit releases in case of beyond design basis accidents, and enclose reactor building equipment and internal structures from possible external impacts. The double containment is designed to achieve the above objectives.

Outer containment is made of non-pre-stressed ferroconcrete and is designed for impact of external air shock wave, aircraft crash and external natural hazards. Outer containment maintains the annulus tightness.

Inner containment is made of prestressed concrete with 6mm sealing steel lining and designed to withstand DBA in combination with safe shutdown earthquake (SSE), as well as BDBA covered in the design, and is capable of limiting radioactive substances generated in the process [4].

Since the design covers 1.4 SSE seismic impact, it is necessary to rate the containment tolerance to this impact. The design assumes that the inner containment must retain its safety function under the given impact to limit radioactive releases.

The earthquake level was rated using PSA data [17], which may compromise the containment integrity. The seismic resistance of reactor building structures and inner containment was rated in order to determine the threshold robustness. The minimum level of seismic load leading to the reactor plant, safety system equipment and pipelines safety function failure was assessed.

Peak ground acceleration (PGA) for 40%-increased SSE is assumed to be 0.54g. If this value is exceeded, the probability of outer containment failure sharply increases and the inner containment of the reactor building may creep resulting in loss of integrity. The safety function for limiting releases to the environment is failed.

Thresholds at which reactor building robustness and containment integrity may be lost, are 0.68 and 0.58 g, respectively.

These values derived with the conditional probability of 0.05 and in the confidence interval of 95% demonstrate that the figures confirm the safety margin incorporated in the design with respect to the containment that will retain its protective safety function at 1.4 SSE.

2.2.3. Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

The sources of secondary effects of seismic impacts on Akkuyu NPP were assessed during engineering surveys and the development of the Site Parameters Report [5]. Tsunami waves are the most dangerous secondary effect of an earthquake.

The Site Parameters Report [5] includes studies of the maximum possible tsunami wave height in the vicinity of Akkuyu NPP site. The impact of tsunami on Akkuyu NPP was analyzed in PSAR Chapter 3 [4].

Parameters of tsunami sources (design-basis earthquakes), which are capable of generating the maximum impact on Akkuyu NPP site, were determined by simulation of tsunami as secondary earthquake-induced impact.

Design-basis earthquakes that induce tsunamis (including the 10,000-year recurrence period assessment) were selected as input data for the computational models of the survey. Probabilistic tsunami hazard assessment is used to calculate the peak coastal tsunami height in the vicinity of Akkuyu NPP with the exceedance probability of 10,000 years.

Initially, when the analyses were carried out considering the native bathymetry of the site, the maximum probable sea level in the vicinity of Akkuyu NPP site (once per 10,000 years) equaled 10.05 m (tsunami - 7.97 m, + tide-0.15 m, + storm wave set-up - 0.6, + seasonal variations - 0.15 m, + sea rise due to global warming - 1.0 m, + barometric effects - 0.10 m, + wind set-up – 0.08 m). A separate study was performed afterwards, considering the construction of offshore hydraulic structures, the maximum probable sea level in the vicinity of Akkuyu NPP site has been additionally estimated amounting to 8.63 m (tsunami- 6.55 m and all other sea level constituents of 2.08 m adverse effects which are comprising of: tide-0.15 m, storm wave set-up - 0.6 m, seasonal variations - 0.15 m, sea rise due to global warming - 1.0 m, barometric effects - 0.10 m, wind set-up – 0.08 m).

The grade elevation of main buildings and structures is 10.5 m (in TUDKA-99 system of elevations) that gives the sufficient safety margin for the NPP against flooding. At the same time, the cooling water system (water intake pipe, pumps, etc.) is designed for the maximum sea level during tsunami and other limit states of the sea level.

The stability against hydrodynamic impacts on on-shore facilities was also assessed as part of the tsunami secondary effect assessment. The following on-shore facilities were assessed for stability:

- breakwater dike,
- bank protection structure ,
- essential service pump stations,
- essential service pipeline tunnel,
- bank protection support wall,
- water intake facility.

Breakwater dike forms a closed circuit at the location of the water intake facilities and the shore line.

The dike is built of rock material of various sizes (mass of 1-400 kg) with 1:2 slope ratio from the seaside and 1:1.75 from the side of inner water area before pump stations. Elevation of the protection dike edge is +10.50, the width of the breakwater dike edge is 27.74 m.

The bank protection structure is designed as a pile of stone of the required mass on the slope of the newly built territory with 1:2 slope ratio and is similar to the breakwater dike structure. The slope shall be protected to +12.50 m from the seaside by 30 ton antifer blocks in two 6.36 m layers.

Soil at the base of the bank protection structure will be replaced with rock material.

Protection dike, water intake facility, essential-service pump stations and essential-service pipeline tunnel have seismic category I.

Heat removal from safety systems to the ultimate heat sink is designed in seismic category I and safety class 2 buildings (cooling water is supplied through two independent trains).

The above-mentioned structures can be destroyed during an earthquake exceeding SSE level. Seismic impact threshold for seismic category 1 structures and components for load-bearing reinforced concrete structures of the essential-service pump station building, HCLPF is 0.74g (1.4SSE = 1.4×0.388 = 0.54g).

The damage of water intake facility is also possible during an earthquake exceeding SSE. Each water intake structure has three reinforced concrete pipelines of 3.80x3.80m square cross-section for each line with water intake portals, which significantly reduces the probability of damage of all the pipelines during an earthquake exceeding SSE.

The damage (failure) of all pipelines cannot lead to the complete blocking of sea water supply to the essential-service pump station for two power units of the NPP at the same time, because water flow through damaged pipelines and damaged breakwater dikes will be maintained under such circumstances.

Nevertheless, hypothetically, heat removal to the primary ultimate heat sink may be lost. In this case, the NPP design provides for the residual heat removal to the alternative heat sink by passive systems during 72 hours.

Thus, in combination with other factors that cause the maximum sea level, an earthquake with its secondary impact (tsunami) will not lead to the flooding of Akkuyu NPP site and the failure of the secured and conventional cooling system.

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

The main goal of protection from external impacts is to preserve safety functions of systems and components that perform these functions and maintain physical barriers that prevent the release of radioactive substances and radiation into the environment.

Buildings and structures are designed taking into account the specified design impacts in accordance with the current regulatory framework. Therefore, OBE and SSE have no radiation consequences, additional strengthening measures are not required.

The outer containment is made of cast-in-situ reinforced concrete and rules out the brittle instantaneous failure (cliff-edge effect).

The acceleration threshold specified above for seismic category I structures is determined with sufficient conservatism. The threshold can be further increased if non-conservative approach is applied. SSE margin is 0.68g (with assumed HCLPF) to maintain the tightness of UJA building containment.

Nevertheless, the following organizational and engineering arrangements are recommended to further improve the safety and the design itself regarding the stability against earthquakes and their secondary effects:

- in addition to regular design solutions for power sources, the design provides an alternative diesel-generator set (ADGS) (PSAR Chapters 8, 15) [4]. Possible options for this solution, including mobile equipment, are advisable to select and consider taking into account its seismic stability. ADGS shall be more stable against seismic impacts comparing to the systems that it is alternative to. If ADGS is normally located beyond the site or the given region, it is necessary to make organizational and engineering arrangements for ADGS connection (delivery, deployment) taking into account possible damage of distribution network access infrastructure,
- relevant operating procedures shall be developed to maintain availability of equipment required to transfer the reactor plant into a safe state after an earthquake,
- to improve stability of the plant against secondary effects of an earthquake (namely the integrity of bank protection structures) it is recommended to develop a procedure for regular inspections of bank protection structures, breakwater dike, water intake facility, tunnels for essential-service pipelines.

Therefore, 1.4 SSE seismic impact with an acceleration of 0.54 (0.388×1.4) will create no cliff edge effect. This margin is sufficient to ensure that the containment retains its integrity when the seismic level is increased.

3. FLOODING

According to the Turkish Regulation on Nuclear Power Plant Sites [7], Section 6, Hydrological External Natural Events, the following flood effects have to be accounted in NPP design:

"Article 18 - (1) Flood causing events and their potential effects in the region shall be taken into account individually for sites on rivers and on the sea coasts including enclosed and semi-enclosed water bodies, gulfs and lakes coasts.

(2) In order to determine floods within the scope of design basis external event, the probabilistic or deterministic methods shall be used. If not possible to use those methods, stochastic method is used. Uncertainties should be considered in analysis.

(3) Oceanographic, hydrological, meteorological and topographical information including seismic data shall be collected relevant to coastal sites. Collected data is compared with using suitable scale maps, tables and graphics, by using aerial photographs and satellite images, probable areas that are subject to flood hazards should be identified.

(4) Hydrological and meteorological data over a minimum of 50 years should be collected.

(5) In addition to hydrological and meteorological events such as the failure of water retaining structures like dam break that may cause flooding separately, flood that may occurs with combinations of events should be analysed.

(6) Parameters of tsunami or seiche that can affect the plant, is determined via deterministic and probabilistic methods and whenever possible, these results should be verified with reviewing of historical records and seiche data on coastal region at site vicinity. Conservative approach is used in case of disharmony.

(7) The nature and breaking mechanism of the waves and for the entire range of water elevations that are expected should be identified, and the hydrostatic and hydrodynamic loading on structures important to safety should be evaluated".

3.1. Design basis

Akkuyu NPP site bay is situated along the Mediterranean coastline in the Mersin Province of Southern Turkey and is located 140 km west-southwest of Mersin Harbor, 50 km southeast of Gülnar and 100 km north of the island of Cyprus (Figure 9).

The site lies in a bay along the Mediterranean coastline in the Mersin Province of Southern Turkey. Site is located adjacent to Akkuyu Bay, a small semi-enclosed body of water connected to the Mediterranean Sea. The topography of Akkuyu NPP site is a flat coastal plain rising 0 to 50 m above sea level surrounded by hills up to 270 m high. Akkuyu NPP site adjoins Aksaz and Akkuyu-Çamalanı Bays with a radius of 3 km.

The western part of Akkuyu Bay is partially protected by a breakwater, which is the natural boundary of the NPP area. There are no wetlands or reservoirs in the Akkuyu NPP site area that may have an adverse effect on Akkuyu NPP site.

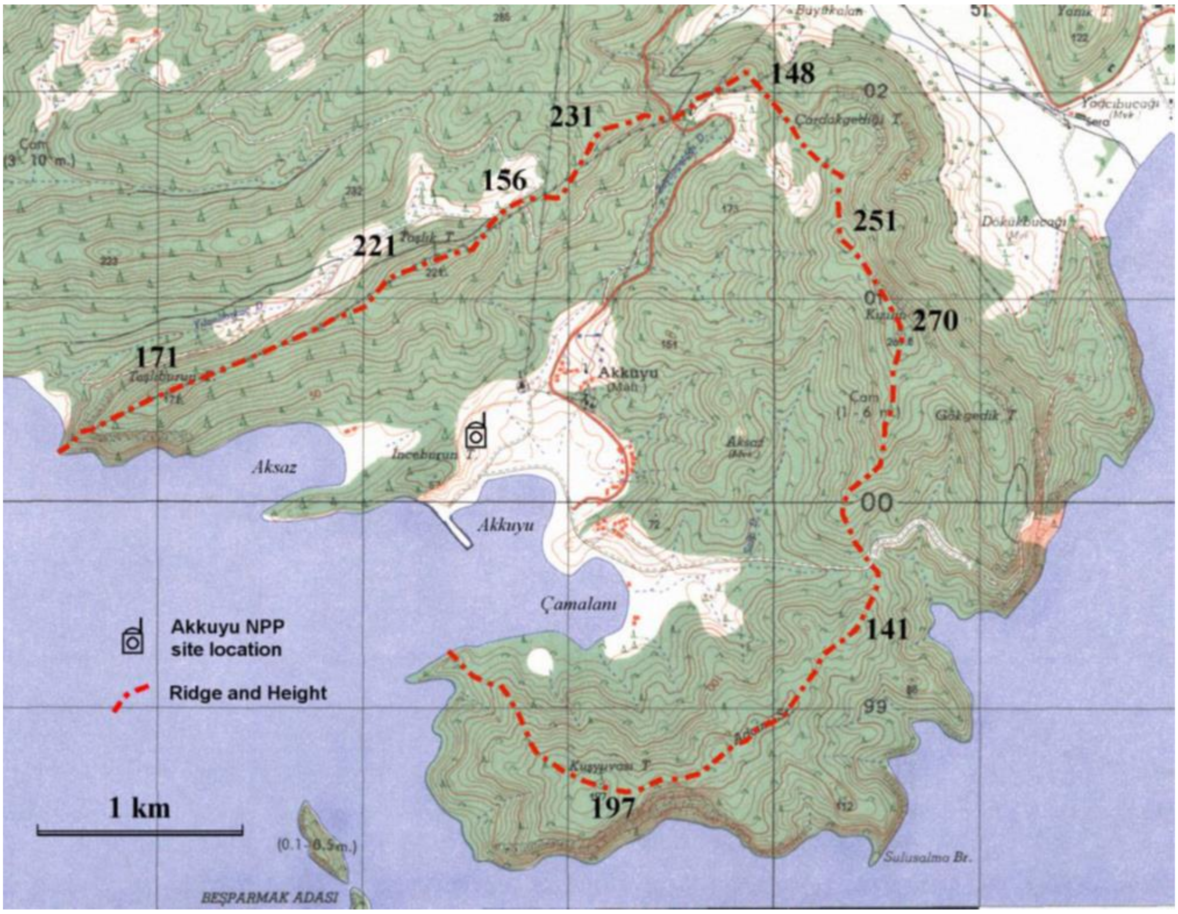

Figure 9 Akkuyu NPP site topography

3.1.1. Flooding against which the plants are designed

The design basis flood elevation for the NPP is determined by considering a number of different flooding scenario combinations. The following flooding scenarios are taken into consideration [19]:

- probable maximum precipitation,
- probable maximum flood from streams and rivers,
- potential dam failures,
- probable maximum surge,
- probable maximum tsunami,
- flooding of the water discharge channel.

Probable Maximum Precipitation

24-hour PMP was calculated which is based on 44 years of daily rainfall data from the Silifke Meteorological Station. According to analysis, 24-hour PMP is 688.5 mm. Subsequently runoff hydrographs and determine peak flows are plotted. The resulting design-basis daily maximum precipitation with recurrence once in 10,000 years is equal to 314.22 mm (Anamur Meteorological Station) and 266.8 mm (Silifke Meteorological Station). The results show that the Akkuyu NPP design PMP 688.5 mm provides more than two-fold safety margin for protection of NPP buildings, structures and equipment against this impact [5].

The resulting design-basis precipitation depth-duration curve (Figure 10) is plotted.

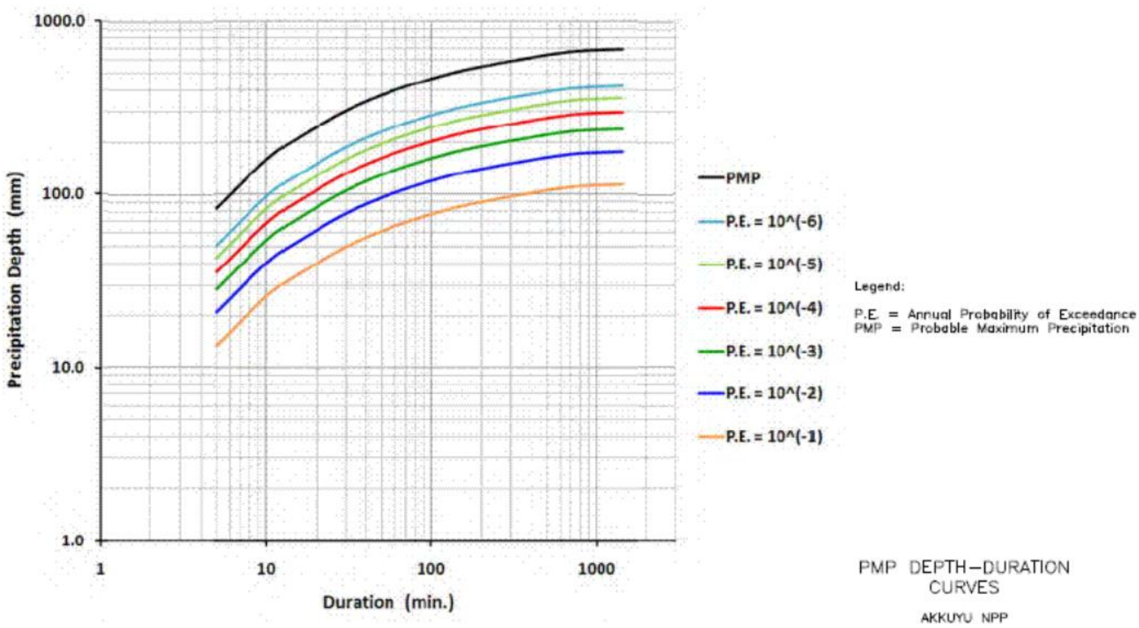

Figure 10 Resulting depth-duration curve

Probable Maximum River Floods

NPP site has temporary streams flowing only during the cold season from November to February, when most of precipitation falls. Sipahili is the continuously flowing river closest to the NPP site. Sipahili River is the most important surface water source in the vicinity and is located at about 7 km distance to the west of the NPP site. During dry season Sipahili River has noncontinuous surface flow. Due to the physical and hydrological separation of the Sipahili River from the Akkuyu NPP site, the runoff of the Sipahili River can by no means impact the region of Akkuyu NPP or cause inundation of the NPP construction site.

Estimates for the creeks resulting from the river basins within the NPP area also show that they pose no flooding threat to the NPP, neither through surface flow nor through an increase in the MSL in the Akkuyu Bay. River basins at the Project area is presented in Figure 11.

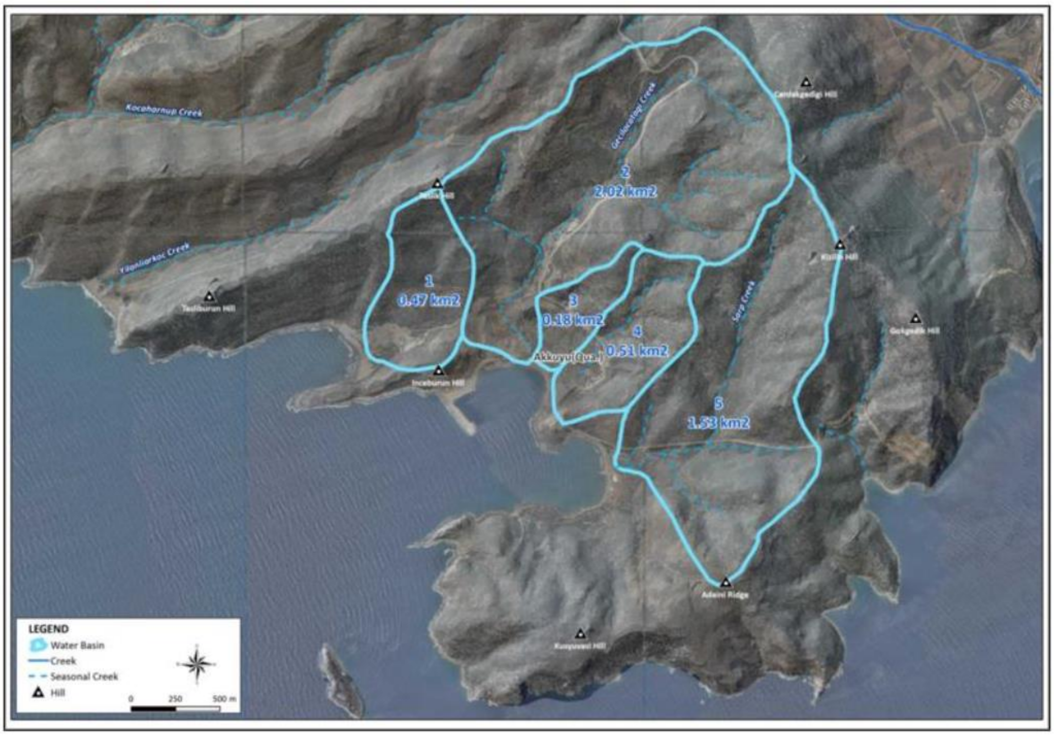

Figure 11 River basins at the project area

Potential Dam Failure

Akkuyu NPP site located within two catchment basins of Zeytinçatağı Creek and Çamalanı-Sarp Creeks surrounded by mountain ranges is virtually isolated from all the dams in the region. There are no dams that could fail and flood the Akkuyu NPP site. Gezende Dam, the closest dam to the Akkuyu NPP site, is 53.3 km away. The next nearest dams are 110.5 km and 161.7 km away from the Akkuyu NPP site. These dams are located beyond the drainage divide of the Akkuyu NPP watershed area. Other dams were not considered in the analysis because they were either a great distance from the Akkuyu NPP site or divided by a diversion feature (e.g. mountain range) [19].

Probable Maximum Storm Surge

Regional cyclonic storms of East Mediterranean Sea and probabilistic local storm winds, which are 'rare phenomena', are considered to estimate the probable maximum storm surge (PMSS) at the Akkuyu NPP. Maximum wind speed is found from historical meteorological data. 40 storm tracks are analyzed to determine the critical cyclone with the highest storm surge level in the Akkuyu Bay. The potential maximum water level due to the PMSS rise is calculated below Akkuyu NPP site level.

Cooling Water Channels and Reservoirs

Safety-important water supply systems of Akkuyu NPP do not depend on availability of water in local creeks and streams. The probability of landslide blocking or limiting flow to Akkuyu NPP is negligible [5]. The probability of frazil ice and ice jams to impact Akkuyu NPP is negligible.

Probable Maximum Tsunami

Tsunami is the most critical event by means of external flooding in Akkuyu Site. Chronicles and geological investigations in the Eastern Mediterranean show that tsunamis occurred more than 3,000 years, due to high seismicity and volcanic eruptions. Paleotsunami study is carried out and a tsunami catalogue is compiled. Critical scenarios of past seismic events and fault mechanisms have been studied. Earthquake induced tsunamis have been modelled. Deterministic and probabilistic tsunami hazard studies have been conducted.

A digital elevation model (DEM) of coastal bathymetry, coastal topography and offshore hydraulic engineering structures (OHES) has been developed to make the above tsunami hazard calculations more precise. Contours of offshore structures were plotted on a map with natural contours using design drawings when developing this model

Refer to [5] for summary details with the identification numbers and descriptions of historic tsunamis occurred in the Eastern Mediterranean and related to the Akkuyu region (Figure 12) with their potential sources. Figure 12 shows the locations of the most significant earthquakes that induced tsunami in Eastern Mediterranean in the past.

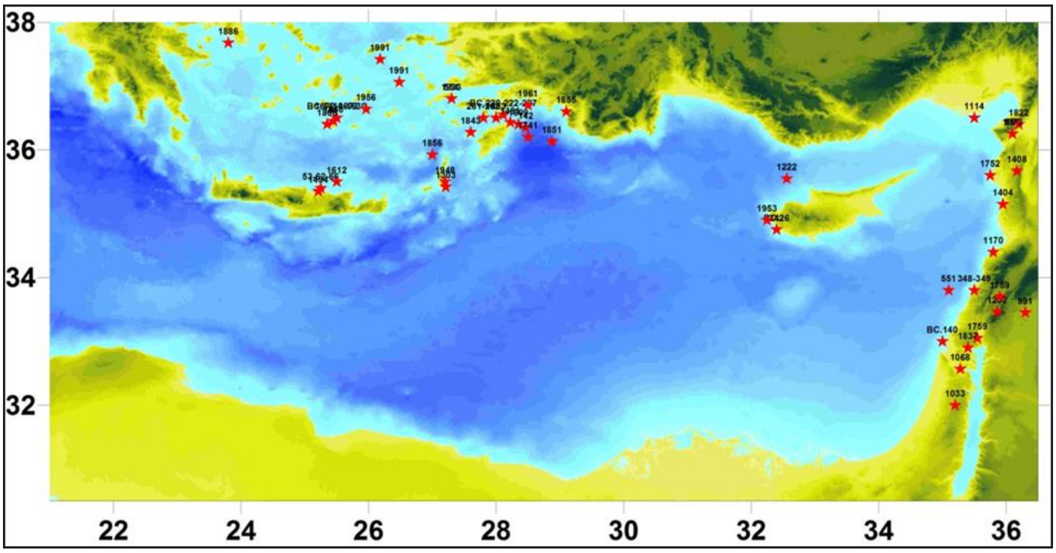

Figure 12 Locations of tsunami initiating events in Eastern Mediterranean classified according to estimated earthquake magnitudes, Richter scale

A digital elevation model (DEM) of coastal bathymetry, coastal topography and offshore hydraulic engineering structures (OHES) has been developed to make the above tsunami hazard calculations more precise. Contours of offshore structures were plotted on a map with natural contours using design drawings when developing this model.

A digital elevation model (DEM) of coastal bathymetry, coastal topography and offshore hydraulic engineering structures (OHES) has been developed to make the above tsunami hazard calculations more precise. Contours of offshore structures (Figure 14) were plotted on a map with natural contours (Figure 13) using design drawings when developing this model.

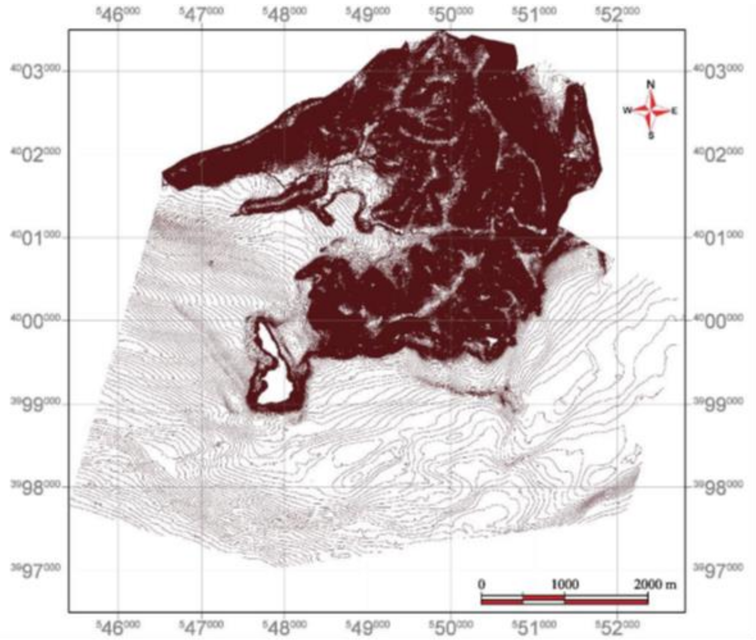

Figure 13 Natural topographic and bathymetric contours

After construction of offshore structures (breakwater, dike dam, bank protecting structure, coast support walls, water intake etc.) calculations showed that 10.000 years critical tsunami height is 6.55 m.

Design basis sea levels in the Akkuyu NPP area of 10^{-4} probability, with combination of several adverse events are determined. These adverse effects are listed below:

- sea level rise due to global warming 1.0 m,
- wind wave set-up 0.08 m,
- tide 0.15 m,
- storm wave set-up 0.60 m,
- barometric effects 0.10 m,
- seasonal fluctuations 0.15 m have been added to this value.

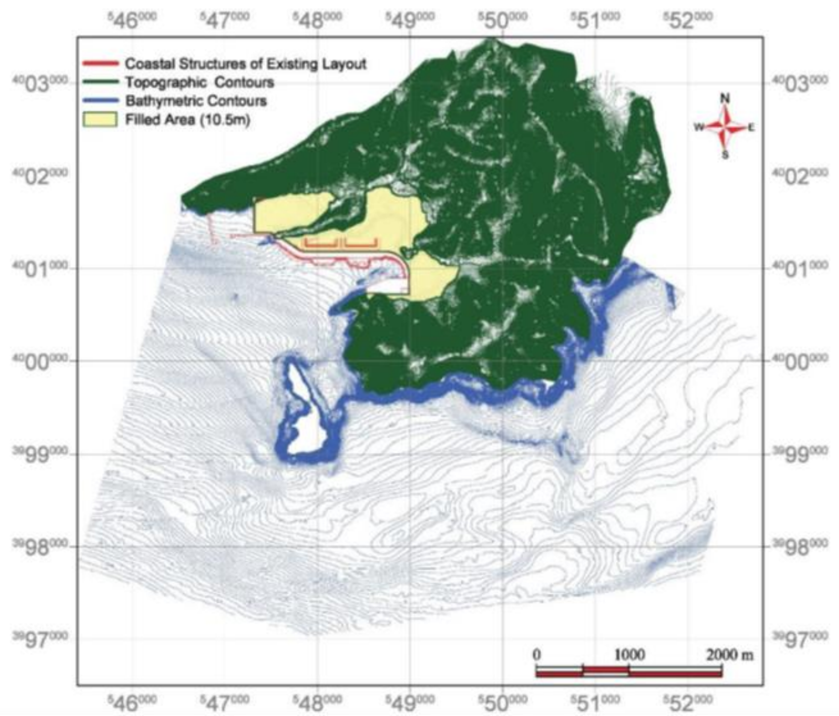

Figure 14 Edited topographic and bathymetric contours with filled area and coastal structures of Akkuyu with designed OHES

In order to provide an additional safety margin, the expression of calculations of maximum sea levels due to tsunami as "mean plus standard deviation" (mean + standard deviation) gives a maximum value of 6.55 m. The total contribution to the wave height due to sea level rise, wind wave set-up, tide, storm wave set-up, barometric effects and seasonal fluctuations is 2.08 m. Thus, the maximum sea level considering designed offshore hydraulic structures is 8.63 m.

Totally 8.63 m tsunami height value is found. Site grade level is 10.50 m which implies an enough safety margin.

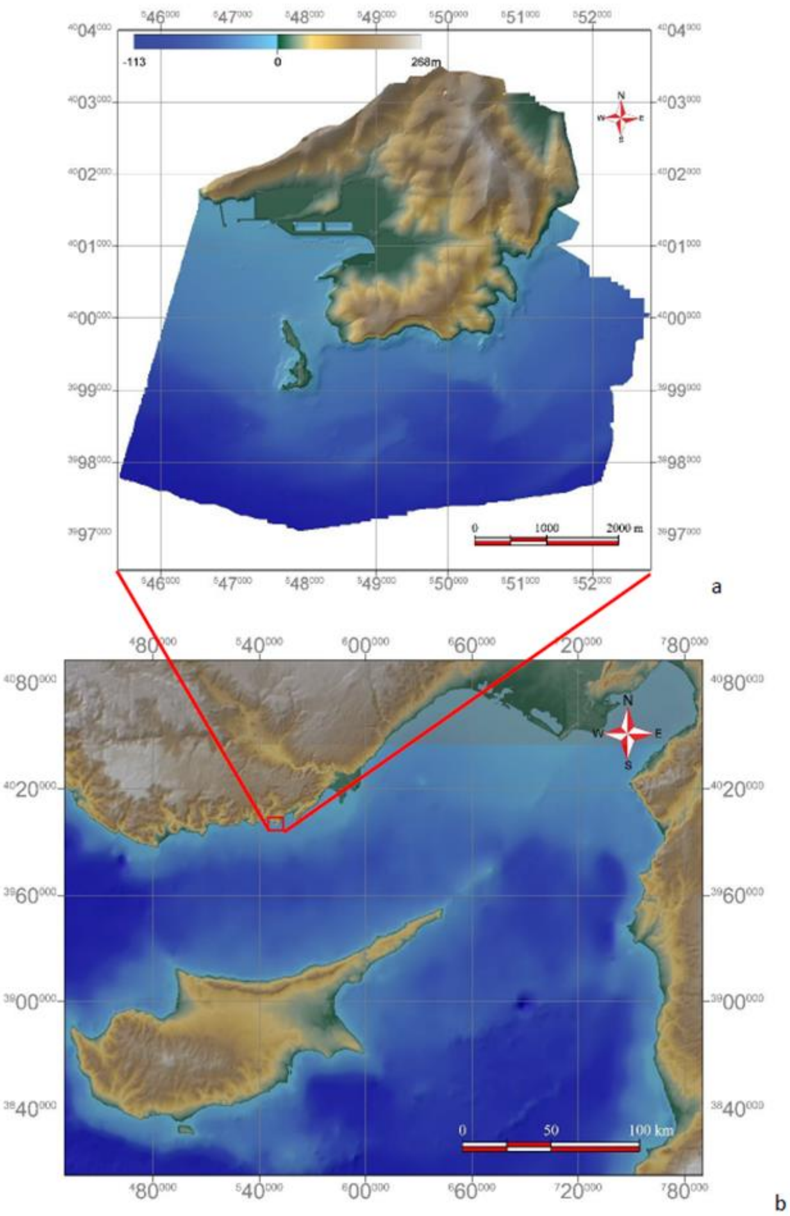

Figure 15 a) Final DEM of Akkuyu with designed OHES, b) DEM of Eastern Mediterranean (Domain B) as Akkuyu with designed OHES DEM is inserted

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

<u>Provisions against the İmpact of local probable maximum precipitation</u>

To protect the site against temporary storm creeks, drainage channels are designed along the site borders: north channel 01UZN with drainage channels in banks and sluiceways, east channel 02UZN and south channel with sluiceways (Figure 16).

Assuming 20-minute duration of extreme rainfall and absolutely flat NPP site, water will flood out in a layer of 2.7 mm in the first case and 9.5 mm in the second one without drainage. With drainage to the sea, only a partial flooding of the site is possible in the area of auxiliary structures that do not house safety-related equipment (in particular, diesel fuel, oil, gas.

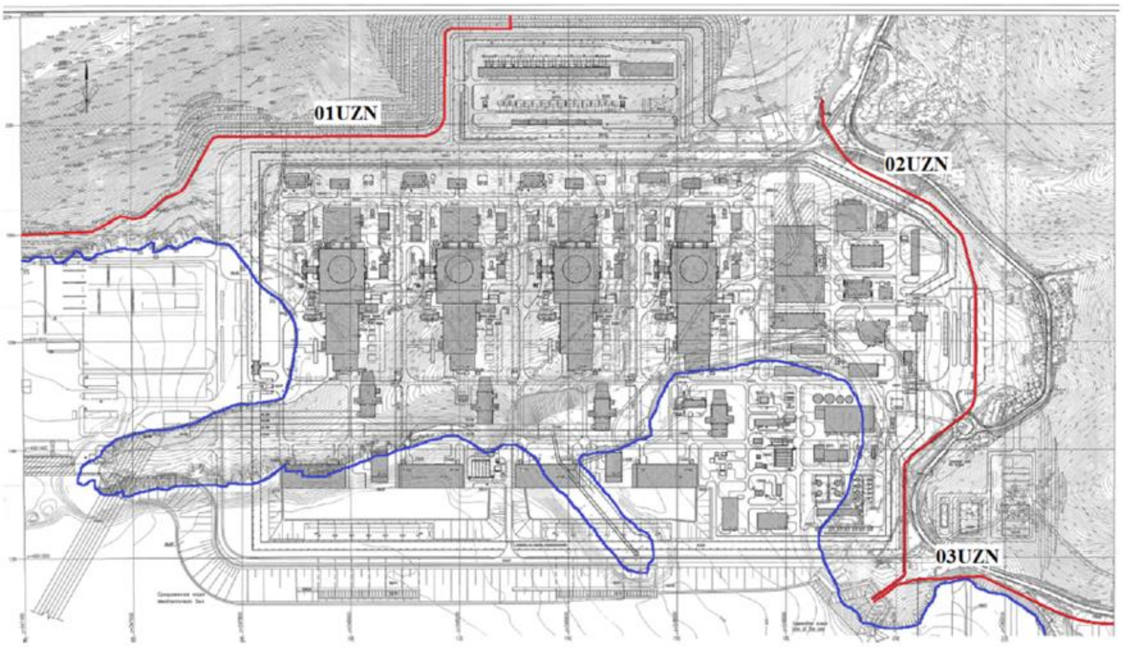

Figure 16 Drainage channel layout, grid pitch 200 m

Provisions against tsunami

The most critical case is flooding due to tsunami in Akkuyu Site. Regarding tsunami effect; Akkuyu Site can be regarded on the basis of "dry site" concept.

The maximum tsunami height is 6.55 m. The total contribution to the wave height due to sea level rise, wind wave set-up, tide, storm wave set-up, barometric effects and seasonal fluctuations is 2.08 m. Thus, the maximum sea level considering designed offshore hydraulic structures is 8.63 m.

Akkuyu NPP site grade elevation is 10.50 m, which is higher than the maximum tsunami height. Breakwater's height which will be constructed is up to the elevation of 12.50 m above mean sea level.

Provisions against maximum probable river flood

Since there is no potential hazard of river flood and Akkuyu NPP is located above the design basis flood level, no risk is expected for Akkuyu NPP structures, systems and components due to stream overflow at the site. The delivery of personnel and necessary equipment to the Akkuyu NPP is also enabled.

Provisions against probable dam failure

As there are no dams, the failure of which would cause flooding of Akkuyu NPP [4,5], no extra provisions are required against this event excluded from consideration in the probabilistic criterion safety analysis [6].

Provisions against probable maximum storm surge

According to the evaluations given in [6], the level of PMSS with the probability of 10^{-6} per year can be 7.54 m, which is 2 meters lower than Akkuyu NPP design site grade elevation

- 10.50m. Therefore, this event is excluded from probabilistic Eb criterion safety analysis. Design provisions are adequate to protect Akkuyu NPP against the probable maximum storm surge. The delivery of personnel and necessary equipment to the Akkuyu NPP is also enabled.

To protect the site against flood for precipication and storm, drainage channels are designed along the site borders. The site is also protected against storm water by an underground storm water drainage system, including gutter inlets on the roads, and a drainage gutter along the site fence to collect storm water.

Provisions related to cooling water channels and reservoirs

The proposed water intake system for the Akkuyu NPP consists of four intake structures drawing water from the Akkuyu Bay. Water is discharged to Akkuyu Bay through water outfall tunnels. Special provisions related to cooling water channels and reservoirs are not required.

The component cooling system (water intake pipe, pump, electrical units, etc.) is designed for the possible maximum sea level rise by 10 meters and its decrease by 8 meters when water recedes during tsunami and other sea level limit states. Therefore, abnormal operation of water intake facilities is ruled out during the maximum estimated flooding.

As well as the provisions indicated above; landslide tsunami, seiche effect of semiclosed Mediterrenean basins have been evaluated but no significant hazard is established.

3.1.3. Plant's compliance with its current licensing basis

The Applicant has conducted flood studies in compliance with Turkish regulations, IAEA requirements and Russian Federation Regulations. The essential factors which may cause the flooding of Akkuyu NPP site and design provisions demonstrate that Akkuyu NPP meets the requirements of the licensing basis given below:

- Turkish Regulation, Decree On Licensing of Nuclear Installations [1]
- Turkish Regulation, Regulation on Nuclear Power Plant Sites [7]
- Russian Regulation, NP-064-05 Consideration of external natural and man-induced impacts on nuclear facilities [22]
- Russian Regulation, NP-001-97 (OPB-88/97) General regulations on ensuring safety of nuclear power plants [18]
- IAEA SSR-2/1 Rev1., Safety of Nuclear Power Plant Design [23]
- IAEA NS-R-3, Site Evaluation for Nuclear Installations [20]
- IAEA SSG-18 Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [21]

3.2. Evaluation of safety margins

As it has been mentioned in the previous sections, the most critical case is tsunami by means of flooding in Akkuyu Site. Regarding tsunami effect; Akkuyu Site with 10.50 m grade can be regarded on the basis of "dry site" concept.

The maximum tsunami height is 6.55 m. The total contribution to the wave height due to sea level rise, wind wave set-up, tide, storm wave set-up, barometric effects and seasonal fluctuations is 2.08 m. Thus, the maximum sea level considering designed offshore hydraulic structures is 8.63 m.

3.2.1. Estimation of safety margin against flooding

The assessment of other flooding events and design provisions showed that they were less significant compared to tsunami. Tsunami with the maximum sea level, including the global sea level rising, tide, seasonal variations, wind wave set-up, storm wave set-up, barometric effects is found as 8.63 m. Where as site grade level is 10.50 m; that implies an enough safety margin exists. Therefore it can be concluded as achieving and maintaining the safe shutdown state, systems and structures designed for flood protection remain in operable condition.

Significant difference between the probability of the maximum sea level in the conservative evaluations (for 10.000 years) on which Akkuyu NPP site grade elevation of 10.50 m is based and the probability of the maximum sea level in the probabilistic safety analysis ($1.1{\cdot}10^{-7}$ year return period) shows a sufficient flood safety margin of Akkuyu NPP. The same conclusion is confirmed by the maximum sea level - 8.63 m taking into account the designed offshore hydraulic structures.

Nevertheless, the flooding of NPP rooms located below the upper plane of the base plate (relative elevation of 0.000 m for the level of 11.15 m according to mean sea level), which house systems responsible for the fulfillment of main NPP safety-related functions, was considered in the stress test in order to evaluate safety margins. At the same time, it is assumed that all equipment located in the above-mentioned rooms becomes inoperable due to flooding, this equipment cannot be restored within 72 hours.

The conservative flooding assessment of safety systems and systems important for safety (SS and SIS), which are located below 0.00 m (11.15 mean sea level), demonstrated that the flooding with these assumptions leads to loss of heat removal from the primary circuit and spent nuclear fuel in the fuel pool resulting from water ingress into the reactor building adjoining structures and to UKA building at minus 5.40 m, and to other elevations below 0.00 m. UKA building at minus 5.40 m houses rooms of KAA secured cooling water system, JNA emergency and planned primary circuit system and fuel pool cooldown system, and other safety systems. At these elevations, JNA system rooms are located close to the rooms with thermal monitoring sensors that monitor parameters of this system. Battery rooms of safety systems and normal operation systems are located in the uncontrolled access area of the adjoining structures. Rooms for DC boards of safety systems are located at minus 2.10 m.

The substructure of the SDGS building at minus 5.00 m houses service basements with auxiliary equipment of the diesel generator and ventilation rooms.

The evaluation of safety margins demonstrates that anticipated operational occurrences at the power units are possible in this situation due to loss of heat removal from the primary circuit and spent nuclear fuel in the fuel pool to the ultimate heat sink.

There is loss of AC power supply to the power unit, loss of all emergency power supply sources. In this mode, PHRS, HA-2,3 and the alternative diesel generator fulfill the function of residual heat removal from the reactor and spent fuel. The scenario of such beyond design basis accident is described in detail in section 5 hereof, and the Akkuyu NPP design provides for residual heat removal from fuel in the reactor and fuel pool during 72 hours without fuel damage.

Moreover, drainage facilities with water discharge to the sea are envisaged for water removal from the territory adjacent to the Site. In addition, sidespill weirs within water treatment facilities ensure removal of stormwater amount, exceeding the water treatment facility. As a further measure, nuclear safety important buildings are fitted with tight doors resistant to possible flooding, which will in turn be located 0.65m above the site grade (11.15 – 10.50 = 0.65m).

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

Considering high safety margin of Akkuyu NPP provided by the design developed in accordance with applicable requirements set by EUR and IAEA, there is no need for additional provisions to improve the NPP stability against flood.

4. EXTREME WEATHER CONDITIONS

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

Loads and extreme impacts to be considered in the design and additional input climatic parameters for designing heating, ventilation and air conditioning systems which were derived for Site Parameters Report are analyzed as part of the stress tests. Expected frequency of the originally postulated tornado, maximum winds, maximum rain and extreme temperatures, maximum snow load were assessed.

Meteorological data representative of the Akkuyu NPP area and adopted as the Akkuyu NPP design basis are defined in Section 4.7 of SPR [5]. These data are also given herein in Table 4. Additional climatic parameters adopted for the designing of heating, ventilation and air conditioning systems are given in Table 5.

These tables show three types of data by source of origin. The first type of meteorological data is taken from ENVY Akkuyu NPP site engineering survey reports. ENVY studied the meteorological data of long-term observations obtained from Anamur and Silifke meteorological stations. These data were used to derive average and extreme meteorological parameters. "The Summary of Average and Extreme Weather Conditions" released by the Turkish State Meteorological Service was also used. The work made in Chapter 4 of SPR [5] proved that Anamur meteorological station is generally more representative of the Akkuyu site than Silifke meteorological station. This is due to the topographic similarity of Anamur meteorological station and Akkuyu NPP. The following tables, therefore, show the data from Anamur meteorological station.

The second type of the data in Table 4 and Table 5 is based on a series of atmospheric measurements of Akkuyu NPP site.

The third type of data is taken from Russian regulations (SNiP and GOST). However, they are limited to those regulatory documents that are applicable to the Akkuyu NPP site conditions. In particular, the 'Cold Season' is not considered, since the cold season in the meaning of Russian regulations is not applicable for the Akkuyu NPP site [5].

Table 4 Loads and extreme impacts to be considered in the design [5]

Parameter	UoM	Value
Typical temperature range:		
annual average	°C	19.1
recorded minimum	°C	-4.8
recorded maximum	°C	44.2
Extreme (possible once in 10,000 years) air temperatures:		

Parameter	UoM	Value
minimum value	°C	-12.7
maximum value	°C	50.4
Probable tornado intensity class (possible once in 10,000 years over 1,000 km^2 area surrounding the NPP site)		
Kp (intensity class)		2.0
Vp (maximum spin velocity)	m/s	60
Up (translational velocity)	m/s	15
ΔPp (pressure drop)	hPa	44
L (tornado pathway (length))	km	9
W (width)	m	90
Maximum typical (observed) wind speed (gust)	m/s	46.8
Extreme wind speed (possible once in 10,000 years):		
gust	m/s	76.1
10-minute average	m/s	42
Maximum wind speed (possible once in 100 years):		
gust	m/s	49.2
10-minute average	m/s	27.1
Maximum wind speed (possible once in 50 years):		
10-minute average	m/s	24.9
Maximum wind speed (possible once in 25 years):		
10-minute average	m/s	22.6
Maximum daily precipitation (possible once in 10,000 years)	mm	302.7
Maximal typical snow load (design snow cover weight, probable once in 25 years according to SP 20.13330.2011. Loads and Impacts. Updated version of SNiP 2.01.07-85* - M., 2011)	kPa	0.8
Snow load (possible once in 100 years)	kPa	0.98
Maximum snow load (possible once in 10,000 years)	kPa	1.6
Standard maximum glaze-ice wall thickness (at elevation of 10 m as per SP 20.13330.2011. Loads and Impacts. Updated version of SNiP 2.01.07-85* - M., 2011)	mm	3
Dust storms	Number of days per year	40 to 60
Lightning	flash/(km^2·year)	4-6 times observed in the Akkuyu NPP area

Table 5 Additional input climatic parameters for designing heating, ventilation and air conditioning systems

Parameter	UoM	Value
Air temperature of the coldest five days and corresponding relative humidity:		
with non-exceedance probability of 0.92	°C %	5.4 53
with non-exceedance probability of 0.98	°C %	3.7 47
Average temperature of the hottest month at 01 or 03 p.m. (at 2 p.m.), and corresponding relative humidity	°C %	31.4 68
Air temperature, the highest value of which was observed respectively for several hours and less in a year and the corresponding relative humidity:		
up to 88 hours per year	°C %	32.9, 35.5
up to 440 hours per year	°C %	30.1 48.7
Average temperature of the coldest month (winter ventilation)	°C	11.3
Average relative humidity of the coldest month (January)	%	72
Absolute maximum observed air temperature, and corresponding air relative humidity	°C %	44.2 25
Absolute minimum observed air temperature, and corresponding air relative humidity	°C %	-4.8 85
Average outdoor air temperature with average daily air temperature ≤8°C (heating season)	°C	7
Period with average daily air temperature of ≤8°C (heating), days	day	4
Average outdoor air temperature with average daily air temperature ≤0°C (heating season)	°C	-1.77
Period with average daily air temperature of ≤ 0°C, days	day	0.15
Average monthly air temperature of the hottest months (July-August) and corresponding relative humidity	°C %	27.9 73
Average daily heat quantity from total solar radiation (direct and diffuse) to:		
horizontal surface (July)	W/m^2	335

Parameter	UoM	Value
vertical surface of western and eastern orientation (July)	W/m^2	157
Total daily solar radiation to vertical surface of western and eastern orientation		
direct	W/m^2	2455
diffuse	W/m^2	1397
total	W/m^2	3852
Total daily solar radiation to horizontal surface		
direct	W/m^2	6505
diffuse	W/m^2	1530
total	W/m^2	8035
Maximum hourly solar radiation to vertical surface of western and eastern orientation (July):		
direct	W/m^2	530
diffuse	W/m^2	180
total	W/m^2	710
Maximum hourly solar radiation to horizontal surface (July):		
direct	W/m^2	815
diffuse	W/m^2	140
total	W/m^2	955
Maximum amplitude of air temperature daily fluctuations in the hottest month	°C	20.8
Prevailing wind direction (height 10 m at the 60-meter mast) at the Akkuyu NPP site		
summer	rhumb	SW, SSW, NNE, WSW
winter	rhumb	NNE, NE, N
annual	rhumb	NNE, NE, N, SW
Average yearly barometric pressure at the Akkuyu NPP site level	hPa	1007.7

Additionally, various combinations of these weather conditions were considered during the stress tests.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

The NPP buildings and structures are rated for resistance to the following initiating events:

- extreme temperatures,
- extreme wind loading, tornadoes,

- extreme rains,
- snow loads and icing,
- dust storms,
- lightning strikes.

Extreme air temperatures

In accordance with section 4.7 of SPR [5], the following extreme air temperatures (possible once in 10,000 years) are assumed:

- minimum – minus 12.7 °C,
- maximum – plus 50.4 °C.

Akkuyu NPP is designed so that process equipment of safety systems and normal operation systems remains functional at the maximum indoor temperature plus 40°C. Control and monitoring system hardware is functional at room temperature not higher than 40°C [4].

Extreme maximum outdoor air temperature of 50.4°C is used for the confirmatory analysis of ventilation systems of safety system rooms. The analysis confirmed that Akkuyu NPP ventilation systems maintain a temperature of not more than plus 40°C in safety system rooms at this temperature [4].

As for low temperatures, Akkuyu NPP is designed so that equipment of safety systems and normal operation systems remains functional at the minimum indoor temperature of plus 5°C [4]. Extreme minimum outdoor air temperature of minus 12.7°C is used for the confirmatory analysis of heating and ventilation systems of safety system rooms. The analysis confirmed that heating and ventilation systems maintain a temperature of not less than plus 5°C in safety system rooms at this temperature [4].

Due to long extremely low temperatures of outdoor air, the air temperature in normal operation system rooms may drop below plus 5°C. Operator intervention is necessary to prevent this by partially disabling ventilation systems [4].

Besides the ventilation and air conditioning system, extreme outdoor temperatures can affect the passive heat removal system. The passive heat removal system is designed for the following extreme outdoor air temperatures:

- typical minimum temperature – minus 14.1 °C,
- typical maximum temperature – plus 50.4 °C (section 3.4.1.4 of PSAR) [4].

Due to long extremely high outdoor air temperatures, the air temperature may rise in normal operation system rooms. This can lead to overheating of electrical and process equipment and its shutdown when appropriate trips and interlocks are triggered. However, the regular operation of the safety system will protect the power unit [4].

Therefore, design-basis extremely high/extremely low outdoor air temperatures (possible once every 10,000 years) are not a cliff edge effect leading to a significant safety deterioration.

Extreme wind, tornado

The extreme wind speed that can probably occur once in 10,000 years is assumed equal to 76.1 m/sec. Tornado intensity class with the probability of this weather event within 1,000 km2 vicinity of Akkuyu NPP once in 10,000 years is assumed as 2 according to Fujita scale (F). Extreme wind loads do not result in the violation of safe operation limits because the buildings and structures are designed for such loads. As for the missile impact in case of tornado intensity class 2, only light items might be sent flying in the air, which would not impact Akkuyu NPP buildings or structures according to PSAR [4].

Wind loads cannot affect the availability of the passive heat removal system, since the system is designed for heat removal under wind loads up to 90 m/s, which is significantly higher than the maximum wind probable to occur once in 10,000 years. Natural circulation in the passive heat removal system can be possibly disrupted by tornado impact. However, the tornado effect is of a transient nature, because the tornado moves along with the tornado generating cloud at the speed of up to tens of kilometers per hour. Natural circulation in the passive heat removal system restores to normal as soon as the tornado leaves [4].

Therefore, Akkuyu NPP is designed against extreme wind with sufficient safety margin. Extreme wind and tornado would not compromise the reliability of heat removal. Insignificant exceedance of extreme wind speed over design limits would not cause cliff edge effect.

Extreme rain

An underground rainwater sewage system is designed to protect the Akkuyu NPP site against rainwater. Water enters the system through gutter inlets located on the roadway. According to the design explained in PSAR, storm water from the area surrounding Akkuyu NPP site is collected in the trench drain located around the fence of Akkuyu NPP site [4].

The design-basis daily maximum precipitation (once in 10,000 years) for Anamur meteorological station is 302.7 mm [4]. However, according to SPR, maximum probable precipitation for Akkuyu NPP site is assumed as 688.5 mm [4]. This value twice exceeding the daily maximum precipitation with 10,000-year recurrence period was adopted as the design parameter for Akkuyu NPP. Such a conservative approach offers a high capacity margin for the storm sewage system.

Besides, all safety-related facility entrances are at least 0.5 m above the plant grade elevation in order to protect the safety-related facilities from intense precipitation at the Akkuyu NPP site [5].

Therefore, Akkuyu NPP is designed against probable flooding and inundation caused be extreme rain with sufficient safety margins, and the exceedance of precipitation level above the design limits would not cause cliff edge effect.

Snow load

Snow cover in the Akkuyu NPP area is a rare event. The characteristics of solid atmospheric precipitation of snow type for reference meteorological stations for 1975-2009 are taken from PSAR and given in Table 6 [4]:

Table 6 Snow precipitation for Anamur and Silifke MS for 1975 to 2009

Parameter	**Month**			
	I	**II**	**XI**	**XII**
Anamur meteorological station				
Number of days with snow	-	0.3	-	-
Number of days with snow cover	-	-	-	-
Maximum thickness of snow cover, cm	-	3	-	-
Silifke meteorological station				
Number of days with snow	0.1	0.4	0.1	-
Number of days with snow cover	-	0.1	-	-
Maximum thickness of snow cover, cm	-	2	3	-

PSAR [4] assumes the extreme snow load of 1.6 kPa (design value once in 10,000 years) as the design basis.

The most dangerous hazard is ice, which might cause damages of overhead transmission lines and NPP blackout. Akkuyu NPP is designed so that the icing thickness, which is exceeded once in 5 years, on circular cross-section elements with a diameter of 10 mm at an elevation of 10 m is 3 mm. according to PSAR [4]. If this value is exceeded, the off-site power supply may be disrupted due to damage of overhead transmission lines due to icing. However, the emergency power supply system remains functional (cables of this system are laid indoors and underground), which ensures the transfer of Akkuyu NPP into a safe state without compromising the reliability of heat removal. Therefore, the icing of overhead transmission lines is not a cliff edge effect leading to a significant safety deterioration.

Dust storms

The number of days per year affected by particulate drift from the Sahara ranges from 40 to 60 days in the Eastern Mediterranean. Based on the results of studies conducted at Erdemli 75 km northwest of the Akkuyu NPP site, the concentrations of PM_{10} and $PM_{2,5}$ (particles of aerodynamic equivalent diameter of less than 10 µm and less than 2.5 µm) changed from 2 to 326 µg/m^3 and from 0.5 to 28 µg/m^3 respectively according to PSAR [4].

PM_{10} concentrations measured from July through August 2011 and from January through November 2012 at the Akkuyu NPP site varied within the range of 1.1 to 187 µg/m^3, the average concentration of PM_{10} was within the range of 23 to 28 µg/m^3 according to SPR [5].

Ventilation systems have filters to protect Akkuyu NPP rooms against dust as explained in PSAR Chapter 2 [4]. The filters will clean the air in case of dust storm. If dust content in the air is high for a long period of time during dust storm, the efficiency of ventilation systems might be affected and the dust may enter the Akkuyu NPP rooms. Personnel shall replace clogged filters at the first sings of low performance of ventilation system. The above mentioned measures ensure the stability of Akkuyu NPP against dust storms. Therefore, dust storm cannot affect the safety of Akkuyu NPP.

Lightning Strikes

According to PSAR [4], the Akkuyu NPP design includes a set of measures for:

- protection against direct lightning strikes,
- protection against secondary effects of lightning - induced voltage surge,
- protection against extreme high voltage and heavy currents through overhead and underground utilities,
- shielding of ACS and MCS accommodating rooms.

The Lightning Protection System Includes:

- lightning rod,
- conductors,
- screens,
- equipotential nets,
- ground wire.

The above mentioned features ensure the safety of personnel, protection of buildings and electrical equipment, electromagnetic compatibility and reliability of monitoring and control systems (MCS). Therefore, lightning strikes cannot affect the safety of Akkuyu NPP [6].

Analysis of Extreme Weather Events

Table 7 provides summary of various effects of extreme weather events made on the basis of SPR and PSAR.

The results of probabilistic safety analysis for external initiating events are given in [6]. The following has been made according to the methodology of Level 1 PSA for external initiating events:

- development of selection criteria and compilation of a complete generalized list of external initiating events (EIE),
- screening analysis based on ‘qualitative’ selection criteria,
- boundary and detailed analyzes of selected EIEs,

- analysis of the severe core damage frequency during equipment failures caused by EIEs or their combination.

EIEs to be considered when assessing the severe core damage have been selected on the basis of extreme values (maxima) of the intensity and frequency of EIE, as well as on the basis of qualitative selection criteria and taking into account the design basis of the NPP unit.

Sources of external events specific to the Akkuyu NPP units have been identified based a detailed analysis of information about the NPP unit site, design features of the unit, the location of unit structures and systems, facilities located in the NPP area. No sources out of the list recommended by RB-021-01 [28] were found among the identified ones.

The list of EIEs for screening analysis based on "qualitative" criteria includes the following extreme weather conditions:

- sandstorms,
- snowstorms,
- strong wind,
- tornado,
- hail,
- lightning strikes,
- external flooding,
- high water level in water bodies,
- tides,
- breakthrough of a natural or man-made reservoir,
- nearshore sea regime (drifts, surges, storm waves),
- seiche,
- tsunami,
- abnormal drop in the water level of landlocked bodies of water,
- high summer temperature,
- drought,
- icing,
- ice phenomena on water courses,
- low winter temperatures,

- avalanche,
- mudflows,
- rainfall,
- snow loads,
- fog.

The following EIEs caused by extreme weather conditions have been selected for the boundary analysis (the remaining EIEs from the above list were screened out at the qualitative selection stage) [6]:

- external flooding,
  - tides and seasonal fluctuations
  - tsunami
  - wind impact
  - combination of external impacts causing rise in the water level in the Mediterranean
  - accumulation of a pool on building roofs due to extreme precipitation
  - site flooding due to water runoff from surrounding hills
- strong wind,
- tornado,
- snow loads.

The following EIEs caused by extreme weather conditions have been selected for the detailed analysis (the remaining EIEs from the above list were screened out at the qualitative selection and boundary analysis stage) [6]:

- accumulation of a pool on building roofs due to extreme precipitation (heavy rainfall),
- strong wind.

The external initiating event "accumulation of a pool on building roofs due to extreme precipitation" leads to the accumulation of a pool on the roof of 10UMA building. The most likely consequence of this scenario will be 10UMA roof leak. It is conservatively assumed that this will lead to the failure of equipment located in 10UMA building.

The boundary analysis shows that the external initiating event “strong wind” causes damage to the offsite power grid equipment. The consequence of such an event is the loss of normal power supply to the NPP.

Probable effects of combined extreme weather events is presented in Table 8.

Table 7 Probable effects of extreme weather events

Initiating event	**Initial impact on NPP**	**Secondary impact on NPP**	**List of systems and components which might be affected**	**Probable pessimistic scenario**	**Result of preliminary safety analysis for this probable scenario**
Extreme wind	Dynamic impact of wind pressure	Missile generation Damage to site building structures, deterioration of PHRS performance	All site buildings and structures, normal operation power supply system, PHRS, ventilation and air-conditioning systems	Loss of off-site power	Safety is ensured by normal operation of active safety systems taking into account the assumed design criteria
Tornadoes	Dynamic impact of wind pressure	Damage of category II buildings and structures per PiNAE-5.6. Temporary deterioration of PHRS performance	All site buildings and structures, including outdoor switchgears, normal operation power supply system, PHRS, ventilation and air-conditioning systems	Long-term loss of off-site power	Safety is ensured by normal operation of active safety systems taking into account the assumed design criteria
Extreme temperature	Environment temperature lowering or increasing to extreme levels	Ultimate heat sink temperature increase/decrease beyond design basis values, glazing on the components of normal power supply equipment.	Essential and non-essential service water supply systems, ventilation and air-conditioning systems, PHRS	Long- term loss of off-site power supply of the unit Unit shutdown due to lowered/increased air temperature in safety-important rooms Loss of normal heat removal	Safety is ensured by normal operation of safety systems taking into account the assumed design criteria
Extreme precipitation and other meteorological events (snowstorm, snow, icing)	High water (snow, ice) level at site, on the roofs of buildings, on power supply utilities	Damage of buildings, structures, power transmission lines, site flooding, water leaking through basements, open small ventilation windows, transoms.	All buildings and structures Normal power supply systems, turbine department systems, service water systems	Long- term loss of off-site power supply of the unit	Safety is ensured by normal operation of safety systems taking into account the assumed design criteria

Initiating event	Initial impact on NPP	Secondary impact on NPP	List of systems and components which might be affected	Probable pessimistic scenario	Result of preliminary safety analysis for this probable scenario
Lightning strikes	Impact of sparking and electromagnetic pulses	Electromagnetic interference, shock wave, surge voltage, on-site fire	Normal power supply systems, control systems	Loss of off-site power supply of the unit, spurious actuation of control systems.	Safety is ensured by normal operation of active safety systems taking into account the assumed design criteria

Table 8 Probable effects of combined extreme weather events

Event 1	Event 2	The list of systems and components which might be impacted	Probable pessimistic scenario	Result of preliminary safety analysis for this probable scenario
Tornadoes	Lightning strike	All site buildings and structures, including outdoor switchgears, normal operation power supply system, PHRS, ventilation and air-conditioning systems	Long- term loss of off-site power	Safety is ensured by normal operation of active safety systems taking into account the assumed design criteria
Extremely low temperature	Extreme rain	All site buildings and structures, including outdoor switchgears, normal operation power supply system, PHRS, ventilation and air-conditioning systems, secured service water system and conventional service water system,	Long- term loss of off-site power supply of the power unit caused by the damage of overhead transmission lines due to icing. Unit shutdown because of low temperature in the rooms important to safety.	Safety is ensured by normal operation of safety systems taking into account the assumed design criteria
Extreme wind	Extremely low/high temperature	All site buildings and structures, including normal operation power supply system, PHRS, ventilation and air-conditioning systems, secured service water system and conventional service water system	Loss of off-site power, unit shutdown caused by low/high temperature in safety important rooms. Loss of normal heat removal	Safety is ensured by normal operation of safety systems taking into account the assumed design criteria

Event 1	**Event 2**	**The list of systems and components which might be impacted**	**Probable pessimistic scenario**	**Result of preliminary safety analysis for this probable scenario**
Lightning strikes	Extreme rain	All buildings and structures, normal power supply systems, control systems, turbine department systems, service water systems	Long- term loss of off-site power supply of the unit, spurious actuation of control systems.	Safety is ensured by normal operation of active safety systems taking into account the assumed design criteria

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

The analysis of extreme weather impact on the Akkuyu NPP made in PSAR [4] showed that all the buildings accommodating safety equipment of 1, 2, 3 safety class and safety class 1 normal operation equipment are stable against the impact of design basis extreme weather events and secondary effects caused by such events.

To monitor the structural health of these critical buildings after heavy storms, hurricane-force winds and other extreme weather events, PSAR requires unscheduled technical inspections and surveys to be made according to Survey Program and Critical Building and Structure Inspection Plan. Such surveys of buildings and structures shall be made by a dedicated company and qualified inspectors. The surveys shall be performed according to developed and approved program using visual inspection and instrumentation methods.

Thus, no additional design provisions are needed to improve the stability of Akkuyu NPP against extreme weather conditions. As far as organizational provisions are concerned, to improve the stability of Akkuyu NPP against extreme weather conditions, Akkuyu NPP shall develop detailed manuals for personnel in case of extreme environment temperature, wind, precipitation and dust storms.

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

Information presented under this chapter depends on information presented in Preliminary Safety Analysis Report of Akkuyu NPP Unit 1 [4]. Since Akkuyu NPP Unit 1 is under construction and some construction license conditions are not yet met, some of the information presented here will be finalized during the operation license phase in the Final Safety Analysis Report.

5.1. Loss of electrical power

The impact of the loss of grid on safety functions and non-safety functions is analyzed for the loss of all off-site power. The stress-test assumes that the required heavy equipment cannot be delivered to the NPP site by any means of transport within 72 hours. Portable light equipment can arrive to the site from other locations after the first 24 hours.

The Akkuyu NPP design consists of four power units with the total installed capacity of 4,800 MW (4 VVER power units with installed power over 1,200 MW each). Off-site power supply shall be provided by Akkuyu NPP connection to the Turkish power grid via planned eight 380 kV transmission lines, including six long lines (over 70 km) and two short transmission lines with a length of about 6 km to connect to the local 154 kV distribution network (through 380/154 kV autotransformers) (PSAR Chapter 8 [4], SPR Chapter 15 [5]).

The existing 154 kV network in the Akkuyu NPP area is used to supply power to the site during NPP construction. It is not certain how 154 kV grid is going to be utilized. At this stage, it is envisaged that stand-by transformers are connected to 380 kV gas-insulated switchgear (GIS) of the Akkuyu NPP. To improve the survivability of 380 kV GIS, the GIS connection is designed as one-and-a-half-breaker (2 busbars and 3 breakers for two feeders). GIS will have ten cells to connect eight lines and two spare power supply lines for further extension of power distribution system.

During normal operation, auxiliary power supply of the unit is provided by two 24/10.5 kV main auxiliary transformers (AT) with capacity of 80 MV·A each connected between a generator circuit breaker and a unit step-up transformer.

When main ATs are disconnected, the auxiliary power supply of each unit of the NPP is provided by two 380/10.5 kV stand-by auxiliary transformers of the power unit (AST) with capacity of 80 MV·A each connected to 380 kV GIS as one-and-a-half-breaker. Auxiliary power supply through ATs is switched to the ASTs automatically, thus maintaining uninterrupted auxiliary power supply of Akkuyu NPP units.

For powering common-plant loads the design includes two 380/10.5 kV transformers, 63 MVA each, connected to 380 kV GIS as one-and-a-half-breaker.

The location of Akkuyu NPP with grid connection via six long 380 kV transmission lines (over 70 km) is shown in Figure 17. Komet connection shown in figure is soon cancelled.

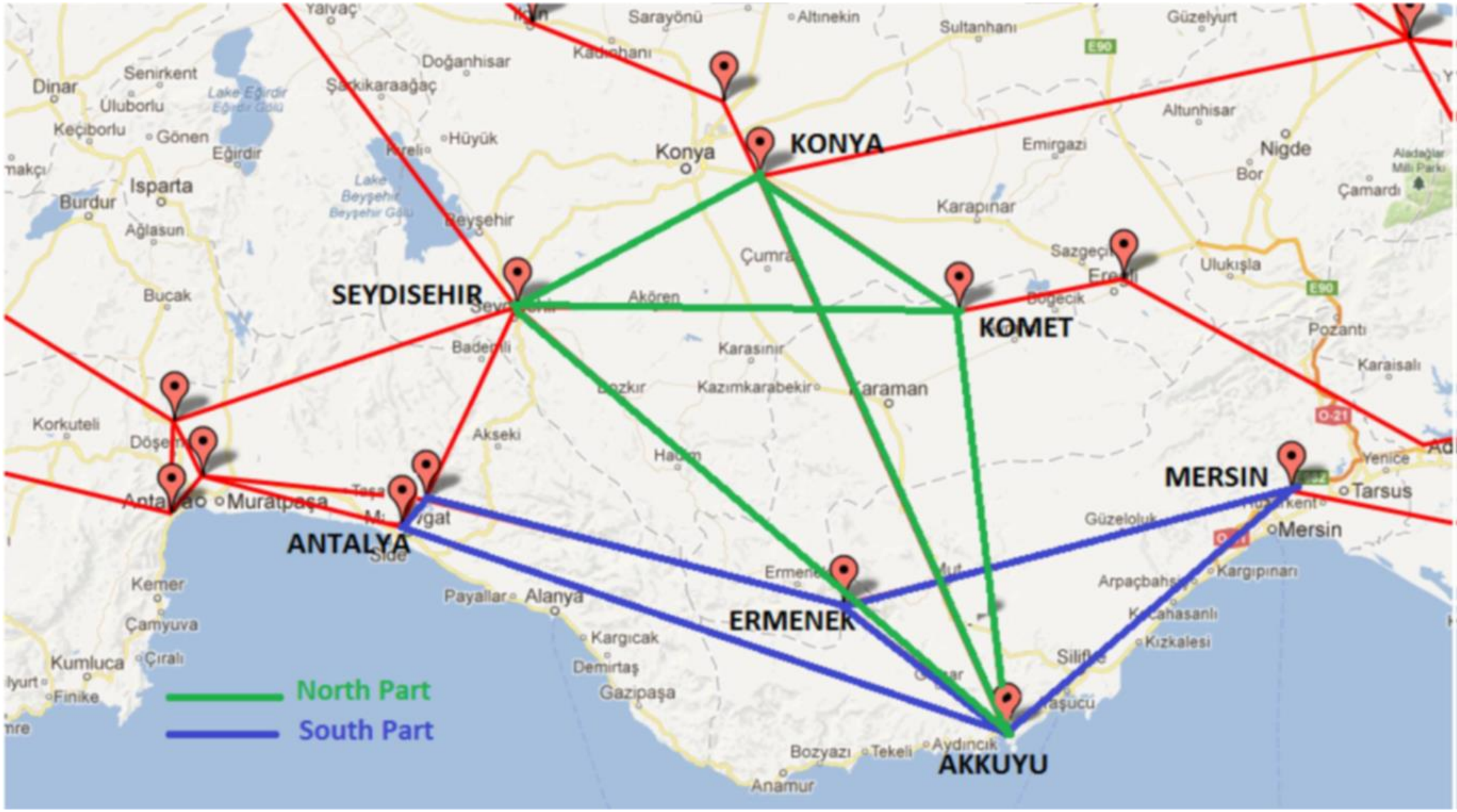

Figure 17 Akkuyu NPP location and 380 kV grid connection

Power from Akkuyu NPP shall be distributed into the Turkish grid via 380 kV transmission lines through the following substations:

- Seydişehir substation: connection to north-west region, where principal load centers are located, via powerful consumption center (i.e. Seydişehir substation),
- Konya substation: connection to the north region via powerful consumption center (Konya substation),
- Mersin substation: connection to the grid of east region via powerful consumption center (Mersin substation),
- Ermenek substation: the nearest connection to the grid of the south region and connection to important consumption centers in the west (Antalya region),
- Antalya substation: direct connection to important consumption centers in the west (Antalya region),
- Akkuyu-1 and Akkuyu-2 substations: connection to local 154 kV distribution network (through 380/154 kV autotransformers).

The design enables manual or automatic disconnection from the grid and transfer to auxiliary power supply. Akkuyu NPP is manually or automatically transferred to auxiliary power supply by relay protection in case of failure in power distribution components or loss of off-site power supply sources.

NPP units may be isolated to BOP in case of blackout. Akkuyu NPP is connected by two power transmission lines to 380/154 kV transformers supplying the local region. Pre-estimated capacity of the transformers is 250 MVA. However, this load is not sufficient for the simultaneous operation of all four units of the NPP. Therefore, in case of blackout three of the four units shall be isolated to BOP for the period of time that depends on the capabilities of the turbine, while one of the units can supply power for local loads together with BOP load.

The design provides for sufficient grid capacity to enable self-start of auxiliary mechanisms at full load shedding of the NPP.

Auxiliary power supply system of each NPP unit is designed to supply consumers that support:

- NPP operation under normal operation conditions,
- power unit operation under emergency conditions including the loss of main and stand-by power supply sources.

Each Akkuyu NPP unit shall have the following auxiliary power supply systems:

- normal operation power supply system (NOS) powering non-safety systems, including common-plant auxiliary power supply system,
- normal operation reliable power supply system (NO RPS) with one 10 kV main diesel generator station (MDGS),
- emergency power supply system (EPS) with two 10 kV standby diesel generating sets (SDGS) provides independent power to consumers of two trains of safety systems (PSAR Chapter 8) [4].

Normal operation reliable auxiliary power supply system (NO RPS) consists of 10 kV and 0.4 kV sections. Two 10 kV sections (BDE, BDF) under normal conditions are powered from the auxiliary normal operation power supply system through two 10 kV section switches. When a section is de-energized (BDE or BDF), it is powered from the second section by turning on jumper switches from ALT. When both 10 kV sections are de-energized, the supply of 10 kV sections (BDE, BDF) is backed up by 6.3 MW unit diesel generator (MDGS). 10 kV sections power 10/0.4 kV 10BGT11, 10BGT21, 10BGT12, 10BGT22, 10BGT23 transformers. MDGS and 10 kV and 0.4 kV switchgears of NO RPS are accommodated in separate buildings of seismic category 2 designed for design basis seismic impact of OBE-level (S1 level according to the regulations of the Turkey).

In normal operation, when MDGS diesel-generator is in 'stand-by', MDGS diesel-generator fuel-supply system ensures that:

- adequate fuel inventory for the operation of each diesel-generator at rated power for 24 hours is maintained and stored in a tank,
- seven-hour fuel inventory is maintained in feeder-tank,
- fuel is replenished in feeder-tank,
- fuel is separated at regular intervals (diesel fuel is cleaned from mechanical impurities and water using a special separator as scheduled or when quality deteriorates (according to regular tests)).

Fuel and oil inventory for the normal operation auxiliary power supply system is sufficient for at least 24-hour operation of MDGS diesel generator station at rated power and 227 g/(kWh) specific fuel consumption at rated power operation.

Each of the two trains of emergency power supply system (EPS) has a 6.3 MW diesel-generator set (SGDS) connected to one 10 kV section and providing power supply for all loads in one train of safety systems in case of the loss of power from the main auxiliary transformers and standby auxiliary transformers. Diesel generators are started in response to voltage drop or frequency drop in 10 kV section of respective train of emergency power supply system.

SDGS and 10 kV and 0.4 kV switchgears of the two trains of emergency power supply system (EPS) are accommodated in separate buildings of seismic category 1 designed for the maximum design seismic impact of SSE-level (S2 level according to the regulations of the Turkey) (PSAR Chapter 8) [4].

Besides, it is possible to connect 0.4 kV alternative air-cooled diesel generator set (ADGS) to two emergency power supply trains (EPS) to power some of the emergency systems' loads and special engineered safety features required to manage beyond design basis accidents (PSAR Chapter 8) [4].

Each diesel generator (MDGS and SDGS) and all switchgears in trains are physically separated and electrically independent from each other and their consumers.

Each diesel generator (MDGS and SDGS) has a control and alarm panel installed in the diesel room. The diesel generators are started automatically. Remote start is enabled from the MCR and ECR of respective power unit through a channel independent from automatic equipment; remote start is also enabled from the control panel located in the diesel room.

The automation level of each diesel-generator (MDGS and SDGS) allows automatic 'standby' mode, automatic start and step loading, diesel rotation frequency control and generator's terminal voltage control during step up loading and long autonomous operation.

The interval between start-up command and readiness for loading does not exceed 15 sec.

Diesel generators are supplied with diesel fuel from separate intermediate tanks located near MDGS and SDGS buildings. Intermediate tanks are refilled during normal operation through the line from the diesel fuel storage pump station 00UEJ of 1,000 m^3; in case of blackout it is provided by trucks from a storage or the nearest petroleum depot; sufficient inventory of diesel fuel and oil is provided for diesel generators of each power unit. Diesel fuel can be delivered by trucks via receivers directly to intermediate tanks of SGDS or MGDS of any of the power units.

In normal operation, when SDGS diesel-generator is in 'standby' mode, SDGS diesel-generator fuel-supply system ensures that:

- adequate fuel inventory for the diesel generator operation for 72 hours is maintained and stored in intermediate fuel tank,
- eight-hour fuel inventory is maintained in feeder-tank,
- fuel is regularly separated in feeder-tank.

The following is ensured for all conditions of NPP unit operation requiring the operation of diesel generators (MDGS):

- fueling of intermediate fuel tank (directly from the storage during normal operation; fuel from trucks through unloading valves - during loss of power),

- fuel preparation (filtering),
- fuel supply to diesel engine during the entire period of its operation,
- automatic fuel replenishment in feeder-tank,
- ventilation of fuel tanks.

During normal operation, when diesel-generator is in 'stand-by', SDGS diesel-generator lubrication system ensures:

- storage of inventory for at least 15 days,
- replenishment of circulation tank,
- oil filtering,
- diesel readiness for starting within 15 sec,
- lubrication of the generator's friction parts,
- oil supply to the generator and from it.

In all the modes requiring the operation of diesel generator, the system provides:

- oil filtering,
- replenishment of circulation tank,
- oil supply to diesel and generator.

Batteries in the floating-charge mode are used for DC power supply of the power unit. Table 9 provides data on Unit 1 batteries, which are similar for each unit. Storage time is specified for design load of the batteries in case of loss of power from rectifiers (loss of respective AC power sources).

Table 9 Storage batteries

System KKS	Building KKS	Description
10BTB10, 10BTB20, 10BTB30, 10BTB40	10UBA	Reliable emergency power supply system 3480 A·h batteries, 2-hour discharge
10BTA11, 10BTA21	11UBP 12UBP	Emergency 3480 A·h batteries, 2-hour discharge
10BTA12, 10BTA22	11UBP 12UBP	Emergency 3480 A·h batteries, 72-hour discharge

10BTC21, 10BTC41	10UBB	CPS (normal operation power supply) 2320 A·h (1 hour) batteries. These batteries serve as a backup power source in case of a short-term voltage drop in the house-load network to keep CPS rods in a predetermined position for three seconds in order to prevent premature triggering of reactor scram.

5.1.1. Loss of off-site power

The power source outside the Akkuyu NPP site is 380 kV transmission networks of the Turkey. Strong wind can potentially damage transmission lines resulting in the loss of off-site power supply to the NPP site. The equipment of the Turkish power grids and substations on rounded surfaces is designed for wind pressure of 70 kg/m^2. Such constant pressure will create wind with a speed of 34 m/s (within 10 minutes at an average). The annual wind speed of 34 m/s (within 10-minute average) is $1.3 \cdot 10^{-3}$ 1/ year (PSAR Chapter 2) [4].

When such a wind (tornado) occurs, one can conservatively postulate the mechanical destruction of grid components. Since the design involves NPP connection to Turkey's grid through six long 380 kV transmission lines, a complete loss of off-site power is possible only in the NPP area, where the transmission lines are closely spaced. Similar destruction can occur on two 154 kV transmission lines of regional grids connected to 380 kV GIS via 380/154 kV autotransformers. Mechanical destruction of transmission lines will make it impossible to quickly restore off-site power, so this scenario may be considered as a long-term blackout (Long term loss of offsite power).

In addition, the loss of off-site power can be the result of such initiating events as earthquake, extreme temperatures, extreme precipitation and other meteorological phenomena (snowstorms, snow, ice phenomena), lightning strikes, accidents on land and water transport with adverse effects on NPP, aircraft crash, explosions and emissions of flammable or explosive substances, damage to storage facilities and tanks of combustible, explosive and toxic substances on the NPP site, external fires (fire itself and carbon plasma indirectly), salt spray.

The loss of off-site power is an initiating event of category 2 design-basis conditions and refers to the group of initiating events leading to a decrease in secondary circuit heat removal. This initiating event causes the following essential events at each power unit of the NPP:

- trip of normal operation systems, generator trip and turbine trip,
- tripping of all reactor coolant pumps and residual heat removal from the core due to natural circulation of primary coolant,

- reactor scram,
- disconnection of emergency power supply system (EPS) sections and normal operation reliable power supply system sections (NO RPS) from normal operation power supply system (NOS),
- start of emergency diesel generators (SDGS), their connection to EPS 10 kV sections and set-by-step starting of safety systems,
- start of main diesel generator (MDGS), its connection to NO RPS 10 kV sections and set-by-step starting of safety-important normal operation systems,
- actuation of active and passive safety systems and safety important normal operation systems according to their purpose and implementation of safety functions.

Safety functions are ensured due to the start of emergency diesel-generators (SDGS) and automatic step-by-step start of safety systems without operator intervention within 30 minutes.

The operation of core residual heat removal systems ensures stabilization of parameters in the transient process, primary circuit cooldown and the achievement of reactor safe state (cold shutdown). The reactor plant is cooled down through the secondary circuit by SG emergency cooling system powered by emergency diesel generators and PHRS passive heat removal system.

In case of LOCA, active and passive safety systems maintain sufficient coolant inventory in the primary circuit and ensure effective residual heat removal. Additional coolant can be taken from the spent fuel pool (SFP) with further transition of safety systems to recirculation through the containment sump. Heat is transferred through the component cooling system and secured cooling water system to the ultimate heat sink (sea water).

Residual heat from the spent fuel pool is removed by spray cooling system powered by the emergency diesel generators. Heat is removed through the component cooling system and secured cooling water system to the ultimate heat sink. Residual heat from the fuel pool may be removed by emergency and planned primary circuit and FP cool-down system as spray cooling system back-up. If cooling water to heat exchangers is not available, heat is removed from SFP by evaporating water in the pool and supplying water from the spray system, HA-2, HA-3 hydro accumulators or SFP cleaning system tanks. Reliable power may be supplied to SFP cleaning system pumps from unit diesel generator. Process control batteries are recharged from operating diesel generators.

Emergency diesel generators (SDGS) support the operation of safety systems for residual heat removal and stabilization of reactor parameters until the normal power supply is restored. Automatic protection actions of safety systems transfer the reactor plant to a safe state. Unit diesel generator supports the operation of safety-important normal operation systems. The power unit can remain in this mode for an unlimited period and the safety limits will not be breached subject to power supply from diesel generators (with appropriate reliability parameters (PSAR, Chapter 1) [4] and their fuel replenishment. At the same time, the design performance of diesel generators (PSAR, Chapter 8) [4] and auxiliary systems (fuel, cooling, starting air, oil, air intake and gas exhaust) (PSAR, Chapter 9) [4] ensures their long non-stop operation (more than 72 hours) taking into account external climatic conditions arising during an accident.

Calculation analysis made with a conservative assumption of the failure of all normal operation systems (without their operation) and one of the safety system trains, for example, with the assumption of failure of one train of emergency SG cooldown, demonstrates that the protective automatic actions of safety systems bring the reactor plant into a safe state. Acceptance criteria for design basis conditions, when safety functions and integrity of barriers (fuel matrix, fuel cladding, reactor coolant pressure boundary and containment) to the spread of radioactive substances into the environment, are fulfilled.

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

Initiating event with the total loss of power from all AC sources (total plant blackout), including SDGS and MDGS is a beyond design basis accident (BDBA). To manage beyond design basis accidents, the design provides for special engineered safety features, including an alternative 0.4 kV air-cooled diesel generator (ADGS) with a capacity of 2 MW to be used as an additional AC power supply source in EPS after 72 hours from the accident; it is connected to EPS 0.4 kV sections of 11BMA, 11BMS, 12BMV, 12BMD and ensures power supply and operation of the limited number of equipment that is required for beyond design basis accident management (PSAR Chapter 8) [4].

The total blackout of NPP with the loss of all AC power supply sources (except power supply from the batteries for 2 and 72 hours) causes tripping of normal operation systems, all RCPs and SG feed-water pumps, reactor scram, generator trip, turbine trip, closing of BRU-K, shutdown of auxiliary power supply pump systems, emergency cooldown of steam generators, primary volume control, PRZ TEHs and the transient that temporarily deteriorates heat removal from the core and fuel pool for a short time.

For 2 hours, while power is supplied from batteries, safety systems automatically function to perform secondary pressure limiting functions (BRU-A), primary and secondary overpressure protection (PRZ PORV, SG PORV), isolation of failed SG (MSIV) and heat removal from the reactor through secondary circuit by actuating PHRS (see Section 1.1.2), the isolation of containment (isolating valves are closed), and safety injection (control of the valves in ECCS hydro accumulator lines) in case of loss of coolant.

From 2 to 72 hours the management of beyond design basis accident may probably require the operation of some of the process equipment of safety systems and special engineered safety features to perform heat removal from the core and fuel pool including PHRS air damper controls, emergency gas removal from SG collectors and the reactor, pressurizer pilot-operated relief valves (PRZ PORVs), fuel pool drainage valves, line valves of stage 2 and 3 ECCS hydro accumulators (for primary circuit and FP emergency makeup). Some of these loads are fed by 2 hours batteries so for the functionality of these functions AAC must be connected within 2 hours in SBO. 2 hours batteries are going to be discussed in further stages.

A list of machinery powered by batteries (2 and 72 hours) is given in (PSAR, Chapter 8) [4].

During beyond design basis accident after 72 hours, it is assumed that alternative measures will be taken guaranteed during the first 72 hours since the beginning of the accident. The connection of the alternative diesel generator (ADGS) to 0.4 kV EPS sections allows restoring the emergency power supply and starting some process equipment of safety systems and special engineered safety features to perform heat removal from the core and fuel pool (makeup of primary circuit and FP), heat removal from the reactor through secondary circuit (PHRS) as well as limitation of radioactive releases from the containment due to reduced pressure (spray system actuation), leak collection and air filtering in the annulus.

The above valves may be controlled by the operator remotely from MCR/ECR within 72 hours depending on the accident escalation.

Alternative diesel generator also supplies power to DC boards and battery recharging (with a discharge time of 2 and 72 hours) via rectifiers in two trains of the EPS.

DC boards supply power to the accident and post-accident monitoring system and ESFAS in two trains of the EPS.

The list of equipment powered by ADGS is given below (Table 10). Additional equipment connected to the diesel generator (for example, ventilation equipment) will be updated at the stage of detailed design.

Table 10 List of equipment powered by ADGS

Equipment	**KKS Code**
Sump water supply valves (opening when switching to the sump)	11JNA10AA003
	11JNA15AA003
Primary circuit water supply valves (opening)	11JND11AA001
	11JND11AA002
	11JND21AA001

Equipment	KKS Code
	11JND21AA002
PRZ water supply valves (closing)	11JND12AA001
	11JND22AA001
Fuel pool water supply valves	11JMN12AA001
	11JMN12AA002
Reactor water supply valves (through HA-2)	11JMN12AA003
Spray nozzle water supply valves	11JMN14AA001
Ventilation tower connection valves	11KAA25AA001
	11KAA25AA002
	11KAA25AA003
	11KAA26AA001
HA-2 line water supply valves (from JMN)	11JNG14AA001
	11JNG24AA001
	12JNG34AA001
	12JNG44AA001
PHRS lock electromagents	11JNB51AA001
	11JNB51AA002
	11JNB52AA001
	11JNB52AA002
	11JNB61AA001
	11JNB61AA002
	11JNB62AA001
	11JNB62AA002
	12JNB71AA001
	12JNB71AA002
	12JNB72AA001
	12JNB72AA002
	12JNB81AA001
	12JNB81AA002
	12JNB82AA001
	12JNB82AA002
Emergency sampling valves	11KUA07AA801
	12KUA07AA802
	11KUA09AA801
	12KUA09AA802
	10KUA06AA001
	10KUA06AA002
	10KUA08AA001
	10KUA08AA002

Equipment	KKS Code
Alternative component cooling loop pump	11KAA25AP001 (heat generation 9 kW)
Emergency boron injection pumps	11JND10AP001 (heat generation 10.4 kW)
	11JND20AP001 (heat generation 10.4 kW)
Spray pump	11JMN11AP001 (heat generation 9 kW)
Ventilation tower:	10KAA25AC001 (4 modules)

Considering the list of equipment powered by ADGS, the power of an alternative diesel generator shall be 2,000 kW (PSAR, Chapter 8) [4].

BDBA management equipment and gear are started by personnel using hand-operated devices. Equipment that is not involved in BDBA management shall be manually disconnected from the corresponding 0.4 kV section.

Since the unit is equipped with a passive heat removal system (PHRS) via steam generators, one of the batteries of each train powers PHRS gauges for 72 hours according to process requirements.

Appropriate training, exercises and drills will be conducted for NPP personnel to acquire relevant skills for connecting ADGS.

NPP design solutions related to the use of passive safety systems allow reactor core cooling in the event of a failure of conventional active safety systems for a period of time sufficient to take measures to restore power supply systems that supply water for cooling nuclear fuel in the core and fuel pool, and also corrective measures that include actuation of additional BDBA management systems and the restoration of failed equipment. Detailed restoration procedures will be prepared as part of SAMG and BDBAMG.

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

During beyond design basis accident associated with complete loss of all AC power sources, including SDGS and MDGS diesel generators, and from an alternative diesel generator (total plant blackout), only DC power is left from batteries that provide power to consumers in these conditions. The specified time threshold for this state is 72 hours from the accident initiation, during which all the batteries of the NPP units are discharged (PSAR Chapter 15) [4].

The initiating event of the beyond design basis accident is the failure of all (including alternative) AC power sources (plant blackout). The accident analysis considers loss of coolant, which is 2.25 m^3/h corresponding to the maximum possible leakage during rated power operation (PSAR, Chapter 15) [4]. This loss rate includes leaks through RCP seals, leaks through PRZ PORV, sampling and uncontrolled leaks.

Total blackout of the NPP leads to the shutdown of unit normal operation systems, shutdown of all RCPs and SG feeder pumps, reactor trip, generator trip, turbine trip and the transient that leads to a short-term deterioration of heat removal from the core and fuel pool. As a result of failure of SDGS emergency diesel generators, the pumps of safety systems are unavailable, including active part of ECCS and SG ECS. In this case, APCS equipment, I&C, inverters and process loads are powered from batteries (PSAR, Chapter 8) [4].

Closing of turbogenerator stop valves causes secondary circuit pressurization and triggers BRU-A on the steam generators; then BRU-A maintain pressure in the steam generators.

PHRS SG 1÷4 start to maintain pressure upon de-energization of safety system sections and failure of DGs (with a delay of 30 s). PHRS rate stabilization and core decay heat reduction causes secondary circuit depressurization and closing of BRU-A after reaching the relevant set point.

As a result of loss of coolant, the coolant level decreases in the PRZ and primary circuit equipment. Calculations show (PSAR, Chapter 15) [4] that steam enters the hot legs of MCP and further steam generators, where it is condensed, in about 10 hours due to the level decrease in the PRZ and reactor. Generated condensate is added to the liquid coolant and then flows via the circulation circuit to cold legs of MCP and to the reactor. Natural circulation is not disrupted in the primary circuit in the time interval considered (72 hours). The primary pressure does not drop to the ECCS HA setpoint within 72 hours.

Primary coolant mass escaped through leaks after 72 hours is about 67 tons. Steam mass dumped through the secondary circuit steam relief valves during the considered time interval does not exceed 29 tons.

Calculations show that the requirement to reach the safe state of the reactor and the acceptance criteria for maintaining the integrity of physical barriers are met (PSAR, Chapter 15) [4]. The core and reactor internals are not damaged, the reactor is in a subcritical state after scram. Further measures should be taken to manage BDBA (PHRS switch to cooldown mode) and restore failed equipment.

Calculation was also made for BDBA with loss of cooling water supply to the fuel pool for 72 hours subject to the following acceptance criteria:

- highest temperature of the containment achieved in emergency conditions does not exceed 1,200 °C,
- fuel pellets do not melt even locally (the fuel temperature should not exceed 2,540 °C for burnt-up fuel and 2,840 °C for fresh fuel).

Acceptance criteria are fulfilled under the considered conditions during the time period since the initiating event until the start of outcropping of FA fuel part (at least 35.6 hours) located in the fuel pool.

Initial and boundary conditions for this accident, including main objectives achieved during the elimination of this accident, are given in [4] (PSAR, Chapter 15) [4].

In addition, there is a possibility of cooling the fuel pool via the alternative component cooling, which is switched to the first train of the essential-service component cooling system and consists of the pump of alternative component cooling heat exchanger for cooling of the additional system of cooling the fuel pond, piping, and valves. Heat exchanger for cooling of additional system of cooling the fuel pool is switched to the pipes of the alternative component cooling by flexible hoses. Water to the heat exchanger is supplied from the basin of the Mediterranean Sea by portable pump unit. Supply and removal of sea water is performed via the flexible hoses.

The safety of operating personnel in MCR and ECR during such an accident is maintained by independent power supply (which is going to be cleared in the subsequent stages) and life support systems allowing to (PSAR, Chapter 12) [4]:

- remove heat from operating equipment,
- maintain the temperature from +21 to +26 °C and humidity 40–60% in MCR, ECR and personnel rooms,
- create air overpressure at least 20 Pa in MCR and ECR,
- supply outside air not less than 20 m^3/h per person in the mode of filter ventilation.

Indoor air temperature in control safety system, recirculation supply and exhaust ventilation system rooms is maintained during normal operation and accidents from +20 to +27 °C, and under extreme conditions from plus 10 to + 40 °C (PSAR, Chapter 12) [4].

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

Below information depends on the preliminary findings of Preliminary Safety Analysis Report of Akkuyu NPP. However, those findings will be finalized during operation license stage with submission of Final Safety Analysis Report.

The analysis of power loss impact on safety functions and non-safety functions made in the stress test demonstrates that the design solutions ensure NPP safety during both design-basis and beyond design basis accidents with loss of power for 72 hours. Heavy equipment delivery to the NPP site by any transport is not required during this period of time.

In case of any unit de-energization with loss of AC power, transients in the primary and secondary circuits are almost parallel with some variations, because their time lag depends on quite short time periods during which the safety systems are triggered (started) by alarms. During this time, the reactor is brought to subcritical state, secondary circuit pressurization causes BRU-A and SG SVs to open for a short time, heat removal through secondary circuit is provided by PHRS. If the SDGS and MDGS are available, they are actuated, and active safety systems and safety important normal operation systems are started step-by-step. If the SDGS and MDGS fail to start, the only alternative AC source at the unit is an alternative diesel generator (ADGS), and if it fails, the batteries of two EPS trains with the discharge time of 2 and 72 hours. In any type of complete blackout, primary and secondary circuit overpressure protection is achieved by the actuation of BRU-A (SG PORV and PRZ PORV). The key event for the subsequent long-term heat removal from the reactor in all types of complete blackout is the start of passive heat removal system through steam generators.

If internal AC power sources from diesel generators (SDGS, MDGS, alternative diesel generator) remain available during complete blackout of 380 kV GIS (loss of off-site power), the power unit can remain in this state for a long period of time without time limitations and safety limits will not be breached subject to power supply from diesel generators (with appropriate reliability parameters (PSAR, Chapter 1) [4] and their fuel replenishment.

If the alternative diesel generator is lost, the safety limits will not be breached (without fuel melting) for up to 72 hours from the initiation of BDBA (PSAR Chapter 15) [4].

The design solutions relate to the use of passive safety systems, and above all, PHRS and ECCS hydro accumulators that allow effective cooling of the reactor core in the event of a failure of conventional active safety systems for a period of time sufficient to take corrective measures that include actuation of additional BDBA management systems and restoration of failed equipment.

The buildings and structures designed to accommodate emergency power supply equipment meet the requirements for their integrity and availability according to their classification and ensure their protection against probable external natural and human-induced hazards in the NPP location area.

5.1.5. Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

Accident management procedures shall include:

- actions during a blackout with complete de-energization of all four NPP power units,
- actions during complete de-energization of one, two or three power units with their disconnection from the power grid with switchover to auxiliary power or/and isolation of the operating power unit to 154 kV load.

The strategy for managing such accidents should be based on the defense-in-depth concept at the design level to avoid escalation (decrease probability) of design-basis and beyond-design-basis accidents into severe fuel melt accidents due to the complete loss of NPP on-site and off-site power.

It is advisable to have alternative diesel generator packages (considering ADGS performance specified in Section 5.1.2) at each NPP power unit, including mobile equipment provided against external events with a long-term impact on the NPP, its systems and components. And in the upcoming stages it is going to be decided whether additional diesel generators are going to be added to which DID level with their mechanism.

5.2. Loss of the decay heat removal capability/ultimate heat sink

Mediterranean Sea water is the primary ultimate residual heat sink of Akkuyu NPP. The alternative heat sink is atmospheric air.

The stress test with loss of residual heat removal to the ultimate heat sink analyzes the impact of loss of primary ultimate heat sink on safety functions and non-safety functions. The stress test assumes that the required heavy equipment cannot be delivered to the NPP site by any means of transport within 72 hours.

Akkuyu NPP cooling water supply system is designed for heat removal from the reactors and the spent fuel pools in any normal operation conditions and anticipated operational occurrences, including design and beyond design basis accidents, and for cooling water supply to other systems of all four NPP units (PSAR Chapters 9,12) [4].

The cooling water supply system of each unit has direct-flow design with a single circulation of sea water through heat exchange equipment.

A breakwater dike shall be built to intake cooling sea water into the NPP service water supply and to protect the site from wave action (including tsunami wave). The breakwater dike forms a closed loop of water intake structures and the coastline for power units 1-2 and 3-4 (50UZQ and 60UZQ).

Water intake structures of power units 1 and 2, and power units 3 and 4 include six culverts (50UPC and 60UPC) laid along the bottom of the Mediterranean Sea at the base of the breakwater dike (50UZQ and 60UZQ) - three culverts for power units 1 and 2, and three culverts for power units 3 and 4.

Cooling water flows through water intake structures with protection against ingress of marine animals and large debris (50UPC and 60UPC) and fish protection structures (50UPX and 60UPX) into the water intake part of the main pump houses 10UQA, 20UQA, 30UQA and 40UQA of the first and fourth NPP power units interconnected at the ends with essential-service pump houses 11UQC and 12UQC (21UQC and 22UQC), (31UQC and 32UQC), (41UQC and 42UQC) and separated from them by expansion joints preventing their cross-effect under external impacts. Non-essential service pump houses are accommodated in the main pump houses of each power unit.

The water intake part of the pump houses is divided into independent water intake chambers with repair gates allowing repair of separate equipment groups without a complete shutdown of the pump house and equipped with a mechanical cleaning system.

The cooling sea water treatment system prevents various deposits on internal surfaces of equipment and pipelines of the system during operation, including hardness salts, organic and inorganic dispersed impurities, algae and bacteria, and biofouling of surfaces.

Each power unit shall have the following basic systems, which remove heat to the ultimate heat sink:

- PA Main Cooling Water System (Normal Operation System),
- PC Conventional Cooling Water System (Normal Operation System),
- PE Secured Cooling Water System (Safety System).

These systems supply service water to all four NPP power units. Heated water is discharged under residual pressure via a discharge pipe through seal pits into the Mediterranean.

PE secured cooling water trains are fully independent from each other: process parts, control systems, support systems, locations of equipment, pipelines, cables, control components, etc. Hence, owing to the physical separation of trains, failure in one train cannot lead to failure in another train.

The alternative heat sink is atmosphere that is used to remove residual heat from the reactor core through the secondary circuit using passive heat removal system (PHRS) (PSAR Chapter 12) [4].

PHRS is a protective system that performs specified functions in all anticipated operational occurrences and accidents requiring passive heat removal from the reactor to maintain it in a safe state. PHRS is a passive system.

PHRS is closed natural circulation loops for the removal of residual heat from the reactor. The system consists of four natural circulation circuits, one for each circulation loop. Each circuit comprises two heat exchangers, steam and condensate lines with valves, air supply and exhaust ducts, air seals and regulators.

The system is not required during normal operation, PHRS is in standby. The air seals are closed and regulators are open in this mode. Regular tests and inspections are made to maintain the system in good order.

In case of category 2 EPS 0.4 kV section de-energizing, air seals shall open with lag time 30 s. Atmospheric air draught is generated via draught shafts, due to natural draught, and cools down the PHRS air heat exchangers.

The RP can be maintained in 'hot' standby state using PHRS within period of time required to restore AC power supply sources, i.e. at least 72 h after accident initiation. If necessary, the PHRS may be transferred by operator to RP cooldown mode by opening regulators powered from category 1 emergency power system.

In case all AC power sources are unavailable with concurrent primary circuit break, 10JNB50 PHRS operates jointly with 10JNG10 1st and 2nd stage hydro accumulators.

In case of small-break initiating events, residual heat is removed with the RP concurrently cooled down using PHRS designed in such case to decrease pressure to setpoints of HA1 and HA2 hydro accumulators. The same task shall be performed by PHRS in case of primary-to-secondary LOCA (e. g., leak as a result of SG header top cover seal failure).

5.2.1. Design provisions to prevent the loss of the primary ultimate heat sink, such as alternative inlets for sea water or systems to protect main water inlet from blocking

The primary ultimate heat sink is the Mediterranean Sea water, it enters the NPP through three culverts for every two power units, total of six culverts at the NPP. Sea water intake is common for them, then the sea water supply is divided: the first three culverts supply water to the water inlets of pump houses of power units 1-2, and the second three to water inlets of pump houses of power units 3-4. There is no cross connection of cooling water supply by RA, PE and PC systems between power units. PE secured cooling water system trains are independent from each other at each power unit.

All six sea water culverts are independent of each other, which reduces the likelihood of their simultaneous blockage. Water inlets of pump houses of power units 1-2 and 3-4 are also independent of each other.

Each culvert is 3.80x3.80m three-point ferro-concrete pipeline of square cross-section for each line with water intake portals. 10.00 m round water intake portal shall be reinforced with concrete. The inlet is 2.50 m high. The top of the water intake portal is at elevation of minus 10.25 m. Steel bars shall be installed in inlet holes all around the intake portal serving as passive elements to protect the service water supply system against marine animals and large debris.

The fish protection structure with the fish protection system is part of the water intake structure (50UPC) and is mounted on each water intake portal. The fish protection system is an air-bubble screen that prevents algae, jellyfish, plankton, and fish from entering the water intake portal. To do this, two rows of ring perforated pipelines are installed around each water intake portal. In case of algae fouling of the water intake portal, personnel will have enough time to clean the portal.

Air is supplied to fish protection structures via pipelines from the compressor building located near buildings 50UPK and 60UPK. The compressor station power supply is not reliable and the fish protection system does not work in the event of NPP blackout. However, considering that cooling water pumps and conventional-load pumps are shutdown during blackout, the flowrate of sea water through culverts is reduced to the cooling water flow to secured loads, and the fish protection is not significantly affected by the blockage of the ultimate heat sink.

Category II buildings and structures can be destroyed by an external shock wave. The most dangerous is the damage to UPC water intake structures and UPX fish protection structures from an explosion in an accident at sea. However, thanks to the design of the structures, the shock wave cannot disrupt service water intake for cooling secured loads of the NPP.

5.2.2. Loss of the primary ultimate heat sink (e.g., loss of access to cooling water from the river, lake or sea, or loss of the main cooling tower)

If the primary ultimate heat sink is lost, the power unit is shut down by process parameters, while the standby power supply of the power unit is preserved. The loss of the primary ultimate heat sink leads to loss of the secured and conventional cooling system of the power unit with the preserved auxiliary power supply system of the power unit. The loss of the ultimate heat sink can be considered as a design-basis accident with the failure of residual heat removal of the reactor similar to the loss of all AC power sources.

If residual heat cannot be removed to the ultimate heat sink (Mediterranean Sea water) through the secured cooling water system, then it can be continuously discharged through SG secondary circuit into the atmosphere (alternative ultimate heat sink) by the passive heat removal system, so that the reactor can be brought into a controlled state that can maintained. The operation of the passive heat removal system is not limited in time and shall be sufficient for removing residual heat from the reactor core and does not require special accident management actions.

Category 1 emergency power supply system powers electrically driven elements of PHRS (10JNB50) during accidents requiring PHRS to operate in SG cooling mode. SG pressure is maintained by direct-acting (from its own media) passive regulators of PHRS without the power supply system required.

Automatic control is the main type of control. The system may be also actuated by the operator from keys in MCR and ECR. During design and beyond design basis accidents, the system is started automatically by alarms or started passively in case of blackout (opening of valves when the holding electromagnets are de-energized).

PHRS is a passive part of the emergency heat removal system through secondary circuit designed to perform specified functions under the following conditions:

- according to the design of protective safety systems, the system has four independent circulation circuits. According to the design performance of equipment, three available circulation circuits are sufficient for the system to perform its functions in full in any conditions requiring it (i.e. PHRS has capacity margin),

- under the most adverse external conditions (ambient air temperature plus 45 °C) and steam pressure in the steam generator is 7.0 MPa, the total capacity of PHRS (8 heat exchangers) is 80.34 MW. At the initial stage of the accident during PHRS rate stabilization, the residual heat is removed by steam dumping from SG into the atmosphere through fast-acting relief valve for steam discharge into atmosphere (BRU-A),
- the design allows passive automatic start of the system (it does not require power supply from external sources or operator intervention).

During normal operation accompanied by secondary circuit pressurization (e.g. partial load shedding of the turbogenerator), PHRS can automatically start, but the system does not reduce the reliability of its functions and does not lead to abnormalities and exceeding of normal operation limits.

Wind loads cannot affect the availability of the passive heat removal system, since the PHRS removes heat under wind loads up to 90 m/s, which is significantly higher than the maximum wind with 0.01% probability (once in 10,000 years).

The natural circulation in PHRS can be compromised during tornado. However, the tornado effect is of a transient nature, since the tornado moves along with the cloud that generates it, and this movement can give speeds of tens of kilometers per hour. Once the tornado has passed, the natural circulation in PHRS is restored.

PHRS is available under external extreme (possible once in 10,000 years) air temperatures, which have a short-term effect on heat removal (due to diurnal fluctuations in air temperature):

- minimum - minus 14.1°C,
- maximum - plus 50.4°C,

In case of loss of coolant accidents, the passive systems – first and second stage hydro accumulators provide sufficient coolant inventory in the primary circuit and the passive heat removal system efficiently removes residual heat. If necessary, third stage hydro accumulators may be engaged.

Residual heat in the spent fuel pools can be removed by water evaporation in the pools and by supplying water from the spray system, HA 2, HA-3 hydro accumulators or SFP cleaning system tanks during the operation of active systems.

The analysis of the beyond design-basis accident with the loss of the primary ultimate heat sink for 72 hours demonstrates that the criteria for emergency core cooling are met.

Akkuyu NPP has once-through cooling system with single circulation of sea water (primary ultimate heat sink) without cooling towers.

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

The loss of the ultimate heat sink can be considered as a design-basis accident with a disruption of residual heat removal similar to the loss of all AC power sources (PSAR Chapter 15) [4].

In order to maintain residual heat removal from the core beyond 72 hours, the designed jump-over line between the pipeline of the first train of 10JMN spray system and the drainage lines of second stage hydro accumulators can be used.

The loss of an alternative heat sink - PHRS passive heat removal system through the SG secondary circuit - might cause escalation of a beyond design basis accident into a severe accident with damage (melt) of nuclear fuel in the reactor.

Residual heat in the spent fuel pools can be removed by water evaporation in the pools and by supplying water from the spray system, HA 2, HA-3 hydro accumulators or SFP cleaning system tanks during the operation of active systems.

Measures shall be taken to restore the functional capability of the systems which remove heat to ultimate heat sink, including the use of additional mobile equipment to provide cooling water supply beyond 72 hours. Termination of heat removal from fuel in the fuel pool will cause a beyond design basis accident with escalation into a severe accident with damage (melt) of nuclear fuel.

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

Below information depends on the preliminary findings of Preliminary Safety Analysis Report of Akkuyu NPP. However, those findings will be finalized during operation license stage with submission of Final Safety Analysis Report.

All initiating events leading to loss of ultimate heat sink eliminate active safety systems and trigger passive safety systems that do not depend on the loss of ultimate heat sink. The operation of the passive heat removal system maintains the reactor in subcritical state and ensures sufficient safety margin. Engineering solutions made in the design prevent fuel damage in case of loss of ultimate heat sink.

The analysis of the beyond design-basis accident with the failure of the primary ultimate heat sink for 72 hours demonstrates that the criteria for emergency core cooling are met.

In case of beyond design basis accidents with full loss of design heat removal to the ultimate heat sink without breaks of primary circulation pipelines, second and third stage ECCS hydro accumulators may be used to supply boron solution to the fuel pool to cool spent fuel.

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

EOP procedures shall include operator actions during beyond design basis accidents with the loss of ultimate heat sink to remove residual heat from the reactor core and fuel pool.

The strategy for managing such accidents should be based on the defense-in-depth concept at the design level to avoid escalation (decrease probability) of design-basis and beyond-design-basis accidents into severe fuel melt accidents as a result of loss of ultimate heat sink.

5.3. Loss of the primary ultimate heat sink, combined with station black out

5.3.1. Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool (e.g., start of water loss from the primary circuit)

Loss of the primary ultimate heat sink combined with the plant blackout is a design-basis accident with nuclear fuel integrity preserved for up to 72 hours after the initiation of BDBA (PSAR Chapter 15) [4].

As a result of primary LOCA with a rate of 2.25 m^3/h at nominal pressure of 16.2 MPa (leaks through RCP seal, leaks through PRZ PORV, sampling lines and uncontrolled leaks), the level in the reactor begins to decrease 10 hours after the accident initiation. After 24 hours, the coolant level in the reactor is reduced to the level of outlet nozzles and steam enters hot legs of the MCP and further into the steam generators where it is condensed and residual heat removed through SG secondary circuit. PHRS maintains natural coolant circulation and primary circuit cooldown. When the primary pressure drops to ECCS HA-1,2,3 setpoints, the primary circuit leakage can be compensated.

The analysis of beyond design basis accidents both with failure of all AC power sources for 72 hours and/or loss of the ultimate heat sink demonstrates that the criteria for emergency core cooling are not failed. The maximum fuel cladding temperature does not exceed the allowable values. There are no conditions for oxidation of fuel claddings. There are no conditions for control rod melting and for deformation of fuel assemblies and fuel elements.

The analysis of the beyond design basis accident with failure of fuel pool heat removal demonstrates that in the conservative scenario of filling spent fuel pool with emergency fuel unloaded from the core with high residual heat (≈ 19 MW), water will start to boil in the pool about 3 hours after the loss of SFP cooling. After reaching the saturation temperature, water boils out of the SFP accompanied by water level decrease in the SFP. As a result of steam generation, natural circulation intensifies in FAs lifting the steam-water mixture in FAs and lowering water heated to the saturation temperature in the inter-assembly space. SFP with emergency fuel will boil off in 35.6 hours before fuel uncovering.

When the water level reaches the fuel part of FAs, fuel elements start heating up due to decreased heat removal from fuel sections located in the vapor medium to steam.

Under the considered conditions, acceptance criteria are met within the time from initiating event to the beginning of uncovering of fuel part of FA (the assumed time is not less than 35.6 h) stored in SFP. These criteria are further maintained, if before the expiry of the specified time from initiating event, borated water will be supplied from the ECCS passive part hydro accumulators into the SFP with the minimum flow rate of not less than 31 m^3/hour.

If there is scheduled unloading of spent nuclear fuel in SFP with less residual heat (≈ 8 MW), the loss of SFP cooling will cause water to boil in SFP in 5 hours, and fuel elements will start to uncover in 93 hours after the initiation of the beyond design basis accident with loss of SFP cooling.

Fuel damage in the fuel pool can be prevented by restoring power supply from ADGS and JMN pump to make-up the SFP with water both from the primary circuit and from containment sumps.

5.3.2. External actions foreseen to prevent fuel degradation

The design solutions ensure the safety of Akkuyu NPP power units during both design-basis and beyond design basis accidents with loss of power and ultimate heat sink for 72 hours without fuel damage both in reactors and spent fuel pools (with operator actions to makeup SFP from the hydro accumulators of ECCS passive part, when required).

According to guidelines for beyond design basis accident management (BDBAMG), during this time, decisions shall be made and implemented to restore the functions of heat removal to the ultimate heat sink and to restore power supply that transfer NPP power units to a safe state ensuring the removal of residual heat of nuclear fuel in reactors and spent fuel pools of power units. Operator action strategies in BDBAMG will be developed and justified at further stages of the project. Operator action strategies in BDBAMG shall include the restoration of on-site and off-site power supply of the NPP, restoration of conventional and secured cooling systems and prevention of beyond design basis accident escalation into severe nuclear accident with fuel meltdown.

5.3.3. Measures, which can be envisaged to increase robustness of the plants in case of loss of primary ultimate heat sink, combined with station black out

NPP management and qualified operating personnel shall be trained and prepared for actions to manage accident with NPP blackout and/or loss of the ultimate heat sink.

It is advisable to have alternative diesel generator packages (considering ADGS performance specified in Section 5.1.2) at each NPP power unit, which, if necessary, may be used in the event of a BDBA with loss of the primary ultimate heat sink in combination with the plant blackout.

6. SEVERE ACCIDENT MANAGEMENT

There is no approved Emergency Plan and no Severe Accident Management Guidelines of the Applicant at the time of writing of this report. The plans and procedures to be developed is expected to reflect IAEA recommendations and world experience after the Fukushima accident. Requirements and expectations on the quality of the emergency plans and emergency preparedness activities of the Applicant are summarized below.

6.1. Organization and arrangements of the licensee to manage accidents

The main responsibilities of the Applicant are to possess sufficient powers, financial, material and other resources necessary for operating NPP, managing accidents at power units and emergency planning for the protection of personnel on site (in severe accidents). The following duties should be performed to fulfill these responsibilities:

- operation of NPP power units (according to the design basis, procedures for operation, maintenance and repair of systems, structures and components), monitoring of operational limits and conditions of power units in accordance with the safe operation regulations,
- accident management at NPP power units in accordance with emergency operating procedures and guides,
- emergency preparedness and response during severe accidents according to NPP personnel emergency protection plan and interaction with third-parties according to the off-site emergency plans (National Radiation Emergency Plan and Provincial Radiation Emergency Plan).

The overall goal of emergency planning (in case of a severe accident) is to establish a management system and take protective measures (to reduce staff and public exposure), possible immediate measures to save lives, as well as severe accident management measures to reduce the spread of radioactive substances off site. Emergency preparedness and response protect NPP personnel and public off the site.

Emergency preparedness and planning measures are determined by the relevant requirements of the Licensing Basis of Regulations, Standards and Guidelines for Akkuyu NPP. Planning philosophy, description of emergency response process, requirements for the development and updating of emergency plans and manuals, emergency training and drills are defined in the National Radiation Emergency Plan (URAP). URAP is in the final draft form and is about to be approved by the relevant authority. Based on these regulations, the operating organization (licensee) shall develop its specific emergency planning documents.

Emergency plans include two main documents:

- NPP emergency plan, which determines the licensee's actions to mitigate the radiological consequences of severe accidents on the site in coordination with off-site public protection measures,
- the off-site emergency plan that regulates zones and distances of emergency planning and defines actions to be taken by authorized local and state authorities to protect the public, property and the environment in the event of an emergency (severe accident). The public and the environment are protected by implementing radiation emergency action plans developed for local authorities (Mersin Province), companies and the facility (NPP) involved in the local response that are associated with the National Radiation Emergency Plan and Disaster Response Plan of Turkey.

The organizational chart and established responsibilities, authorities, duties and roles of all officials and departments, as well as the procedure of their internal and external cooperation of Akkuyu NPP have been developed and presented in the Chapter 13 of the Preliminary Safety Analysis Report (PSAR) of Akkuyu NPP [4]. NPP shall be operated by a sufficient number of qualified personnel and Chapter 13 of PSAR includes the Akkuyu NPP organizational chart which will be updated at subsequent design stages. Operations are carried out by operating shift personnel, who should ensure the safety of NPP units, reliable operation of all systems, structures and components, as well as accident management. According to the Regulation on Specific Principles for Safety of Nuclear Power Plants [24], this organization and other details related to the on-site emergency preparedness and response will be given in the on-site emergency plan of Akkuyu NPP. The on-site emergency plan of Akkuyu NPP should be prepared before the delivery of the nuclear fuel to the NPP. The emergency plan to be prepared by the operator shall meet the requirements of Akkuyu NPP Licensing Basis. According to the Appendix A 12.3 of the EPR-Method 2003 Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency document of the IAEA [25], the on-site emergency plan to be prepared by the nuclear power plant operator should generally include at least the following topics:

- classification of the emergency and the identification of the underlying events and plant parameters when this classification is made,
- formation of on-site emergency response organization and determination of responsibilities,
- principles of response activities to be carried out on-site and preparations for response activities,
- arrangements for alarms, notifications and communication,

- emergency management and realization of the situation assessment,
- emergency worker’s safety and protection from radiation,
- radiological monitoring to be carried out in the facility, on-site and near the site during emergency situations,
- information to be provided for off-site emergency response organization to inform the public about the emergency situation,
- centers and points to be used during emergency response, equipment and other additional units,
- termination of the emergency and remedial actions,
- implementation instructions of the emergency response organization,

In the event of NPP operational occurrences that could lead to a radiation emergency, operational and administrative management of the NPP shall take the following organizational emergency response measures:

- operating shift personnel shall determine the nature of the accident at the NPP unit, possible consequences associated with the release of radioactive substances, and report an emergency situation (in the order of subordination) up to the plant shift supervisor,
- the plant shift supervisor shall identify the emergency situation, make an announcement according to the list of NPP operational occurrences, which could lead to a radiation emergency, provide information on the operational occurrence within the established time limits, and supervise measures for protecting the plant personnel at the initial stage (before the emergency committee is assembled),
- having been notified, the NPP director (chief engineer) shall instruct on assembly of an emergency committee and, after assessing the operational occurrence and predicting its development, make a decision to declare ‘Alert’, ‘Emergency’ at the NPP and implementation of the on-site emergency response plan,
- heads of the NPP departments shall supervise alerting of personnel (via the local alert network and communications) and take the necessary protective measures for the personnel (use of personal protective equipment and iodine prophylaxis, shelter in administrative and industrial buildings and special facilities on site or evacuation off site),

- emergency committee shall manage emergency actions, NPP services and resources to localize the operational occurrence and mitigate its consequences on-site, as well as cooperation with local and state authorities to provide emergency measures to protect the population off-site.

Implementation of the on-site emergency response plan in the event of an accident at the Akkuyu NPP provides:

- plant personnel alert,
- start of NPP emergency management bodies,
- radiation, chemical and general reconnaissance,
- radiation and engineered protection of personnel,
- medical protection of personnel,
- physical protection of NPP,
- logistics,
- evacuation measures.

6.1.1. Organisation of the licensee to manage the accident

The details of the emergency management organization to be established on-site during the emergency shall be explained in the on-site emergency plan. In particular, information such as who will be the emergency manager, who will be in charge until the emergency manager arrives, who will be responsible for communicating the decisions that are taken by the emergency manager, and who will be responsible for ensuring their implementation should be given. The main activities to be carried out by the emergency response manager in support of the on-site emergency response organization and the timing objectives for these activities should be determined. It should be taken into account that the emergency response organization is ready to full capacity when the timing objectives for the actions to be implemented during the emergency are determined.

The emergency classification is one of the most important points to perform an effective emergency management. According to the IAEA technical document "Generic assessment procedures for determining protective actions during a reactor accident" IAEA-TECDOC-955 [26], there are three classes of emergency for nuclear power plants: general emergency, site area emergency and alert. Declaration of an emergency related to one of these classes immediately and directly leads to the initiation of emergency response.

Criteria for classification of emergency are the pre-defined emergency action levels (EALs), which are immediately used for making decisions on the implementation of emergency measures corresponding to the class of emergency. The EALs are determined by the applicant organization and evaluated by TAEK.

URAP (National Radiological Emergency Plan) suggests the following emergency classes related to possible off site consequences due to nuclear power plants:

- General emergency is an accident with actual or significant risk of radioactive release or radiation exposure which presupposes off-site emergency response activities. When this class of emergency is declared, measures for mitigation of consequences and protection of people must be taken urgently on-site and off-site,
- Site area emergency is a significant deterioration of protection level for people on site or near the NPP. When this class of emergency is declared, remedial actions and measures for on-site protection of people must be taken urgently, and arrangements shall be made to take the off- site protective measures if necessary,
- Facility emergency is a significant deterioration of protection level for people on site. When this class of emergency is declared, remedial actions and measures for on-site protection of people must be taken urgently,
- Alert is an accident with an indefinite or significant deterioration of protection level for the public or people on site. When this class of emergency is declared, measures for evaluation and mitigation of consequences and enhancement of preparedness of on-site and off-site responsible organizations shall be taken urgently.

Moreover, to manage accidents at NPP power units, the operator shall organize the development and analytical justification of the following documents: Emergency Operating Procedures (EOP), Beyond-Design-Basis Accident Management Guide (BDBAMG) and Severe Accident Management Guide (SAMG). These documents should be submitted to regulatory body and approved. After the approval of these documents, operating shift personnel of NPP shall be trained in applying emergency operating procedures and guides (EOP, BDBAMG and SAMG) using a full-scale simulator of the Main Control Room (MCR).

Emergency Operating Procedure (EOP) consists of procedures divided into two parts designed to manage anticipated operational occurrences and design-basis accidents. Each EOP procedure (PSAR Chapter 13) provides a brief description of the reactor's initial state, symptoms of operational occurrence or accident, design automation algorithms, MCR/Emergency Control room (ECR) operator actions (step-by-step) and final state of the power unit. The goal of step-by-step operator actions is to diagnose the initiating event, select the appropriate EOP procedure, monitor automated control actions and transfer the power unit to a controlled (hot) or safe (cold) state, and take these actions manually in the event of automation failure. The status of critical safety functions (CSS) and their effectiveness shall be monitored by MCR/ECR operators using information from the safety parameters display system (SPDS). EOP procedures for the management of design-basis accidents are applied by operators for initiating events of accidents that trigger reactor scram and/or safety systems. These EOPs include event-based emergency operating procedures; symptom-based emergency operating procedures (in terms of optimal recovery procedures).

Beyond-Design-Basis Accident Management Guide (BDBAMG) includes symptom-based procedures that define MCR/ECR operator actions in case of initiating events and additional failures that were not covered in the design basis accidents. When managing a beyond design basis accident, it is necessary to prevent fuel damage (melting) in the reactor and fuel pool, protect physical barriers to the spread of radioactive substances and take actions to mitigate beyond design basis accidents. Symptoms (power unit parameters and their numerical values) are used to diagnose the current status, monitor and restore critical safety functions, determine entry conditions, necessary operator action and transition conditions to be updated after the BDBAMG calculation justification.

In the symptom-oriented approach, the main objectives of BDBAMG are achieved by monitoring and restoring the following critical safety functions (CSF):

- F-0 “Operability”,
- F-1 “Reactor Core Subcriticality”,
- F-2 “Reactor Core Cooling”,
- F-3 “Primary-To-Secondary Heat Removal”,
- F-4 “Primary Circuit Integrity”,
- F-5 “Containment Integrity”,
- F-6 “Primary Coolant Inventory”.

Severe Accident Management Guide (SAMG) includes general provisions, accident management rules and symptom-based guidelines (recommended) that define MCR/ECR operator actions in case of heavy fuel damage (melting) in the core and melt escape beyond the reactor vessel. SAMG elements are diagnostic flowchart, serious threat tree, MCR operator manuals, long-term monitoring manuals, ancillary computation tools, SAMG exit guidance.

6.1.2. Possibility to use existing equipment

The design of each Akkuyu NPP power unit includes the following engineered safety features for managing severe accidents:

<u>Hermetic enclosure system (double containment):</u> Confinement and limitation of radioactive releases through the primary (inner) prestressed reinforced concrete containment

and protection from external hazards (airplane crash, air shock and extreme weather conditions) by means of a secondary (outer) cast-in-place reinforced concrete containment.

Beyond-vessel core melt catching and cooling system (JKM): Retention, cooling, and subcriticality of the melt (after escaping the damaged reactor vessel) to maintain the integrity of the primary containment.

Containment hydrogen concentration monitoring and emergency removal system (JMTJMU): Monitoring and preventing the formation of explosive mixtures in the primary containment with passive catalytic hydrogen recombiners.

PRZ PORV extra control line (JEF) and emergency gas removal system (KTP): Pressure reduction in the primary circuit down to 1 MPa by discharging medium from the PRZ into the containment in case of severe accidents before the melt escape from the reactor vessel.

Spray system (JMN): Injection to limit the pressure and temperature in the containment, introduction of chemicals to bind radioactive iodine in the vapor-gas mixture and limit leakage of radioactive substances through the containment, supply of coolant to the reactor or fuel pool to cool fuel subject to emergency power supply from ADGS.

Annulus ventilation and filtration system (KLB22): Collection and filtration of leaks through the primary containment to reduce the release of radioactive substances into the environment subject to available emergency power supply from ADGS.

Emergency power supply system (with total loss of power supply from all AC sources and full discharge of all batteries after 72 hours): Connection of 0.4 kV alternative aircooled diesel generator set (ADGS) to provide power to consumers needed to manage a severe beyond design basis accident.

Special service water supply facilities (KAA25 alternative component cooling pump and PEC10 mobile pumping unit): Cooling of spray system (JMN) components and heat removal to the ultimate heat sink (seawater) subject to emergency power supply from ADGS.

Provision and Management of Feedstock and Materials (fuel for diesel generators, water, etc.): Sufficient water and fuel shall be provided for diesel generators (SDGS, MDGS and ADGS) during the plant operation. Diesel generators of the emergency power supply system (SDGS) at each NPP power unit are provided with fuel from intermediate tanks with a volume sufficient for the operation of each diesel generator within 72 hours. Intermediate tanks are refilled during normal operation through the pipeline from the common-plant diesel fuel storage 00UEJ of 1,000 m3; during NPP blackout it is provided by trucks from a warehouse or the nearest petroleum depot with sufficient inventory of diesel fuel and oil for diesel generators (SDGS and ADGS). Diesel fuel can be delivered by trucks via receivers directly to intermediate tanks of SGDS of any of the power units. Fuel supply systems are autonomous for every diesel generator.

Communication and Information Systems: NPP communication and alarm systems are designed for efficient, reliable and stable operative control in its daily operation and emergency situations, receiving centralized annunciation signals and conveying them to the NPP personnel, timely informing personnel about emergency situation at NPP, on-site civil defense forces and teams, duty services of civil defense and emergency (emergency response) management body, as well as population, managers, and personnel of facilities situated in the coverage zone of local annunciation system. The use of internal communication hardware allows personnel to access offsite communication networks, local PA system and accident management facilities along main and back-up communication channels. The list and amount of NPP communication and alarm facilities is determined in accordance with the requirements of guiding documents for NPP communication systems. The systems shall consist of two highly reliable sets for off-site and on-site communication channels. Off-site communication facilities include main and standby (including direct ones) channels of all necessary kinds of communication: telephone, (including governmental) data transmission and others. To ensure the survivability of the system, it includes various independent communications lines (directions): wired (including fiber-optic), radio relay, radio, and satellite.

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

6.1.3.1 Unhabitability of NPP control rooms

The units of Akkuyu NPP has two control rooms (MCR and ECR), which have communications (communication systems, radiation monitoring, post-accident monitoring and safety parameters) with the on-site emergency management center of (PECP). NPP PECP is designed to coordinate the management of severe accidents at power units, implement emergency response plans to protect personnel (on-site) and cooperate with off-site authorities and centers.

MCR/ECR air-conditioning and life-support systems have two independent trains (safety class 2 and seismic category 1) that perform the following functions:

- maintaining the temperature 21-26 °C and humidity 40-60% in MCR/ECR,
- creating overpressure of at least 20 Pa in MCR/ECR,
- supplying outdoor air at least 60 m^3/h per person during normal operation and 20 m^3/h per person during filter ventilation,
- life support of rooms and personnel safety at MCR/ECR for managing accidents and taking emergency protective measures (for on-site events with toxic, chemical and radioactive substances in outdoor air).

MCR/ECR are fully isolated during emergencies for a time sufficient to measure the concentration of harmful (radioactive and chemical) substances in outdoor air near air intakes of MCR/ECR air conditioners. In addition, this state is introduced when there is a danger of contamination of outside air with toxic substances, carbon monoxide (in case of fire) and other harmful substances that are not trapped in filter ventilation mode.

In the state of MCR/ECR full isolation, the operator closes sealed valves on supply and exhaust ducts and starts the life support cylinder station, which has emergency power supply from batteries or an alternate diesel generator (in the event of plant blackout). The life support cylinder station has 15 cylinders with a capacity of 185 liters each with an air pressure of 24.5 MPa and a purge line that delivers 20 m^3/h of air per operator (total 80-130 m^3/hour per control room) and creates overpressure not less than 20 Pa in MCR/ECR. This allows operating personnel to perform their control functions with overpressure in MCR/ECR for 6 hours.

In the event of unhabitability or failure/damage to control systems at the MCR, operators can move to ECR of the respective power unit or PECP, which have independent accident management systems, as well as communication, air conditioning and life support systems.

The protected emergency command post (PECP) serves the following functions:

- NPP crisis centre (technical support center of the plant), where the emergency management team is assembled to maintain communication with MCR/ECR operators, provide general emergency response management, implement emergency response plans for NPP personnel protection and cooperation with third-party organizations in the event of emergency,
- emergency localization and mitigation center (when an accident escalates beyond the local incident to a regional scale, in cases of emergencies of peace and wartime),
- data center of parameters critical for the safety of power units, monitoring and forecasting of radiation situation on-site and off-site (in the emergency preparedness zones and distances),
- simulator for on-site emergency response managers.

NPP PECP is made of cast-in-place reinforced concrete, protected from external hazards and has the following independent supporting systems necessary for performing emergency functions:

- PECP automated radiation monitoring system (ARMS),
- HVAC systems,
- water supply and drainage systems,

- power supply system with two diesel generators as a backup source (for loss of off-site power),
- lifting gear for the repair of diesel generators,
- information, communication and lighting systems (weak current circuits),
- fire extinguishing systems.

6.1.3.2 Potential failure of accident and post-accident monitoring system

During accidents with total blackout of the plant (when the batteries are discharged within 72 hours at NPP power units), MCR/ECR operators may lose the following emergency and post-accident monitoring systems:

- monitoring the integrity of protective barriers (fuel matrix, fuel cladding, primary circuit pressure boundary and containment) to the spread of radioactive releases using an automated radiation monitoring system (ARMS) in the containment and on the NPP site,
- monitoring hydrogen concentration in primary containment rooms (JMU),
- monitoring the state of isolation valves, temperature and pressure parameters in the primary containment (for prolonged heating during a severe accident),
- monitoring parameters in the beyond-vessel core melt catching and cooling system (JKM), including monitoring the temperature rise when the melt escapes beyond the damaged reactor vessel.

Monitoring system sensors have sufficient measuring ranges and are certified for environmental emergency conditions that are constantly checked during inspections and tests to maintain control over the integrity of protective barriers and parameters in the containment, Containment integrity gauges (monitoring of hydrogen concentration, temperature and pressure) shall be resistant to severe accidents.

It is necessary to take organizational measures to restore emergency power supply to accident control and post-accident monitoring systems by connecting an alternative diesel generator (ADGS) or special mobile devices if early signs of a possible loss of control and monitoring of containment of NPP power units

6.1.3.3 NPP site location parameters complicating emergency response measures

The factors related to the existing infrastructure of the Akkuyu NPP site which might complicate the evacuation of personnel and public during an emergency are:

- limitations of ways to settlements located near the NPP,

- terrain features that contribute to formation of local centers of radioactive contamination in case of radiation emergency at the NPP.

The limitation of ways to human settlements located near the NPP will significantly complicate measures to protect the public in the event of radioactive release spreading in northern, north-east or north-west direction and radioactive contamination of Adana-Antalya highway, which runs about 3 km from the NPP.

In case of tsunami, access roads to Akkuyu NPP and the nearest settlements are at a higher altitude than the NPP, so it would not be destroyed or blocked. The Akkuyu nuclear power plant is the end point of access roads, therefore, in case of a radiation emergency, there will be no traffic towards the plant, and nearby settlements should be evacuated.

The main reason that could complicate emergency response measures (including delivery of human and material resources to the site) is the potential loss of road infrastructure due to high radioactive contamination at the point of intersection of Adana-Antalya highway with plant access roads and to settlements located in the urgent protective action planning zone.

These difficulties can be mitigated by taking the the off-shore shipping route into account as an alternative evacuation route using a berth for receiving heavy loads. Moreover, the development of additional infrastructure during the NPP construction (alternative transport routes to the NPP and nearby settlements taking into account topographic features) and relevant emergency plan measures will ensure evacuation as a priority safety measure, and supply of human and technical resources for accident mitigation.

6.1.3.4 Natural factors that may affect the emergency response

The following natural hazards among which the prevailing external events are earthquakes and tsunamis may impede the emergency measures in the Akkuyu NPP area:

- geological/topographical risk factors (landslides, rock displacement hazard),
- hydrological risk factors (coastal floods, intensive local precipitation),
- meteorological risk factors (inversion, fog, typhoons/tornadoes),
- seismic risk factors (seismic ground motion, fault displacement).

Despite the fact that these hazards could impede response to radiation emergency, they are not critical and can be mitigated by means of appropriate emergency preparedness, such as:

- improve the robustness of emergency planning (by distributing emergency teams and medical facilities; increasing seismic protection of buildings and roads, strengthening the foundations and support structures for engineering activities),

- backup emergency planning measures (redundancy of communication systems and alarm systems, utilities, access ways to the plant and evacuation routes),
- detailed analysis of natural hazards in the development of emergency plans for public and personnel protection in case of radiation emergency.

On the other hand, many weather events may also influence the implementation of emergency plans. Inversions and fogs can deteriorate visibility making it difficult to evacuate. Heavy storms, hail, tornadoes can damage evacuation routes and equipment if these weather events occur during the emergency response. Moreover, routes of evacuating people from settlements in the Akkuyu NPP site vicinity can be blocked as a result of road damage by landslides or rock falls. Communication and power lines can also be damaged. Moreover, landslides can damage the infrastructure for emergency plans, shelters for the public and special aid facilities (medical aid and decontamination stations). Therefore, it is necessary to provide numerous emergency evacuation routes, mobile power supply sources, shelters, and also develop maps of areas prone to landslides (soil or rock).

6.1.4. Conclusion on the adequacy of organisational issues for accident management

The main conclusions on the adequacy of Akkuyu NPP organizational structure for the management of power unit accidents and emergency response (in case of a severe accident) can be made after the development of emergency operating procedures and guides (EOP, BDBAMG and SAMG), as well as on-site emergency response plans for personnel protection and the compatibility of the on-site emergency plan with the off-site emergency plans (national and provincial radiation emergency plans). Additional information on the adequacy of the organizational structure can also be obtained through training and drills (emergency exercises) of operating personnel on the full-scale simulator of the MCR and review of emergency response plans (readiness of emergency response teams).

6.1.5. Measures which can be envisaged to enhance accident management capabilities

The following measures shall be envisaged to improve accident management capabilities:

- organizational arrangements to improve the ability to use existing equipment, eliminate factors that may prevent severe accident management, and emergency planning to identify logistical and human resources for preparedness and response to radiation emergencies,
- development of an emergency classification system to be interrelated with a system of operating criteria (emergency action levels that are measured parameters of the plant) and with an emergency classification system defined in IAEA documents and national radiation accident response plan,

- development and analytical substantiation of severe accident management operator actions (SAMG), Emergency Operating Procedures (EOP) and Beyond-Design-Basis Accident Management Guide (BDBAMG),
- analysis and necessary measures in the event of simultaneous accident at several NPP power units and the impact of several hazards on the entire site, the loss of NPP protected emergency command post, the loss of emergency and post-accident monitoring systems, communication and alarm systems in the event of a complete blackout for more than 72 hours, the use of mobile means to restore emergency power supply, the forced need to implement radiation-risky protective measures to evacuate personnel and population in emergencies (in case of a severe accident),
- development of additional technical means of communication with off-site agencies to provide organizational support for accident management measures, accessibility of resources of various off-site civil structures and agencies, fire-fighting equipment, civil defense shelters and rooms suitable for deployment of crisis centers and personnel when evacuating off site,
- development and implementation of additional measures to ensure the stability of communication channels and interface between various components of the accident management and emergency response system both on-site and off-site (local and national level),
- development of a reliable on-site and off-site radiological monitoring system which is operable in emergency conditions.

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

6.2.1. Before occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes (including last resorts to prevent fuel damage)

In case of a beyond design basis accident with loss of the core cooling function, the operator shall take actions to restore the critical safety functions aimed at preservation of physical barriers and prevention of fuel melting in accordance with the symptom-based procedures of the Beyond Design Basis Accident Management Guide (BDBAMG) to be developed and substantiated at a later stage of the project.

At the current stage of the Akkuyu NPP project, the analysis of beyond design basis accidents was conducted to determine initiating events in order to demonstrate the integrity of physical barriers without fuel melting within 72 hours (without operator’s actions) on the basis of assessment of compliance with the following acceptance criteria (PSAR Chapter 15) [4]:

1) Pressure in the primary circuit and steam lines of SG shall not exceed 115% of the design value, i.e. not higher than 20.29 MPa and 9.71 MPa, respectively,

2) Fuel pellets do not melt even locally (temperature is less than 2,540 °C for burn-up fuel and less than 2,840 °C for fresh fuel),

3) Criteria for emergency core cooling:

- maximum temperature of fuel element cladding in accident conditions shall not exceed 1,200°C,
- equivalent fuel element cladding oxidation level shall not exceed 18% of the initial cladding thickness,
- channels for coolant flow inside fuel assemblies shall not be blocked to such extent that deteriorates the cooling capability because of ballooning, damage of fuel element cladding, and deformation of other fuel assembly' details and reactor internals,
- melting of CPS control rods shall not be permitted,
- movement of CPS control rods in the reactor shall not be disturbed because of possible deformations in fuel assemblies, CPS CR and reactor internals,
- interaction between different components of fuel assemblies shall not cause their melting,
- the amount of hydrogen, which is generated during the interaction of fuel element cladding with the coolant, shall not exceed 1% of the maximum possible amount that could be generated if the whole section of the fuel element cladding (enveloping the fuel pellets) reacts completely with water and converts to ZrO_2 ($Zr+2H_2O=ZrO_2+2H_2$). The real hydrogen amount generated shall be estimated taking into account all reactions resulting in hydrogen generation,
- safe state of the core shall be attained in such a way as to create conditions for maintaining the reactor in subcritical state, its cooldown in shutdown state after an accident, and for dismantling of the core and reactor internals.

The fulfillment of the core cooling function for initiating events of different categories was assessed in the beyond design basis accident analysis (PSAR Chapter 15) [5] at high pressure in the primary circuit (without pipeline break), small-break loss of coolant (DN 70) and large-break loss of coolant (DN 850) with loss of all AC power sources (at full NPP blackout) within 72 hours.

The analysis of the initiating event “Failure of all AC power sources within 72 hours” (without pipeline break) taking into account the total coolant leak of 2.25 m3/hour at nominal pressure in the primary circuit of 16.2 MPa demonstrated that all acceptance criteria for integrity of barriers are met (the functions of reactor subcriticality and core cooling are fulfilled). The controlled state of the reactor is achieved at the maximum temperature of fuel element claddings not more than 352°C and at the stabilization of the first circuit pressure above 6 MPa. Scram (CPS CR drop) shuts the reactor down and brings the reactor to subcriticality state, and the reactor power decreases to the level of residual heat. The turbine trip leads to a short-term opening of BRU-A valves (steam discharge into atmosphere) and SG PHRS actuation in the secondary circuit pressure maintenance mode (failure of automatic cooldown is assumed). SG PHRS within 72 hours ensures passive heat removal via the secondary circuit of SG into atmosphere and maintenance of natural circulation in the primary circuit (with loss of primary circuit coolant (approximately 67 t) through leaks). Thus, during this accident the safe (cold) state of the reactor may be achieved by actuation of SG PHRS cooldown automatically or by operator actions (PSAR Chapter 15) [4].

The analyses of accidents with loss of coolant and failure of ECCS active part (small breaks DN 25-100 and large breaks DN 850) with loss of all AC power sources within 72 hours demonstrated that all acceptance criteria for integrity of barriers are met (the functions of reactor subcriticality and core cooling are fulfilled). The most representative small-break loss of coolant (DN 70) achieves the maximum temperature of fuel element claddings of 880°C, the maximum local fuel element cladding oxidation depth of 1.52% (total weight of oxidized zirconium - not more than 0.25% of the total amount) and the primary circuit depressurization below 0.2 MPa. The large-break loss of coolant (DN 850) achieves the maximum temperature of fuel element claddings of 467°C, the maximum local fuel element cladding oxidation depth of 0.01% and the primary circuit depressurization below 0.2 MPa. Reactor scram (CPS CR drop) shuts the reactor down and brings the reactor to subcriticality state, and the reactor power decreases to the level of residual heat. The turbine trip leads to a short-term opening of valves for steam discharge into atmosphere (BRU-A). Actuation of SG PHRS cooldown ensures SG depressurization, closure of BRU-A and passive heat removal through the secondary circuit into atmosphere. During the first seconds of the loss of coolant accident, the reactor vessel is almost fully evaporated, and then, when primary pressure decreases below 5.9 MPa, it is refilled with boron solution from stage 1 hydro accumulators (ECCS HA-1). When primary pressure drops below 1.5 MPa, stage 2 hydro accumulators (ECCS HA-2) start boron solution injection and the reactor vessel continues to fill. Throughout the operation of HA-2 (not less than 24 hours), the reactor core is filled with water and cools down. Upon depletion of ECCS

HA-2, ECCS stage 3 hydraulic accumulators (HA-3) are started to cool the reactor core within 72 hours (PSAR Chapter 15) [4].

Throughout the operation of HA-3, the reactor vessel is filled up to the level of cold leg nozzles. When HA-3 hydro accumulators are depleted (in 76 hours from the accident initiation), the water level in the reactor vessel starts to decrease, and core cooling continues in saturation (boiling) conditions. Loss of primary coolant with no safety injection leads to the beginning of reactor core uncovering. At the 79th hour of the accident, the reactor level is decreased below the top of the core. The core overheating starts 82 hours after the accident, and the core degradation (fuel melting) processes start in 84 hours (SPR) [5].

Thus, during this accident, the core uncovering and fuel melting start when all ECCS hydro accumulators are depleted. A severe accident may be prevented only by personnel actions to restore emergency power supply (from any AC sources), safety injection into the primary circuit by pumps of active safety systems, filling of the primary circuit and transferring to the closed circuit of heat removal from the core.

6.2.2. After occurrence of fuel damage in the reactor pressure vessel

In case of a beyond design basis accident (with fuel melting in the core), the operator has to start actions in accordance with the strategies of the Severe Accident Management Guide (SAMG) to be developed and substantiated at a later stage of the project.

The occurrence of fuel damage in the reactor vessel was estimated in Akkuyu NPP PSA level 1. The total fuel damage frequency (FDF) in the core for all groups of internal initiating events for all operating modes of the power unit and the 18-month fuel cycle equals $2.79 \cdot 10^{-7}$ per reactor per year (PSAR Chapter 15) [4].

The total contribution of all primary circuit leaks inside the containment during unit power operation to FDF is approximately 67 %, and the biggest contribution of large-break loss of coolant (with equivalent break diameter of 279 mm) is 23 %. Therefore, the analysis of the most representative severe accident with loss of coolant (DN 850) and full loss of all AC power sources (over 72 hours) was made for the Akkuyu NPP design (PSAR Chapter 15) [4].

The severe accident analysis (after fuel melting in the reactor vessel) was made to demonstrate the integrity of the fourth physical barrier (containment) and check severe accident management measures by assessing the fulfillment of the following acceptance criteria:

- hydrogen concentration in the vapor-gas mixture that generates in the primary containment rooms after core melt shall not reach the explosion-hazardous values (the hydrogen detonation limit shall not be exceeded),
- the primary and secondary pressure shall not exceed corresponding strength-governing design values and acceptance criteria of beyond design basis

accidents (not more than 115% of the design value, i.e. not more than 20.29 MPa and 9.71 MPa for the primary and secondary circuits, respectively),

- if it is impossible to cool down the melted core and structural materials (corium) in the reactor vessel, the pressure in the primary system shall not exceed 1 MPa at the moment of reactor vessel melt-through,
- it is necessary to ensure subcriticality of the melted core in the reactor vessel and in the core melt catcher (CMC),
- it is necessary to limit radioactive releases from the containment to environment.

At the stage of fuel damage in the core, the analysis of the accident with loss of coolant (DN 850) and full loss of all AC power sources for over 72 hours (PSAR Chapter 15) [4] it is stated that:

- hydrogen generation due to core material oxidation starts 83 hours after the accident at fuel element cladding temperature exceeding 700°K,
- exceedance of fuel element cladding temperature of over 1,773°K leads to the commencement of vigorous zirconium oxidation and sharp increase in temperature,
- when fuel element cladding temperature reaches 2,250°K, the oxide film collapses starting the next accident stage - core degradation (melting),
- 89 hours after the accident, a significant corium (melted fragments of fuel and structural materials) escape outside the core, support structures and core barrel melting and corium precipitation on the reactor vessel bottom start.

The reaction of corium with possible water residues on the reactor vessel bottom leads to sharp water boiling (with primary pressure increase by no more than 0.5 MPa), which leads to the increase in hydrogen generation due to further oxidation of zirconium and steel. The estimated total escape of hydrogen to the containment at the in-vessel stage of the accident is approximately 1,100 kg.

6.2.3. After failure of the reactor pressure vessel

At the beyond-vessel stage (after reactor vessel failure), the analysis of LOCA (DN 850) and complete loss of all AC power sources beyond 72 hours showed that the destruction (melt-through) of the reactor vessel bottom occurs 92 hours after the accident initiation with corium residual heat rate about 11.8 MW (PSAR Chapter 15) [4].

After the reactor vessel failure, the corium (melt) escape to the core melt catcher (CMC) of JKM system ensures the integrity of the containment as a barrier to the spread of radioactive substances. First, the metal fraction of the melt escapes to the CMC, which contains only steel of melted reactor vessel. Then, as the reactor vessel fails, the oxide portion of the corium generated inside the reactor begins to enter the CMC. Upon reaching the CMC, the maximum melt temperature is estimated at about 2,550°K, the average temperature of the oxides is 2,500 °K and the metal temperature is 2,000°K. In total, about 259 tons of corium (86.2 tons of uranium dioxide, 19.2 tons of zirconium dioxide, 12.5 zirconium, 141.4 tons of steel and its oxides) with oxidation rate of about 53%, escapes into the CMC during the accident.

After the corium escapes into the CMC, the corium reacts with sacrificial oxide materials (filler) that are located inside the CMC. Heavy melt fractions (containing uranium dioxide) drop down and mix with molten blocks of filler, which leads to the corium saturation with light oxides (with complete oxidation of zirconium). It is expected that 2.2 hours after the start of melt escape into the CMC, the melt pool will invert with its light oxide part moving upward to the melt surface.

Radiation heat transfer from the melt surface causes melting of flange thermal shielding of the CMC and activation of passive valves to supply water to the melt surface. Cooling water begins to flow to the melt surface (through passive valves) approximately 3 hours after the corium escape to the CMC.

The melting spread is assumed to stop 12 hours after the corium escape to the CMC (104 hours after the accident initiation), and gradual cooling and solidification of corium on the surface begins. The maximum corium temperature in the center of the CMC stabilizes around 2,600°K, the average temperature of the oxide layer is 2,100°K, and of the metallic phase is 2,000°K. The metal layer temperature decreases towards the outer surface of the CMC by outside cooling and melt surface cooling from above.

Within 24 hours after the corium escape into the CMC, the outer surface of the CMC is cooled by supplying water from the containment pit and the melt surface is cooled from above by supplying water from the RVI inspection shaft. Generated steam is removed through steam dump channels. Within 48 hours after the reactor vessel failure, steam escape from the CMC will be approximately 720 tons, hydrogen - 44 kg.

The accident analysis shows that passive catalytic recombiners of JMT system allow hydrogen concentration to be reduced in the containment rooms by 10.5 times. The hydrogen concentration on average in the containment shortly reaches the lower flammability limit (hydrogen burning), but the lower detonation limit (concentration 18%) is not reached. The analysis shows that the maximum volume concentration of hydrogen does not exceed 6.3% (at a high vapor concentration of 60% and a low oxygen concentration of 5%), therefore the hydrogen explosion safety criteria are met.

After using the water supply in the RVI inspection shaft, the water supply to cool corium in the CMC can be provided from off-site sources (by connecting fire engines to the water supply pipeline to CMC).

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

6.3.1. Elimination of fuel damage / meltdown in high pressure

Akkuyu NPP design includes various engineered safety features to depressurize the primary circuit below 1 MPa in order to prevent damage/melting of fuel in the core and melt escape from the damaged reactor at high pressure to maintain the containment integrity.

If emergency AC power supply is available, primary circuit can be depressurized by:

- steam generator emergency cooldown system (JNB10) for primary circuit cooldown (heat removal through the secondary circuit to cooling and sea water),
- emergency boron injection system (JND10-20) for injection into the PRZ in case of primary-to-secondary LOCA.

In case of loss of all AC power supply sources (total blackout), the primary circuit can be depressurized by the following passive systems that need battery power (short-term):

- steam generator passive heat removal system (JNB50) for primary circuit cooldown (heat removal through the secondary circuit to atmospheric air),
- PRZ PORV extra control line and emergency gas removal valves (KTP) to discharge steam from the PRZ into the pressurizer relief tank and containment.

When emergency AC power supply is restored (from an alternative diesel generator) during loss of coolant for more than 72 hours, core damage under high pressure may be prevented by actuating one train of the spray system (JMN) to supply boron solution into the reactor (via HA-2 pipelines).

According to the safety analysis, all scenarios that lead to severe accidents and have considerable contribution to the core fuel damage frequency occur at low primary pressure.

If primary circuit cooldown cannot be achieved by removing heat through the secondary circuit of steam generators (JNB10 or JNB50), primary circuit is depressurized by injection into the PRZ from the emergency bore injection system (JND10-20) or steam dumping from the PRZ through PRZ PORV extra control line and opening of the emergency gas removal valves (KTP).

In addition, in case of reactor vessel failure, the core melt does not escape into the containment reactor cavity, but enters the special beyond-vessel core melt catching and cooling system (JKM) - core melt catcher (CMC). Therefore, direct heating of containment atmosphere and structure is prevented in the Akkuyu NPP.

Operator actions to implement the primary circuit depressurization strategy shall be taken out in accordance with the Severe Accident Management Guide (SAMG) to be developed and justified at a later stage of the design.

6.3.2. Management of hydrogen risks inside the containment

Akkuyu NPP design has a containment hydrogen monitoring and emergency removal system (JMU-JMT), which includes monitoring system sensors and passive catalytic hydrogen recombiners in all containment rooms (PSAR Chapter 12) [4].

Passive catalytic hydrogen recombiners are located in places (rooms) of possible hydrogen accumulation with the required capacity, which allow performing a given function at any state of the gas-vapor mixture in such a way that it does not require mixing the medium in the containment to create a homogeneous atmosphere.

In-containment hydrogen hazard is managed by following requirements defined in NP-040-02 [27] of the Licensing Basis:

- during normal operation, formation of explosive hydrogen-containing mixtures is prevented in systems, components and rooms housed inside the containment (provided by design solutions for normal operation and compliance with regulations and manuals),
- detonation and deflagration (burning) of hydrogen-containing mixtures in containment rooms are prevented during design-basis accidents,
- in case of beyond design basis and severe accidents, detonation of hydrogen-containing mixtures is prevented, and deflagration (burning) is allowed provided that the localizing safety systems perform the functions defined by the NPP design.

Hydrogen explosion safety criteria and design limits are determined on the basis of the Shapiro-Moffett diagram, i.e. the safety criterion for design-basis accidents is the state of vapor-gas mixture inside the primary containment (the ratio of hydrogen, oxygen and steam concentrations):

- DBA safety criterion is the atmosphere condition in the containment defined outside area B (outside the deflagration “peninsula”),
- BDBA and severe accident safety criterion is the atmosphere condition in the containment defined outside area A (outside the detonation “peninsula”).

The following is assumed as design limits for the analysis of hydrogen explosion safety during design-basis accidents:

- during design-basis accident - no more than 2% of the volume concentration of hydrogen on average in the containment, that is 50% of the lower flammability level (LFL),
- post-accident period of the design-basis accident, the volume concentration of hydrogen in containment rooms shall not exceed 0.5%.

The lower detonation limit (LDL) at a hydrogen concentration of 18% ensuring the steam-gas mixture outside the detonation “peninsula” is adopted as a hydrogen safety criterion for beyond-design basis and severe accidents.

Criteria for no deflagration (burning) of the hydrogen-containing mixture in the containment during beyond design basis accident are determined on the basis of the Shapiro-Moffetti diagram, i.e. no burning during beyond-design-basis accidents is achieved when any of the following three requirements are fulfilled:

- if the average volume concentration of hydrogen does not exceed 4%, then the average volume concentrations of vapor and oxygen are unlimited,
- if the average volume concentration of hydrogen is in the range of 4 to 10%, then the average volume concentration of vapor should be at least 6 %,
- if the average volume concentration of hydrogen exceeds 10%, then the average volume concentration of oxygen should be less than 5%.

If no deflagration (burning) criteria are not met, a check should be made of the permissible loads on containment structures that arise when hydrogen is burned under these conditions.

For severe accidents, it is necessary to be guided by acceptance criteria to prevent the explosion of hydrogen-containing mixtures (outside the detonation “peninsula”) threatening the failure of containment.

In addition, hydrogen hazard management can also be accomplished by some operator actions that increase the concentration of water vapor in the vapor-gas mixture (reducing the concentration of oxygen) and redistribute hydrogen among containment rooms.

All operator actions to implement the containment hydrogen hazard management strategy shall be taken in accordance with the Severe Accident Management Guide (SAMG) to be developed and supported at a later stage of the design.

6.3.3. Prevention of overpressure of the containment

Akkuyu NPP design considers the following physical phenomena, engineered safety features and severe accident management strategies that can prevent overpressure to maintain the containment integrity:

- prevention of reactor vessel failure (melt-through) under high pressure in the primary circuit, which is achieved by primary circuit depressurization below 1 MPa (6.3.1 hereof),
- actuation of steam generator passive heat removal system (JNB50) for primary circuit cooldown to ensure sufficient heat removal through the SG secondary circuit to the atmosphere, reduce primary circuit pressure and prevent the reactor failure under high pressure,
- steam explosion is ruled out at the in-vessel stage of a severe accident due to design features of the reactor, which reduce the possibility of intensive reaction of the molten core with water, and organizational provisions to prohibit water supply to the primary circuit after core meltdown, i.e. fallout of corium on supporting structures, core barrel bottom and reactor vessel bottom does not threaten the primary circuit integrity (primary circuit pressure increases no more than 0.5 MPa),
- steam explosion is ruled out at the beyond-vessel stage of a severe accident, because the core melt catcher (CMC) is designed to prevent water accumulation in the core catcher prior to its escape into the CMC, i. e. the main cause of a possible steam explosion (direct contact of melt with water) is eliminated after the reactor vessel is damaged and melt enters the CMC.

In addition, it is necessary to evaluate the impact of reactor passive heat removal systems (HA-2, HA-3, PHRS), core melt catcher and CMC cooling water supply systems on increasing overpressure in the containment no more than the design value during beyond design-basis and severe accidents.

All operator actions to implement the containment overpressure prevention strategy shall be taken in accordance with the Severe Accident Management Guide (SAMG) to be developed and supported at a later stage of the design.

6.3.4. Prevention of re-criticality

An analysis of a severe accident with DN 850 large-break LOCA with total blackout for more than 72 hours was made in the Akkuyu NPP design to evaluate the molten core subcriticality criterion (PSAR Chapter 15) at all stages of the accident: fuel melting in the reactor vessel, reactor vessel failure and melt escape to CMC.

The analysis of this accident showed that the reactor scram (CPS CR drop) and core evaporation due to loss of coolant reduce multiplication factor Keff below 0.587 (due to reactivity feedbacks). ECCS hydraulic accumulators (HA-1, HA-2 and HA-3) are actuated to fill the reactor core with boric acid solution increasing Keff to 0.861 (with a concentration of 16 g/kg H_2O) and to 0.953 (with a concentration of 8 g/kg H_2O). After all the hydraulic accumulators are depleted, water level drops in the reactor and the core heats up (before damage), Keff again decreases to 0.58. Therefore, without a moderator, VVER reactor core (even with the maximum U-235 enrichment of fuel up to 5%) is deeply subcritical. At this stage of the accident (before fuel damage), the reactor core shall be filled with boric acid solution with a concentration of at least 16 g/kg H_2O for reliable subcriticality.

After reaching a temperature of 1,500°K at the in-vessel stage of a severe accident, absorbing rods (CPS CR) start to melt, zirconium rapidly oxidizes with the depressurization and swelling of fuel cladding, which practically does not affect Keff. Further increase in temperature of fuel cladding to 2,250°K leads to core melting and damage.

When fuel elements are destroyed, Keff mainly depends on the state of core bottom, where water can survive if the center is blocked. In the event of core melt pools (above the blockage level), all water vapor is displaced and the temperature rises, which further decreases Keff unlike core debris.

When a molten pool with a temperature of 2,873°K generates at the top of the core (blockage at a height of 0.4 to 1 m from the core bottom with the survived grid of FA rods surrounded by steam with a pressure of 0.5 MPa), conservative Keff will be less than 0.617. After melt-through of support structures, core barrel bottom and melt fallout on the reactor vessel bottom, Keff will be less than 0.682, provided there is no water and other absorbers (B, Dy, Gd).

According to the analysis of the in-vessel stage of a severe accident, it can be concluded that the melt will be subcritical during core degradation in the absence of water inside at the fuel level.

For the beyond-vessel stage of a severe accident (after the melt escape from the damaged reactor vessel), analyzes of criticality in the core melt catcher (CMC) made by Monte Carlo method using SAPPHIRE 2006 3D model (with conservative assumptions and allowing for code uncertainties) (PSAR Chapter 15) showed that:

- corium cooling in the CMC is deeply subcritical,
- from the moment of corium ingress in the CMC until its cooling (solidification), Keff varies in the range from 0.601 to 0.456,
- maximum Keff can be achieved after the long-term cooling of corium to a temperature of 373°K, which will be accompanied by corium cracking with pore filling with water without boric acid,

- Keff will not exceed 0.564 (for the first fuel load) and 0.639 (for stationary refueling), even with the most unfavorable corium cracking and filling with water.

6.3.5. Prevention of basemat melt through

To prevent melt-through of the foundation slab and maintain the primary containment integrity, Akkuyu NPP design employs a beyond-vessel core melt catching and cooling system (JKM), which provides retention, cooling and subcriticality of the melt after it escapes the damaged reactor vessel (PSAR Chapter 12, item 12.2.3.5). The main component of this system is a core melt catcher (CMC).

The core melt catcher (CMC) performs the following basic safety functions (PSAR Chapter 12):

- catching corium - liquid and solid fragments of the molten core and reactor structural materials,
- retaining the bottom of the reactor vessel with corium during its plastic strain,
- preventing corium from escaping beyond the CMC,
- preventing boiling of the core melt,
- cooling corium in the CMC and removing steam from the heat exchange zone (from the outer surface of the CMC and from the melt mirror inside the CMC),
- minimizing escape of hydrogen and radioactive substances from the melt into the containment,
- maintaining corium subcritical in the CMC,
- non-exceeding the maximum allowable stresses in the basic building structures of the CMC,
- limiting the temperature of structural concrete of the core barrel (for prolonged heating - no more than 90°C in local areas, for short-term heating - no more than 200°C),
- performing safety functions without operator control actions,
- protecting support structures and radiation protection of the reactor from thermal radiation from the melt (at the stage of long-term corium cooling).

After the reactor vessel melts through, fragments of molten core and structural materials (corium) escape immediately to the bottom slab (a multilayer guide structure made of special concrete), the main function of which is to transition corium to the core melt catcher (CMC).

Water from the containment pit, which enters the pit as a result of loss of coolant, and water drain from first, second and third stage ECCS HAs are used to cool the outer surface of the CMC in case of a severe core meltdown accident. Water is also used from the fuel pool, which is drained into the pit via a special pipeline.

Water supply in the RVI inspection shaft (320 m^3) is sufficient to supply water to the melt surface for 24 hours (with flow rate at least 11 m^3/h). Water is supplied from the RVI inspection shaft by opening JKM system valves, which the operator must open before the discharge of batteries (in case of a BDBA threat). Water from the RVI inspection shaft enters closed passive valves located in the CMC cantilever truss, and water from the containment pit enters closed passive valves in the CMC housing (without time limit). Passive valves are in standby and triggered by temperature rise above the melt surface in the CMC.

The general layout of the core melt catcher is shown in Figure 18.

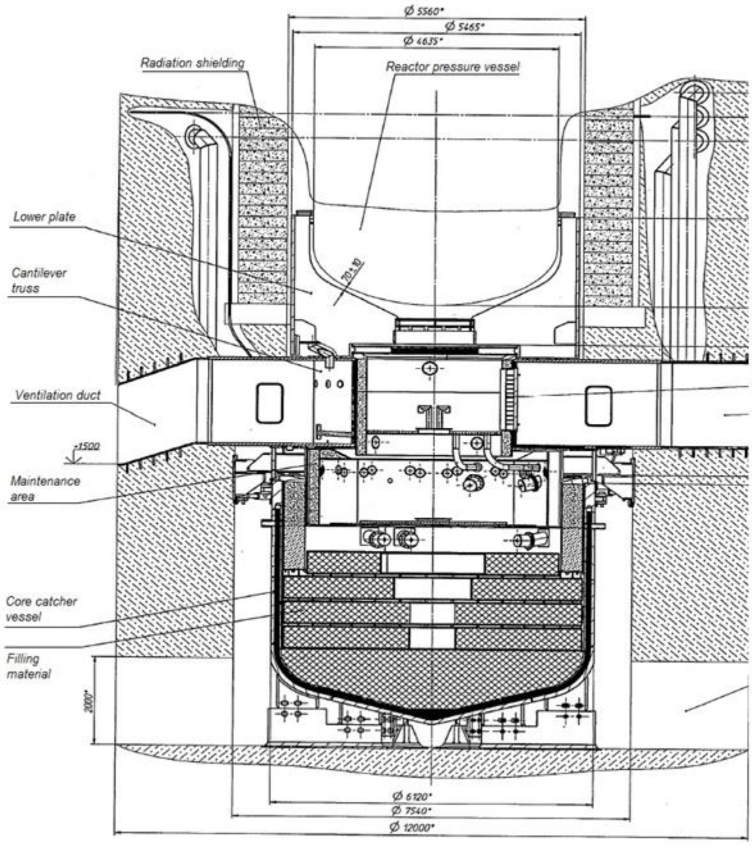

Figure 18 Core melt catcher

6.3.6. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity

In case of a beyond design basis accident with the loss of all AC power sources (blackout), the NPP design includes DC sources in two trains of the emergency power supply system (two batteries with a discharge of 2 and 72 hours in each train), which maintain uninterrupted power supply of equipment used to protect the containment integrity:

- containment isolation valves,
- SG passive heat removal system valves (SG PHRS),
- PRZ PORV extra control line and emergency gas removal system valves (KTP),
- ECCS HA drain line valves (HA-1, HA-2 and HA-3),
- valves on water drain lines from the fuel pool to the containment pit,
- BRU-A, MSIV and SG PORV control line valves,
- control safety systems and post-accident monitoring systems, including containment hydrogen monitoring system (JMU).

When emergency AC power supply is restored (from an alternative diesel generator), one train of the spray system (JMN) can be actuated to inject into the containment to reduce pressure below the design value (when controlling the hydrogen concentration within acceptable limits).

The supply of compressed air to equipment used to protect the containment integrity during a beyond design basis accident is not required.

6.3.7. Measuring and control instrumentation needed for protecting containment integrity

In case of beyond design basis and severe accidents, the Akkuyu NPP design includes special measuring instruments and a post-accident monitoring system. The containment integrity during accidents is monitored by MCR operators.

Special measuring instruments (monitoring systems) necessary to monitor the containment integrity are located in UJA reactor building and UJB annulus space of the hermetic enclosure system (containment). The sensors of these measuring channels are designed for a wide range of measurement parameters and are certified for stability in special environmental conditions of containment rooms (temperature, pressure, humidity, gamma radiation dose rate, hydrogen concentration, etc.) that can occur during beyond design basis and severe accidents. In addition, all special containment integrity measuring instruments have seismic category 1.

During an accident, the reactor operator monitors the containment integrity through the following monitoring and post-accident monitoring tools on the MCR panels and displays:

- status indication (open/closed) of isolation valves on containment penetrations,

- display of automated radiation monitoring system (ARMS) sensors on NPP power units for measuring the gamma radiation dose rate inside the containment, which allows estimating the rate of fuel damage in the core up to total meltdown,
- display of ARMS sensors for measuring the volumetric activity of inert radioactive gases (IRG) in the annulus air, which allows estimating the rate of leakage through the primary containment structures and air purification on filters that are provided during the operation of annulus ventilation and filtration system (KLB22) to reduce radioactive releases into the environment,
- display of ARMS sensors on the NPP site for measuring the gamma radiation dose rate and volume activity of radionuclides in the air, which allows estimating the rate of accidental release from the containment into the environment,
- display of environmental parameters in containment rooms (temperature, pressure, humidity, gamma radiation dose rate, hydrogen concentration, etc.).

Operator actions to implement monitoring strategies necessary to protect the containment integrity shall be taken on the basis of instrumentation and post-accident monitoring information in accordance with the Beyond Design Basis Accident Management Guide (BDBAMG) and the Severe Accident Management Guide (SAMG) to be developed and supported at a later stage of the design.

6.3.8. Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

Akkuyu NPP design has independent active and passive safety systems for each power unit, which are used to perform the functions of heat removal from the core and fuel pool in the containment.

A probabilistic safety analysis (PSA-1) for internal and external initiating events (natural and human-induced) has been made for the Akkuyu NPP at this stage, which also includes common-cause failure tolerance analysis of systems, structures and components. One of the causes for the simultaneous accident at several NPP power units may be external initiating events on the site (earthquakes, flooding or extreme weather conditions) that lead to failures of supporting systems at several power units (complete blackout and/or loss of heat removal to the ultimate heat sink).

To improve the reliability of functions (heat removal from the core and fuel pool), the NPP design employs passive systems that are more resistant to common-cause failures of the supporting systems. Passive systems do not require emergency power supply from alternating current sources (diesel generators) and heat removal to the ultimate heat sink (sea water). The analysis results of such events for each NPP power unit are discussed in Section 5 hereof.

In addition, each NPP power unit has independent passive systems for managing severe accidents: hermetic enclosure system (double containment), beyond-vessel core melt catching and cooling system (JKM), in-containment hydrogen monitoring and emergency removal system (JMT -JMU), PRZ PORV extra control line (JEF) and emergency gas removal system (KTP) and an emergency power supply system (connectability of an alternative diesel generator and mobile installations).

Akkuyu NPP power units are technically and structurally independent. Accident management and implementation of on-site emergency plans are coordinated from the NPP general crisis center located in the protected emergency command post (PECP) and has communication with MCR/ECR of all NPP power units.

6.3.9. Conclusion on the adequacy of severe accident management systems for protection of containment integrity

Conclusion on the adequacy of severe accident management systems to protect the containment integrity of power units (in case of a severe accident) can be made after the development and analytical substantiation of emergency operating procedures and guides (BDBAMG and SAMG). Additional information on the adequacy of the organizational structure can also be obtained through training and drills (emergency exercises) of operating personnel on the full-scale simulator of the MCR to verify BDBAMG and SAMG. Some conclusions can be borrowed from independent safety assessment reports for Akkuyu NPP licensing documents.

6.3.10. Measures which can be envisaged to enhance capability to maintain containment integrity after occurrence of severe fuel damage

The following measures shall be envisaged to enhance the capability to maintain the containment integrity after heavy fuel damage:

- development and analytical substantiation of severe accident management operator actions (SAMG) to protect containment integrity,
- analysis and measures to be taken in the event of a simultaneous accident at several NPP power units and the impact of several hazards on the entire site, the loss of emergency power supply - complete blackout for more than 72 hours (loss of all AC and DC sources), loss of instrumentation and post-accident monitoring systems, and also additional measures for the use of mobile means to restore emergency power supply and severe accident management,

- development of additional equipment for managing severe accidents at power units remotely from the NPP crisis center (PECP) in the event of loss of control or life support facilities at MCR and ECR.

6.4. Accident management measures to restrict the radioactive releases

6.4.1. Radioactive releases after loss of containment integrity

At the current stage of the Akkuyu NPP project, the analyses which were performed for the severe accident management measures to radioactive releases during accident conditions are based on the assumption that the core integrity is preserved. According to this assumption, the off-site radiological effects are determined by using the source term related to the maximum emergency release for an accident with the core integrity is preserved.

The estimation of the inventory of the fission products involves two periods. The first period takes place during in vessel stage of the accident, starting from the loss of primary coolant to core degradation and reactor vessel melt-through. The second period takes place during beyond-vessel stage of the accident when the corium (core melt) is released from the reactor vessel, which occurs after reactor vessel melt-through but before the corium enters the core catcher (CMC) of JKM system.

During the estimation of the source term related to accidental releases experimental data were used for the following parameters:

- fuel temperature and heating rate,
- composition of vapor-gas medium,
- vapor-gas flow temperature and speed,
- burnup of fuel,
- pressures above the core during its heating.

The main radionuclides (radioactive inert gases, iodine, cesium, strontium, barium, cerium and ruthenium) are taken into account while estimating the fission products in the releases. The different fission products which transfer from the fuel to outside of the containment can be summarized as below:

- volatile fission products (noble gases, iodine, tellurium and cesium), which at high temperatures are complemented by the escape of fission products of medium and low volatility due to formation of their volatile compounds in the fuel matrix and corium. Fission products escaping from the fuel during a severe accident enter the containment atmosphere, which leads to the heating of the vapor-gas medium by residual energy of the fission products,

- the in-vessel stage of the accident, corium temperature does not exceed 2,200 °K, and during this period, together with the gas-vapor medium, the containment atmosphere will be filled by fission products in gas- or aerosol forms (dose-forming radionuclides): inert (noble) radioactive gases (IRG) in the amount of up to 70%; iodine, molybdenum, cesium and tellurium, up to 60%; up to 40% of barium; strontium, ruthenium, lanthanum - up to 20%; cerium - up to 15% of their content in the fuel. Iodine, cesium and other radionuclides (except the IRG) enter the containment atmosphere as aerosols,
- after the reactor vessel failure, the core melt (corium) enters core melt catcher. Then, corium gradually solidifies. Water vapor, hydrogen and aerosols are released to the containment atmosphere.

In case of containment leakage or depressurization through emergency filter, the repeated release of aerosols is possible from the containment inside surfaces, where aerosols had been already deposited, and a significant release of radioactive aerosols will ocur.

6.4.2. Accident management after uncovering of the top of fuel in the fuel pool

During a beyond design basis accident with loss of the spent fuel pool cooling function, the actions to restore the critical safety functions aimed at preservation of physical barriers and prevention of fuel melting in the fuel pool in accordance with the symptom based procedures of the BDBAMG to be developed and substantiated at a later stage of the project shall be taken by the operator.

Accident analysis for initiating event with the loss of fuel pool cooling and the plant blackout within 72 hours has been made to demonstrate the integrity of physical barriers; the analysis was based on the following acceptance criteria:

- maximum temperature of fuel element cladding in accident conditions shall not exceed 1,200°C,
- fuel pellets do not melt even locally (temperature is less than 2,540 °C for burn-up fuel and less than 2,840°C for fresh fuel),

It is assumed that fuel is not damaged in the fuel pool and the specified acceptance criteria are met if the fuel parts of fuel assemblies (FA) do not uncover. The analysis of accident with loss of fuel pool cooling showed that acceptance criteria are met (no fuel uncovering in the fuel pool) within at least 35.6 hours at the maximum integrated residual heat rate of 19.3 MW (in case of accident after emergency core unloading into fuel pool). For other cases of spent fuel storage in fuel pool, the integrated residual heat rate is significantly lower and time available for operator actions until fuel uncovering is significantly longer. Acceptance criteria for this type of accident are further met, if within the specified period the operator either restores the operation of active fuel pool cooling systems (JNA, JMN) or starts boron solution supply from ECCS second stage hydro accumulators (HA-2) of 960 m^3 and ECCS third stage accumulators (HA-3) of 720 m^3 or supplies water to the fuel pool from any external sources (mobile pump unit or fire engines) with the minimum required flow to compensate the evaporation (8.56 kg/sec).

6.4.3. Conclusion on the adequacy of measures to restrict the radioactive releases

The analyses performed in the PSAR of Akkuyu NPP for the estimation of the source terms of the releases as a result of accident conditions and the off-site radiological consequences of these accidents showed that engineered safety features of Akkuyu NPP design ensure that radioactive releases and radiologic consequences during accidents are below the limits recommended by EUR and IAEA.

However, it should be noted and mentioned in the report that the analyses related to the releases during accidents and radiological consequences related to these releases which are given in the current version of the chapter 15.7 of the PSAR are being performed again by the Applicant since the software and approaches used for the calculation of radiological consequences of the accidents are not fully compliant with the regulatory requirements. The following points will be taken into consideration during the performance of the radiological consequences during accidents:

- during the repetition of the calculations related to the radiological consequences, an atmospheric dispersion model that takes into account complex topographical conditions, breeze effect and wet deposition will be used,
- moreover, the categorization of initiating events, release locations, release durations and the time behavior for each source term will be checked.

The final conclusion on the adequacy of measures to limit radioactive releases of Akkuyu NPP during accident management at the power units and the emergency response (in case of severe accident) can be made after the completion of NPP licensing process (after all necessary safety analysis are done) and preparation of on-site emergency plan.

7. GENERAL CONCLUSION

Due to the fact that the Fukushima accident has increased awareness and sensitivity on the potential dangers of earthquakes and tsunamis, the studies related to external events in the region of Akkuyu site, at which 4 units of VVER-1200 will be constructed, have become more important. Although many studies about the earthquake and tsunami potential for this site have been carried out before the Fukushima accident, TAEK has requested from the Applicant to update these studies. Within this context, site related parameters are updated. On the other hand, existing regulations related to earthquakes and tsunamis, calculation methods and steps of hazard analysis are being reviewed.

Taking into account the site parameters report, preliminary safety analysis report, probabilistic safety analysis report and Stress Tests Report submitted to TAEK by the Applicant, an evaluation for the design issues were briefly presented in this report. In these evaluations, the Applicant states that safety criteria and design limits of Akkuyu NPP are established in accordance with the applicable Turkish regulations, Russian and international regulations such as IAEA safety standards. Since the Akkuyu NPP Unit 1 is under construction and the project is at early stage, there is preliminary information about the plant response and the effectiveness of the preventive measures to be implemented in severe accident management strategies, as provided by the Applicant. In addition, there is no approved Emergency Plan of the Applicant at the time of writing of this report. The plans and procedures to be developed are expected to reflect IAEA recommendations and world experience after the Fukushima accident. As far as the information provided by the Applicant indicates, the Akkuyu NPP design is expected to comply with the improved nuclear safety requirements based on the lessons learned from the Fukushima accident. Also, it should be noted that Akkuyu NPP Unit 1 construction license conditions have not been met by the Applicant, yet. Construction license conditions are required to be met before submission Final Safety Analysis Report.

7.1. Key provisions enhancing robustness (already implemented)

Key provisions enhancing the robustness of the design are listed as follows:

- Akkuyu NPP design includes safety systems that can maintain or recover the critical safety functions under conditions far beyond of DBAs.
- In case of failure of a critical safety function, independent and diverse systems are designed for reactor scram and maintaining the reactor subcriticality for an unlimited time period. Actuation of control rods for scram is based on gravitational forces and the core power has self-limitation properties due to negative coefficients of reactivity.
- In case of failure of core cooling, if the active systems for emergency cooling fail water is provided to the primary circuit by passive hydro-accumulators.

- Under the conditions of DBA and BDBA, the primary circuit heat removal is provided by the steam generator emergency cooling system based on active principles. In case this system fails, heat is removed through passive heat removal systems.
- Protection against common cause failures due to internal and external hazards is ensured by means of spatial and physical separation, and diversification of the safety systems included in the design.
- Emergency power supply system is reliable and is redundant. It consists of two emergency DG for each of the safety trains and there are additional DG stations for normal operation systems important for safety.
- In case of loss of all the power sources, including in-house sources, external sources and DGs, all critical safety functions can be performed by the passive systems for a long period of time. The residual heat removal is assured through the passive heat removal system.
- Each Akkuyu NPP unit has two independent ways of heat removal from the reactor core to the ultimate heat sink:
    - through essential service water system to the sea, and
    - through passive heat removal system to the atmosphere.
- The total loss of primary ultimate heat sink is an initiating event with consequences similar to the total loss of AC power supply, described above. Safety of the plant is assured in this case as well.
- As a result of the passive safety systems operation during a severe accident, the pressure in the containment is maintained below the design value.
- The minimum volume of water is maintained in the spent fuel pool to guarantee a long time period before fuel uncovery.
- The passive hydrogen recombiners, with their capacity and location in the containment, prevent the possibility of hydrogen accumulation and hydrogen explosion hazard in both design basis and beyond design basis conditions.
- The severe accident management principles foreseen in the design correspond to the requirements for nuclear installations of latest generation and the design provides necessary technical measures for implementation of the required severe accident management strategies.

Further safety analyses will be performed during the project implementation. The emergency plan and severe accident management guidelines will also be developed.

7.2. Safety issues

Akkuyu NPP design is developed within the scope of Russian Federation regulation, and is in general in compliance with up-to-date safety requirements that are established by Turkish Regulations, IAEA and EUR requirements. Safety systems and engineered safety features for beyond-design-basis accident management implemented in the design shall provide adequate core cooling, spent fuel pool cooling and ultimate heat sink. Detailed assessment process with respect to IAEA, Turkish and Russian requirements is being carried out by TAEK.

7.3. Potential safety improvements and further work forecasted

In addition to the construction license conditions and additional findings during review and assessment, the following measures shall be taken to improve the safety of power units:

- consider the possibility of supplementing the design with supporting alternative equipment (mobile pumps, DG) with standbys, which can be connected to either of the two safety trains and to the equipment of normal operation systems important for safety; or, as an alternative, consider the possibility of setting up an additional intermediate substation supplying power in a crisis situation to critical equipment of the power unit subject to voltage in the off-site power grid (both 10 kV or 0.4 kV switchgears and directly consumers),
- make organizational and engineering provisions for connection (delivery, deployment) of planned alternative power supply equipment taking into account possible damage of the site distribution network access infrastructure,
- pay attention to seismic resistance of the planned alternative power supply equipment; it shall have greater seismic resistance than regular power supply systems,
- develop relevant operating procedures to maintain availability of the equipment required to transfer the reactor plant into safe state after an earthquake,
- develop accident management (EOP, BDBAMG, SAMG), emergency preparedness and response (personnel protection plan, public protection plan) documentation,
- it is advisable to develop a procedure for regular inspections of bank protection structures, breakwater dike, water intake facility, tunnels for essential-service pipelines to improve resistance of the power plant against secondary effects of an earthquake (namely the integrity of bank protection structures),

- pay attention to the development of the infrastructure to improve emergency response (alternative routes of materials / personnel delivery during accidents, evacuation routes).

It should be also noted that the Applicant shall develop and implement internal procedures for supervising compliance with the requirements of TAEK during NPP commissioning in accordance with the requirements of the Licensing Basis.

ANNEX I

Multilateral Conventions, Treaties and Bilateral Agreements of Turkey

1. Convention on Nuclear Safety, 1994
2. Paris Convention on Third Party Liability in the Field of Nuclear Energy (29 July 1960), 1961
3. Protocol to Amend the Convention on Third Party Liability in the Field of Nuclear Energy of 29 July 1960 (28 January 1964), 1967
4. Protocol to Amend the Convention on Third Party Liability in the Field of Nuclear Energy of 29 July 1960, as Amended by the Additional Protocol of 28 January 1964 (16 November 1982), 1986
5. Treaty on the Non Proliferation of Nuclear Weapons (NPT), 1979
6. Agreement Between the Government of the Republic of Turkey and the IAEA for the Application of Safeguards in Connection with NPT, 1981
7. Protocol Additional to the Agreement Between the Government of the Republic of Turkey and the IAEA for the Application of Safeguards in Connection with NPT, 2001
8. Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, 1990
9. Convention on Early Notification of a Nuclear Accident, 1990
10. Convention on the Physical Protection of Nuclear Material, 1986
11. Comprehensive Nuclear-Test-Ban Treaty, 1999
12. Agreement Between the Government of Canada and the Goverment of the Republic of Turkey for Co-operation in the Peaceful Uses of Nuclear Energy, 1986
13. Agreement Between the Goverment of the Republic of Turkey and the Government of Argentine Republic for Co-operation in the Peaceful Uses of Nuclear Energy, 1992
14. Agreement Between the Goverment of Korea and the Goverment of the Republic of Turkey for Co-operation in the Peaceful Uses of Nuclear Energy, 1999
15. Agreement Between the Government of French Republic and the Goverment of the Republic of Turkey for Co-operation in the Peaceful Uses of Nuclear Energy, 2011
16. Agreement for Cooperation Between the Republic of Turkey and the United States of America Concerning Peaceful Uses of Nuclear Energy, 2006

17. Memorandum of Understanding for Technical Cooperation and Exchange of Information in Nuclear Regulatory Matters Between Turkish Atomic Energy Authority and the State Nuclear Regulatory Committee of Ukraine, 2008
18. Agreement Between the Government of the Republic of Turkey and the Cabinet of Ministers of Ukraine on Early Notification of a Nuclear Accident and Exchange of Information on Nuclear Facilities, 2001
19. Agreement Between the Government of the Republic of Turkey and the Government of the Republic of Bulgaria on Early Notification of a Nuclear Accident and on Exchange of Information on Nuclear Facilities, 1997
20. Agreement Between the Government of the Republic of Turkey and the Government of Romania on Early Notification of a Nuclear Accident, 2008
21. Agreement Between the Government of the Republic of Turkey and the Government of the Russian Federation for Cooperation in the Use of Nuclear Energy for Peaceful Purposes, 2011
22. Agreement Between the Government of the Republic of Turkey and the Government of the Russian Federation on Early Notification of a Nuclear Accident and Exchange of Information on Nuclear Facilities, 2011

ANNEX II

Laws, Decrees, Regulations and Guides Concerning the Safety of Nuclear Power Plants

Laws, Decree Laws and Presidential Decrees

1. Law on Turkish Atomic Energy Authority, 1982
2. Decree Law No. 702: Regulation of the Law on the Organization of the Nuclear Regulation Authority and its duties in some laws (Articles 1-15)
3. Decree Law No. 703: Regulation of the Law on the Organization of Turkish Atomic Energy Authority
4. Presidential Decree No. 4 on the organization of authorities and institutions linked and related to Ministries (Articles 785-792)

Decrees

1. Decree on Licensing of Nuclear Installations, 1983

Regulations

1. Regulation on Working Procedures of Atomic Energy Commission, 1983
2. Regulation on the Establishment and Working Procedures of Advisory Committee on Nuclear Safety, 1997
3. Regulation on Radiation Safety, 2000
4. Regulation on National Practices during Nuclear and Radiological Emergencies, 2000
5. Regulation on Safe Transport of Radioactive Materials, 1997
6. Regulation on Basic Requirements on Quality Management for the Safety of Nuclear Installations, 2007 (Rev’d 2009)
7. Regulation on Nuclear Safety Inspections and Enforcement, 2007 (Rev’d 2008)
8. Regulation on Issuing Document Base to Export Permission for Nuclear and Nuclear Dual Use Items, 2007
9. Regulation on Specific Principles for Safety of Nuclear Power Plants, 2008
10. Regulation on Design Principles for Safety of Nuclear Power Plants, 2008
11. Regulation on Nuclear Power Plant Sites, 2009

12. Regulation on Protection of Outside Workers in Controlled Areas from the Risks of Ionizing Radiation, 2011
13. Regulation on Physical Protection of Nuclear Materials and Nuclear Facilities, 2012
14. Regulation on Nuclear Material Accounting and Control, 2012
15. Regulation on Radioactive Waste Management, 2013
16. Regulation on Clearance in Nuclear Facilities and Release of Site from Regulatory Control, 2013.
17. Regulation on Working Procedures of Atomic Energy Commission, 1983
18. Regulation on Environmental Impact Assessment, 2008
19. Regulation Regarding Equipment Procurement Process and Approval of Manufacturers for Nuclear Facilities, 2015
20. Regulation on Construction Inspection of the Nuclear Power Plants, 2017
21. Regulation on Operating Organisation, Qualitifications and Training of Operating Personnel and Operating Personnel Licensing in Nuclear Power Plants, 2017
22. Regulation on Management in Nuclear Installations, 2017
23. Regulation on Radiation Protection in Nuclear Facilities, 2018

Documents and Guides

1. Guide on Format and Content of Site Report for Nuclear Power Plants, 2009
2. Directive on Principles of Licensing of Nuclear Power Plants, 2010
3. Guide on Owner and Authorization Application for Nuclear Installations, 2014
4. Guide on the Construction Activities in Nuclear Installations that are Authorized as per the Authorization Stages, 2016
5. Directive on Determination of Licensing Basis Regulations, Guides and Standards and Reference Plant for Nuclear Power Plants, 2014

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2. Akkuyu NPP Stress Tests Report Revision B02, Akkuyu Nuclear JSC, 2018
3. List of Licensing Basis for Akkuyu Nuclear Power Plant, TAEK Atomic Energy Commission, 2014
4. Akkuyu NPP Unit 1, Preliminary Safety Analysis Report, Akkuyu Nuclear JSC, 2017
5. Akkuyu NPP Site Parameters Report, 2017
6. Probabilistic Safety Analysis, Volume 1, Book 15, Assessment of Fuel Damage Frequency Under External Natural and Human-Induced Impacts, AKU-VAB0101-BAA0015, Revision B01
7. Regulation on Nuclear Power Plant Sites, 2009
8. Guide on Format and Content of Site Report for Nuclear Power Plants, 2009
9. SSG-9 Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series, 2010
10. NP-031-01, Standards for Design of Seismic Resistant Nuclear Power Plant, Russian Federation, 2002
11. MP 1.5.2.05.999.0027-2011 ‘Seismic Design Standards for Nuclear Power Plants. Guidelines, Russian Federation, 2011
12. MR 1.5.2.05.999.0025-2011 ‘Seismic Analysis and Design of Nuclear Power Plants. Guidelines, Russian Federation, 2011
13. NS-G-3.6, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants, IAEA Safety Standards Series, 2004
14. ASCE 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary, American Society of Civil Engineers, 1998
15. ASCE 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities American Society of Civil Engineers, 2005
16. NP-006-098, “Requirements to Contents of Safety Analysis Report of Nuclear Power Plant with VVER Reactors”, Russian Federation, 2003
17. Probabilistic Safety Analysis, Volume 1, Book 16, Assessment of Fuel Damage Frequency Under Seismic Impacts, AKU-VAB0101-BAA0016, Revision B01
18. OPB-88/97-NP-001-97, General Regulations on Ensuring Safety of Nuclear Power Plants, Russian Federation, 1997

19. Akkuyu NPP, Turkey, Akkuyu NPP Stress Test Report, Akkuyu Nuclear JSC, 2012
20. NS-R-3, Site Evaluation for Nuclear Installations. IAEA, Vienna, 2003
21. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, Safety Standards Series, IAEA, Vienna, 2011
22. NP-064-05, Accounting of External and Man-Induced Impacts to Nuclear Energy Use Facilities, Russian Federation, 2005
23. SSR-2/1 Rev. 1, Safety of Nuclear Power Plant Design, IAEA Safety Standards Series, 2016
24. Regulation on Specific Principles for Safety of Nuclear Power Plants, 2008
25. EPR-Method 2003, Method for Developing Arrangements for Response to a Nuclear or Radiological Emergency, IAEA, Vienna, 2003
26. TECDOC-955, Generic assessment procedures for determining protective actions during a reactor accident, IAEA, Vienna, 1997
27. NP-040-02, Rules of Hydrogen Explosion Protection in Nuclear Power Plants, Russian Federation, 2002
28. RB-021-01, Assessment of reactor core severe damage frequency (for natural and man-induced external events), Russian Federation, 2001