Environmental Impact Assessment Report

for the Decommissioning of Units 1 to 4

at Kozloduy Nuclear Power Plant

CHAPTER 1

ANNOTATION OF THE INVESTMENT PROPOSAL – ACTIVITIES AND TECHNOLOGIES

CONTENTS

1. Annotation of the Investment Proposal – activities and technologies

1.1 Background of the Investment Proposal (IP)

In November 1999, the Bulgarian Government and the European Commission signed a Memorandum of Understanding in which the Bulgarian Government was committed to shut down and decommission Units 1 to 4 of Kozloduy Nuclear Power Plant (KNPP) at the earliest possible date, beginning with the Close down of Units 1 and 2 at the end of 2002. A commitment for Close down of Units 3 and 4 at the end of 2006 was signed at later date. In consequence all four units were shut-down at the agreed time.

Taking into account the financial consequences of early closures, as well as the need of competitive energy sector, the European Commission has offered a multi-annual assistance package for Bulgaria's energy sector dedicated for assistance to the nuclear energy sector for the decommissioning of Units 1 to 4 of Kozloduy NPP (KNPP), as well as for assistance to the energy efficiency field. In connection therewith the Kozloduy International Decommissioning Support Fund (KIDSF) was established in June 2000, administered by the European Bank for Reconstruction and Development (EBRD).

The financing of the decommissioning activities after terminating of the EU financial aid will be provided by the Decommissioning fund.

In accordance with the Working plan for the activities leading to the final termination of the operation of KNPP Units 1 and 2 and pursuant to CMD 848/19.12.2002 KNPP Units 1 and 2 were disconnected from the energy system on 31.12.2002. KNPP Units 3 and 4 were disconnected from the energy system on 31.12.2006.

Currently Units 1 to 4 have a license issued by the Bulgarian Nuclear Regulatory Agency (BNRA) as RAW management facilities subject to decommissioning.

Currently, the SNF of the four units is removed from them and is stored at the SNF storage site.

Units 3 and 4 of Kozloduy Nuclear Power Plant EAD were declared as RAW Management Facility subject to decommissioning and their property is entrusted for management to SE”RAW” by Decision of the Council of Ministers (DCM) of Republic of Bulgaria (DCM 1038/19.12.2012).

The Investment Proposal (IP) subject is the decommissioning of KNPP Units 1-4. State Enterprise “Radioactive Waste” (SE RAW) is the Proponent of this IP.

The initial Decommissioning strategy envisaged preparation and operation of the Safe Enclosure (SE) area of the reactor buildings and the auxiliary buildings for a period of 35 years, and the last stage included deferred dismantling. In the process of analysis and assessment, considering all technical, economic and social aspects and based on the international experience, the strategy was updated to the continuous dismantling decommissioning strategy.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

According the Updated Decommissioning Strategy, after expiration of the transitional period under the valid licenses, there are two decommissioning stages:

- Stage 1: Preparation of Safe Enclosure (SE) and Operation of SE for 5 years and Dismantling outside the SE area, and
- Stage 2: Dismantling inside the SE area

In order to execute the decommissioning process in these two stages, a preliminary stage is planned, determined as Pre-decommissioning stage, during which the necessary supporting projects are planned. The most important of them are the following infrastructure projects:

- Size Reduction and Decontamination Workshop;
- Sites for management of materials generated by the decommissioning activities of Units 1-4 at Kozloduy Nuclear Power Plant (Decay Storage Sites for Transitional RAW and Site for Conventional Waste (Material) from Decommissioning);

The two-stage decommissioning strategy (Stage 1 and Stage 2) and the above-mentioned projects require a licensing procedure for getting a permit from the competent authority BNRA.
The activities during the **Pre-decommissioning stage (PDS)** (from reactor final shut down to the **Stage 1**) are under the license for operation in "E" mode and are not part of the requirement for a decommissioning permit. These are mainly *Preparation of documentation* for issuance of decommissioning permit and *Preliminary activities* for decommissioning (provision and development of suitable utilities for dismantling, cutting, fragmentation, screening, size reduction, decontamination and measurements to release from control, and removal of: inflammable and hazardous materials and conventional waste, thermal insulation, operational RAW, spent ion exchange resins retrieval and conditioning, system isolation and draining). These activities and projects will ensure the necessary facilities for implementation of the proposed activity in accordance with the nuclear safety and radiation protection requirements, standards and rules.

Decommissioning of KNPP Units 1-4 begins with **Stage 1** in accordance with the Updated decommissioning strategy [7]. This stage covers the activities related to the Preparation and Operation of the Safe Enclosure (SE) of the Reactor Buildings (RB) and dismantling of the equipment outside the SE area.

Stage 2 consists of activities related to the deferred dismantling of the equipment within the SE area and release of buildings for use for other purposes.

The last stage of the decommissioning process is Close down and land reclamation.

The decommissioning of Units 1-4 will take place simultaneously with the operation of Units 5-6 and the DSFS storage, which have passed successfully separate EIA procedures [37, 87].

A major project related to the Units decommissioning activities is the construction of a National disposal facility for low and intermediate level radioactive waste near KNPP. This project has a separate EIA, approved by the MoEW.

For the planned projects "Facility for treatment and conditioning of solid radioactive waste with high volume reduction factor" and "Construction of heat generation plant (HGP)" separate procedures for EIA elaboration are performed.

NPP decommissioning is described in Appendix I of the ESPOO convention [184]. In this regard an EIA Procedure in Transboundary context was initiated and the IP was notified to Republic of Romania, as concerned party according the EIA convention in Transboundary Context. Upon receipt of the relevant notification, the Romanian government decided to participate in the EIA procedure. In the course of the scoping process Republic of Romania has formulated its general and specific considerations which have been considered in the present EIAR.

The ToR for determining the EIA scope and content [9] and the necessary structure of the report serve as the basis of the present study.

1.1.1 General description of the site of Kozloduy NPP and the site of Units 1 to 4 planned to be decommissioned

The Kozloduy Nuclear Power Plant was erected in north-western Bulgaria on the right bank of the Danube River close to the Town of Kozloduy. Detailed description of the plant location is included in section 1.1.3.

From an administrative point of view Units 1-4 are separated from Units 5 and 6. Auxiliary Buildings 1 and 2 (AB-1 and AB-2) belong to Units 1, 2 and Units 3, 4 respectively, whereas AB-3 belongs to Units 5 and 6.

Kozloduy NPP consists of the following set of facilities:

- Main Building (Reactor Building and Turbine Hall) – the Main Buildings of Units 1 and 2 and of Units 3 and 4 are shared;
- Auxiliary Buildings – AB-1 and 2 service Units 1-4 of Power Production 1, whereas AB-3 services Units 5 and 6 of Power Production-2;
- Water Demineralization Plant -1: services Units 1-4;
- Water Demineralization Plant -2: services Units 5 and 6;
- CPS-1 and 2 service Units 1-4, whereas CPS-3 and 4 service Units 5 and 6;
- DGS-1 and 2 service Units 1-4, whereas DGS-1, 2 and 3 cells service Units 5 and 6;
- Switch Yard – consists of three parts: 110 kV, 220 kV, 400 kV;
- Administrative Buildings;
- Auxiliary Buildings.

The structures and equipment of the main building have foundation at different elevation depending on their design function. Due to the presence of loess soil during the construction of the buildings, the soil was excavated up to the absolute elevation

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

of 32.0 m to 24.15 m. Depending on the respective building foundation elevation a loess – cement based embankment was formed; after that the waterproofing was implemented and the relevant building foundation was fulfilled.

The set of equipment of the Main Building of Units 1 to 4 consists of the following:

- Turbine Hall;
- Longitudinal rack of the electric devices;
- Device compartment;
- Crosswise rack of the electric devices;
- Ventilation centre;
- Emergency boron compartment for each Unit.

Units 1 and 2 and Units 3 and 4 respectively were built as twin-units. The main peculiarity of this type of construction is the design layout of two reactors in a common building with a shared Central Hall. There is a single Turbine Hall (TH) for the WWER-440 units. There is no significant difference between the premises and the situation of the equipment in the premises. The number of premises is consistent with the primary and auxiliary equipment that has to fit into them. A significant difference between the premises is the presence of an Emergency Control Room (ECR) in Units 3 and 4.

Some of the installed systems are shared by two different units, whereas other installed systems are shared by all WWER-440 units.

The RB structures from elevation -3.35 m to elevation +10.50 m, which serve for biological protection, are mainly of cast-iron reinforced concrete. The basements of the turbine generators in the Turbine Hall are also of cast-iron reinforced concrete. The covers of the roof and between the floors of the Main Building are large reinforced concrete slabs.

The civil structures of the hermetic rooms of the RB are calculated to stand an overpressure in the hermetic rooms equal to 0.1 MPa.

The Auxiliary Building is designed for decay storage of liquid and solid radioactive waste and contains facilities for processing of liquid and gaseous radioactive waste, gas purging systems, ventilation systems of the auxiliary building and equipment for preparation of boron solutions.

The Auxiliary Buildings (ABs) are connected via corridors with a band of w indows to the Reactor Buildings (RBs). The general ventilation extraction duct from the Reactor Building to the stack is at elevation +10.50 m and passes through the upper part of the gallery.

A common ventilation stack is built for two units with height 150 m.

1.1.2 General characteristics of the KNPP Units 1 to 4

NPP "Kozloduy" consists of 6 units. The Reactors of Units1-4 are of WWER-440/230 type and they were put into operation in the period from 1974 (Unit 1) to 1982 (Unit 4). The Reactors of Units 5–6 are of WWER – 1000/B320 type.

The common characteristic features of Units 1-4 are as follows:

- The Reactors are the pressurized water type, with two circuits – primary and secondary. The electrical output of Units is 440 MWe.
- The fuel used in WWER-440 is UO_2 with enrichment up to 3.6 % for ^{235}U.
- The Units have six loops in the primary circuit, with six main circulation pumps and six steam generators.
- Each of the WWER-440 reactors and its primary circuit is located in a concrete system of hermetically sealed rooms.

The construction of Kozloduy NPP has been carried out in three stages:

- First stage - construction of Units 1 and 2; design name Kozloduy – I;
- Second stage - construction of Units 3 and 4; design name Kozloduy – II;

Units 1 to 4 have the following data of the operation history given in the table 1.1.2-1 below:

Table 1.1.2-1 Data of the operation history of KNPP Units 1-4

Characteristic	**Unit 1**	**Unit 2**	**Unit 3**	**Unit 4**
Beginning of the construction	April 1970	April 1970	October 1973	October 1973
Physical start-up – minimum controllable level	30.06.1974	22.08.1975	4.12.1980	25.04.1982
Energetic start-up	24.07.1974	24.08.1975	16.12.1980	17.05.1982
Start-up: First TG Second TG	30.07.1974 02.08.1974	24.08.1975 26.09.1975	17.12.1980 11.01.1981	17.05.1982 23.05.1982
Bringing the Reactor to full power	25.10.1974	0711.1975	27.01.1981	17.06.1982
Realized fuel cycle up to 2000 inclusive for Units 1 to 2 and 2006 for Units 3 and 4	18	19	22	21
Number of equivalent full power days till the end of 2000	5474.0	5416.87	4832.75	4735.8
Average factor of the output usage during unit operation for the operation period till the end of 2000	82.46 %	80.45 %	83.86 %	84.3 %
Quantity of the electrical power produced for the time of operation till the end of 2002 for units 1&2 till the end 2006 for units 3&4 (MWh)	66 675 000	68 905 000	68 703 000	66 711 000

Main technical and technological characteristics of the Units

KNPP-1 and KNPP-2 both consist of two energy units with an electrical capacity of 440 MW each.

The main layout of Units 1 and 2 is included in Chapter 11 in figure 11.1.2-1, while figure 11.1.2-2 presents the technological scheme of the primary circuit.

The thermal scheme of the units is shown on figure 11.1.2-3. The remaining units have a similar main layout and respectively technological scheme of primary circuit and thermal scheme. The secondary circuit purpose is to take up the thermal energy of the primary circuit and to transform that thermal energy into electrical one, ensuring high efficiency of the process. Then, the electrical energy is transferred through the Switch Yard to the Integrated Energy System (IEE), and to the consumers.

Demineralized water under a pressure of 12.5 MPa (operation pressure in nominal mode) is used as a moderator and coolant in the Reactor. The fuel for the Reactor is slightly enriched uranium; the concentration of Uranium-235 is up to 3.6 %.

The primary circuit, which is radioactive, consists of a Reactor and six circulation loops. The secondary circuit is not radioactive, consists of the following: steam supply part of the steam generator, a turbine and auxiliary equipment of the Turbine Hall.

Units 1 to 4 of KNPP are equipped with two turbine generators of K-220-44 type, each of them of 220 MW output. They operate with steam under a pressure of 4.4 MPa.

The voltage of the generators, TBB-220-2 type, is 15.75 kV, the power factor is 0.85; with water cooling of the stator and hydrogen cooling of the rotor.

The unit house load power supply is provided by a voltage of 6 kV and 380/220 V.

Service water supply is provided by direct flow with water taken from the Danube River.

Main equipment and layout of the main components

Primary circuit

The primary circuit is radioactive and includes the main circulation circuit and a number of auxiliary systems. Here below are the main ones among them:

- Main circulation circuit - Reactor, Main Circulation Pipelines, MCPs, MIVs and SGs;
- Pressurizer system for the compensation of the temperature changes of the coolant volume;
- System for water purification of the primary circuit;
- Make-up system for the primary circuit;
- Boron injection system.

The main circulation circuit consists of a pressurized water Reactor and six circulation loops. Each circulation loop consists of a SG, MCP and pipelines of DN 500 mm.

Reactor

The source of heat energy for the NPP is a heterogeneous pressurized water reactor of WWR-440 type of 1375 MW heat capacity and it operates with thermal neutrons. The

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Reactor is a vertical cylindrical vessel, which comprises of a pressure vessel and a removable upper head with a spherical cover.

The Reactor consists of the following main elements:

- Reactor Pressure Vessel;
- Pressure Vessel Internals — shaft, bottom of the shaft, basket, protection tube bank;
- Upper Head;
- Fuel Assemblies;
- Control Rod Drives.

Steam generator

The capacity of the SG is 452°t/h with a dry saturated steam under a pressure of 4.7°MPa. The SG has a single shell and it is an evaporator with horizontally arranged tube bundles.

Main circulation pump

The MCP is a vertical glandless centrifugal pump with integrated asynchronous motor with three-phase AC and a short-circuited rotor. The rotor of the motor and the bearings are in the operation cavity of the pump. The pump body is tightened to provide the leak tightness of the primary circuit.

Main isolation valves (MIV)

MIV are installed on each of the loops – two per loop, one on the hot leg and one on the cold leg.

Main circulation pipelines

Main circulation pipelines connect the main equipment of the reactor system with the reactor. The material used for the pipelines is stainless steel and the pipe diameter is 496 mm (560x32).

Pressurizer

The pressurizer is connected to the main circulation pipelines by two pipe connections to the “hot” part of a loop and with one connection to the “cold” part of a loop.

There are three safety valves mounted at the pressurizer. The valves release steam through nozzles into the relief tank. The gases released by the safety valves are sucked in the gas purification system.

Secondary Circuit

The secondary circuit includes the section outside the SG tube bundle, a turbine and auxiliary equipment of the Turbine Hall and its serving systems.

The main ones among them are as follows:

- Turbine;
- Lubrication system of the turbine generator;

- Regenerative systems;
- Condensers;
- Deaerator and feed water system;
- Main steam pipelines;
- Residual heat removal system;
- Technological steam supply and collection of the condensate;
- Services water supply;
- Generator;
- Generator sealing system;
- Generator stator cooling system;
- Gas-oil cooling system.

Service Water Supply

The service water is supplied from central pumping station 1 for Units 1-2 and central pumping station 2 for Units 3-4, and includes the following:

- service water supply of safety-related consumers, including intermediate circuits of cooling;
- service water supply of consumers, which are not safety-related;
- circulation water supply.

Main Generator

The purpose of the TBB-220-2 generator is to produce electrical power and it is connected to the rotors of the turbine, after the 2 LPT. An auxiliary TBS -6-2 generator, intended to produce electrical energy for the plant home loads is directly connected to main generator.

Transport and Technological Part

The transportation and technological equipment is designed to perform all of the operations related to the receiving of fresh fuel, reactor core refuelling and transportation of the spent fuel assemblies to the SFSP of the Units.

Purification of the radioactive medium and Radioactive Waste Storage

The special water treatment system of the NPP radioactive water is designated to maintain the water regime in the main and auxiliary circuits, to process and purify the radioactive water and to store the radioactive waste.

Contaminated waters (radioactive and contaminated with salts) come into the system.

<u>SWT-1</u>

Special Water Treatment system 1 is designed to purify the primary circuit coolant and consists of three filters: a mixed bed filter in a hydrogen form, a cation filter in a mixed potassium-amine form and an anion filter in a boron form.

SWT-2

Special Water Treatment system 2 is designed to purify the primary circuit drainage water. The system consists of three successively operating filters, two cation exchange filters and one anion exchange filter in a boron form. Capacity of the facility – 40 t/h purified water stored in contaminated condensate tanks. SWT-2 is one (operating); there is no standby unit. Special Water Treatment system 2 is identical for Units 3 and 4 as well.

SWT-3

Special Water Treatment system 3 is designed for treatment and purification of drain waters and of contaminated condensate from mechanical, chemical and radioactive contamination. The system consists of two evaporators, which operate under atmospheric conditions. For the purpose of the additional purification of the distillate, two interchangeable filtration installations are provided. One of them has got two cation exchange and one anion exchange filters, whilst the other one has got one cation exchange filter and one anion exchange filter. Capacity of the Facility is 6°t/h; the purified water is discharged into the Danube River after control. SWT-3 is common to Units 1 and 2 - the two evaporators can operate independently from one another. SWT-3 can also process the water from contaminated condensate tanks. SWT-3 is identical for Units 3 and 4 as well. SWT-3 is fully operable.

SWT-4

Special Water Treatment system 4 is designed to purify periodically the SFSP water and to purify the emergency boron tank water. SWT-4 consists of successively operating mechanical, cation exchange and anion exchange filters. Capacity of the facility – 40 t/h; the SFSP purified water is returned into the SFSP; the purified water of emergency boron tank is directed back into the emergency boron tank and can be directed to the contaminated condensate tanks for storage.

SWT-4 is common for Units 1 and 2.

The system for SWT-4 is identical for Units 3 and 4 as well.

SWT-5

The SG blow down water after the blow down coolers is processed to the filter part of SWT-5 for regulation of the SG water chemistry regime.

SWT-6

Special Water Treatment system 6 is designed to purify boron water from primary circuit in order to receive clean solution of Boron acid for re-use in the technological cycle. This system is installed only at Units 3 and 4.

Liquid Waste Storage Facility at AB-1

This storage facility serves for long-term storage of the following:

- Concentrate from the evaporators of SWT-3;
- The exhausted filtering materials of the ion exchange filters of the SWT 1 to 5.

The filtering materials enter into the liquid waste storage facility along the pipelines used for hydraulic discharge. The residue concentrates are moved along the pipelines using compressed air.

In the liquid waste storage facility there are:

- Five storage tanks (1÷5 ECT) each of 400 m^3 usable capacity to store radioactive concentrates;
- Two storage tanks (1, 2 HAST) each of 400 m^3 useable capacity to store high radioactive spent ion exchange resins from the SWT-1, 2, 4;
- Two reinforced concrete storage vessels (1, 2 LAST) each of 180 m^3 useable capacity to store low radioactive ion exchange resins from the SWT-3 and 5.

The storage tanks of the high radioactive waste are made of stainless steel; the vessels are arranged into reinforced concrete tanks which are lined with stainless steel.

The liquid waste system is in operation and its current state is as follows: 1 to 5 ECT full of 80 %; 1, 2 LAST - full of 216 m^3 of low level resins; 1 full of 131 m^3 high level resins; 2 HAST is an inoperable vessel and it is empty.

Liquid Waste Storage Facility at AB-2

Received liquid RAW are stored in tanks located in the AB-2, at elevations from -2.0 m to +6.30 m between axles 7a – 19. Each of the tanks is located at own room.

Storage facilities for liquid radioactive waste include the following tanks:

- Two storage tanks for high radioactive spent ion exchange resins (1, 2 HAST) from the SWT-1, 2, 4. The treated water is with high activity of 10-5 ÷10–2 Ci/l. There is a possibility to store also high activity sorbents from SWT-6 as well as high activity sorbents from SFS.
- Two storage tanks for low radioactive spent ion exchange resins (1,2LAST) from SWT-3 and SWT-5;
- Five storage tanks (1÷5ECT) for radioactive concentrates from evaporators of SWT-3;

1, 2 HAST are located in BK-013/1, 2 room. Each tank is a cylindrical vessel with a diameter of 9000 mm and height 6500 mm. Made of stainless steel 0X18H10T. The vessels are arranged in rooms which are lined with stainless steel 0X18H10T. The weight of each HAST is 12.4 t and have capacity of 420 m^3.

1, 2 LAST are designed to keep exhaust filters ion exchange resins from the SWT-3 and SWT-5. The treated water is with activity of 10-8 ÷ 10–10 Ci/l

1, 2 LAST are located in room BK-015/1, 2 from AB-2 at elevations from -2.0 m to +6.30 m. LAST are with size 4.8x4.5x8.4 m and have capacity of 180 m^3 each. The tanks are rectangular, reinforced concrete and are lined with stainless steel 0X18H10T.

1, 2, 3, 4, 5 ECT are designed to keep radioactive concentrates from evaporators of SWT-3 with activity of 10-2÷10–4 Ci/l. Tanks are five in number and are made of stainless steel 0X18H10T. The tanks are located in rooms BK-016/1÷BK-016/5, which are lined with stainless steel. Each tank is a cylindrical vessel with a diameter of 9000 mm and height 7000 mm. The weight of each ECT is 14 t and has a capacity of 500 m^3.

Two montejus are part of liquid waste storage facility at AB-2. Each montejus is cylindrical vessel with diameter of 2000 mm. Made of stainless steel 0X18H10T. The weight of each montejus tank is 3.27 t and has capacity of 10 m^3.

Resins and activated carbon filters are unloaded by hydraulic to storage for liquid waste. The unloading of evaporator concentrate is done with compressed air.

Liquid waste storage facility at AB-2 is in operation.

Solid Waste Storage Facility

Storages in AB-1

The storages are located in a building having reinforced concrete structure, an area segregated from auxiliary building -1 (AB-1) that services Units 1 and 2.

The storages for solid RAW are of the bunker type with a hatch on top, seven in number, with volume ranging (from 80 m^3 to 230 m^3) and useful capacity of 1010 m^3.

“Storage pit” – Reactor Building of Units 1 and 2

It is used for temporary storage of solid RAW category 2-III originating from the operation of nuclear reactors.

It is located in the Reactor Hall of Units 1 and 2 (room A-301), total useful capacity of 81 m^3.

Current state:

- Room BK 111, volume -160 m^3 – the cell is empty and cleaned;
- Room BK 030, volume - 180 m^3 – the cell is 40 % full. The stored waste is 70 % mixed type of RAW, packed in polyethylene bags and 30 % of them are in bales.
- Room BK 031/1, volume - 120 m^3 – the cell is 90 % full. The stored waste is of aerosol filters;
- Room BK 031/2, volume - 120 m^3 – the cell is 20 % full. The stored waste is a heating chamber of SWT-3, 3 pcs of MIV bodies and lids of the MIS;
- Room BK 031/3, volume - 120 m^3 – the cell is 95 % full. The stored waste is of metal parts and cables;
- Room BK 031/4, volume - 230 m^3 – the cell is 90 % full. The stored waste is of mixed RAW, packed in polyethylene bags;
- Room BK 031/5, volume - 63 m^3 – the cell is 90 % full.

The Storage Pit-1 is 65 % full.

Solid Waste Storage Facility for intermediate and low level waste in AB-2.

The storage facilities are located in room A-301 at AB-2.

The solid RAW are stored at "Storage pit" with volume of 81.6 m^3, located in main room A-301. The stored inside RAW are with activity of 10°mSv/h.

Solid RAW are stored in temporary storage facilities type bunker, with hatch. The capacity varies between 80 m^3 and 230 m^3. Total capacity for temporary storage is 1010 m^3. In reinforced concrete structures of AB-2 are stored:

- Low activity aerosol filters;
- Solid waste;
- Soft (supple) waste packed in plastic bags.

The system is in operation.

Decontamination facility for Unit 3 and 4

The system for solutions and preparation of pure condensate for decontamination is used to prepare solutions for decontamination of equipment and facilities in the RB. The decontamination system consists:

- System pure condensate for decontamination;
- System acid for decontamination;
- System alkali for decontamination.

The main equipment of the system relating to the preparation of acid and base for decontamination is located at room BK-201 of AB-1 and is serviced by the personnel of Units 1 and 2. The submission of pure hot condensate for decontamination is also from AB-1.

The system is in operation.

System for purification of gaseous waste

During the NPP operation a constant blow down of the radioactive gases is performed. The blow downs, which are purified from the radioactive gases, enter the suction side of the gas blower and they are released into the atmosphere through the ventilation stack which is provided with a radiation monitoring for the releases of noble gases, ^{131}I, ^{133}I and of β aerosols. The system consists of two channels, which allows for the regeneration of the zeolite filter of the stand by channel without interrupting the operation of the system. The system is in normal operational state.

Heating, Ventilation and Conditioning

The technological ventilation of the NPP is designed to establish normal working conditions for the personnel and to remove the excess of heat and humidity from the operational rooms.

The technological ventilation systems are designed as autonomous systems for the following categories of rooms in the Controlled Area and the Supervised Area depending on the designation of the rooms and the equipment installed inside them:

- Non-serviced;

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

– Semi-serviced;

– Serviced.

The non-serviced compartments, during the operation of the equipment are characterized by high gamma background and possible radioactive airborne contamination.

Special extraction ventilation is provided in the non-serviced hermetic rooms that have been designed for over pressure. The extraction ventilation establishes enough under-pressure in the hermetic rooms and thus prevents the distribution of the radioactive air into the other operational rooms. The extracted air stream is processed by high efficiency aerosols filters (HEPA) and by charcoal filters (removal of iodine) before releasing into atmosphere.

The release of air from all the ventilation systems into the atmosphere is performed through a ventilation stack with a height of 150 m and provided with a radiation monitoring system (KALINA chain).

Systems for technological ventilation are planned for the KNPP Units 1-4 main and auxiliary buildings.

The ventilation systems are in normal operation.

WWER 440 Units Power Supply System

The KNPP is designed to supply power that meets the requirements of the power system of the Republic of Bulgaria. According to the design of KNPP their power supplied is at 220 and 400 kV voltage. NPP 880 MW first line power supply is at 220 kV voltage. NPP 880 MW second line power supply is at 400 kV voltage.

At each pair of Units in KNPP there are 4 generators of the type TBB-220-2 (two per each Unit) installed with 220 MW power and 15.75 kV voltage.

The generators of Units 1-2 are connected to the 220 kV outdoor switchyard bus bars and those of Units 3-4 are connected to the 440 kV bars through the relevant unit transformers.

House Load Power Supply Scheme (HL)

During the normal power operation of the unit the normal power supply of 6 kV house loads for Unit 1 is achieved by two power transformers 1TO and 2TO (for Units 2 – 3TO and 4TO), with the follow characteristics:

– U_H=15.75/6/6;

– S_H=25MVA.

There are three windings and the primary winding is connected as a "firm connection" to the generator output. The secondary windings are connected to the operational power supply switchgears PA and PB. Switchgears PC, whose main consumers are MCP, are supplied with power from the auxiliary generator (house loads generator).

The power supply of the second category consumers (allowable power interruption up to 3 min) is provided by diesel-generators with a power of 1600 kW each and voltage of 6 kV. Six diesel generators are installed at Units 1-2 (3 for each Unit), four of them supplying power to a separate Safety System channel. The two third ones function as a back-up of the other two.

Main Control Room (MCR)

Two main control rooms are constructed (one per each Unit), which are located at an elevation of 9.60 m for Units 1-4. Control of all main and auxiliary mechanisms and elements of the reactor unit and the two turbo generators, including 220 kV and 400 kV breakers, is envisaged to be achieved from the MCR. Operational house load transformers at 15.75/6.3 kV, breakers at 6 kV with standby power supply of 6 kV switchgear, unit operational and standby transformers at 6/0.4 kV, elements of reliable unit power supply network (house load generators, reversible motor-generators, diesel-generator facilities and reliable power supply lines) are controlled from the MCR.

Dosimetric Control

Responsible for the radiation protection of KNPP staff is the Dosimetric control department which performs the following:

– Permanent and planned technological control of the radiation level and concentration of radioactive substances in the KNPP rooms and the adjacent areas;
– Control of the radioactive releases from the NPP into the environment (water, air);
– Control of the level of radioactivity of the main technological fluids and control of the equipment tightness to avoid radioactive releases into clean compartments and into the circuits;
– Automated control of the environment (AIS IHM and AIS ERM);

In-house Dosimetric Control

In the scope of the in-house dosimetric control the following is included:

– Control of the gamma dose rate in the rooms and reactor hall;
– Control of the concentration of the radioactive gases and aerosols in the rooms air;
– Signalling in the rooms;
– Individual dosimetric control of the NPP personnel;
– Control of the contamination of the personnel in the sanitary locks;
– Control of the contamination of the equipment surface and the rooms;
– Control of the radioactive releases from NPP.

The in-house dosimetric control covers the rooms of the main building controlled area.

The in-house dosimetric control is realised by multi-channel device System, portable instruments, a set of devices, and by sampling as well. KALINA monitoring system is used for control of the ventilation releases into the stack.

Special Process Control

The main function of the special process control is to control the activity of the fluids and control of the tightness of the equipment.

The special process control covers the main technological circuits of the unit, the technological blow down system and the technological ventilation system.

The control is realized by radiometers and sensors, included in the set of the device "System 8004-1". The system for radiation dosimetric control and for special process control "System 8004-01" is in operation using all its measuring channels in RB, AB-1 and AB-2. The radiometer for control over the air-born discharges to the atmosphere through the VS-1 and VS-2 - RCS 02-03 "KALINA" is in operation with all its measuring channels. The laboratory control in parallel of the air-born discharges into the atmosphere through the VS-1 and VS-2 is done once per 24 hours.

External Dosimetric Control

The main objective of the nuclear safety at Kozloduy NPP with the reactors of WWER-440 type is the protection of the operation personnel and of the population in the neighbourhood of KNPP from the effect of ionizing radiation in harmful doses. The external dosimetric control is dedicated to provide this principle. According to the Regulation [17] two statutory areas are set up around KNPP. The first area is called Precautionary Action Zone (PAZ) and the second one is called Surveillance Zone (Monitored Area (MA)).

The external dosimetric control includes the following two automated systems for radiation monitoring within a radius of 1.8 km:

- Automated Information System for In-house Monitoring (AIS IHM)
- Automated Information System for External Radiation Monitoring (AIS ERM)

The Automated Information System for External Radiation Monitoring includes main and control stations for monitoring of Gamma background, ground concentration of Iodine -131 in the air and water stations for SE Operation waste waters. Automated Information System for In-house Monitoring includes fourteen boards for Gamma-radiation background control in operation and an extension with two new stations or sixteen boards in total.

On the KNPP site there are differentiated radiation protection zones and access to each zone depends on the radiation conditions. In correspondence to the requirements for radiation safety, all premises of the main and auxiliary buildings have been regrouped into two zones:

- Controlled (restricted access) Area, where the personnel may be exposed to harmful radiation factors;
- Supervised Area, adjacent to the Controlled Area.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

The Controlled Area covers the following: RB compartments, SWT compartments and the ventilation extraction centre in the Main Building, the compartments used for the storage of liquids and dry radioactive waste, the special gas purification system and the tanks storage facility in the Auxiliary Building.

The degree of the biological shielding has been determined in the Technical Design taking into account the calculated exposure rates depending on the category of each room, the type of work performed as well as the duration of the servicing personnel's stay in the rooms.

The monitoring of the radiation environment is performed by the section Operational health physics control of the Operation department.

Special-statutory Areas (SSA)

According to the existing regulations in the field of nuclear energy some areas of special statute were established around the nuclear and ionizing radiation facilities. At the present time these zones are Precautionary Action Zone (PAZ) and Surveillance Zone (Monitored Area (MA)). The PAZ (re-estimated presently with a radius of 2 km, in relation to the shut-down of Units 1 to 4 at KNPP) is a statutory defined area, established to limit public exposure in the event of accidents.

The Surveillance Zone (Monitored Area (MA)) is an area around NPP outside the boundaries of the precautionary action zone in which the monitoring related to urgent radiation protection measures is set up. For KNPP the Monitored Area is a territory outside the PAZ where radiation monitoring on the environment and population takes place for the purposes of the urgent radiation protection measures. On the border of the Controlled Area limits are set for the effective dose for the population in case of Beyond Design Basis Accident (BDBA). In accordance with the relevant criteria the MA is set with a radius of 30 km around KNPP.

For the purposes of estimation and justification of the Special-statutory Areas (SSA) dose rate criterion for maximum individual dose at the border of the zone less than <50 mSv/a [11] is applied for the Precautionary Action Zone (PAZ). Under these conditions, taking into account the Design Basis Accident with verified and validated model, the boundary of Precautionary Action Zone (PAZ) is defined as 2.0 km towards the geometrical centre of the distance between the ventilation stacks (VS) of Units 5 and 6 of KNPP.

Within the boundary of the Precautionary Action Zone (PAZ) there are no settlements, whilst on the eastern border a small settlement is located – the village of Harletz; to the west the town of Kozloduy is situated.

Emergency Planning Zones

The emergency planning zones are defined in the On-site Emergency Plan as one of the measures for the protection of the population.

According to the Regulation for emergency planning and emergency preparedness in case of nuclear and radiation accident [27], the emergency planning zones are:

- On-site Emergency Planning Zone – Protected Area;
- Precautionary Action Zone;

- Urgent Protective Action Planning Zone;

Boundaries for the zones are defined based on the following criteria:

- The On-site Emergency Planning Zone – site boundaries are defined according to the design of the nuclear facility;
- For the Precautionary Action Planning Zone – the individual annual effective dose in accident conditions should not exceed 5 mSv outside the area border;
- For the Urgent Protective Action Planning Zone – the estimated maximum individual dose for a member of the public at the border of the zone and outside it during the beyond design basis accident are: Effective dose - 5 mSv during the first year after the accident; Absorbed dose - 50 mGy for the thyroid gland;

Based on the performed accident calculations for the maximum design basis accident and beyond the design accident of units WWER-440 (B-230) and WWER-1000 (B-320) as well as on their radiological consequences, in correspondence to the dose criteria for decision making about personnel and population protection at the early and middle phase of the accident development according to the Regulation for emergency planning and emergency preparedness in case of nuclear and radiation accident [27], the following emergency planning zones are defined:

- Zone No 1 – KNPP On-site Emergency Planning Zone;
- Zone No 2 – Precautionary Action Zone (PAZ) within the conventional radius of 2 km and with geometric centre between the ventilation stacks of KNPP Units 5 and 6;
- Zone No 3 – Urgent Protective Action Planning Zone (UPAPZ) within the radius of 30 km around KNPP;

Depending on the accident conditions in the emergency planning zones different measures for the protection of the personnel and population will be put in place. If required, the planned accident activities could be performed outside the boundaries of the zones.

All activities related to decommissioning of Units 1-4 will be concentrated mainly in Zone 1, but any of the planned activity during the decommissioning will potentially concern Zone 2 and Zone 3 as well.

In-house Radio-Ecological Monitoring (Environmental Radiation Monitoring)

The In-house radio-ecological monitoring is set forth in the long-term program of Kozloduy NPP [150], established in compliance with the regulatory documents, the good international practice as well as with the experience accumulated in the Radio-ecological (RM) Monitoring Department and it is coordinated with MoEW, MH and BNRA. Analyses of air, soils, waters, sediments, vegetation, and gamma-radiation background are performed. The objective of the monitoring is to define clearly and precisely the radiation characteristics of the environment, to check the compliance of the effective state to the Bulgarian legislation in force in this field, to estimate the modifications and trends in the radiation situation around KNPP.

The radiation monitoring in the scope of this program includes gamma-radiation dose rate control and laboratory analyses of samples collected from the main environmental components within the KNPP 100-km area.

Areas under Surveillance

For the purposes of the localisation of the radiation impact produced by NPP to the environment and life world, the following areas under surveillance are fixed:

- Kozloduy NPP industrial site;
- Precautionary Action Zone (PAZ) – 2.0 km of radius (re-estimated presently with a radius of 2 km, former 3 km, in relation to the shut-down of Units 1 to 4 at KNPP);
- Surveillance zone (Monitored area) (MA) – 30 km of radius;
- Benchmark outposts – up to 100 km of radius.

In these areas not only in-house monitoring, but also radiation monitoring under the responsibility of the regulatory authorities (Environmental Executive Agency - EEA, National Center on Radiobiological and Radiation Protection – NCRRP etc.) is performed.

Monitored Items

The monitored items in the separate areas are:

- Gamma-radiation background;
- Radioactivity in the main environmental components:
- Atmospheric air – aerosols and atmospheric depositions;
- Water – natural water reservoirs, drinking, ground and waste waters;
- Bottom sediments and algae;
- Soil;
- Vegetation.
- Radioactivity of foodstuffs and agricultural crop in the region.

Controlled parameters

The following main radiation parameters are studied:

- Equivalent gamma-radiation dose rate;
- Radioactivity and radionuclide composition of the samples

Basic requirements to the radiation monitoring of the environment are:

- Measurements and samples collection from points with unfavourable characteristics
- Carrying out of parallel monitoring in reference points where influence from NPP is not expected;

- Analysis of radionuclides being typical for WWER reactors;
- The Minimum detectable Activity –MDA (also called lower level detectable (LLD)) to be sufficiently low for detection of background activity;
- Quality assurance during all activity stages to be present.

In order more realistic results to be obtained, a verification between the results from the in house (self) monitoring and the monitoring performed by the Environmental Executive Agency (EEA) – MoEW and the NCRRP – MH is applied.

In the Monitored Area (MA) 36 local control stations (LCS) are installed for surveillance of the terrestrial ecosystems and 7 – for the aquatic ecosystems. The stations are of the following three types:

- Type A. These are 11 stations where the radiation characteristics of the air are controlled, including sampling and analysis of aerosol deposited on filters, samples collection from atmospheric depositions, soil and vegetation. The equivalent gamma dose rate and the integral gamma dose rate are measured by means of TLDs;
- Type B. These are 25 stations for sample collection from atmosphere depositions, soil and vegetations. The equivalent gamma dose rate and the integral gamma dose rate are measured by means of TLDs;
- Type C. These are 7 stations where samples from water, water sediments and algaes are collected. These stations cover the water basins in the region of the rivers Danube, Ogosta and Tsibritsa. The equivalent gamma-dose rate is measured too.

The control stations of the above indicated types are located in the KNPP industrial site zone, in the PAZ and in the 100- km area surrounding KNPP. Within the territory of these areas samples from potable water, foodstuff and green crop are collected and analysed.

Fig. 1.1.2-1 shows the layout of these stations for environmental radiation monitoring of Kozloduy NPP.

The radiation monitoring covers also items from KNPP industrial site. The scope includes measurements of the gamma-radiation background (7 control stations, 10 TLD installed along the NPP fence and 10 TLD installed along the SFSF fence), sampling and analysis of atmosphere depositions, vegetations and soils (at 7 control stations). Monitoring of ground water is performed by means of 80 test pits (piezometers) aligned following a special layout, monitoring of water in the spray pool and of the rain water and the waste water discharge system (see fig. 1.2.2-2 Radiation monitoring stations on the industrial site).

Detailed description of the controlled items, of the control type and periodicity, of the measurements and analyses conditions and of the method used in analysis, of the main technical tools and equipment are given in the Program for environmental radiation monitoring at Kozloduy NPP [150].

Within the PAZ on-line control is performed by the automated information system for external radiation control (AISERC) Berthold through 10 measuring stations. 2 of them situated out on the industrial site. The monitoring priority is the gamma-radiation background and the ^{131}I content in the near ground surface air. Associated to these stations, 3 automated meteorological stations perform the surveillance of the air quality in the near ground surface layer. For the identification of the air quality in the higher atmosphere layers (at 25 km of altitude) in order to determine the air-born pollutant migration, including radionuclides, an Automated system for aerologic sounding (ASAS) is used.

In the frame of the National Environmental Monitoring System (NEMS) the EEA performs environmental radiation monitoring covering the whole national territory and maintains a data base for this. The system provides in time and reliable information about the state of the environmental components and the factors of impact. On the ground of the above information analyses, assessment and forecasts are made for justification of the activities related to the environmental protection against adverse impacts. The indicators targeted for surveillance are: gamma-radiation background, natural and anthropogenic radionuclides content in soils, depositions, harmful products, specific alpha- and beta-activity, uranium, radium and tritium content in the surface and ground waters, radon exhalation at uranium mines, from the mining and thermal power industries and in the surrounding settlements, radionuclide content of the atmospheric air.

For the purposes of the on-line gamma-radiation background surveillance on the Bulgarian territory, a National automated system for permanent control of the gamma-radiation background (BULRaMo) is functioning since 1997 in accordance with the engagement for safe use of nuclear energy, the control over the environmental state and the on time mutual information in case of nuclear facilities accidents or transboundary effect.

The system is composed of 26 Local Monitoring Stations dedicated to gamma-radiation background monitoring, covering the whole country territory. More congested are the monitoring stations around Kozloduy NPP. All monitoring stations are equipped with the appropriate instrumentation and communication tools. To the stations, in case of incidents, mobile monitoring stations are attached in view of operational activity.

The AISIRC and BULRaMo systems are connected and functions as a whole with NEMS.

In case of accident situation the field measurements, samples collection and tests are performed using mobile laboratory for measurement of the gamma-dose rate in movement and in control points and of the aerosols activity of ^{137}Cs, ^{134}Cs, ^{131}I in the near ground surface air, the content of ^{137}Cs, ^{134}Cs and ^{131}I activity in polluted surfaces samples are performed.

Personnel situation of KNPP Units 1-4

In 2008 about 4485 employees worked in Kozloduy NPP. For Units 1 to 4 there were 1092 employees in 2008 whereas in 2002 there were still 1420 employees. During 2011 700 employees were at EP-1. Besides, on the site of the power plant there are different subcontracting companies (External Organization) operating so the number of the staff that is potentially exposed to the ionizing radiation was 3463. It must be

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

noted that this number does not include personnel from other structural units performing activities directly connected to Units 1-4, such as the "Safety" department, the "Security" department, the "Quality" department, the "PTC" department, "Diagnostics and Control", who will also be affected by the transition of the Units from "E" mode to decommissioning.

The necessary personnel for the different decommissioning stages of KNPP Units 1-4 is the following:

- Pre-decommissioning stage (PDS) – 695 people;
- Decommissioning stage 1 – 650 people;
- Decommissioning stage 2 – 212 people.

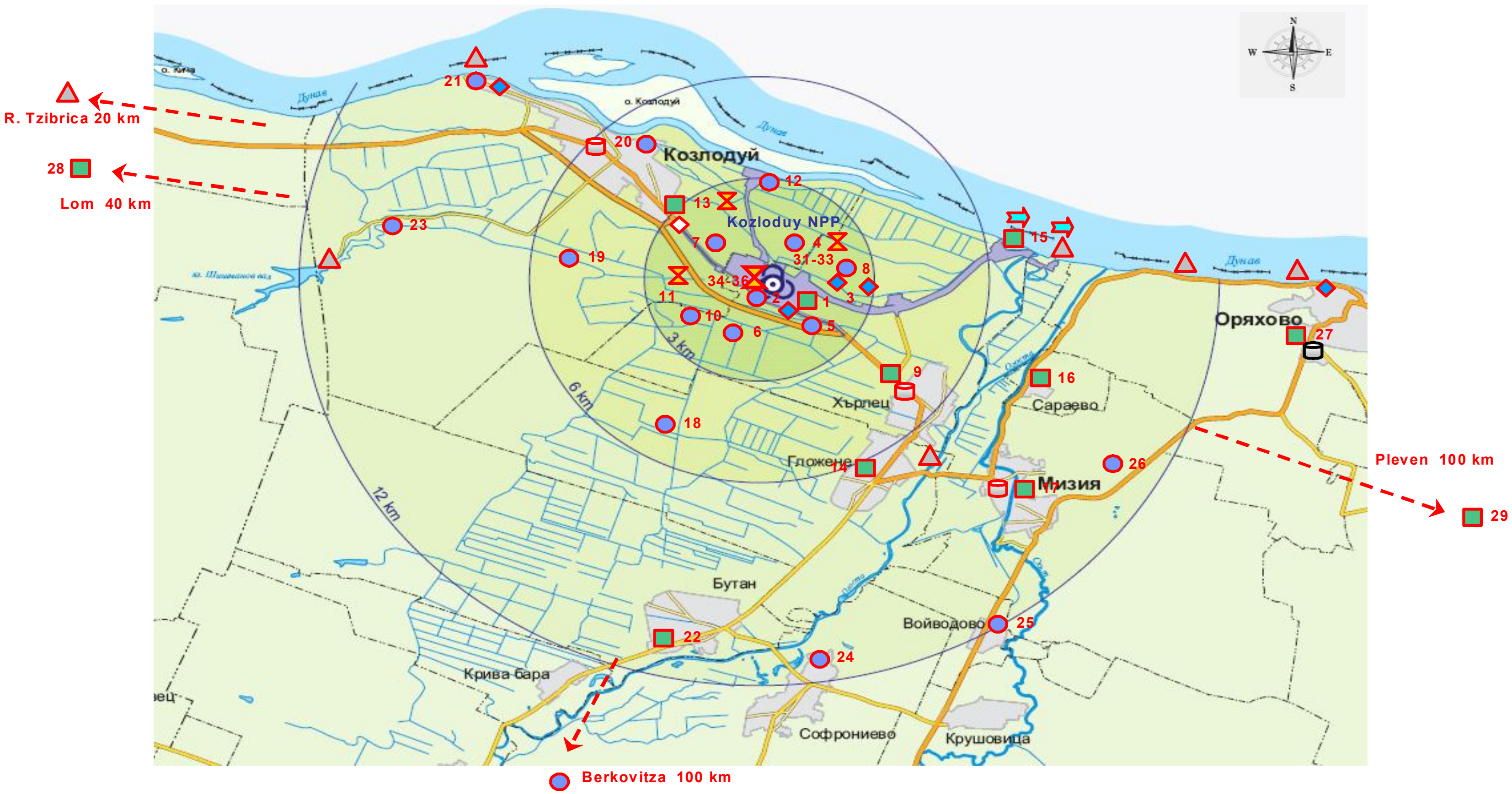


Fig. 1.1.2-1 Location of measuring stations for Environmental radiation monitoring at Kozloduy NPP.

- Station type ”A” aerosol, atmospheric depositions, soil, vegetation, gamma dose rate and the integral gamma dose TLDs – 11 numbers
- Station type “B”: atmospheric depositions, soil, vegetation, gamma dose rate – 15 numbers
- Station type “C”: water, water sediments, algaes, gamma dose rate – 7 numbers

nutrition chain products: -potable water - milk - fish - foodstuff and green crop

The Environmental Radio-Ecological Monitoring programme on the Romanian territory is described in the respective section of Chapter 3

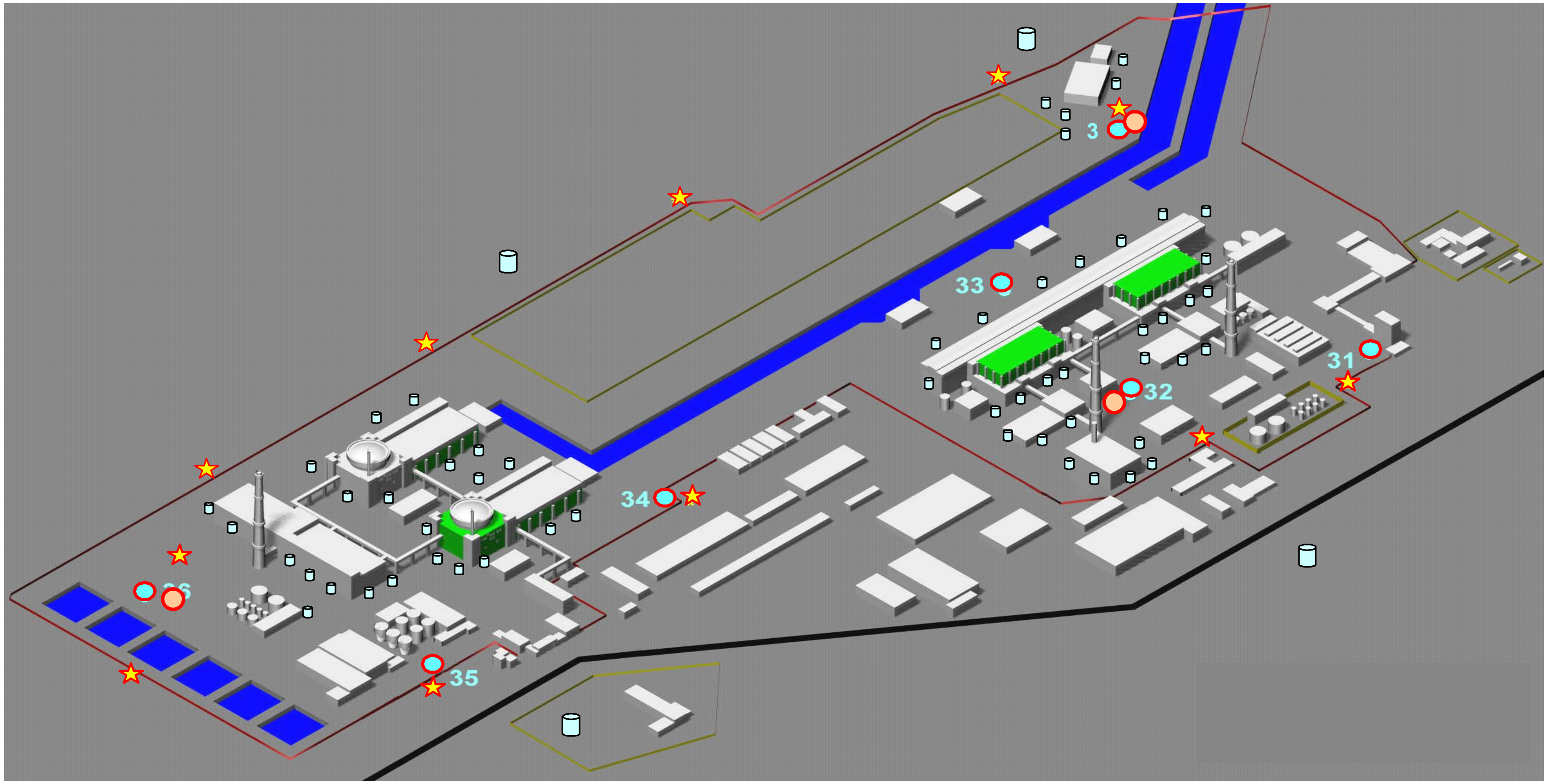


Fig 1.1.2-2 Location of measuring stations for in-house radio-ecological monitoring and TLD at Kozloduy NPP site.

Station type “B”: atmospheric depositions, soil, vegetation, gamma dose rate – 7 numbers
TLDs (type: TLE-4) located at the fence of NPP – 10 numbers
More than 180 boreholes for monitoring of soils, ground waters at the site of Kozloduy NPP and 4 bench mark boreholes outside of the plant.
Station for Aerosols monitoring – 3 numbers

1.1.3 Radiological status of the units

The radiological status of the Units 1 & 2 was investigated in the frame of the project "Project development for overall "Kozloduy NPP" PLC Units 1 and 2 radiation investigation" (Contract No 257000/24.06.2005, performed by EWN and ENPRO Consult LTD) and the results are presented in [59 ÷ 69]. Below a short description of the methodology is given and the main radiological results are summarized.

Methodology

The methodology is based on the approach to assess/consider and to carry out the following main items:

- operational history
- contamination paths
- contamination classification
- selection of measurement and sampling places
- execution of measurements and samplings
- evaluation of measurement and sampling results

The overall objective with these radiological investigations was to obtain an overview of the radiological status of the plant systems, buildings and ground areas, i.e. dose rates and contamination levels.

This methodology, described in detail in [59], was successfully applied for the decommissioning project of the 5 WWER-440 units of the Greifswald NPP and was the basis for the radiological investigation of the KNPP Units 1 and 2.

Turbine Hall (TH)

The radiological situation of Units 1-2 is comparable with the initial situation of the Greifswald NPP, because of the same type of nuclear power plant and a similar operation history.

The dose rate measurements were performed in a raster/grid of 6 m x 6 m on the levels -3.6 m, 0 m, 3 m, 6 m, 9.6 m, 14.7 m and 21 m. If necessary, i.e. radiation was detected, the raster/grid was reduced to 1 m x 1 m. In this case also contamination measurements were performed. The executed measurements did not show unusual results. At the levels -3.6 m and +14.7 m the dose rate is slightly higher than the normal level. The maximum detected values are 2.8 μSv/h and 8.0 Bq/cm² (see figure 1.1.3-1).

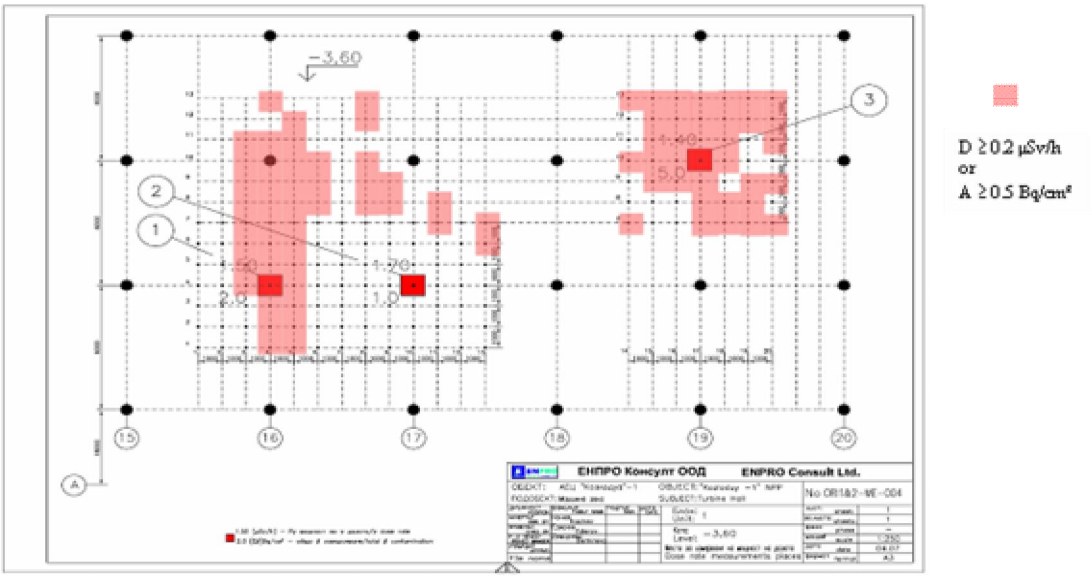


Fig. 1.1.3-1 Visualization of the results on the horizontal sectional drawings (raster/grid 1 m x 1 m) for the level -3.6 m of Unit 1.

Depending on the valid KNPP regulations, the whole area mentioned above should be either decontaminated before the start of the dismantling works or the dismantled material, especially material from non- or potentially contaminated systems, must be protected during the dismantling phase aiming to avoid any cross contamination.

The detected surface contamination inside the systems is also similar to the initial situation at Greifswald NPP. In total 95 samples were taken at 78 sampling places. 5 material samples were without a defined surface. The ^{60}Co surface contamination ranges between:

- 22 x > 0.1 Bq/cm² and ≤ 0.5 Bq/cm²
- 10 x > 0.5 Bq/cm² and ≤ 1.0 Bq/cm²
- 5 x > 1.0 Bq/cm².

Figure 1.1.3-2 shows some results of surface contamination of secondary circuit of KNPP Unit 1.

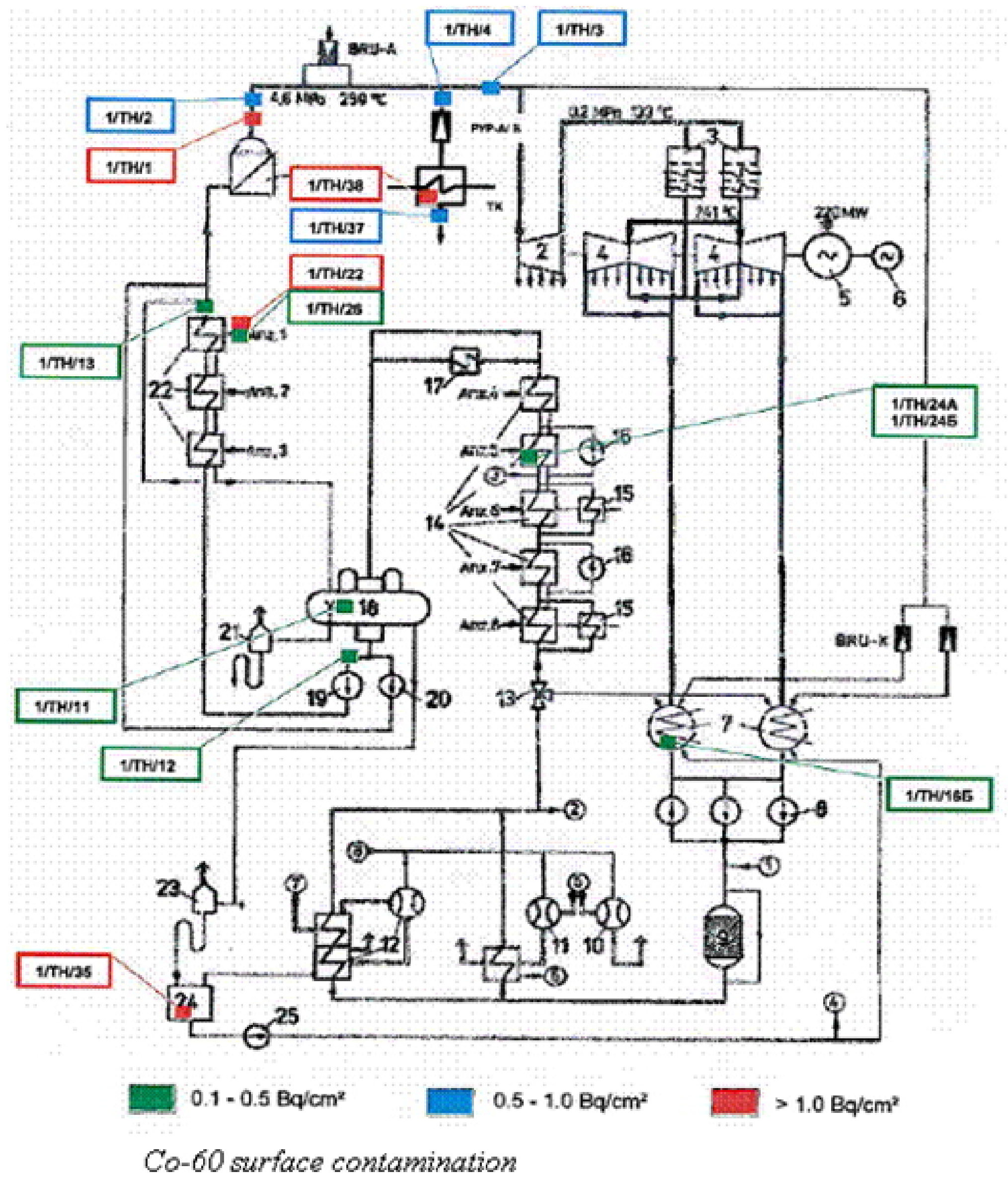


Fig. 1.1.3-2 Basic flow sheet of the secondary circuit with marked surface contamination [Bq/cm²] – Unit 1.

The assumption that some parts of the secondary circuit of the KNPP are slightly contaminated was confirmed. Thus, also the proposed methodology for classification of the systems into 3 categories in view of the dismantling and the material management including the free release measurement should be confirmed as well.

According to the sampling results the list of system classifications was updated, i.e. the classification of a system was changed in case the contamination was higher than 0.5 Bq/cm² (see example in table 1.1.3-1).

Table 1.1.3-1 Table with system classification (extract)

			not contaminated	**potentially contaminated**	**contaminated**
			Secondary Circuit		
35	Turbine *(Components of the turbine - high pressure and low pressure parts, rotor, condenser etc. - are part of the main steam system)*	1, 2			
	Turbine control and protection system		x		
	Operating (secondary) condensate				x
	Turbine drainage system			x	
	Turbine stuffing box steam ejector system			x	
	Turbine/generator bearing- and seal oil system		x		
	Oil drip tray of turbine bearing- and seal oil system			x	
37	Main steam system	1, 2			x
45	Final cool down and residual heat removal system	1, 2			x
77	Nitrogen system – high and low pressure	1-4	x		

For the nuclide vector determination of the TH Units 1 and 2 in total 14 complete analyses were used (In EWN 10 complete analyses were used for the TH Units 1-5). In table 1.1.3-2 the nuclide vector for the complete TH Units 1 and 2 is given:

Table 1.1.3-2 Nuclide vector for Turbine Hall Units 1 and 2

Nuclides	**Nuclide Vector TH01**
^{60}Co	14.0
^{137}Cs	34.0
^{241}Am	1.0
^{55}Fe	15.0
^{63}Ni	27.0

Nuclides	Nuclide Vector TH01
^{241}Pu	8.0
^{90}Sr	1.0

The detailed information about measurements, samplings, results and determination of nuclide vectors is given in the Technical Reports [61], [62] and [63].

Controlled Area

The results of the performed dose rate measurements are comparable with the situation at the Greifswald NPP, some years after its shut down. In many KNPP rooms the dose rate is relatively low. In some rooms the dose rate is rather high as for example in the rooms with filled tanks, filters and pipes. In total, measurements were performed in more than 400 rooms and sites (no access is provided to about 80 rooms, so these measurements will be executed later). The dose rate results available now are important and provide necessary information needed for the dismantling planning, in order to estimate and finally to limit the dose commitment as well as to plan shielding measures, decontamination measures (for example the hot spot removal).

In general, the level of detected surface contamination inside the systems is similar to the contamination level in EWN (1993/94). In total, 82 samples were taken at 67 sampling places. The ^{60}Co surface contaminations [Bq/cm²] of equal sampling places from Units 1 and 2 were compared – there are no major differences between both Units.

The assumption that there are some contamination paths to the secondary circuit (Turbine Hall) was confirmed. Thus, also the proposed methodology for classification of the systems into 3 categories in view of the dismantling and material management, including the free release measurement, is confirmed. For the nuclide vector determination of the Controlled Area 25 complete analyses, i.e. the nuclide specific alpha, beta and gamma evaluation, were executed by the radiochemical department of the KNPP. (In EWN 40 complete analyses were used for the Controlled Area Units 1-4.) In table 1.1.3-3 the nuclide vector for the Controlled Areas of Units 1 and 2 is given:

Table 1.1.3-3 Nuclide vector for the Controlled Areas of Units 1 and 2

Nuclides	Nuclide Vector CA01
^{60}Co	24.0
^{137}Cs	19.0
^{241}Am	1.0
^{55}Fe	33.0
^{63}Ni	19.0
^{241}Pu	3.0
^{90}Sr	1.0

The detailed information about measurements, samplings, results and the determination of nuclide vector are given in the Technical Reports [64], [65] and [66].

Buildings

In the frame of the above mentioned project a "Programme for the Radiological Investigation of Buildings and Concrete Structures" [67] was prepared for KNPP. The results of this programme will only give a first rough impression about the surface contamination and possible contamination penetration in the concrete. More detailed investigations are necessary with regard to building decontamination, free release procedures and concrete disposal strategies. All opportunities to receive more information about the contamination of buildings and concrete structures should be used during the decommissioning and dismantling phases. The status of the execution of the "Programme for the Radiological Investigation of Buildings and Concrete Structures" is on-going.

Land/soils

In 2008, EWN delivered a measurement programme to KNPP with the objective to get an overview about the present radiological situation, i.e. the status of sealed (e.g. streets) and not sealed (e.g. soil) land.

In connection with the preparation of this measurement programme KNPP provided to EWN available dose rate results of 28 point to point measurements from the KNPP site Units 1 to 4 and dose rate results from the street in front of the transport corridor of the controlled area Units 1 & 2. Actually, all these measurement results give only a very rough picture regarding the ground contamination situation of the four units. Therefore, these results could only be the starting point for the execution of this measurement programme, i.e. the dose rate measurements should be continued, supplemented by contamination measurements and measurements with the in-situ measurement device. At the moment within the KNPP site soil monitoring is performed systematically according to the Radio ecological programme and results show no pollution of the soil. This also confirms the results from the "Land Measurement Programme" execution [68].

Underground layers

In 2008, EWN delivered a measurement programme to KNPP with the objective to check the contamination status of the relevant underground installations. In case of inadmissible contamination, the necessary measures should be defined to avoid a contamination spreading to the environment. These could be e.g. technical measures for detection of the source of contamination and closure of the contamination path, cleaning measures on the corresponding system areas and/or periodical control measurement to guarantee the status quo.

In connection with the preparation of this measurement programme KNPP provided to EWN available results of the analysed samples of different shafts (see table 1.1.3-4). The results are given in Bq/l, i.e. water samples were taken and evaluated. These results already show the presence of contamination (lines 1, 2, 3, 13, 14, and 16).

Table 1.1.3-4 Activity results from the Liquid samples from KNPP EP-1 shaft measurements

No	Object	Note	Activity Bq/l					
			^{134}Cs	^{137}Cs	^{60}Co	^{54}Mn	^{58}Co	^{110}mAg
1.	"MS"-1		< 2.0	2.4 ± 0.4	< 1.9	< 2.0	< 1.8	< 2.4
2.	"MS"-2		3.0±0.4	88.5 ± 3.5	2.8± 0.4	< 2.1	< 1.9	< 2.8
3.	"MS"-90		< 2.2	69.0 ± 2.8	2.1± 0.3	< 2.0	< 2.3	< 2.7

No	Object	Note	Activity Bq/l					
			^{134}Cs	^{137}Cs	^{60}Co	^{54}Mn	^{58}Co	^{110}mAg
4.	"MS"-97*	The shaft is dry						
5.	"MS"-156		< 1.8	< 2.2	< 1.8	< 1.8	< 2.0	< 2.0
6.	"MS"-165		< 1.8	< 1.7	< 1.7	< 1.8	< 1.7	< 1.8
7.	"MS"-170		< 1.8	< 1.8	< 2.1	< 1.8	< 1.8	< 1.8
8.	"MS"-171		< 1.8	< 1.9	< 2.2	< 1.9	< 1.9	< 1.8
9.	"SFS-1		< 1.5	< 2.5	< 1.6	< 1.8	< 1.9	< 2.2
10.	SFS-2*	The shaft is dry						
11.	CS-18/1*	The shaft is dry						
12.	MS-1*	The shaft is dry						
13.	MS-100		< 1.8	14.3 ± 1.0	< 1.8	< 1.9	< 1.9	< 2.4
14.	MS-108		< 2.0	4.6 ± 0.6	< 1.9	< 2.1	< 2.0	< 2.7
15.	MS-162		< 1.9	< 1.8	< 2.1	< 1.7	< 1.8	< 1.8
16.	AB-1		16.7±1.2	8230.0±183.0	224.0±6.4	< 5.9	<5.9	< 9.2

***Sampling is not possible**

Record 19/21.11.2007 for the results from the gamma-spectrometric control of liquid samples from the revision and control shafts at KNPP EP-1 for November 2007, according to the site monitoring program

However, all results in the table indicate the contamination of water/liquids in 2007 (i.e. the water quality can change) but do not show the situation regarding surface contamination of shafts/pipes. Thus, it is not possible to evaluate the contamination situation of the complete system.

Therefore, the investigations proposed in the programme - dose rate measurements, surface contamination measurements (e.g. of the shaft material) and sludge or sediments samples (perhaps accumulated activity) - have to be performed in a manner allowing complying with the objectives of the report [69].

Units 3 and 4

Within the planned project "Evaluation of the Radiological Inventory of KNPP Units 1-4" it will be required to perform the radiological investigation also of Units 3 and 4 (at present the phase EO I is going on). Following the used methodology for Units 1-2 the basis for the evaluation will be a careful analysis of the operational history of Units 3 and 4.

Based on the existing information from KNPP and the EWN experience - decommissioning of 5 WWER-440 units - it is expected that the contamination values, the dose rate levels and the nuclide vectors of Units 3 and 4 will be very similar as the ones for Units 1 and 2.

Summary

The available radiological data of the KNPP and the information received from the "Project development for overall 'Kozloduy NPP' Plc. Units 1 and 2 radiation investigation" are comparable with the initial situation of the Greifswald NPP and were expected because of the same type of nuclear power plant and a similar operation history. Based on the radiological status of KNPP Units 1 to 4 it can be concluded that the level of impact on the environment during the decommissioning phases and the dismantling activities will be very low. Of course, a necessary condition is the exact consideration of all valid licenses, regulations, ordinances etc. applicable for these activities.

1.2 Connections with other existing/planned activities

The EIA report for the Decommissioning of Units 1-4 at KNPP will consider the aggregate impact of all design activities that are planned to be executed during the Pre- decommissioning stage (PDS), Stage 1 and Stage 2 of the decommissioning and during close down and land reclamation. Thus, the EIAR will analyse and describe the expected impact on the population and the environment resulting from all processes related to the decommissioning of KNPP Units 1-4.

There is taken into account the effect from the implementation of the IP in the context of other existing/planned activities on the IP site, interrelated to the decommissioning.

Existing activities (existing facilities)

The decommissioning of KNPP Units 1-4 will take place simultaneously with the operation of Units 5-6 and of Dry Spent Nuclear Fuel Storage (DSFS), which have passed separate EIA procedures [37, 87].

- Units 5-6

These units are currently in operation. There is an intention to increase their power rate as well as for a lifetime extension.

- Dry Spent Nuclear Fuel Storage (DSFS)

In DSFS on the KNPP site spent nuclear fuel assemblies will be stored in a specially designed cask system. The design life time of the facility will be at minimum 50 years. The spent fuel assemblies will be sealed in special storage casks ensuring their safety for the facility life time. This project has been subject to a separate EIA procedure finalized with positive MoEW decision.

Planned activities (other Investment Proposals)

- Project for National Disposal Facility for low and intermediate level radioactive waste (NDF)

The purpose of this facility is the storage of short lived low and medium level RAW. The facility is at the stage of preparation of technical design and safety analysis report. This project has passed separate EIA procedure and this EIAR has been approved by MoEW.

- Project – Facility for the Treatment and Conditioning of RAW with a High Volume Reduction Factor

The project shall ensure the supply of a facility with a high volume reduction factor for the treatment of solid low and medium activity RAW, which are currently stored at the KNPP site [51]. The facility shall be capable to reduce the volume of un-compacted as well as of compacted and super-compacted RAW. The facility will be installed inside the perimeter of KNPP site fence. The proposed location is within the Auxiliary Building – 2 (AB-2), elevation +6.30 m, Room BK 301. The Plasma System is a high energy technology able to treat a large range of wastes. In plasma technology a thermal plasma field is created by directing an electric current through a low pressure gas stream. A separate EIAR is in course of elaboration for the facility.

- Project – Heat Generation Plant for steam and hot water

The purpose of this project is the design, construction and commissioning of a Heat Generation Plant for generation of steam and hot water as a backup source for steam and network water for the KNPP Units 1-6 consumers. Pursuant to MoEW letter (ref. No B-1214/29.07.2009) this Investment Proposal is subject to a mandatory EIA procedure) Thus, the elaboration of a separate EIAR is planned for this project.

1.3 Location

For the decommissioning activities for Units 1-4 it is planned to include the territory of these units (fig. 1.3-1), the transportation of the generated waste to sites for decontamination and the safe temporary storage of the waste.

Fig. 1.3-1 KNPP Units 1-4 site

The planned territory for the needs of KNPP Units 1-4 decommissioning includes only the existing site of these units, meaning that the necessary area for the decommissioning is currently used by KNPP. The planned decommissioning activities are not expected to differ significantly from those currently performed at KNPP Units 1-4 site. After completion of all decommissioning measures and activities, the buildings planned for free release will be used for industrial purposes. Free release of territories for agricultural or forest needs is not expected.

KNPP is located at a distance of 120 km (as the crow flies) and at a distance of 200 km (via motorway) from the City of Sofia. The following municipalities are included in the 30-km zone around the site: Kozloduy, Valchedrum, Hairedin, Mizia (entirely) and Lom, Byala Slatina, Oryahovo (partially). A sparsely-populated part of the territory of Romania is also included in the 30-km zone around the site – namely 23 settlements including 2 towns – Dabuleni and Bechet and 21 villages- Nedeia, Gighera, Zaval, Ostroveni, Sarata, Calarasi, Listeava, Piscu Sadovei, Sadova, Gangiova, Macesu de Jos, Macesu de Sus, Sapata, Plosca, Bistret, Brandusa, Goicea, Barca, Horezu Poenari, Toceni and Valea Stanciului. The KNPP nearest populated settlements are the following: Town of Kozloduy – at 2.6 km to the south-west; Village of Harletz – at 3.5 km to the south-east; Village of Glojene – at 4.0 km to the South-East; Town of Mizia – at 6.0 km to the south-east; Village of Butan – at 8.4 km to the south; Town of Oryahovo – at 8.4 km to the east of the site.

The Kozloduy Nuclear Power Plant was erected in north-western Bulgaria on the right bank of the Danube River 5 km to the south-east of the Town of Kozloduy. The site is situated at the 694th km of the Danube River and at a distance of 3.7 km to the south of the fairway of the river and from Bulgaria's border with Romania (figure 11.1-3). The KNPP site is located in the northern part of the first non-flooded (loess) terrace of the Danube River at an absolute elevation of +35.00 m. The area of the site is planar with an average altitude from +28.00 m to +36.00 m according to the Baltic altitude system. The plane and the site are protected from the Danube by a levee reaching absolute elevation of +30.40 m. To the north it borders with the Danube lowland. To the south of the site, the slope of the watershed plateau is relatively high (100-110 m), to the west it is around 90 m, and to the east it is lower and descends to 30 m above sea level. The regional location of KNPP is shown on fig. 1.3-2.

Fig. 1.3-2 Regional location of KNPP site

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.4 Necessary areas

The total area of the site is approx. 3.2 km^2 and together with the channels for circulation and technical water supply reaches 5.2 km^2. The structures of the auxiliary buildings of Units 1 to 4 of Kozloduy NPP are situated within an area of 1.4 km^2. From an administrative point of view Units 1-4 are separated from Units 5 and 6.

Fig. 1.4-1 shows the plan of the Units 1-4 site.

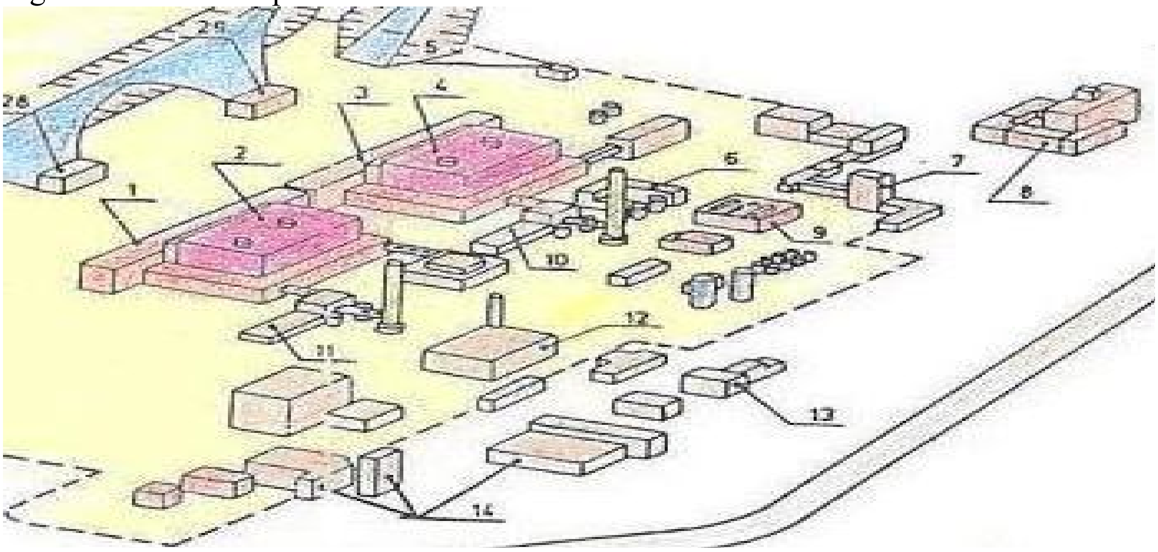


Fig 1.4-1 Plan of KNPP Units 1-4 site

Legend – 1. Turbine Hall Units 3-4; 2. Reactor Building Units 3-4; 3. Turbine Hall Units 1-2; 4. Reactor Building Units 1-2; 5. Nitrogen-oxygen station; 6. Chemical water purifying system Units 1-4; 7. Administrative building; 8. Training Centre; 9. Mechanical workshops for Units 1-4; 10. Auxiliary building for Units 1-2; 11. Auxiliary building for Units 3-4; 12. Spent nuclear fuel storage pond; 13. Information center; 14. Auxiliary administrative buildings.

1.5 Stages of implementation of the Investment proposal

Updated Decommissioning Strategy of Units 1-4 of Kozloduy NPP

On 25th April 2005 R. of Bulgaria signed the Agreement of Accession to the European Union, officially entered into force on 1st January 2007. As a part of this agreement, the Bulgarian government undertook to shut down the KNPP Units 3 & 4 by the end of 2006. Aiming to follow this obligation, an Updated Decommissioning Strategy for Units 1-4 of Kozloduy NPP [7] was prepared in June 2006.

The main cause for updating Decommissioning Strategy is the fact that Units 3&4 were shutdown at the end of 2006 and consequently the decommissioning process for these units has been become necessary. For this reason a new strategy was prepared taking into account the decommissioning of the Units 1-4, not only of Units 1 and 2. Also economical, social and other factors were taken into account.

The Decommissioning Strategy for Kozloduy NPP Units 1 to 4 is based on a review of the existing decommissioning design for Kozloduy Nuclear Power Plant Units 1 and 2, on a review of the latest international trends in decommissioning and on extensive decommissioning experience from projects in UK, France and other countries.

On this basis a totally new Decommissioning Strategy including Units 3 and 4 was established.

The Updated Decommissioning Strategy for Continuous Dismantling of Kozloduy NPP Units 1-4 is based on the following assumptions:

- Units 3 & 4 shutdown at the end of 2006;
- A decision on decommissioning of Units 1&2 as well as of Units 3&4 is made;
- The Dry Spent Nuclear Fuel Storage (DSFS) is commissioned;
- The Project of retrieval and treatment of the solidified phase of ECT is completed;
- The National Repository is commissioned.

The development of the updated strategy is based on a comprehensive analysis of the international experience, of the possible decommissioning options along with their advantages and disadvantages. The results of the analysis are presented in Attachment 12 of the Updated Decommissioning Strategy [7]. A comparative scheme between the updated and the original options is presented in Figure 1.5-1. It can be seen there that in the original strategy a gap of 35 years was planned between the preparatory work and the start of the dismantling so there is a break in the waste management processes. The continuous dismantling strategy envisages an earlier start of the preparatory work and the move to dismantling work is without a significant gap. The waste management processes are used continuously and the work load is more uniform. If the decommissioning permit for Units 1&2 comes in force in 2013 the dismantling activities could be started in the same year, which is four decades earlier than the time planned under the original strategy. The grounds of the updated decommissioning strategy propose an even, smooth and continuous use of the human and the financial resources as well as of the RAW treatment facilities.This continuous dismantling approach is a combination of both possible decommissioning options, namely:

- Immediate dismantling of certain facilities and equipment;
- Deferred dismantling of certain facilities and equipment.

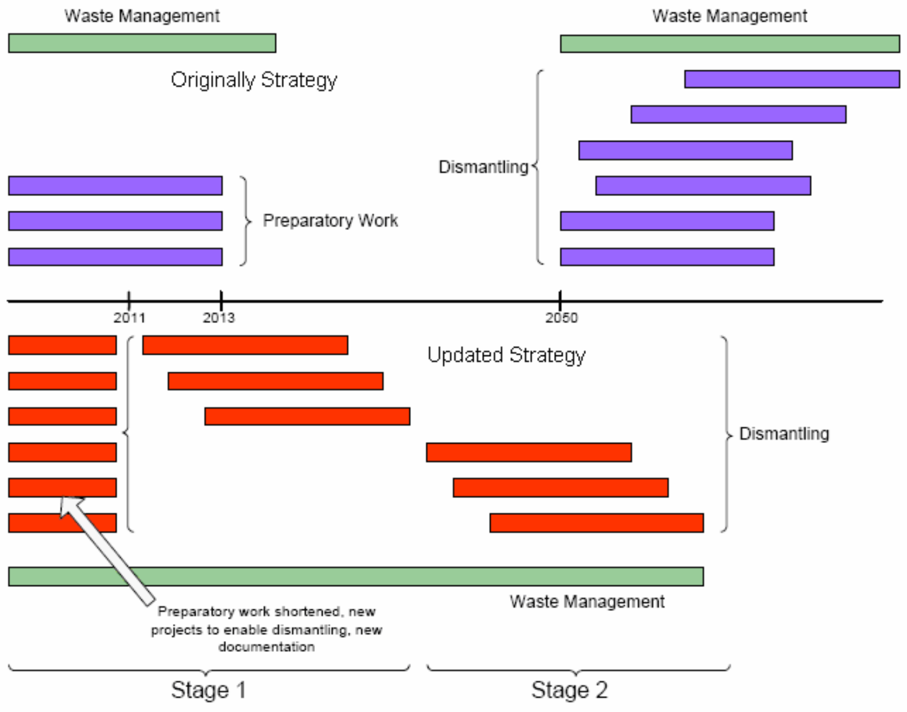


Figure 1.5-1 Comparative Scheme of both strategies.

The Updated Strategy Implementation does not allow any gaps in the usage of the decommissioning required resources (human, financial and technical).

The Auxiliary Buildings could be excluded from the SE area in order to enable continuation of the operational waste backlog processing.

The duration of the SE may be flexible in order to enable smooth continuation of the dismantling activities once the dismantling outside the SE area has been finished.

The Updated Decommissioning Strategy for Kozloduy NPP Units 1-4 was approved in June 2006 [7].

General Description of the Chosen Strategy

The Updated Strategy for Continuous Dismantling for decommissioning of Units 1-4 of Kozloduy NPP includes 2 Stages:

- Stage 1: Preparation and Operation of Safe Enclosure of Reactor Building 1, Reactor Building 2 and dismantling of the equipment that is outside the Safe Enclosure Area;
- Stage 2: Deferred dismantling of the equipment within Safe Enclosure and release of the buildings from regulatory control.

Stage 1 includes the phases of Preparation for Safe Enclosure and the Operation of the Safe Enclosure, as well as dismantling of equipment that is outside the Safe Enclosure Area. In compliance with the provisional schedule, the preparation for SE of the RB 1 and RB 2 will take two years, following the appropriate procedures described in [7] and will start will start upon entering into force of the Decommissioning Permit.

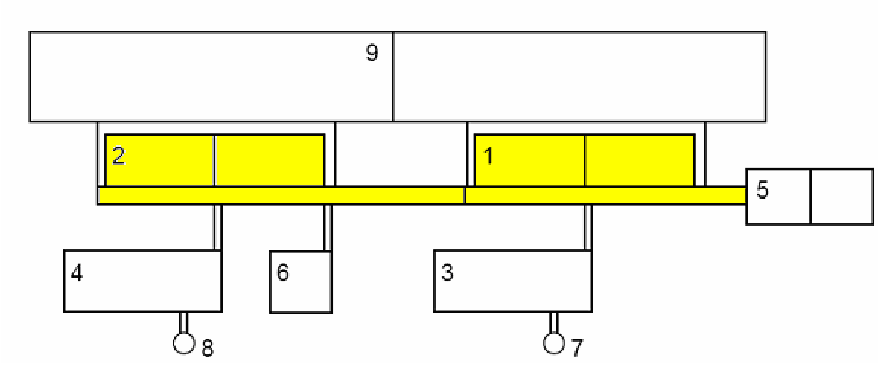
The dismantling of equipment outside the Safe Enclosure Area is planned to start in parallel with this and with the dismantling of equipment from the non-contaminated buildings and the Turbine Hall of Units 1 to 4, namely:

- Non-contaminated equipment dismantling;
- Dismantling of the turbines;
- Dismantling of the Secondary Circuit equipment.

These activities may be carried out in parallel or sequentially based upon availability of financing from the Nuclear Facilities Decommissioning Fund, capacity for sorting, fragmentation, and free-release measurement facilities as well as upon the employment policy of KNPP. The end of Stage 1 is predefined by the completion of the dismantling outside the Safe Enclosure Area.

Dismantling of equipment in the Safe Enclosure Area will be done during Stage 2. The end of Stage 2 will be predefined by the completion of the dismantling and of the activities related to the clearance of the building by the Regulator [36]. Under the Continuous Dismantling Strategy, the Auxiliary Buildings (AB) and the Ventilation Stacks (VS) are excluded from the SE area which would allow their use as radioactive waste management facilities before the start of dismantling activities in SE area.

The scope of SE will be limited to the Reactor Building 1 and Reactor Building 2 and the interconnecting passageways as illustrated in figure 1.5-2.



1 – Reactor Building I (Units 1 & 2)
2 – Reactor Building II (Units 3 & 4)
3 – Auxiliary Building 1
4 – Auxiliary Building 2
5 – Laboratory/sanitary building 1
6 – Laboratory/sanitary building 2
7 – Ventilation Stack 1
8 – Ventilation Stack 2
9 – Turbine Hall

*The yellow area is the SE area but also Laboratory/Sanitary Building 1

Figure 1.5-2 Planned Safe Enclosure Area under the Updated Strategy.

1.6 Project Implementation Structure

The integral process of preparation and decommissioning, according the adopted Alternative for Continuous Dismantling can be regrouped in the following main stages with the respective relevant activities, namely:

- Pre-decommissioning Stage (PDS)
- Stage 1 of the Decommissioning
- Stage 2 of the Decommissioning
- Stage of close down and Land Reclamation.

The summarized schedule and the description of the decommissioning activities of Units 1-4 in compliance with the chosen Decommissioning Strategy is detailed in the Decommissioning Plan of the Units [36]:

With the implementation of the above mentioned stages different type of RAW management activities will be executed.

Upon sorting of the dismantled materials, depending on the level of contamination they could be:

- Free-released from regulatory control (without or after decontamination) and evacuated outside the KNPP site;
- Temporary stored for natural radioactive decay;
- Submitted as RAW for the appropriate treatment and conditioning.

1.6.1 Main activities, installations and facilities

The purpose of this section is to describe the activities from the Pre-decommissioning stage, Decommissioning Stage 1 and Decommissioning Stage 2, and the Close down and Land Reclamation Stage and the applied safety principles.

All performed activities must comply with the health protection and the safety of the personnel and the population and the protection of the environment. The main safety principles are given in the Decommissioning plan [36].

1.6.1.1 Main activities, related to the IP implementation

Decommissioning activities are pre-planned, including preparation of schedules for the equipment dismantling activities.

For each decommissioning activity or group of activities preparation of a work package/procedure, according to the activity complexity, is planned containing detailed description of the activities, along with the following information:

- Number, qualification and experience of the participants;
- Training activities for the participants;
- Equipment and facilities for activity execution;

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

– Necessary protective equipment;
– Support activities;
– Instruments for radioactivity measurement;
– Necessary time for activity execution;
– Expected dose rate according to the ALARA principle.

SERAW will prepare these work packages for Stage 1 in the corresponding terms, after which the information will be included in the final version of SAR during decommissioning. The preparation of the packages will be based on the main safety principles given in the document [36]. This ensures the conditions for application of these principles to the activities during SE preparation, SE and dismantling of systems and equipment outside of the SE area. The application of these principles for radiation protection of the personnel during normal operation is given in Chapter 10, and for all other cases – in Chapter 6 of document [36].

In the Plan for decommissioning of KNPP Units 1-4 [36] the activities planned for Stage 1 of the decommissioning of these units are described in detail. Stage 1 includes the phases Preparation for SE and operation of SE, as well as dismantling of equipment outside of the SE area.

Dismantling of the equipment outside of the SE area begins with the equipment in the non-contaminated buildings and in TH of Units 1-4 and includes:

– Dismantling of non-contaminated equipment;
– Dismantling of turbines;
– Dismantling of the secondary circuit.

These three types of activities can be executed simultaneously or consequently depending on the availability of funds in the Fund for decommissioning of nuclear facilities, the capacity of the facilities for sorting, size reduction and free release measurement, as well as based on the KNPP employment policy. End of Stage 1 will be determined by the completion of dismantling outside the Safe Enclosure area.

During Stage 2 dismantling of the equipment in the Safe Enclosure area will be performed. The end of Stage 2 will be determined by the completion of the dismantling of the reactors and activated components and the free release of the buildings.

During the execution of the above stages: Pre-decommissioning stage, Decommissioning Stage 1 and Decommissioning Stage 2, and the Close down and land reclamation stage different types of waste management activities will be performed.

After sorting of the dismantled materials, depending on the contamination level, they can be:

– released from control and transported outside of KNPP site, without or after decontamination;
– stored for natural radioactive decay;

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

- submitted as RAW for the appropriate treatment and conditioning.

The work break down structure includes the main structural components, which provides the general frame and the structure of the transient process from operation and fuel withdrawal, to the decommissioning and dismantling. It is prepared in compliance with [7].

The objective of this work break-down structure is to support planning, management and control of the decommissioning activities. It provides the frame of the managerial and financial supervision of the decommissioning program and allows for development of schedules.

When the higher level activities are defined the duration, the deadlines and the costs are added to the Work Breakdown Structure. On the other hand, this will allow analysis of the human resources and a detailed schedule will be prepared.

The decommissioning activities work breakdown structure in compliance with [7] is divided into main key phases:

- Preparatory activities for decommissioning;
- Facility Shutdown Activities;
- General Equipment and Material Procurement;
- Safe Enclosure of Reactor Building-1 and dismantling of the equipment that is outside the Safe Enclosure Area – Stage 1;
- Deferred dismantling of equipment inside the Safe Enclosure Area and free release of buildings for reuse for other purposes –Stage 2;
- Waste Processing, Storage, and Disposal;
- Site Security, Surveillance and Maintenance;
- Project Management, Engineering, and Site Support;
- Fuel and Nuclear Material Management.

Each of these key phases includes second level sub-tasks.

<u>Delivery of materials and equipment</u>

All procurement activities include the main equipment required for the decommissioning of the units and materials needed to perform these activities. This covers 4 groups:

- Main dismantling equipment (except the equipment for dismantling of the reactor pressure vessels and the reactor internals);
- Decontamination equipment and supporting tools;
- Radiation protection equipment;
- Equipment for safety assurance and maintenance of the Units over a long period of time.

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.6.1.1.1 Activities during PDS

Activities during operation termination - "E" mode operation and removal of the nuclear fuel from the Units

According to the Decommissioning Strategy Updated in June 2006 [7], by the time of issuance of the Decommissioning Permit by BNRA (planned for the beginning of 2013 for Units 1-2 and end of 2013 for Units 3-4), Units 1 and 2 are envisaged to be operated in "E" mode. It is necessary for Units 1-4 to be maintained in a safe operational status and to maintain their acting and additionally specified requirements in the technological specification, to follow the operational procedures as well as to keep the necessary personnel.

Preparatory activities for decommissioning – including radiological inventory and removal of hazardous and other waste remaining from units operation

The following activities below were performed according to the updated decommissioning strategy [7]:

- Preparation of a plan for the transient period, including a program for reduction of the number, re-qualification and training of the personnel;
- Identification of systems and equipment, which are not important for safety and not containing radioactive substances above the free release levels and which dismantling may be started prior the issuance of the Decommissioning Permit;
- Radiological inventory of the units for the purpose of dismantling activities planning and receiving of decommissioning permit;
- Dosimetric mapping for potentially contaminated areas;
- Detailed planning of the future decommissioning activities.

Activities on collection, sorting, treatment and transportation of waste resulting from:

- Decontamination of the Reactor Refueling Shaft (RSS);
- Decontamination of Emergency Water Storage Tank (EWST), Spent Fuel Storage Pond (SFSP) and the racks for the Spent Fuel;
- Cleaning/removal of the sludge, deposited on the EMT, SFP floors;
- Complete draining of EWST and SFSP;
- Removal of thermal insulation and it is expected that a significant part of it is radioactive;
- Special laundry/showers drains;
- Spent solutions from cleaning of the floor;
- Removal/change of ventilation filters;
- Decontamination or removal of some "hot spots", which could result in high exposure of the personnel both during SE preparation or SE operation;

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
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- Dismantling of equipment and devices at the boundary of the isolated contaminated circuits the integrity of which cannot be ensured over the SE period;
- RAW from the cleaning operations/decontamination.

Waste management

A typical feature of these Units is the availability of a large volume of untreated RAW, generated during Units operation. The predominant part of these RAW are low and intermediate level wastes containing short-lived radionuclide, as per IAEA terminology. One precondition for this is that the first two Units of the nuclear power plant have been designed in compliance with the concept of collection and storage of RAW at NPP site by the stage of the decommissioning of the units.

The main waste management activities, presented in [36], which will be performed by the personnel until 2018, are:

- Removal of hazardous and burnable waste;
- Removal of the thermal insulation;
- Preparation for activities for withdrawal of RAW that had been accumulated during Units operations;
- Laboratory analysis of RAW;
- Operation of the temporary storages for RAW;
- Commissioning of a free release measurement facility;
- Commissioning of RAW incineration facility;
- Retrieval and conditioning of spent ion exchange resins;
- Solid waste fragmentation;
- Decontamination of equipment and metal waste aiming for their free release;
- Treatment of low level liquid waste;
- Elaboration of annual plans for the quantities of RAW, which will be submitted to the State Owned Enterprise "Radioactive Waste", aiming to define the annual costs of the submitted waste;
- Submission to the State Owned Enterprise "Radioactive Waste" of untreated and conditioned RAW for treatment, storage and disposal;
- Conventional Waste Management;
- Operation of the temporary storages for RAW;
- Downloading of data for RAW and for conventional waste into the Decommissioning Management Data Base System (DeManS) and maintaining the accountancy.
- SNF Management after its removal from the Units.

According to [7] the activities for Spent Nuclear Fuel management until 2012 include:

- Storing of the Spent Nuclear Fuel in the SFSP of Units 1-4, in the "Wet Storage Facility" and in the "Dry Storage Facility" at Kozloduy NPP site;
- Safe management and storage of activated waste, related to nuclear fuel operation (irradiated dummy fuel assemblies, control rod extensions and Control Rod System driving mechanisms);
- Information and nuclear fuel data management by the use of an information system for nuclear fuel control and accounting;
- Physical security of the nuclear material.
- Pre-dismantling Decontamination

During decommissioning of the Units, decontamination is used to reduce the radiation exposure of the personnel and the population by removal of the decay and activation products contained in deposits, the passivation layer and dust from the facilities. Another reason for decontamination is to provide a possibility for eventual reuse and recycling of the materials, as well as the compliance with the acceptance criteria for radioactive waste (RAW) for final disposal [17]. The objectives of the decontamination during decommissioning of Units 1-4 of Kozloduy NPP are:

- To reduce the dose rates when performing the activities for the preparation of the SE, the operation of the SE and the dismantling;
- To reduce the contamination of the equipment and the materials down to levels that meet the free release levels as stipulated in Article 129, paragraph 1 of [17].
- To avoid the uncontrolled spread of contamination during the SE.

The above benefits shall be balanced towards:

- The radiological burden on the personnel involved in the decontamination activities;
- The costs of the decontamination activities and the expenditures for the management of the secondary RAW.

The major difference between the decontamination operations conducted during the decommissioning and the routine decontamination conducted during the units operation is due to the fact that the decontamination induced corrosion is no longer a concern. Therefore, decontamination formulations using more concentrated or more aggressive reagents could be used.

The decontamination results are expected to be:

In the Turbine Hall (TH): The decontamination of the contaminated equipment in the TH is aimed at the free release of the total quantity of ferrous and non-ferrous metals, the reduction of the non-metals (mainly concrete) quantity, which is a subject of final disposal as RAW and creating of possibilities for reuse of the TH for other purposes. The decontamination of the metals will be done in the Size Reduction and Decontamination Workshop (SRDW) after dismantling and cutting of the equipment.

In the Auxiliary Building 1 and Auxiliary Building 2: The decontamination of the contaminated equipment from Auxiliary Building (AB-1) is aimed at the free release of at least 90 % of the metals and reduction of the quantities of non-metals (mainly concrete and epoxy resins from the floor coatings), which are a subject of storage and final disposal as RAW. The decontamination of the metals will be done in the SRDW after dismantling and cutting of the equipment.

In the Reactor Building 1 of Units 1-2 and Reactor Building 2 of Units 3-4: The decontamination of the contaminated equipment from the Reactor Buildings is aimed at free release of not less than 80 % of the surface contaminated metals. Decontamination of activated equipment is not envisaged.

Decontamination methods

A detailed description of the decontamination methods, as well as of the equipment necessary to perform it, is presented in [39].

Primary circuit decontamination

The Cerium Process will be used for the primary circuit decontamination. The Cerium process is of regenerative type and exhibits a decontamination factor DF>>1000.

Decontamination of tanks and pools

According to the Technical Design for the Decommissioning of KNPP Units 1-2 [6], as well as to the Decommissioning Plan for these units [36], decontamination of tanks and pools that are contaminated with radioactive substances is envisaged. Such decontamination of tanks and pools that are contaminated with radioactive substances is envisaged for Units 3-4 as well.

The equipment supplied under Project 4a in [11] will be used for the decontamination of the Reactor Refueling Shaft, SFSP and the SFSP racks and other similar big tanks, the treatment of the water in them, as well as the conditioning of the resultant RAW.

The supplied equipment shall ensure the following:

- Decontamination factor ≥ 20 for surface decontamination;
- Decontamination factor ≥ 50 for water decontamination;
- Post decontamination ≤ 14 μSv/h.

The equipment supplied under Project 4b in [36] will be used for the emergency water storage tank decontamination.

The main steps of the SFSP decontamination are as follows:

- To perform a radiological survey of the Spent Fuel Pools (SFP), the SFP racks and reactor refueling shafts of Units 1 and 2;
- To decontaminate the walls, floors and other components of the pools and the tanks.
- Decontaminate the SFSP racks;
- Remove the sludge and the solid waste from the bottom of the pools;

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- Filtration of the contaminated water from the pools with the objective to remove the insoluble impurities;
- Separate, concentrate and place in containers the radioactive waste generated as a result of these activities.

The decontamination objectives are the following:

- To minimize the dose burden of the personnel during the future decommissioning activities, following the requirements of [16];
- Minimize the spread of the radioactive contamination during the decommissioning activities.

The following facilities have been supplied (Project4a, see [36]):

- A facility for decontamination of the pool walls and floor and a water filtration system;
- Floating surface foam collector with integrated suction pump and filter;
- Underwater vacuum cleaning system;
- Underwater object trap;
- High pressure pump and a device for directing the water jet;
- Underwater lighting system;
- Underwater TV monitoring system;
- Gamma monitoring system;
- Shielded containers for transportation of filters.

The following technology will be used for the decontamination of the SFSP, the SFSP racks and the reactor refueling shaft (according to the Project 4a in [36]):

<u>Initial identification of the radiological characteristics</u>

Initial identification of radiological characteristics is done for the surfaces that are going to be treated. The purpose of this task is to determine the initial radiological working and to identify the possible hot spots, as well as to provide a quantitative assessment of the decontamination at the end of the process.

Water and sludge samples have to be taken for the radiological survey, using KNPP equipment and procedures.

The radiological survey includes measurements above the racks, the accessible walls and bottoms of the SFSP, as well as the walls of the reactor refueling shaft. This is done by the use of an underwater system for radiation monitoring, which consists of a waterproof gamma probe with a physical shielding, connected to the system for dose rate measurement, which allows for a remote control, data display and recording. The probe will be moved in the pool by the bridge crane which is in the reactor building or by another similar facility. The measurement results are recorded, indicating whether they are related to a certain trajectory or are single spots measurements. The position

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

of the measurement is marked with respect to a given point (for example the corner of the pool).

Decontamination of the SFSP racks

The decontamination of the racks is performed in the following sequence:

- The racks are dismantled in the SFSP by the use of dedicated KNPP tools. Each part is measured at several positions and afterwards is withdrawn from the SFSP and rinsed with low pressure demineralized water during the withdrawal. After this initial cleaning, if the radiation levels are below 10 μSv/h gamma dose rate at 0.1 m from the surface, the racks are free released (according to the design decontamination procedure of Project 04a of [36]);
- If the dose rate exceeds the specified value, a second cleaning is done. This is performed by the use of the brushes of the pool cleaning system.
- After the second cleaning, if the dose rates are below the admissible ones, the rack is transported to the storage place after drying;
- If necessary, a third cleaning by the use of an underwater hydro-laser is performed;
- A floating surface foam collector should be operating during the decontamination, in order to filter the surface water layer;
- The sludge and other insoluble impurities, generated during the cleaning of the racks are sucked by an underwater vacuum system. The solid particles of various sizes are trapped by rough filters for particles having size up to 3 mm, followed by a cartridge filtration system. The filtration system (rough filters and cartridge filters) should remove the solid radioactive waste from the water and the water should be returned back into the pool. The filtration systems have indicators for differential pressure in order to monitor the clogging of the filter cartridges. The radiation status of the cartridge filter is measured periodically to follow-up the accumulated radioactivity. The cartridge filters are withdrawn when the pressure difference or the radiation status reach the specified limit values.
- Measurements of the radiological characteristics are performed after the decontamination of each part with the objective to calculate the achieved decontamination factor. If “hot spots” still remain, they shall be decontaminated once more.

Decontamination of the walls and bottoms of the SFSP and Reactor Refuelling Shaft

SFSP will be decontaminated after the withdrawal of the racks from it. A radiological map shall be prepared in advance, in order to identify the “hot spots”. The decontamination of the pools is performed under water by simultaneously running, separate systems, in order to avoid reoccurrence of suspension from the sludge and to minimize the doses and the spread of the contamination. The decontamination is done under the following process sequence:

- Depending on the radiological map, the decontamination is started from the areas of highest radioactivity;

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

- Cleaning of the SFSP walls and bottom using the "Facility for decontamination of the pool bottoms and walls and the water filtration system". The radioactive contamination is removed by the use of two rotating brushes and is sucked through the axes of the brushes into the filtration system;
- In case of need, a high pressure water jet, created by a high pressure pump and directed by a special device (hydro-laser), is applied;
- The water purification is done by the use of a surface foam collector, underwater catcher of objects and underwater vacuum cleaner. The sludge is trapped in cartridge filters, which are replaced when the limit values of the pressure drop and/or of the dose rate are reached. The small objects are collected in bags, whilst objects heavier than 40 kg can be placed into an underwater container;
- Shielded containers are envisaged for the safe transportation of the spent filters. All manipulations with the spent filters are performed under water. The container dose rate in open air shall not exceed 2.0 mSv/h (at contact) and 0.1 mSv/h at 1 m distance from the external surface of the container.
- Underwater systems for lighting and TV monitoring are envisaged in order to facilitate the underwater activities,
- Surface dose meters for continuous monitoring and gamma probes are envisaged for radiation monitoring during all manipulations. Radiological measurements are done for establishing of the achieved decontamination level.
- When the expected degree of decontamination is achieved for the SFSP walls and bottom, the whole quantity of water is filtered through the SWT-4 filters.
- For the purpose of the refueling shaft decontamination, the water level shall be equal to the level during the refueling. After measuring the radiological characteristics of the shaft in case the contamination level is higher than the admissible one, high pressure water decontamination should be done in order to achieve the admissible contamination levels. The walls and the floor are treated by brushes in order to eliminate the "hot spots".

The decontamination of the SFSP and the refueling shaft shall be done under stringent following of the dedicated procedures and instructions.

1.6.1.1.2 Activities during Decommissioning Stage 1

Site management and site maintenance

All the activities related to the project management and the site maintenance are included in this section.

<u>Control of the Units being decommissioned:</u>

- Control of the Units being decommissioned:
- Cost planning and control;

- Documentation control and activities reporting;
- Quality Assurance;
- Administration of contractors;
- Public Relations.

Technical Support

- Supply of electricity and other needed fluid (oil, water, steam, compressed air etc.);
- Supplies and warehousing;
- Support/auxiliary services;
- Radiation protection and monitoring;
- Safe and healthy work conditions;
- Trial operations/construction of temporary facilities;
- Changes in the site configuration;
- Maintenance of the site, roads, parking places, sewages etc.

Administrative Support

- Human resources planning/management;
- Training;
- Budget and finances;
- Quality Assurance;
- Contracts – preparation of technical specifications for tenders, selection of contractors/subcontractors and preparation of contract.

Operation

- Supervision and maintenance of Safe Enclosure of Reactor Building 1 of Units 1 and 2 and Reactor Building 2 of Units 3 and 4;
- Operation and maintenance of buildings, installations and operable equipment;
- Physical security and fire protection at the site.

Project management and engineering

The main engineering activities are related to the support and development of the decommissioning project, including:

- Development of dismantling strategy, dose assessment, selection of the best developed scenario;
- Designing of changes in the buildings (entrances, change rooms, people walk way routes and waste handling);

- Maintaining of project management data and operation of this system;
- Elaboration of designs for new buildings;
- Coordination of the elaboration of application documents for the Decommissioning Permits for the Units, licensing of the individual sites, installations and facilities and keeping connections with the regulatory authorities;
- Keeping data from project management and the system operations and downloading of the data in the DeManS Database [6, 35];
- Assistance to the general engineering studies and the radiological inventory of the Units;
- Assistance to the Project Management Unit;
- Emergency Planning.

The purpose of the Safe Enclosure Strategy as per the IAEA document [57] is to bring the nuclear facility to a state of long-term safe storage and at the same time to reduce the maintenance needs. This strategy allows certain conditions to be met or certain activities to be implemented prior to starting the final dismantling of the equipment. The Safe Enclosure does not result in deterioration of the overall safety of the unit.

The Safe Enclosure Area is defined in the Updated Decommissioning Strategy for Units 1-4 of Kozloduy NPP [7] and is illustrated in figure 1.5-2. It includes the Reactor Building 1, where Reactors 1 and 2 are located and the Reactor Building 2, where Reactors 3 and 4 are located, and the interconnecting passageways and LSB1.

The following goals are defined for the SE stage in order to fulfill the fundamental safety objective, namely - the protection of the people and the environment from the harmful impact of the ionizing radiation.

- Ensure relevant radiation protection;
- Concept for dose intake limitations (ALARA principle);
- Ensure industrial safety of the workers;
- Maintain all discharge levels in compliance with the levels allowed in the license.
- Activities during the Safe Enclosure preparation phase

A. Activities during the phase of Safe Enclosure Preparation

According to Chapter 7 of the Decommissioning Plan [36], the activities performed during the preparation of the Reactor Buildings of Units 1-4 for SE are grouped as follows:

- Activities outside the SE area;
- Activities at the boundary of the SE area;
- Activities inside the SE area.

Activities at the boundary of the SE Area

- Isolation of all the unnecessary connections of the RB with the TH and AB-1;
- Dismantling of equipment and devices at the boundary of the isolated contaminated circuits, the integrity of which cannot be ensured over the SE period;
- Perform construction activities for SE Preparation.

Preparation of the systems for SE (activities within the SE area)

- Tagging with labels of systems, which remain in operation during the SE of the units;
- Isolation of all the systems, which are not needed for the SE:
- Isolation of the operable part of the systems, which remain in operation during the SE, from the part which will be not in operation:
- Removal of fluids from the systems;
- Removal of hazardous waste;
- Removal of combustible materials;
- Removal of the iodine filters;
- Replacement of the aerosol filters which remain in operation during the SE;
- Switch-off/disconnection of all the electrical systems, which do not remain in operation during the SE;
- Removal of the insulation from the systems, which will be decontaminated;
- Conservation of systems and equipment, which can be used during or after SE (handling systems, lifting equipment);
- Arrangement of ventilation areas:
- Adaptation of the existing systems, which are needed for the SE;
- Installation of the new systems, which are needed for the SE;
- Decontamination of the reactor cavity;
- Primary Circuit decontamination and the equipment related to it (cerium decontamination of loops);
- Decontamination of SFP and the racks for the spent fuel (the equipment supplied under Project 4a will be used for the purpose);
- Decontamination of the EWT: Decontamination of EWT walls and the floors (the equipment supplied under Project 4b will be used for the purpose);
- Decontamination, draining, flushing, drying and isolation of technological systems, which are not needed for the SE;

- Decontamination or removal of some “hot spots”, which could result to high exposure of the personnel both during SE preparation or SE operation;
- Dismantling and moving contaminated equipment to the SE area;
- Decontamination of rooms and surfaces.

Collection, sorting, treatment and transportation of waste resulting from:

- Decontamination of the Reactor Refueling Shaft (RSS);
- Decontamination of Emergency Water Storage Tank (EWST), Spent Fuel Storage Pond (SFSP) and the racks for the Spent Fuel (Cleaning/removal of the sludge, deposited on the EMT, SFP floors; Complete draining of EWST and SFSP)
- Removal of thermal insulation and it is expected a significant part of it to be radioactive;
- Special laundry/showers drains;
- Spent solutions from cleaning of the floor;
- Removal/change of ventilation filters;
- Decontamination or removal of some “hot spots”, which could result in high exposure of the personnel both during SE preparation or SE operation;
- Removal of resins from the high-activity sorbent tanks and low-activity sorbent tanks and of the solidified phase of the evaporator concentrate from ECT;
- Dismantling of equipment and devices at the boundary of the isolated contaminated circuits the integrity of which cannot be ensured over the SE period;

B. Activities during the phase of SE Operation

Safe Enclosure of the Reactor Building 1 and Reactor Building 2

This phase is characterized by minimum activities regarding operation and management as follows:

- Walk-downs and equipment tests on periodical basis;
- Replacement of aerosol filters;
- Maintenance of buildings and equipment that remain in operation;
- Renewal of the operational system during SE (if needed);
- Installation of blower heaters in the SGC (if needed).

Collection, sorting, treatment and transportation of waste generated during the SE operation

Considerably less and low active waste is expected to be generated during the SE operation:

- Ventilation filters;
- The condensate water, which is collected in the special sewage;
- Rain water, collected around AB-1 and AB 2;
- In-situ decontamination prior dismantling, where needed;
- Post-dismantling decontamination;

C. Dismantling of the equipment outside of the SE area (Turbine Hall)

The Plan for dismantling of equipment outside the SE Area of Kozloduy NPP shall follow the usual practice of the nuclear power units decommissioning projects. The equipment dismantling in the turbine halls and other Auxiliary Buildings is performed by using mostly manual or remote conventional "off the shelf" cutting tools and machinery. Initially, smaller components and equipment not related to safety will be removed in order to free additional space for removal and manoeuvring of larger size components.

Turbine Hall dismantling

<u>Preparatory activities</u>

- Removal of hot spots and floor decontamination – contamination that exceeds the limits shall be eliminated prior to beginning of the dismantling of non-contaminated or potentially contaminated systems, in order to avoid cross contamination;
- Emptying the fuel oil and oiling materials tanks – these tanks which are subjects to storage, shall be washed and vented in order to eliminate the gases that are potentially contained in them;
- Arrangement of two areas for size reduction of the dismantled equipment. Each one of the zones will have:
    - Big cutting machines: A band-saw for horizontal and a band-saw for vertical cutting (remotely controlled machines);
    - Small machines for manual cutting: Circular saws and disk saws, thermal cutting, shears;
    - Ventilation system for servicing of working places for manual cutting and air purification prior to discharge outside the TH.
- Removal of the insulation (asbestos) prior to pipelines dismantling;
- Provision of access for the heavy equipment;
- Preparation of the transport scheme for the containers full of waste
- Provision of storage site for the waste containers.

<u>Dismantling activities</u>

The sequence of operations during dismantling of the separate units and components is described in the relevant Work Packages.

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
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The Preliminary dismantling program for the TH is presented in table 1.6.1.1.2-1; the same program will be executed during dismantling of TH of Units 3-4. The table specifies the degree of contamination of a given system/equipment, which is considered during the evaluation of the dose budget for a given operation. The assessment of the possible contamination is done in compliance with the methodology, presented in [36]. Three categories have been implemented in the methodology as follows:

- I category – Non-contaminated material: Control measurements of the material (random checks) are needed for the free release of the material;
- II category –Potentially contaminated material Measurement of all the material is required when making a decision for free release;
- III category –Contaminated material: The material is treated as radioactive waste and in single cases it may be free released after decontamination and measurement.

Table 1.6.1.1.2-1 Preliminary dismantling program for the TH for Units 1&2, according to [36]

Code	System; Equipment
1.1.1.	Removal of burnable and harmful materials and thermal insulation dismantling in TH of Units I and II
1.2.1.	Dismantling of Unit transformers 1÷4
1.3.1.	Dismantling of Generators 1÷4
1.4.1.	Dismantling of the condensate and the temporary boiler tract of TG 1÷4
1.5.1.	Dismantling of the feeding tracts of Units 1 and 2
1.6.1.	Dismantling of the main steam pipelines of Units 1 and 2
1.7.1.	Dismantling of turbines and auxiliary equipment
1.8.1.	Dismantling of condensers of turbines 1-4
1.9.1.	Dismantling of internal building structures

Generation of RAW during SE Preparation and SE Operation

The assessment of the quantities and type of RAW generated during the preparation of SE and the operation SE of KNPP Units 1-4 is presented in sections 1.11 and 1.12.

1.6.1.1.3 Decommissioning activities during Stage 2 (Dismantling within the SE area)

Large size equipment from Primary Circuits of the units is envisaged to be dismantled intact and to be stored in specially constructed "decay storage" at the site.

The reactor pressure vessels are envisaged to be dismantled last and intact (without cutting) in the reverse order of their installation. Their withdrawal from the reactor cavities and the transfer to the transport corridor will be done by the use of the 250 tone cranes in the reactor buildings. If possible, the reactor internals will be loaded back in the reactor pressure vessel and they will be transported to the storage facility and will be stored in one common container, together with the reactor pressure vessel. It is envisaged to store the Reactor Pressure Vessels in special steel or reinforced concrete containers, specially prepared for this purpose.

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Contaminated components

Detailed information about the dismantling activities is not available at the present time and it is necessary to executed detailed planning of these activities according to EWN experience [50]. General information about dismantling methods used is given below.

The EWN dismantling strategy comprises the removal of all components/parts (e.g. steam generators and pressurizer) as large as possible for decay storage and their later treatment.

An important issue for the project realization and project controlling was the implementation of an integrated software package “Decommissioning Management System”. This DeManS comprises among others the main parts: project planning and calculation, mass and radiological inventory, material flow control module and Environmental Information System.

This information is the base for the detailed planning of activities:

- The stipulation of the dismantling technology,
- The requirements for radiation protection (ALARA principle),
- The requirements for maintenance of industrial health, and
- The requirements/measurements for the reduction of environmental impacts.

The dismantling technologies are comparable or the same ones used during repair and maintenance activities.

Additionally, new technologies were introduced, e.g. laser cutting for dismantling of synthetic material or band saw facilities for dismantling of large concrete structures.

Post dismantling decontamination and treatment of RAW

Decontamination of dismantled equipment

For decontamination of the dismantled equipment from Units 1 and 2 of Kozloduy NPP during the decommissioning the possibilities described below have been considered [36, 39].

High-pressure water jet decontamination

The necessary equipment for the high-pressure water jet decontamination is the following:

- Hermetic chamber allowing for transportation of materials through its doors and ceiling;
- High-pressure water compressor (up to 200 MPa);
- Turn table;
- Crane.

The chamber is connected to the filtrating ventilation system and the water treatment system.

Abrasive decontamination

The necessary equipment for the wet abrasive decontamination is the following [19]:

– Hermetic chamber allowing for transportation of materials through its doors and ceiling;
– Compressor up to 1 MPa;
– Turn table;
– Crane.

The chamber is connected to the filtrating ventilation system and the abrasive material recycling.

Chemical decontamination by the use of phosphoric acid

The phosphoric acid decontaminates carbon steel surfaces by creating a film of ferrous phosphate. The phosphoric acid is used for direct treatment of contaminated ferrous or steel parts or surfaces.

The equipment, needed for the chemical decontamination by the use of phosphoric acid is [39]:

– A tank with volume 2.5 m^3
– A tank with volume 5 m^3;
– Crane.

The chemical decontamination premises shall be connected to the filtrating ventilation system and to the special sewerage.

Electrochemical decontamination by the use of oxalic acid

The equipment needed for the electrolytic decontamination by the use of oxalic acid is:

– Two electrolytic baths having 2.00 m^3 each;
– Rectifier 2000 A;
– Rectifier 1000 A;
– Crane.

The decontamination premises shall be connected to the filtrating ventilation system and to the active drains.

Ultrasonic decontamination

The ultrasonic equipment is used for the ultrasonic decontamination of metal parts, assemblies and tools in solutions of bases and acids. The equipment is intended to operate at temperatures of the chemical solutions from 20 °C to 100 °C. The solutions that are used for the ultrasonic decontamination are the same as for the chemical decontamination.

In most cases the expected result of the decontamination is achieved by applying the techniques in combination and in sequence.

Activated components

Up to now there are not any data of dismantled activated components available for the Kozloduy NPP Units 1-4. Therefore, EWN data and experience at Greifswald NPP will be used, according to [50]. The below figures present the used technology for dismantling and transportation of the activated components. This process was completed at the end of 2009 at Greifswald NPP.

<u>Reactor Pressure Vessel (RPV)</u>

Dry cutting

This technology was tested (model dismantling) in Greifswald NPP (KGR) with a non- activated RPV.

Fig. 1.6.1.1.3-1 Transportation of the RPV

Storage of the RPVs with the internals of the reactor cavity and reactor cavity bottom and shielding cylinder

This procedure was realized in KGR with the RPVs 1, 2 and 5. The storage place was in Hall 7 of the ISN.

Reactor internals (RI)

Wet cutting

This technology was tested (model dismantling) in Greifswald NPP (KGR) and realized by wet cutting of the reactor internals of Units 1 and 2. The wet cutting place was commissioned in an emptied steam generator box. The duration for the wet cutting procedures of the internals of Units 1 and 2 was about 2.5 years.

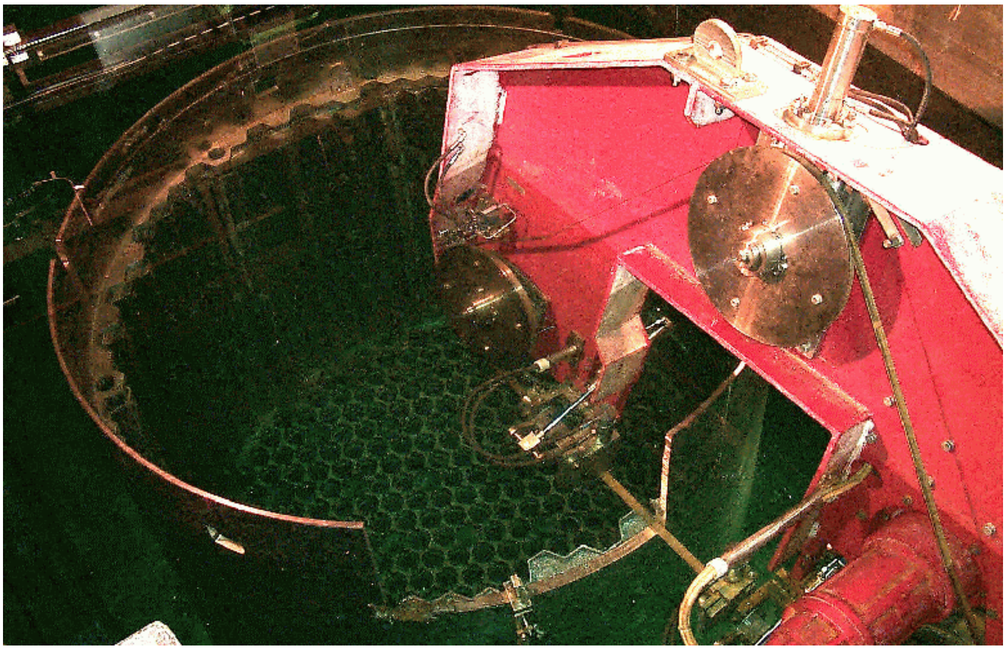

Fig. 1.6.1.1.3-2 Wet cutting, manipulator with band saw

Storage of the complete internals in Devices for Shielding and Transport (DSTs)

This procedure was realized for all internals of the RPV 5 and the internals core baskets and protected tube units of the RPVs 3 and 4.

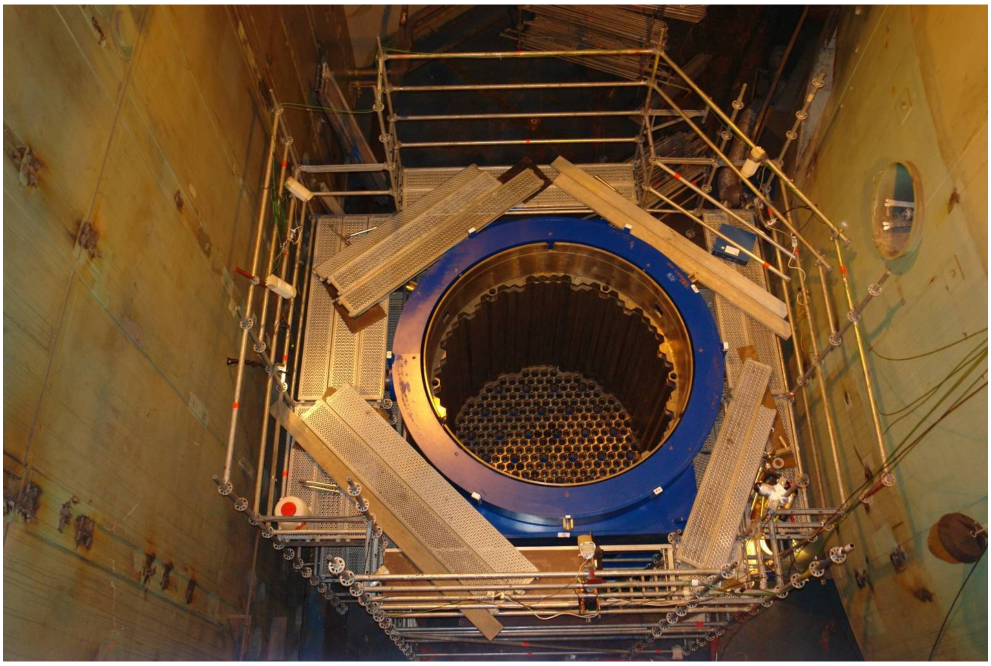

Fig. 1.6.1.1.3-3 Core basket in a DST

Fig 1.6.1.1.3-4 Transport of a DST into the ISN

Fig. 1.6.1.3 -5 Storage hall for RPVs

1.6.1.1.4 Close down and land reclamation stage

The long-term state of the industrial site where KNPP Units 1-4 are being decommissioned is defined as “brown lawn”. It will be achieved by performance of the following activities: dismantling of the equipment not intended for further use; free release of the buildings and facilities remaining in operation; processing and taking out of all RAW from the site and bringing it to a condition suitable for nuclear purposes or other economic activities.

EWN Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.6.1.2 Projects facilities related to Decommissioning of Units 1 - 4 of Kozloduy NPP

The projects that are currently planned to be executed during the Pre-decommissioning stage (PDS) and for which the EIAR will determine the possibility to expect or not impacts on the population and the environment can generally be classified in several types:

- Projects for supply of equipment – for measurement and control, for size reduction and dismantling, for treatment and conditioning;
- Projects for management, control, survey, analysis, training, including delivery and implementation of software and other improvements and modernisations of the existing management systems;
- Projects for construction of facilities.

In order to support the decommissioning, projects for supply of facilities related to the RAW management are executed. These projects are described in detail in the Decommissioning Plan for KNPP Units 1-4 [36].

1.6.1.2.1 Size Reduction and Decontamination Active Workshop (SRDW)

The project ensures the construction of a Size Reduction and Decontamination Workshop (SRDW) [156] of materials that are generated during dismantling of contaminated equipment during the decommissioning of Units 1-4 of Kozloduy NPP.

The main purpose of the SRDW is to achieve levels of radioactive contamination of the processed materials allowing free release, or levels at which the processed materials are Category I RAW, in order to minimize the temporary storage and volume of RAW for submission to SE RAW for further management, including disposal at the National RAW Disposal Facility (NDF).

The designation of the workshop is:

- Size reduction of dismantled materials by cutting;
- Decontamination of dismantled materials by mechanical, chemical and electrochemical methods;
- Packing of the materials in pallets/containers for decontamination and for transportation to the free release measurement facility.

According the TS of the Project [156], SRDW must be designed in accordance with the design seismic characteristics of the KNPP site. SRDW must be located between TH-1, RB-1 and LSB-1.

The available place between the buildings will be with dimensions 72.00 m x 52.00 m. The site will be located in the internal perimeter of KNPP.

The SRDW must be positioned according to the detailed development plan and under observation of the hygiene, fire safety and technological requirements. SRDW must be a permanent building, with technologically justified production area, providing effective protection from the spreading of radioactive materials into the environment through own Personnel redressing and decontamination facility (showers, archways, changing rooms, etc.) and ventilation tube for releasing of the spent air in the atmosphere. The workshop must be divided into different functional areas, where the relevant facilities and equipment for size reduction, decontamination and the related supporting activities, as well as for processing of the waste generated during the decontamination, processing of cables and others must be located.

The dismantled contaminated materials will be received in the SRDW in 20’ ISO containers or in palettes, placed in 20’ ISO containers. In cases of non-fixed contamination the materials will be packed with polymer materials in order to minimize the container contamination. The palettes with the processed material will be transported in 20’ ISO containers outside of the SRDW to the facilities for free release measurement.

According to the Bulgarian regulations the regulatory radiation levels listed below represent the design criteria for the new facility:

- Equivalent dose rate at 1.0 m from the outside walls of the SRDW: not higher than 1 μSv/h;
- Equivalent dose rate for the population: not higher than 0.025 μSv/h above the natural radiation background.

The contractor must comply with the requirement that the gamma dose rate for a full working day is:

- In the rooms in the decontamination area - $\leq$ 10 μSv/h;
- In the rest working areas - $\leq$ 5 μSv/h.

The description and the main functions of the facility are presented in detail in ToR [156]. The following working areas are envisaged to be arranged in the Workshop:

- Areas for fragmentation of contaminated components and for size reduction;
- Area for mechanical decontamination of contaminated components;
- Area for chemical decontamination;
- Area for secondary RAW treatment;
- Area for removal of cable insulation;
- Areas for temporary storage, gamma monitoring and weighing of the materials;
- Servicing areas and rooms.

The sizes of these areas depend on the sizes of the fragmentation and decontamination equipment as well as on the mass, shape and the size of the dismantled equipment. The post dismantled/size reduction radiological measurements data shall be recorded in the DeManS Database in order to follow-up the material streams. After size reduction and additional measurements, a decision could be taken on the necessary and possible decontamination procedures. The objective of the size reduction is to obtain segments with sizes that are consistent with the overall sizes of the transportation pallets and the decontamination tanks. The decontamination itself is performed by the methods described in [39], depending on the objectives. More detailed technical description for this facility is available in the project ToR [156].

A Similar facility is operating in Greifswald NPP.

1.6.1.2.2 Design and construction of Sites for management of the materials generated by the decommissioning activities of KNPP Units 1-4

In compliance with the Regulation on safety during decommissioning of nuclear facilities [12], a construction of suitable infrastructure for temporary storage of waste and materials from decommissioning is needed before beginning with the dismantling activities.

In order to ensure safe, effective and economically efficient management of the materials and waste generated during the decommissioning of KNPP Units 1-4 the construction of their safe temporary storage facility is planned.

According the Technical Specification of this project, based on the Pre-design study for arrangement of depots and sites for safe temporary storage of Category I RAW, non-radioactive waste and materials from Decommissioning [203], it is planned to set up an open-air site for temporary storage of radioactive materials (RAM) (materials generated by the dismantling activities sentenced to follow a clearance procedure) dedicated to the following purposes: temporary decay storage for Category 1 RAW (maximal storage period, upon free release measurement, 5 years) and temporary RAM storage prior their transportation for decontamination to the SRDW. The RAM will be conditioned in suitably equipped 20’ ISO containers.

Similar facility is currently in operation at Greifswald NPP site.

Within the project frame is also planned the organisation of conventional waste (free-released materials from the dismantling) storage, with the especially dedicated site for these purposes. On this site skip containers used for the conventional materials (metals) will be stored prior to their transportation outside KNPP site perimeter.

According the project documentation the selection was foccused on three possible sites on the territory of KNPP Units 1-4, two of them dedicated to RAM sentenced to follow a clearance procedure and one site for storage of conventional waste (free released materials from the dismantling).

The sites are located within the boundaries of KNPP site inside the restricted access (controlled) area. They are open-air sites, equipped with rain (storm)/surface water drainage.

Site 1: Site beside the receivers

Site 1 shall be constructed in two separate sections:

- Site 1a - north from Receivers Site occupying an area sized 60 x 30 m and
- Site 1b – south from the Receivers Ste occupying an area of 1816 m^2 (see Figure 1-2).

The sites 1a and 1b are dedicated for storage of 20’ ISO containers. The containers arrangement shall be up to three lines patern on a height. The use of a Reachstacker and/or truck crane is foreseen for their handling,

Site 1a, north of Receivers Site

The site is placed at several meters from the neutralization pits. Some of the important underground communications at the Site, north of the Receivers site are the following:

- In its northern part – technological tunnel No.2, technological tunnels No.21 and 20 to the neutralization pits, waste water pipelines, stop valve of a heat transmission line, potable water pipelines and one pipeline for demineralised water, heating water,
- process air;
- in its southern part - underground collector, cable duct with 8 pcs. 6 kV and 5 pcs control cables (CC) and a hydrogen line;
- a pipe of the industrial sewage passes in the middle of the site

Site 1b, south of the Receivers site

Site 1b is located to the south of the Receivers'site and occupies an area of 1816 m^2. The layout of the site is Γ-shaped. The major part of the site area is grassland. To the east of the receivers on this site there is local embankment of 1.30 m of height and 60°m^2 of area. To the south and to the west of the the site auxiliary roads of the KNPP road network are neighboring. Site 1b will be connected to the existing road network from the south in order to profit optimally of the proposed location for the containers arrangement. The underground communications are as follows: Nitrogen and Hydrogene lines from the receivers, a cable route, 4 cables 0,4 kV, telecommunication line etc.

Within the borders of this area falls the pit Ш 1332 and piezometres ПМ 129.

On site there are about 10 trees which will be partially eradicated. The maximum possible conservation of the trees growing at the western site of the area shall be provided also for serving as natural border between the site and the road sidewalk.

The terrain of Site 1 is a property of Kozloduy NPP EAD.

Site 2: "KOTLOVAN" site

Site 2 is envisaged for temporary storage of 20' ISO containers with Category 1 RAM/RAW to be decontaminated in SRDW. The size of the site is 55 x 58 m and it is located between the DGS-2 (diesel generator station -2) and SLB-2 (sanitary/laboratory building). There is an excavation pit there for a building, which size is 50 x 50 m and its depth is about 3.5 m. The containers arrangement shall be up to three lines patern on a height. The use of a Reachstacker and/or truck crane is foreseen for their handling. Site 2 has to be connected to the existing road infrastructure by construction of a new road section and road junction.

Before the implementation of the drainage and the concrete pavement the remaining monolithic cup types residus of foundations will be cleared up fromt the site. In the area of this site there is no underground communications.

The terrain of Site 2 is a property of Kozloduy NPP EAD.

Site 3: Transformers' site

Site 3 is designated for temporary storage of skip containers with conventional materials (metals) (free-released dismantling materials) prior to their transportation outside the KNPP site perimeter The containers arrangement shall be up to one line patern on a height. The use of single beam gantry crane is foreseen for their handling.

The site construction is planned on the existing Transformers'site of Units 1 and 2 and the occupied area will be with a size 292 x 38 m. Site 3 shall be connected to the existing road infrastructure. The Site 3 will be devided into three separate zones. The site construction requires the design, supply and implementation of single beam gantry crane - three (3) units, for the service of each of the zones. Each crane shall be provided with railway. The site is drained by a channel parallel to the Turbine hall at a distance of 4 m. According the vertical planning the site is divided into zone in reference of the different terrain levels. The exisiting railway on the transformer's site is not foreseen to be used.

The terrain of Site 3 is a property of Kozloduy NPP EAD.

Sites' functional requirements

Sites 1a, 1b, and 2 will be provided with drainage/dewatering system for rain/surface water. The rain water will be collected in pits/tanks and following radiation monitoring, it will be directed to the existing KNPP rain water sewage system for evacuation. The design of Sites 1a, 1b and 2 foresees the disposition of a tank-car for pumping the water close to the collection tank. The existing on site drainage system on site will be used in an optimal way.

The sites for temporary storage of RAM from the decommissioning activities at Units 1-4 shall be designed and operated in a way to minimize the occupational, public and environmental risk and in conformity with the ALARA principle.

All sites will be dimensioned with the sufficient loading capacity (mechanical resistance and stability), long structural lifetime of the pavement and foundation dimensioned in reference of the operational and seismic stress.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

The site shall provide safe transport and handling equipment manoeuvring and containers safe handling.

The design shall consider all exisiting underground technological communications (pits, technological tunnels, piezometers, pipelines, cable routes etc) as well as their work conditions and auxiliary equipment.

The size of the sites are conformed with their construction requirements regarding distance from neighboring buildings and requirements for distance from building with specific fire safety cathegory.

1.6.2 Auxiliary facilities and equipment

In order to support the decommissioning activities of KNPP Units 1-4 the implementation of different projects and supply of the necessary facilities is planned.

New facilities for RAW management in Kozloduy NPP

Supply of Liquid Radioactive Waste Treatment Facility (Danube Facility)

The Project shall ensure the supply of equipment for the treatment of the water from the special laundry, the baths and the sewage water from KNPP Units 1 and 2 and for RAW conditioning. At the present moment these wastes are treated by the system SWT 3 of the KNPP Units 1 and 2. This system is going to be shut down during the SE stage of these units.

The facility was commissioned in 2012.

This is a module type facility. Reagents for radionuclide precipitation are dosed in the receiving tank in order to increase the degree of the water purification.The radioactive water from the receiving tank is initially subject to a two-stage filtration in order to remove the dispersed and the colloid particles. At the first stage the RAW is filtered through a strainer screen with an average mesh size of 10 µm. At the second stage the RAW is filtrated by a porous polymeric cartridge micro-filter with an average pore size of 5 µm. Reverse rinsing of the strainer screens is performed periodically depending on the hydraulic losses across the screens (not more than 0.1 MPa), the reverse rinsing water is delivered to the precipitin vessel of the evaporation module.

The final removal of the dispersed and the colloid particles from the liquid RAW takes place in an ultra-filtering module with membranes with an average pore size of several hundreds of the Angstrom unit (1.E-8m). Thus the stream is divided into two parts: a concentrated effluent that is enriched with dispersed particles and purified water.

The concentrate periodically is delivered to the RAW receiving tank. The purified water enters a reverse-osmosis module where it undergoes purification from the dissolved impurities.

In-depth purification of RAW of the soluble substances up to 8-10 g/l is performed in the reverse-osmosis module. The RAW passes through the reverse-osmosis membranes, under high pressure (up to 8 MPa), which have an average pore size of a few tens of the Angstrom unit. The concentrated product is then taken to the precipitator of the evaporation module. The purified water is delivered to the control tanks of the Analysis Facility prior to its delivery to the KNPP control tanks.

According to the Designer estimation, the purified water will have a salt concentration of less than 20 mg/l and an activity of less than 370 Bq/l. The concentration of the RAW takes place in the evaporation module. The water needed for the evaporation is delivered from the precipitator. The salt concentration in the still bottoms will be 300 g/l according to the Designer estimation. The condensate generated in the evaporation module is taken away to the receiving tank and the concentrated media is periodically taken away for cementation in 200 l barrels. The barrels are placed in a special container and are submitted to SD RAW-Kozloduy.

<u>Project – Supply of Water Decontamination and Waste Treatment Mobile Equipment</u>

The project provided mobile equipment for surface decontamination of the Reactor Refueling Pool (RRC), Spent Fuel Storage Pond (SFSP), SFSP racks and other similar large tanks, for tanks' water treatment and for secondary RAW conditioning [54]. Cartridge filters will be used to filter the water. The spent filters will be submitted to SD RAW-Kozloduy. A detailed description of the facility is included in the Decommissioning Plan for Kozloduy NPP Units 1 and 2 [36].

<u>Project – Ion exchange Resins Retrieval and Conditioning Facility</u>

The Project shall ensure the supply of equipment for the retrieval and treatment of spent ion-exchange resins from the existing storage facilities. The facility is a mobile installation with the ability to be dismantled and placed at various places on the KNPP site where spent ion-exchange resins are stored.

The facility consists of several mutually connected systems.

The mobile removal system controls the removal of the wastes (resins, sand, precipitations, gravel etc.) from the tanks. The system includes a container which protects the equipment that is in direct contact with the RAW or their environment. Once the resins get into the removing stream they are exposed to a recirculation process which guarantees a high degree of the homogenization. The retrieving stream is then directed back to the resin tanks via an ejector nozzle which pushes the fluid into the resins and water which are contained in the tank. The method of spraying facilitates the homogenization of the overall tank contents. After homogenization the fluid is sent to the mobile system for dosing.

The system for waste treatment is divided into two sub-systems:

- Mobile dosing system which intakes RAW via a mobile removal system, doses the necessary quantities of resins and water into a barrel, which beforehand has been filled with cement;
- Mobile cementing system which is intended to mix the resins and cement in order to obtain an immobilized product.

The barrels with the cemented radioactive product are placed in a special container and are submitted to SD RAW-Kozloduy. The facility was commissioned in 2012.

The use of this facility will lead to the removal of the sorbents from LAST and HAST until 2016.

Project – Free Release Measurement Facility

The Project shall ensure the supply of equipment for gamma-activity measuring to be used for the control of dismantled equipment and other materials for free release.

The equipment for controlling free release shall:

- Measure 200/400 l barrels and pallets;
- Have a capacity of 15 t per shift;
- Measure γ-activity of $E\gamma > 150$ keV.

The free release facility includes the facility itself, the electric car, the control room, a pallet set and barrel adapters.

The facility is supplied and the hand-over and the metrological tests are done. Systemized information for the radioactive contamination of the equipment and the materials is required for the correct performance of the facility (defining the nuclide vectors as a result of the radiological survey of the units [35], as well as the elaboration of the relevant documents, regulating the operations for the free release measurements.

Radiological inventory equipment

The equipment is delivered and commissioned in EP-1. The radiological survey of Units 1-4 will be done by the use of this measurement equipment.

Other new projects concerning the RAW management during decommissioning

It is proposed to implement the following new projects related to the RAW management during decommissioning of Units 1-4 according to [36]:

Project – Supply of equipment for retrieval of the liquid phase from ECT in AB-1

This equipment will be used to transfer the liquid phase from ECT in AB-1 to the transportation corridor (TC-1) of the RB and will be discharged in the special 12 m^3 road tank supporting the further treatment in the State Enterprise “RAW”-Kozloduy.

Project – Supply of equipment for extraction and treatment of the solid phase in the ECT

A characterization of the solidified phase in all ECTs in AB-1 and 2 will be done prior the retrieval of the RAW. Not less than 90 % RAW of Category 1 and 10 % RAW of Category 2a will be received as a result of the treatment. Characterization of all the end products from the treatment is envisaged.

Project – Comprehensive radiological survey of KNPP Units 1-4

The project envisages a comprehensive assessment to be done of the radiological status of the equipment, structures, and rooms and RAW as well as to evaluate the total radionuclide contents and the quantity of the materials in the units under decommissioning. The project will provide an adequate management of the material streams (dismantled equipment, components, material, radioactive waste, structures, buildings etc.) during the decommissioning by ensuring the process optimization [35].

Project – Supply of various types of containers for transportation and storage of materials generated during KNPP dismantling activities

The project envisages supply of containers for the various streams of RAW generated during decommissioning.

Projects assuring the safety during decommissioning activities

Project – Optimization of the Liquid and Gaseous Releases Monitoring System

The purpose of this project is to provide compliance with the requirements of the Recommendation of the European Commission - 2004/2/EURATOM and with these of the BNRA regarding the monitoring of KNPP releases [53]. These requirements will be followed by updating of the monitoring system for liquid and gaseous releases. The purpose of this system is to improve and optimize the existing system for monitoring (control) of the liquid and gaseous releases from KNPP Units 1-4. The automated information system for radiation control of unbalanced and waste waters (AISRCDWW) is designed to measure the specific volume activity of unbalanced and waste waters released by KNPP in the hydrosphere.

1.6.3 Others

In addition to the activities described earlier in this chapter and in order to optimize the entire process of KNPP Units 1-4 decommissioning, and to minimize and avoid the adverse impacts on the environment, personnel and population, below are presented the remaining of the most important activities, installations and facilities that must be executed with the highest priority:

- Elaboration of detailed dismantling instructions for each dismantling activity;
- Earlier implementation of Projects 4A “Equipment for decontamination and treatment of the waste water” and 4B “Supply of equipment for cleaning of EWST” in order to execute the pre-dismantling decontamination of the facilities and to reduce the radiation exposure of the personnel and the population;
- Updating of the existing database in the DeManS system, which is important for the decommissioning management;
- Earlier completion of Project “Quantitative evaluation of the accumulated materials and radiological survey of KNPP Units 1-4” in order to obtain detailed information on the radiological status of the buildings, rooms, systems and components of Units 1-4;
- Waste treatment activities, such as thermal and/or mechanical cutting must be performed in closed cabins and must be implemented similar to the areas in SRDW;
- Implementation of a mobile filter system for the areas of dismantling and vessel treatment;
- Earlier implementation of Project “Isotope composition, treatment and processing of contaminated soils”;

- Earlier completion of Project “Facility for treatment of liquid RAW” in order to optimize the management of liquid RAW;
- Installation of the system for free release measurement for dismantled facilities and materials (under Project 6a) in order to optimize the waste flow at the site of the units;
- Timely completion of Project “Supply of various types of containers for transportation and storage of materials generated during KNPP dismantling activities” meaning supply of the necessary quantity of containers for transportation and storage of dismantled materials before the start of the actual dismantling activity of the facilities and components.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.7 Main resources and materials – bill of materials and qualitative assessment

The shut-down of KNPP Units 1 to 4 means decreasing of the consumption of industrial chemicals used for maintenance of the water chemical regime at primary and secondary circuit, for regeneration of ion-exchange resins as well as for activities for purification and decontamination. The shut-down of Units 1-4 means also decreasing of quantities of resins used for WTP.

Detailed description of the purpose, needed quantities and their decreasing due to the shutdown of the Units 1 to 2 is given in [6]. Despite some differences between Units 1 and 2 compared to Units 3 and 4, it is assumed that the decreasing of the quantities of chemical substances due to the shutdown of Units 3 and 4 will be equal to those for Units 1 and 2.

In table 1.7-1 are presented the main hazardous chemical substances and mixtures expected to be used within the IP implementation with the estimated amounts and purposes.

Table 1.7-1 Main chemical substances and reagents and their quantities

No	Substance	Measure	Quantity	Application
1	Ammonia NH_3	t/year	6	Improving agent in the primary and secondary circuit
2	Sulphuric acid H_2SO_4	t/year	30	Regeneration of the cationic resins in the water treatment systems
3	Hydrochloric acid HCl	t/year	400	Regeneration of the cationic resins in the water treatment systems
4	Nitric acid HNO_3	t/year	10	Regeneration of cationic resins in WTS 1 - 5
5	Hydrazine hydrate $N_2H_4.H_2O$	t/year	5	Elimination of the oxygen by the coolant of the secondary circuit
6	Potassium hydroxide KOH	t/year	6	Regeneration of the anionic resins of WTS 1, 2 and 4, as improvement agent for pH of the primary circuit as well as for cleaning the equipment and rooms
7	Sodium hydroxide NaOH	t/year	300	Regeneration of the anionic resins in the water treatment systems.
8	Ferrous chloride $FeCl_3$	t/year	80	Desalinated water production
9	Calcium hydroxide Ca $(OH)_2$	t/year	230	Decarbonisation of desalinated water
11	Diesel fuel	t/year	150	Used for periodical testing of the 12 diesel generators supplying power to Units 1, 2, 3 and 4

In chapter 11, Attachment 11.1.4 is given a Reference to chemical substances and products in use at Kozloduy NPP EAD with their respective classification, category of danger, quantity and purpose. The main chemical substances and products and their yearly consumption by the company are included there, remaining in use since 2006 at the amounts exceeding 300 kg. There are also included the chemical reagents used only at Units 5 and 6 in EP-2 not related to the Decommissioning of Units 1-4 at KNPP and subsequently being not subject to the EIA. These chemical substances are given in positions 5, 12, 13, 14, 15, 19, 22 and 26 of the Reference.

Below the classification of the used raw materials and supplies and their quantitative assessment is given.

Stage 1 "SE Preparation"

It is expected that during Stage 1 "SE Preparation" the consumption of reagents will be mainly related to the purification and decontamination activities. Until the mobile "Danube" facility for purification of low radioactive waters is supplied, the concentration of the liquid waste by evaporation, planned in the original design of the WWER-440 units, is the only way for their processing.

Due to that, the chemicals to be used will correspond to the ones used for purification of the waste waters. The use of chemicals necessary for generation of small quantities of desalinated water will also continue.

Stage 2 "Dismantling in the SE area"

Dismantling of the equipment in the SE area is planned for Stage 2 of the decommissioning. Part of the dismantled equipment with low radioactivity will be decontaminated using one of the methods described in 4.1.3.2 [39].

The quantity of used chemicals will depend on the quantity of dismantled equipment and the applied decontamination method. All chemicals related to the generation of chemically desalinated water are also used. The chemicals related to the used decontamination methods are given below [39]:

- H_3PO_4 - phosphoric acid (60 %): used in chemical decontamination. Consumption around 4000 l per year;
- $H_2C_2O_4$ - oxalic acid (3-6 % solution) – used for electro chemical decontamination. Consumption around 1700 kg per year.

The quantity of used chemicals during solid decontamination with cerium [39] depends on whether the SG tubes are 1) included or 2) not included in the dismantled equipment. This determines the total surface to be decontaminated and respectively – the necessary chemicals.

The total surface of stainless steel for case 1 (the SG tubes are not included) is 22331 m^2, and for case 2 (the SG tubes are included) is 82571 m^2.

For both cases the carbon steel surface is 400 m^2. The corresponding number of the batches for decontamination is: for case 1) – 944 batches, and for case 2) – 2514 batches.

The following chemicals are used:

- $Ce(SO_4)_2$ - oxidizer used in solid decontamination with cerium for dissolution of the contaminated layer (0.05 m solution); Consumption– for case 1- 60 kg; for case 2 – 160 kg;
- H_2SO_4 – used for pH correction during oxidation in solid decontamination with cerium;
- Liquid O_2 in bottles – for production of ozone (O_3), which is used for regeneration of Ce4+ for the three layers (outside, inside and the main alloy (100.12 g/m^2). Designed to ensure effectiveness of the cerium regeneration > 65 %. Based on that effectiveness the ozone consumption is calculated: 26 gO_3/m^2.
- H_2O_2 is added to the used decontamination solution to reduce Cr6+ to Cr3+ (16.27 g/m) and Ce4+, which is in the solution to Ce3+. Consumption: 850°g/m^3.
- $K_4Fe\ (CN)_6.H_2O$ – to join in complex Co and Cs with $K_4Fe\ (CN)_6.H_2O$. Consumption: 1 kg/m^3 solution.
- NaOH – used to precipitate metals as hydroxides (pH is increased by adding NaOH until reaching pH = 3) and for purification of the spent solutions of Ce and sulphuric acid.

The consumption of these chemicals depends on whether the SG tubes are included (case 1) or excluded (case 2) in the calculation of the total area for decontamination. The total surface of stainless steel for case 1 (the SG tubes are not included) is 2233 m^2, and for case 2 (the SG tubes are included) is 82571 m^2. For both cases the carbon steel surface is 400 m^2.

Diesel fuel – the expected annual consumption will be 100 m^3 based on the EWN estimation (i. 4.0).

Pursuant to the requirements of the EPL classification of KNPP has been made in terms of storage and use of chemical substances, and the Plant has been classified as "company with high risk potential". In relation to this, a permit has been issued for operation of such a facility. The control in this area is performed by RIEW – Vratsa.

It is expected that, on condition that the valid permits and regulations applicable to handling of hazardous substances and chemicals are complied with, the impact on the environment will be insignificant. Trans-boundary impact is not expected.

Necessary materials during decommissioning of KNPP Units 1-4

The assessment of these materials has been made based on the EWN experience [50] from the decommissioning of the units in Greifswald NPP. Table 1.7-2 shows the necessary materials during decommissioning of KNPP Units 1-4.

Table 1.7-2 Necessary materials

Type of materials	Expected annual quantities
Lubricants	4 000 kg
Industrial gases	
Acetylene	12 365 kg

Oxygen	19 784 kg
Diesel fuel	100 m^3
Protective clothing	40000 kg

1.8 Used energy sources

The main energy sources that can be used during decommissioning activities are water, electricity and heat energy.

The main electricity consumers are the cooling systems pumps, the systems for ventilation, lighting and OLS.

The annual **heat energy** consumption for the entire site is decreased from 403 000 MWh after the shutdown in 1990 till 2000 when it has reached a value of 138 580 MWh and has been approximately constant since. *There are no significant impacts from the decommissioning process***.**

A further decrease in the electricity and heat energy consumption is expected during the decommissioning process.

According to the EWN experience [50], the annual consumption of electricity and heat energy after the final shutdown of the units is as follows:

– Electricity - 50 000 MWh;

– Heat energy - 100 000 MWh.

Another part of the energy consumption is the energy from the transport activities during decommissioning, mostly for the internal and external transportation of the decommissioning waste. This quantity of primary energy is evaluated to be in the worst case around 150 kWh per ton of waste.

Regarding the impacts from the consumption of water, electricity and heat energy during decommissioning activities, it can be stated based on the EWN experience [50] that changes in comparison with the post-operational phase are not expected. The main share from the water consumption is that of the water for sanitary needs of the personnel.

Main consummer of electrical and heat power are the systems remaining in operation in the Units post-operational mode (e.g. the ventilation systems of the reactor buildings)

1.9 Water supply sources

The main water supply sources and their quantities are presented in table 1.9-1.

Table 1.9-1 Annual quantities of water used for technical and sanitary water supply

Location of water abstraction	**Allowed quantity [thousands of m^3]**	**Used quantity [thousands of m^3]**					
		2006.	2007.	2008.	2009	2010	2011
Danube surface	5 000 000	3 334 722	2 323 800	2 629 876	2 593 459.523	2 564 530	2 660 788

waters							
Six shaft wells (SW 1-6)	788.400	-	-	0	0	0	24.779
Shaft well "Ranei – 5"	1 600	190	314	75	15.929	24.000	2.729
Shaft well "Valiata"	788.400	291	183	204	192.270	193.000	216.700
City water supply	-	1 805	1 846	1 259			

Source: Annual report on IHNRM of the environment in Kozloduy NPP, 2006-2011 [88, 89, 90, 191].

The water consumption from the underground sources is insignificant in relation to the total quantity of water used by KNPP. It is relatively constant and is determined by the decrease of the level of ground waters and by the striving to keep it constant.

- Sanitary water supply

- Drinking and sanitary water supply from the city network

The water from the city water supply network of Kozloduy is mainly used for drinking. There is a contract for that signed with "Water Supply and Sewerage" LTD – Vratza.

Water from internal sources

In the period 1995 – 1998 the sanitary and drinking water for Kozloduy town and KNPP is supplied from three boreholes "Ranei" located in the terrace of the Danube River upstream Kozloduy town. The maximal supplied flow rate is 115 l/s.

Sanitary water is also supplied to EP-1 from one borehole in the "Valiata" area. This water is not drinkable, but it can be used for household needs. The design flow rate is 25 l/s.

In the Bank pump station municipal water is supplied from borehole "Ranei 5".

Technical water supply to KNPP from the Danube river

The scheme for KNPP technical water supply from the Danube river is direct current. It provides:

– circulation water – for cooling of the turbine condensers;

– technical water – for cooling of other facilities.

Using *Bank pump stations* (BPS 1, 2 and 3) technical water from the river with debit up to 180 m^3/s is supplied to the *cold channel* (Cold Channel). Four *circulation pump stations* (CPS) supply water vie pipelines to the consumers in the main buildings. Through a system of pipelines and low pressure channels (LPC) the water is transferred to the *hot channel* (Hot Channel 1), and from there – to the Danube river. Following the design for additional technical water supply a *second hot channel* – Hot Channel 2, is constructed, which transfers the water from the consumers of Units 5-6 to the Danube river.

The Cold Channel 1 and the Hot Channel 1 run parallel to each other and together they form the so-called dual channel. The dual channel is the main facility of the KNPP technical water supply.

Generally, after stopping of reactors 3 and 4 as well in 2006, the technical water consumption has sharply decreased, and it is significantly lower than the allowed quantities. However, compared to 1998, when all reactors 1-6 were in operation, and before the stopping of reactors 1-2, a positive change in the water consumption has not been determined, despite the reduced quantity of chemically desalinated water (CDW) and unbalanced water.

The quantity of water from the Danube river used for the production of CDW in the period 2002 – 2008 has decreased by 56.7 %, mainly due to the stopping of Units 1-4 (the decrease for EP-1 is 83.5 %, and for EP-2 is 29.7 %).

During decommissioning of Units 1-4 municipal and technological water will be used.

According to KNPP data (KNPP letter, HR department October 2009) [155], in 2008 the number of people employed in operation, maintenance and repair, engineering support and others, of EP-1 was 1082. Considering the politics of KNPP management of retaining the personnel engaged in the production by including them in the decommissioning activities, it can be estimated, based on that employment and on the average specific water consumption /person/year, calculated for the last 7 years (available information), that the approximate consumption of municipal water during decommissioning will be approximately 1500 thousands of m^3.

The main consumption of technological water during operation of reactors 1-4 is for cooling water and demineralized water.

During decommissioning there is no need for cooling water. According to the EWN data [50] and based on the gained experience form the decommissioning of Greifswald NPP, the consumption of the chemically purified water will remain the same after cease of operation. Demineralized water is needed for all decontamination processes. Increased water consumption is expected due to increase in consumption in the laundries, bathrooms and for cleaning of floors and corridors.

1.10 Generated waste gases - quantitative and qualitative assessment

1.10.1 Conventional air pollutants

Point sources

Direct point sources related to decommissioning and dismantling doesn't exist. The exhaust air of the Controlled Area is a point source, but according to EWN experience [50], conventional air pollutants during dismantling activities, e.g. welding, are not detectable.

Indirect, in connection with the dismantling activities, existing or future point sources are:

- Emergency Diesel Engines for KNPP Units 1-4, trial operations (existing)
- Plasma Melting Facility (project 5c, see EIA-R P 5c)
- Heat Generation Plant (project, separate EIA-R planned).

According to EWN experience [50] the annual emissions from the Emergency Diesel Generator sets in the phases of operation, post operation and in the first 5 years of decommissioning are:

- NO_x 3000 kg/a
- SO_2 125 kg/a
- VOC 110 kg/a
- PM10 220 kg/a.

It is expected that the emission values from the Emergency Diesel Engines for the KNPP Units 1 – 4 are in the same order of magnitude.

It is important to know that with the progress of decommissioning the emissions can be reduced by replacing the old Emergency Diesel Engines with new ones, which cause lower specific emissions. Thus since 2006 the Emergency Diesel Engines emissions have decreased by 98 % [50] and are the following:

- NO_x 52 kg/a
- SO_2 2 kg/a
- VOC 2 kg/a
- PM10 4 kg/a.

In case of construction and operation of a heat plant at 100 % load and generation of 100 000 MWh/year, according to the EWN experience [50] the generated harmful emissions for one year are the following:

- Dust- 30 kg;
- NO_x – 13800 kg;
- CO – 1500 kg.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Planned sources

Planned sources during the dismantling activities are:

- Dismantling activities in the Turbine Hall (cutting, welding, insulation removal)
- Dismantling/Demolition of buildings and concrete structures, and
- All decommissioning and dismantling related transport activities within the KNPP site.

According to the EWN experience [50] the emissions from Turbine Hall dismantling and from dismantling and demolition of concrete structures will not impact the quality of the atmosphere.

More important are the emissions caused by transport activities. In [50] it was calculated that the emissions are 300g NOx and 10 g PM 10 per transportation of a ton of waste from decommissioning.

Another point is the annual average diesel fuel demand from the dismantling period. This demand is about 100 m^3 per year, with the assumptions (year 2010) [185]:

- Trucks with diesel engine: Euro V
- Fuel demand: 30 l/100 km

The emissions in g/km will be: NOx 3,55; PM 0.059.

It can be calculated that: 100 m^3 Diesel per year corresponds to 333 333 km.

In reference to these assumptions the resulting annual transport emissions from dismantling activities are:

- NO_x 1200 kg/a
- PM 20 kg/a

1.10.2 Emissions of RNG and short-lived iodine isotopes

According to the design data after the removal of the fuel no emissions of radioactive noble gases - RNG (Kr, Xe isotopes) and of short-live iodine isotopes (^{131}I, ^{133}I, and ^{135}I) are expected.

Radiological impact on the critical individuals as a result of radiation by noble gases or inhalation of ^{131}I will be absolutely negligible compared with the impact resulting from the respective discharges during the normal operation.

In Units 1-4 KNPP Technical Specification [152] the following limits are proposed for annual emissions through the ventilation pipes of EP-1:

Table 1.10.2-1 Limits of annual discharges through the ventilation stacks of EP-1 of Kozloduy NPP

Emission components	VS-1	VS-2	EP-1	VS-1
RNG, [TBq]	100	100	200	5600
Iodine-131, ^{131}I [Bq]	3	3	6	65
LLA, [GBq]	3	3	6	50
Tritium, ^{3}H[TBq]	10	10	20	250
^{14}C, [GBq]	1000	1000	2000	38000

The results from the measurements of the released activity from γ-aerosols through VS-1 and VS-2 (EP-1) for 2011, according to [193], show that the activities of the long-lived aerosols released through the VS of Units 1-4 is 0.09 % from the admissible level, and the value is three times lower than that in 2010.

The released activity from Strontium 90 (^{90}Sr) in 2011 is nine times lower than that in 2010 and is the lowest for the last five years.

The release of α-emitters has decreased about seven times compared to the previous year.

The releases of tritium (^{3}H) and ^{14}C through the VS of Units 1-4 in 2011 are 0.36 % and 0.33 % respectively from the annual limits.

The measurement of ^{85}Kr through VS-2 starts in 2011 and the measured values are below the detectable minimum.

The summarized data from the gaseous releases from EP-1 in 2011 show that tritium has the main contribution, the ^{14}C releases are ten times lower than those of the tritium, while the activity of γ-aerosols releases is 4 times lower than that of the tritium.

Long-lived aerosols during decommissioning

The activities that will be carried out during the preparation of SE could be compared more or less with the ones during the period of long outage, with a reduction of the maintenance/inspection activities, but with an increase of the cleaning/decontamination activities and activities for conditioning of the radioactive waste. Table 1.10.2-1 shows the annual release through the VS of the units and according to the decommissioning project the proposed limit of annual emissions during the preparation of SE of LLA in VS-1 is 3 GBq. According to the EWN experience [50] and the data presented by EWN all emissions of LLA during the entire decommissioning process of the units are not higher than 20 MBq. Considering that the prevailing winds in the area are mainly from the northwest, west and north, the possibility of transfer of aerosols in the Romanian direction is very limited .

1.11 Generated waste waters - quantitative and qualitative assessment

1.11.1 Flows and quantities of the waste waters

There are three main flows forming the waste waters of the four units, which are namely: conditionally clear waters – waters for production needs, mainly used for cooling; municipal waters, and the occasional rainfall waters. The quantity of the first ones is greatly reduced. About half of them are unbalanced waters, desalinated waters and other production waters. The rest is cooling water, which is not necessary when the units are decommissioned. Due to the fact that during decommissioning cooling waters are not used, and the quantity of the unbalanced and desalinated waters is greatly reduced, it can be estimated that the quantity of waste waters will be reduced by more than 50 %.

The quantities of the waste waters generated in 2009, measured in the "HTF&BS" workshop, compared to the permitted ones, are presented in table 1.11.1-1.

Table 1.11.1-1 Flows and quantities of waste waters from KNPP in 2009

Channel	Water origin	Permitted quantity [m^3]	Generated quantity [m^3]
Flow №1 (TCC – MDC)	Production, residential and rainfall waters EP-1	3 900 000	609 552
Flow №2 (DN300 – MDC)	Residential waters after water treatment complex /EP-2	450 000	194 313
Flow №3 (DN1000 – MDC)	Treated production waters from TH, DGS and etc.	6 600 000	4 564 512
Flow №4 (Switchyard – MDC)	Residential sewage waters from Switchyard	1 095	Not measured
Flow HC1 – Danube river	Cooling and Production sewage waters from EP-1 and EP-2	3 280 000 000	2 383 714 608

The quantities of the waste waters generated in 2010, measured in the "HTF&BS" workshop, compared to the permitted ones, are presented in table 1.11.1-2:

Table 1.11.1-2 Flows and quantities of waste waters from KNPP in 2010

Channel	Water origin	Permitted quantity [m^3]	Generated quantity [m^3]
Flow №1 (TCC – MDC)	Production, residential and rainfall waters EP-1	3 900 000	620 000

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Channel	Water origin	Permitted quantity [m^3]	Generated quantity [m^3]
Flow №2 (DN300 – MDC)	Residential waters after water treatment complex/EP-2	450 000	65 000
Flow №3 (DN1000 – MDC)	Treated production waters from TH, DGS and etc.	6 600 000	1 980 000
Flow №4 (Switchyard – MDC)	Residential sewage waters from Switchyard	1 095	1 000
Flow HC1 – Danube river	Cooling and Production sewage waters from EP-1 and EP-2	1 050 000 000	1 670 452 000
Flow HC2 – Danube river	Cooling and Production sewage waters from EP-1 and EP-2	2 230 000 000	746 004 000

The quantities of the waste waters generated in 2011, measured in the “HTF&BS” workshop, compared to the permitted ones, are presented in table 1.11.1-3:

Table 1.11.1-3 Flows and quantities of waste waters from KNPP in 2011

Channel	Water origin	Permitted quantity [m^3]	Generated quantity [m^3]
Flow №1 (TCC – MDC)	Production, residential and rainfall waters EP-1	3 900 000	680 000
Flow №2 (DN300 – MDC)	Residential waters after water treatment complex/EP-2	450 000	16 000
Flow №3 (DN1000 – MDC)	Treated production waters from TH, DGS and etc.	6 600 000	1 895 000
Flow №4 (Switchyard – MDC)	Residential sewage waters from Switchyard	1 095	1 000
Flow HC1 – Danube river	Cooling and Production sewage waters from EP-1 and EP-2	1 050 000 000	2 114 288 000
Flow HC2 – Danube river	Cooling and Production sewage waters from EP-1 and EP-2	2 230 000 000	507 647 000

Channel	Water origin	Permitted quantity [m^3]	Generated quantity [m^3]
RHC "Ledenika"	Waste waters from modular treatment facility	9 000	9 000

Conventional pollution of the sewage waters

Pollution of the sewage waters with Stage 1 and Stage 2 from the decommissioned Unit 1-4 is a result of:

- Biogenic pollution of residential hygiene sewage waters, which means exceeding of the norms for BOD_5, total nitrogen and total phosphorous. No change of the existing condition is expected.
- Pollution with chemical substances and agents. Due to the maintenance of the consumption of the chemically treated water for decontamination of facilities and dismantled equipment in approximate same quantity as well as after the reactors shutdown, it is forecasted to keep the reagents needed for regeneration of the ion exchange filters used for the demineralization of the water (sulphuric, hydrochloric and nitric acids, sodium hydroxide), calcium carbonate and ferric chloride used for coagulation of the hydrochloric extracted during the regeneration of hydrochloric from the sewage waters from the production of demineralised water as well as of the phosphoric and oxalic acids used respectively in the chemical and electrochemical decontamination.
- Contamination with PAHs and laundry agents used for decontamination, laundry and showers;
- Pollution with oil products by the Transport Department located on the territory of the power plant, black oil inventory, oil inventory, which consumption could be probably increased due to the activation of the transport activities on the territory on NPP.
- Pollution with biogenic elements, suspended solids and metals by leachate from the Site for Conventional Municipal and Industrial Waste (RCMIW).

As a whole, it could be forecasted that the contamination of waste water will be significantly decreased in comparison with the operation period and will be kept constant and equal to the state after the final shutdown of Units 1 to 4.

Radioactive contamination of waters

The monthly radiological control of the waste waters from the plant includes the following points:

- Danube river at the town of Kozloduy – harbor;
- Unbalanced waters, Units 5-6 – clear area;
- Unbalanced waters, Units 5-6 – special area;
- Unbalanced waters, Units 1-4;

- Outlet channel (HC);
- Incoming channel;
- New channel “Valiata”;
- Old channel “Valiata”;
- Danube river at the town of Oriahovo – harbor.

Monthly reports on the volume and activity rates of the unbalanced waters resulting from the internal radiation monitoring of the plant are also submitted to the Environmental executive agency.

The obtained values of the radiation parameters are significantly lower than the admissible limits (fig. 1.11-1).

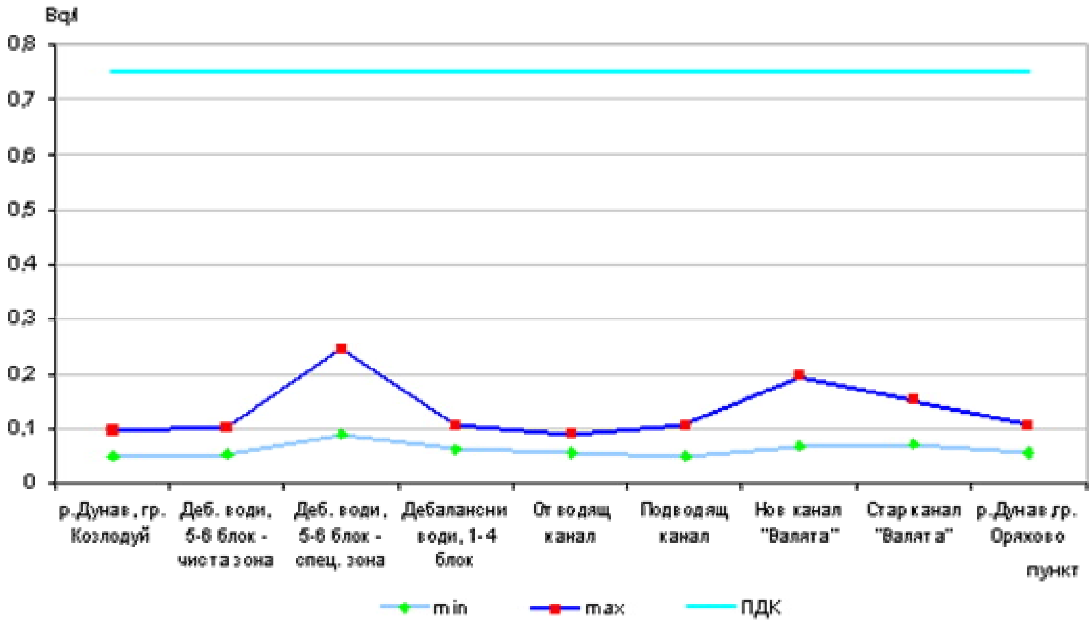

Fig. 1.11-1 Total beta activity of unbalanced waters from KNPP, Bq/l
Source: EEA.

The results from the analyses are comparable with the ones registered in previous years and do not show change in the radiological characteristics of the Danube river in Bulgarian territory resulting from KNPP operation.

The data received in 2011 compared to the results from previous years do not show adverse trends in the radiation conditions and the ecological status of the environment resulting from KNPP operation.

The technology for treatment of liquid RAW used in KNPP since 2005 to the moment includes preliminary treatment through homogenization, distillation and filtration, leading to the generation of RAW called Evaporator concentrate (EC). The clean unbalanced water after the evaporation and condensation is released from the technological cycle into the environment after chemical and radiation control.

The technology for purification of the radioactive waters through evaporation guarantees that, if the standards for radioactivity in released waters are met, the content of hazardous substances does not exceed the individual emission limits according to the permit issued to KNPP.

According to the "Project for supply of a facility for treatment of low activity liquid RAW" [55] it is planned that the waters from the special laundry, bathrooms and the sewage waters from KNPP Units 1-4, after shutdown of the SWT-3 of Units 1-2 during the SE of these units, will be treated by the "Danube" facility, which is modular type and should be commissioned after 2009. The radioactively contaminated waters successively pass through a cascade of filters located in the four specially constructed modules of the "Danube" facility. As per the assessment of the designer the treated water will have salt content less than 20 mg/l and activity less than 370 Bq/l.

Regardless of the LRAW decontamination type they are collected in a tank after passing through the oil/mud separator. Liquid RAW are neutralized by sodium hydroxide (5.5 – 8 pH) before their transfer to the sewage system of the unit and finally to SWT-3, and after the dismantling of SWT-3 - to the "Danube" facility.

It is expected that the quantity of liquid RAW discharged in the special sewage system (if the water treatment system of the water jet decontamination facility is not in operation) not to exceed 20 m^3/week and to be considerably less in the case when the system is in operation.

In regard to the EC2004/2/EURATOM recommendation and NRA recommendation concerning the monitoring on the KNPP effluents discharge, the Project "*Liquid and gaseous releases optimization monitoring system"* was implemented. The purpose of this system is to optimize the current monitoring and control system over the liquid and gaseous releases from KNPP. It includes recording of the volume activity of the unbalanced and wastewaters discharged from KNPP in the hydrosphere; transfer of the recorded data through the communication system realized on the basis of the current communication system in KNPP and archive of the collected information.

1.11.2 Liquid radioactive releases

According to the Regulation for Safe Management of Radioactive Waste [14], the liquid radioactive wastes are categorised in compliance with the provisions of art.5 of the Regulation, dependent of the solid RAW characteristics obtained after their conditioning. In cases where a technology for liquid radioactive waste conditioning is not available, the categorization shall be done taking into account high-tech conditioning technologies

The categorization of the solid RAW is given in section 1.12.1.

In consideration of the specific methods applied for RAW treatment and in accordance to the contract relations with SERAW, additional RAW categorization is set up at Kozloduy NPP EAD, in application of art.7 of the Regulation for Safe Management of Radioactive Waste [14]. The additional categories contribute for a detailed adherance to art.5, item 2 of the Regulation and are based on on directly measured operative parameters, within the limits proposed by SERAW, SD RAW-

Kozloduy, and meet the provisions of the Procedure for acceptance of RAW in the RAW management facilities at KNPP" The following additional categories of liquid RAW are established:

- Category 2-H – with activity up to 3.7 x 10^5 Bq/l;

- Category 2-C – with activity between 3.7 x 10^5 Bq/l and 7.2 x 10^7 Bq/l;

- Category 2-B – with activity more than 7.2 x 10^7 Bq/l.

The liquid RAW of additional category, depending of its origine can be characterized as:

- liquid radioactive concentrate
- ion exchange resins
- slimes and slushes
- oils.

Operational liquid RAW

Table 1.11.2-1 presents the quantities of liquid RAW generated in KNPP for the period 2009-2011 and the results show sharp decrease in the quantity of this waste generated by Units 1-4 (EP-1).

Table 1.11.2-1 Generated liquid RAW in KNPP for the period 2009-2011, m^3

Project	2009	2010	2011
EP-1	237	179	96
EP-2	159	126	173
TOTAL	**396**	**305**	**269**

According to [193] the total annual activity of the liquid releases in 2011 from EP-1 (excluding tritium) is 0.07 % from the annual control level, and the released activity including tritium is 9.09 % from the annual control level. The control levels are presented in table 1.11.2-2.

The released activity from the α- and γ-emitters is comparable with the values for 2010.

The main contributors to the activity of the liquid releases are radionuclides tritium (^{3}H) and ^{137}Cs. Around 90 % from the activity in the municipal waste waters is caused by ^{137}Cs.

Liquid discharges related to the activities during the Safe Enclosure Preparation

A). Limits of radioactive liquid discharges

During the preparation for SE it is expected low and medium level RAW to be generated from:

- Spent solutions from the decontamination of elements from the primary circuit, SFSP racks and EWST;

- Spent solutions from the decontamination of materials for lining of the thermal insulation;
- Secondary waste:
- Waste waters from the bathroom and the special laundry;
- Waters from the special sewage system.

The activities that will be executed during the stage of SE preparation could be compared with the ones in case of continuous outage during the refuelling.

Therefore, during the SE preparation stage the limits of discharge of liquid fuels will be equal to the limits applicable during the normal operation of Kozloduy NPP. Presently, these limits are given in Table 1.11.2-2 and are as follows:

Table 1.11.2-2 Annual limits* for releases of liquid RAW of Units 1-4 during the SE preparation

Sources	Control levels (Bq/year)	Maximal permissible level (Bq/year)
Total β - γ activity, without ^{3}H	8.88×10^{10}	4.44×10^{11}
^{3}H	2.96×10^{12}	2.109×10^{13}

*^{54}Mn, ^{58}Co, ^{60}Co, ^{134}Cs and ^{137}Cs

B) Generation and release of liquid RAW

Radioactive discharges into the Danube river from the monitored radiation sources as a result of the above operations are comprehensively examined and assessed in *NPP Kozloduy Units 1&2/Technical Design for Decommissioning/TASK 3 - Interim Report 1/Environmental Impact Report/Chapter II [6],* as well as in the *Decommissioning plan of Units 1 and 2 of Kozloduy NPP – Chapter 13 KPMU/DPL/013* [36]. Due to this, only the final conclusions will be presented here.

Table 1.11.2-3 Expected quantities of liquid RAW (LRAW) from Units 1 and 2 during SE Preparation

Source of LRAW	Dimension	Quantity *	Type of RAW	Expected Activity (at the moment of RAW generation)
Decontamination of the RRS, SFSP, SFSP racks, EWST	m^3 m^3	1252 (from SFSP) 1226 (from EWST)	Liquid RAW 1. Water solution and suspense 2. Concentrate of SWT-3 evaporator with boric acid concentration 12 g/l	Co-60: 4.2E+08 Bq/t; 137Cs-: 5.1E+08 Bq/t; 239Pu-: 3.7E+06 Bq/t

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Source of LRAW	Dimension	Quantity *	Type of RAW	Expected Activity (at the moment of RAW generation)
Decontamination of primary circuit and SWT-1 equipment	m^3 m^3	Approx. 430 Approx. 1415	1. Spent Decontamination solution 2. Cleaning solution	
Showers,	m^3/year	At least 4800	Water solution of cleaning surface active substances (soap, shampoo)	
Special laundries	m^3/year	3120	Water solution of cleaning surface active substances (detergent) and decontamination reagent.	
Floors, corridors and drainage channels, sewage system and rainfall waters from AB-1	m^3/year	300	Water and water solution of the cleaning surface active substances and decontamination reagent.	

*All amounts given shall be multiplied by 2 for obtaining the amount with SE of Units 3 and 4.

The liquid RAW from pre-dismantling decontamination activities, waters from the decontamination of the materials of the Units, as well as the liquid waste from the special laundries and the cleaning of the floors and corridors are treated like samp waters and they are processed at evaporation facilities. After radiological control and without treatment, only the sewage waters from the showers can be discharged into the Danube River.

As per data from Kozloduy NPP the forecast quantities of the evaporation concentrate generated from the treatment of LRAW are shown in the table 1.11.2-4 below.

Table 1.11.2-4 Forecast quantities of the evaporation concentrate from Units 1–4 during the period 2010-2014, m^3

Year of generation of liquid RAW	Origin	Evaporation concentrate
2010	From AB-1, AB-2, from treatment of the EWST solution, from releasing the SFSP from the SNF	200

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

Year of generation of liquid RAW	Origin	Evaporation concentrate
2011		200
2012		200
2013	From treatment of the SFSP and EWST solutions	120
2014		120

Source: 1. Letter ИЕ-327/24.09.2009 by Kozloduy NPP EAD, Electricity Generation -1, (Input data about Design 16 –EIA-R with decommissioning of Units 1-4) [130, 155]).

The following table summarizes the forecast annual discharges in the Danube river of β-γ - activity and of ^3H from the main emitters through the liquid waste of Units 1 and 2 during the stage of SE Preparation.

Table 1.11.2-5 Forecasted annual discharges into the Danube River of β-γ - activity and of ^3H from the main emitters through the liquid waste of Units 1 and 2

Discharge type	Measure	Year after the final shutdown of the reactors				
		1	2	3	4	5
β-γ activity	Bq/year	9.4 E+09	3.8 E+09	3.8 E+09	4.0 E+10	3.8 E+09
	% from the control level (4.44 E+10 Bq/g)	21	8.6	8.6	91	8.6
Tritium (^3H)	Bq/year	5.6 E+11	N*	N	2.56 E+12	N
	% from the control level (7.75 E+12 Bq/g)	7.2	N	N	33	N

N* - negligibly low activity.
Source: NPP Kozloduy Units 1&2/Technical Design for Decommissioning/TASK 3 - Interim Report 1/Environmental Impact Report/Chapter II [6].

As it is shown in the table the annual discharges of liquid waste during the stage of preparation of SE of Units 1 and 2 vary from 3.8 GBq to 40 GBq [36]. They are a result of the activities executed during the year. Highest activities of the liquid discharges are foreseen during the fourth year after the final shutdown of Units 1 and 2, due to contamination of the Reactor Refuelling Shaft, of the SFSP, the SFSP racks, removal of the precipitate deposited on the bottom of the SFSP, drainage and conditioning of the content of the EWST of Units 1 and 2, when the discharges of β-γ - activity reach 91 %, and ^3H activity is -33 % from the control levels (respectively 18°% and 4.6 % from the respective maximal annual permitted levels).

Decontamination of the equipment after the shutdown of Units 3 and 4 will double the emitted activities. This should be considered when planning the needed activities.

All discharges of β-γ - emitters, ^3H and H_3BO_3 are estimated based on the application of proved technologies for waste treatment (highly effective system for filtering with

EWN Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

dipping and use of ion-exchange resins). The aim is to avoid generation of huge quantities of conditioned solid waste with high content of borates and low specific activity, which can be receive from spent solutions for decontamination of pools, racks and EWST at evaporators of SWT-3.

Considering the status of the Units and the fact that not many activities will be performed during the SE operation, as well as the small number of the operators/technicians included in these activities, it has been estimated that the generation of LRAW during the SE will be considerably less than that during normal operation and SE preparation of the Units.

Liquid discharges related to activities during the Safety Enclosure Operation

A) Limits of radioactive liquid discharges

During the SE operation, the discharge limits of liquid waste into the Danube are reduced by 10 % from the corresponding values during normal operation of Units 1 and 2.

B) Generation and discharge of liquid RAW

The detailed assessment of the liquid radioactive waste releases during the SE Operation Phase is given in the Plan for decommissioning of Units 1-2 [36] and can be summarized in table 1.11.2-6 as follows:

Table 1.11.2-6 Annual discharges of liquid RAW from Units 1 and 2 during the SE Operation

Origin	Volume of the discharged liquid waste, m^3/y	β-γ Activity excluding H^3, Bq/y	Note
Showers	80	8.10^5	$6.0\ 10^4$ Bq/year, if the waste is treated in the mobile facility
Special laundries	200	6.10^7	
Floors, corridors and drainage channels	10	$1,10^7$	
Sewage drainage system and rainfall waters from AB 1	570	$2,1.10^6$	
Total	860	$7,3.10^7$	

Such quantities of LRAW will be released from Units 3-4 during the SE operation as well.

Liquid RAW discharges are expected to be:

– 7.3E10^7 Bq/y, i.e. 1.6 % of the operational limit, conservatively assuming that the special laundry waste are not processed by the mobile evaporator unit;

– 1.3E10^7 Bq/y, i.e. 0.3 % of the operational limit, conservatively assuming that the special laundry waste is not processed by the mobile evaporator unit.

These forecasted releases are extremely low by comparison with those forecasted during the SE Preparation phase, even when expecting to be doubled due to the SE of the Units 1 – 4.

Tritium discharge

Practically, no tritium (^{3}H) release is expected during the SE operation phase.

Sources and characteristics of RAW during Stage 1 of the decommissioning

Tables 1.11.2-7 and 1.11.2-8 present summarized data for liquid RAW generated during SE preparation and SE control of Units 1-2 and Units 3-4 respectively.

Table 1.11.2-7 Generation of liquid RAW during SE preparation and SE control of Units 1-2, according to the DP [36]

Operation	Type of waste	Quantities Units 1 and Unit 2	RAW description	Expected activity (as per the moment of RAW generation)	Number of packages
Decontamination of the RRS, SFSP, SFSP racks, EWST	Ion exchange resins	12 m^3	A mixture of resins with acidic and basic groups. Organic polystyrene resins. Physicochemical characteristics are similar to those of the initial sorbents used in operating activities.	^{60}Co: 4.2E+08 Bq/t; ^{137}Cs: 5.1E+08 Bq/t; ^{239}Pu: 3.7E+06 Bq/t	12 RCC
	Liquid RAW	1252 m^3 (SFSP) 1226 m^3 (EWST)	Liquid RAW 1. Water solution and suspense 2. Concentrate of SWT-3		32 RCC

Operation	Type of waste	Quantities Units 1 and Unit 2	RAW description	Expected activity (as per the moment of RAW generation)	Number of packages
			evaporator with boric acid concentration 12 g/l		
Solid decontamination of primary circuit	Spent decontamination solution	471 m^3			35 RCC
	Precipitated hydro-oxides	16.67 t			5 RCC
	Cleaning solution	1413 m^3			4 RCC
Secondary RAW	From showers	400 m^3/month	Water solution of cleaning surface active substances (soap, shampoo)	4 MBq	After measurement is released in the Danube river
	From Special laundries	3120 m^3/y	Water solution of cleaning surface active substances (detergent) and decontamination reagent.	1.45E+10 Bq/y (for the cemented waste) 4.44E+08 Bq/ y (for the three types of filters)	18.1 m^3/y (104 drums, 11 RCC) 12 mesh filters/3 y– 0.04 m^3 72 microfilters /y – 0.18 m^3 11 ultra filtrate and reverse osmotic /y – 0.087 m^3
Cleaning of floors	Waters from the special sewage system	300 m^3/y	Water and water solution of the cleaning surface active substances and decontamination reagent.		1 RCC /y

Table 1.11.2-8 Generation of liquid RAW during SE preparation and SE control of Units 3-4, according to the DP [36]

Operation	Type of waste	Quantities for Units 3 and 4	RAW description	Expected activity (as per the moment of RAW generation)	Number of packages
Decontamination of the RRS, SFSP, SFSP racks, EWST	Ion exchange resins	12 m^3	A mixture of resins with acidic and basic groups. Organic polystyrene resins. Physicochemical characteristics are similar to those of the initial sorbents used in operating activities.	^{60}Co: 4.2E+08 Bq/t; ^{137}Cs: 5.1E+08 Bq/t; ^{239}Pu: 3.7E+06 Bq/t	12 RCC
	Liquid RAW	1252 m^3 (SFSP) 1226 m^3 (EWST)	Liquid RAW 1. Water solution and suspense 2. Concentrate of SWT-3 evaporator with boric acid concentration 12 g/l		32 RCC
Solid decontamination of primary circuit	Spent decontamination solution	471 m^3			35 RCC
	Precipitated hydro-oxides	16.67 t			5 RCC
	Cleaning solution	1413 m^3			4 RCC
Secondary RAW	From showers	400 m^3/month	Water solution of cleaning surface active	4 MBq	After measurement is released in

Operation	**Type of waste**	**Quantities for Units 3 and 4**	**RAW description**	**Expected activity (as per the moment of RAW generation)**	**Number of packages**
			substances (soap, shampoo)		the Danube river
	From Special laundries	3120 m^3/y	Water solution of cleaning surface active substances (detergent) and decontamination reagent.	1.45E+10 Bq/y (for the cemented waste) 4.44E+08 Bq/ y (for the three types of filters)	18.1 m^3/y (104 drums, 11 RCC) 12 mesh filters/3 y – 0.04 m^3 72 micro filters/y– 0.18 m^3 11 ultra filtrate and reverse osmotic/y – 0.087 m^3
Cleaning of floors	Waters from the special sewage system	300 m^3/y	Water and water solution of the cleaning surface active substances and decontamination reagent.		1 RCC/y

During the Dismantling and Decontamination of the dismantled equipment at Stage 2

The EWN experience concerning the dismantling and decontamination of the dismantled equipment shows that the flow rate of the waste water generated during these operations amounts to 300 m^3/y [50]. The Liquid RAW (LRAW) is collected in special tank. After neutralization and chemical and radiation control the content of this tank is pumped and transferred in the evaporator system for generation of RAW and condensate.

The quantity of the consumed water is less than the annual waste water flow rate because the condensate from the evaporator is reused.

During the wastewater treatment in the evaporator no conventional water pollutant or nutrients are intercepted in the water stream. All LRAW are treated except water from the showers, which, after the dosimetric control, will be discharged directly into the Danube River.

On the ground of EWN experience during the decommissioning of the units in Greifswald NPP it is shown that at the time of decommissioning the total activity discharged into the water for one year (without tritium) is 120 MBq, and regarding the tritium it is 50 GBq. This means that in comparison with the operational period of Units 1-4 (2484 MBq – 1998) the total activity (excluding tritium) was decreased approximately 20 times.

The technologies for decommissioning selected on the basis of the international experience and the proper distribution of the decommissioning activities in time guarantee the limitation of the expected direct effects of the activities on the water quality within the accepted norms. Environmental impact is negligible and can not cause any transboundary impact.

1.12 Generated solid waste - quantitative and qualitative assessment

1.12.1 Radioactive waste

The categorization in force as per Article 5 of the Regulation for safe management of radioactive waste [14], establishes three general categories of solid RAW as follows:

1. Category 1 - transitional radioactive wastes, which could be released from control after appropriate treatment and/or temporary storage for a time period not longer than 5 years, when their specific activity has decreased below the levels for free release from regulatory control;

2. Category 2 – low- and intermediate-level waste containing concentrations of radionuclide for which special measures for heat removal during storage and disposal are not required; the radioactive waste of this category is additionally categorised in:

   a. Category 2a – short-lived low and intermediate level waste containing mainly short-lived radionuclide (half-life shorter or equal to the ^{137}Cs half-life), and long-lived alpha-radionuclide with a specific activity that is lower than or equal to 4E+06 Bq/kg per package and lower than or equal to 4E+05 Bq/kg in the whole radioactive waste volume;

   b. Category 2b – long-lived low- and intermediate-level waste containing long-lived alpha-radionuclide (half-life longer than the ^{137}Cs half-life) with a specific activity exceeding the category 2a limit;

3. Category 3 – high-level waste having radionuclide concentration such that the heat removal has to be taken into account for storage and disposal.

According to the Regulation for safe management of radioactive waste [14], the liquid radioactive wastes, are categorised as per art.5 of the Regulation, in reference to the solid radioactive wastes categorisation on the expected solid RAW characteristics obtained after their conditioning. In cases where a technology for liquid radioactive waste conditioning is not available, the categorisation shall be done taking into account high-tech conditioning technologies.

In consideration of the specific methods applied for RAW treatment and in accordance to the contract relations with SERAW, additional RAW categorization is set up at Kozloduy NPP EAD, in application of art.7 of the Regulation for Safe Management of Radioactive Waste [14]. The additional categories contribute for a detailed adherance to art.5, item 2 of the Regulation and are based on on directly measured operative parameters, within the limits proposed by SERAW, SD RAW-Kozloduy, and meet the provisions of the Procedure for acceptance of RAW in the RAW management facilities at KNPP" The following two main groups are presented:

- Additional categories of solid RAW (category 2a);
- Additional categories of liquid RAW.

The additional categorization of the solid radioactive waste (for RAW category 2) from Kozloduy NPP EAD is as follows:

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E- mail: office@eninbg.com

- Category 2-I – with equivalent dose rate from gamma radiation at 0.1 m from the surface of the waste between 1 μSv/h and 0.3 mSv/h;
- Category 2-II - with equivalent dose rate from gamma radiation at 0.1 m from the surface of the waste between 0.3 mSv/h and 10 mSv/h;
- Category 2-III - with equivalent dose rate from gamma radiation at 0.1 m from the surface of the waste between above 10 mSv/h.

The solid RAW from above additional categories are classified as “compactable” (fabrics, wool and polyvinyl chloride based waste, polyethylene waste and other polymeric waste) and “non-compactable” (metal, wood, building material waste etc.).

Origin of the operational waste

The main source of solid RAW are objects contaminated by artificial nuclides, media, materials and consumables, being in use, or already used in the technological process of electricity generation. Solid RAW are generated during the servicing and maintaining of the facilities in the Units. The analysis of the activity by the Specialized Department “RAW-Kozloduy” shows, that 3 to 5 times more radioactive waste is generated during the planned maintenance of units, than when the units operate on full capacity. As per the morphological characteristics, the RAW could be divided into metal waste, construction waste, spent filters, contaminated tools, wood, protective cloths, rubber, paper, etc.

Table 1.12.1-1 Solid RAW characterization by kind of the material.

Name of RAW	Characteristics
Textile	Protective working cloths, used in the CA, towels, cheese-cloth, used for cleaning and hygiene purposes
Polymers	Mainly polyethylene, used as covering material to protect surfaces from contamination, latex gloves etc.
Wool	Replaced thermal insulation of primary circuit vessels and pipelines
Rubber	Contaminated rubber sealing, rubber gloved etc.
Wood	Contaminated planks, used for the scaffolding during construction and maintenance activities, wooden packs for equipment and tools
Metal	Elements from the technological equipment, which had been decommissioned during maintenance, metal sheets from replaced thermal insulation
Shavings	Waste material from the treatment of radioactive contaminated equipment in the mechanical workshops in the CA
Construction waste	Waste from concrete, bricks, rough coats, floor coatings etc. generated during the construction and maintenance activities
Cables	Surface contaminated cables
Paper	Cardboard box, packing paper etc.
Aerosol Filters	Spent aerosol filters from the special gas purification systems

The generated metal RAW share according to the analysis of the Specialized Department "RAW-Kozloduy" is as follows:

- Non-ferrous metals 10-15 %;
- Stainless steel about 15-25 %;
- Carbon steel 60-75 %.

Characterization of RAW

Each one of the RAW streams by storage sites is characterized in the possible scope and according to the physical, chemical, radiation and other RAW characteristics.

The characterization is done in connection with the implementation of the Regulation for safety management of RAW [14], aiming an assessment to be done prior to their further management by the Specialized Department "RAW-Kozloduy", as well as the necessity to maintain updated inventory lists.

Storage of solid RAW

Solid radioactive waste generated during operational and maintenance activities at Units 1 and 2 are stored in temporary storage rooms in AB-1 and the reactor building of Units 1 and 2 as well as in AB-2 and the reactor building of Units 3 and 4. Detailed description of the locations and quantities on storage of the available solid RAW for Units 1 to 4 is given in section 1.1.2.

General description – solid RAW generated by the Plant during operation and maintenance

The solid RAW is located at two places – in the auxiliary buildings and in the reactor building.

Tables 1.12.1-2 and 1.12.1-3 present the solid RAW generated in KNPP for the period 2009-2011

Table 1.12.1-2 Compactable solid RAW 2-I and 2-II category, m^3 for the period 2009-2011

Site	2009	2010	2011
EP-1	166.96	148.58	51
EP-2	710.18	604.78	587
SNF storage	22.75	21.0	23
TOTAL	**899.89**	**774.36**	**661**

Table 1.12.1-3 Non-compactable (metal) solid RAW 2-I and 2-II category, m^3 for the period 2009-2011

Site	2009	2010	2011
EP-1	6.75	2.38	2.138
EP-2	37.54	12.48	18.131
SNF storage	0	0	0
TOTAL	44.29	14.86	20.269

Note. The amounts of uncompressible (metal) RAW 2-I and 2-II category are given as total amounts. The reason is that the uncompressible (metal) RAW are sorted and kept in 2 m^3 containers in interim storage. From there they are transferred by special transport to SD RAW-Kozloduy.

Categorization of the radioactive materials (RAM) from the decommissioning of KNPP Units 1-4.

For the overall waste mass on site from the decommissioning of nuclear facilities three categories have been adopted as follows:

I category – Potentially non-contaminated material: Control measurements of the material (random checks) are needed for the free release of the material;

II category – Potentially contaminated material: Measurement of all the material is required when making a decision for free release;

III category – Contaminated material: The material is treated as radioactive waste and in single cases it may be free released after decontamination and measurement.

The materials of Categories 2 and 3 are radioactive waste (The definition “radioactive residual material” is used in Germany for all interim stages before final storage). They can be exempted from nuclear regulation after free release measurement procedure (measurement and decision of the radiation regulatory authority when the values are below defined limit values) and possible previous decontamination.

Another type of Category 3 material is operational waste from operation and post operation.

Besides the waste from the NPP dismantling and Category 1 material and Categories 2 and 3 material after free release, important amounts of the disposed waste are generated by other dismantling activities on the site. According the EWN experience more than 10 % are scrap metals. About 90 % of this conventional waste was recovered (e.g. by recycling). The quota of about 10 % not recoverable waste is disposed environmentally friendly by deposition in landfills (dumps) or burning in waste incineration plants. All subcontractors for waste activities are regularly audited.

For the impact assessment, the resulting amount of waste after the dismantling and waste treatment processes is important and is described in the EEIR [50]. This waste from decommissioning can be divided in:

- Conventional waste for utilization;
- Conventional waste for removal (e.g. landfill or incineration);
- RAW for decay storage (e.g. large components for later cutting);
- RAW for reuse or utilization in nuclear facilities;
- RAW prepared for final storage.

The detailed amounts of these types have to be defined in the frame of the preparation of the decommissioning processes, by use of the material data base for detailed planning of the different dismantling activities.

In attachment 11.4.2 is given coarse estimate based on the data from KNPP (KPMU/DPL/013), the EWN experience (KGR) [50] and the calculations from the EIA-Report for the Bohunice NPP [18].

In attachment 11.4.2 is shared the EWN experience (KGR) [50] on the real amounts of waste from the decommissioning as per the current state.

According to the Decommissioning plan of KNPP Units 1-4 [36] during the decommissioning the following streams of materials and wastes will be generated:

- Noncontaminated equipment of the TH;
- Contaminated equipment from the TH;
- Materials from AB and RB for SRDW;
- Materials from the SRDW directed to the free release measurment facility;
- Materials from SRDW not compliant to the free release criteria;
- Solid waste Category I from the equipment for extraction and treatment of the solid phase in the ECT
- Secondary waste from SRDW;
- Asbestos containing noncontaminated waste;
- Asbestos containing contaminated waste;
- Other hazardous substances;
- Non radioactive waste.

Sources and characteristics of the solid RAW during Stage 1 of the decommissioning

<u>RAW from the SE preparation</u>

During SE preparation low and medium level RAW are expected to be generated from:

- Special clothing;

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- Solid RAW (combustible, non-combustible-compactable, non-combustible-non-compactable waste);
- Spent aerosol filters.

RAW from the SE operation

During the control of the SE low and medium level RAW are expected to be generated from:

- Spent aerosol filters.

For the purpose of this plan it is assumed that during the decommissioning of Units 3-4 waste with the same quantitative and qualitative characteristics will be generated as during decommissioning of Units 1-2. The initial RAW assessment during preparation and control of the SE for Units 1-2 is made in [12]. Detailed description of the sources, type and characteristics of RAW from preparation and control of the SE in the Updated strategy for "continuous dismantling" is given in [13] and in the tables below. The activity of RAW is assessed at the moment of their generation. The RAW quantities are assessed for application of the decontamination method, described in the DP.

Sources and characteristics of the solid RAW during Stage 1 and Stage 2 of the decommissioning

During dismantling of two WWER-440 units approximately 450000 t of material are expected to be generated. According to the Greifswald NPP experience [50], 300000 t of materials will be non-contaminated (approximately 2/3 of the total quantity) and 150000 t of materials will be contaminated or potentially contaminated.

Table 1.12.1-4 and 1.12.1-5 respectively show the quantity of generated RAW as it would be if decontamination of the metal from Units 1-2 and Units 3-4 respectively is not performed. The RCC number has been calculated according to the practice of SD "RAW – Kozloduy" (2 t big size non-metal RAW are placed in 1 RCC).

Table 1.12.1-4 shows the quantity of generated solid RAW for Units 1-2 during Stage 1 and Stage 2 of the decommissioning process.

Table 1.12.1-4 Generation of solid RAW during Stage 1 of the decommissioning (preparation of SE and SE Operation) and Stage 2 dismantling of the equipment of Units 1-2, according to the DP [36]

Operation	Type of waste	Quantities for Units 1 and 2	RAW description	Provisional activity (as per the moment of RAW generation)	Number of packages
	Filters	624 pcs.	Collect the whole sludge and contamination, which can be cleaned during	beta, gamma-nuclides: 6.7E+11 Bq; alpha-	24 RCC

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E- mail: office@eninbg.com

Operation	Type of waste	Quantities for Units 1 and 2	RAW description	Provisional activity (as per the moment of RAW generation)	Number of packages
			the operations	nuclides: 3.9E+09 Bq	
Dismantling of thermal insulation	Insulation	600 m^3	Wool	^{60}Co: 25 Bq/g ^{137}Cs: 150 Bq/g	22 RCC
	Special clothing	10 m^3/y	Cotton		15 drums (1.5 RCC)
Technological solid RAW	Burnable	80 m^3 /y	Wood, textile, polymers	0.53GBq/t 4.6GBq/t	134 drums/y (5 RCC/y) (If burning – 8 drums/y, 1 RCC/y)
	non-burnable	4 t/y	Concrete, metal, constructional materials		1 RCC/y
Conditioning of HEPA filters	HEPA filters	14 pcs./y, (1.5 m^3/y after disassembling)			9 drums/y, 1 RCC/y
Conditioning of dismantled equipment	Solid RAW from dismantling of the primary circuit equipment	1950 t	Stainless steel 0X18H10T, carbon steel		1950 RCC
	Solid RAW from dismantling of the AB-1 equipment	1200 t			600 RCC
	Solid RAW from dismantling of SWT-3 and other auxiliary systems	300 t			150 RCC

Table 1.12-5 shows the quantity of generated solid RAW for Units 3-4 during Stage 1 and Stage 2 of the decommissioning process.

Table 1.12-4 Generation of solid RAW during SE Preparation and SE Operation of Units 3-4, according to the DP [36]

Operation	Type of waste	Quantity for Units 3-4	RAW description	Provisional activity (as per the moment of RAW generation)	Number of packages
	Filters	624 pcs.	Collect the whole sludge and contamination, which can be cleaned during the operations	beta, gamma-nuclides: 6.7E+11 Bq; alpha-nuclides: 3.9E+09 Bq	24 RCC
Dismantling of thermal insulation	Insulation	600 m^3	Wool	^{60}Co: 25 Bq/g ^{137}Cs: 150 Bq/g	22 RCC
	Special clothing	10 m^3/y	Cotton		15 drums (1.5 RCC)
Technological solid RAW	Burnable	80 m^3 /y	Wood, textile, polymers	0.53 GBq/t 4.6 GBq/t	134 drums/y (5 RCC/y) (if incine-ration will be done – 8 drums/y, 1 RCC/y)
	Non-burnable	4 t/y	Concrete, metal, constructional materials		1 RCC/y
Conditioning of HEPA filters	HEPA filters	14 pcs./y, (1.5 m^3/y after disassembling)			9 drums/y, 1 RCC/y
Conditioning of dismantled equipment	Solid RAW from dismantling of the primary circuit equipment	1950 t	Stainless steel 0X18H10T, carbon steel		1950 RCC

Operation	Type of waste	Quantity for Units 3-4	RAW description	Provisional activity (as per the moment of RAW generation)	Number of packages
	Solid RAW from dismantling of AB-1 equipment	1200 t			600 RCC
	Solid RAW from dismantling of SWT-3 and other auxiliary systems	300 t			150 RCC

Radiometric characteristics of the solid RAW, defined on grounds of regular measurements (1994-1999) of 200-l-drums prior to super compression in the SE "RAW-Kozloduy" by the use of a gamma scanner [Bq/kg] are presented on Table 1.12.1-6. These data can be referred by a high confidence level to the solid RAW, which are being stored in rooms for decay storage in AB-1: BK 031/3, BK 031/4.

Table 1.12.1-6 Radionuclide contents of solid RAW

No	Type of material	Pγ av., mSv/h	A spec. av., Bq/kg	Radionuclide contents, %				
				^{60}Co	^{110}Ag	^{134}Cs	^{137}Cs	^{54}Mn
1.	Miscellaneous	0.69	3.87E+5	59	4	4	24	5
2.	Polymers	0.33	2.98E+5	64	7	9	9	7
3.	Wool	0.30	5.17E+5	63	9	6	9	8
4.	Textile	0.29	4.02E+5	55	8	7	18	7
5.	Shavings	0.24	3.64E+5	66	6	9	8	8
6.	Rubber	0.24	1.80E+5	65	3	5	20	5
7.	Wood	0.10	1.83E+5	51	1	8	34	5
8.	Metal	0.09	1.30E+5	58	2	8	21	9
9.	Construction waste	0.05	2.96E+5	58	2	5	30	4
All measured RAW		0.25	3.23 E+5	57	6	6	20	6

Quantities of the materials from dismantling in the Turbine Hall

The information about the type and quantity of the dismantled materials in the TH is presented in the tables below.

Table 1.12-6 presents the classification of the materials; table 1.12.1-8 presents the expected quantities of main materials from dismantling in the Turbine Hall, and table

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
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1.12.1-9 includes data about the potentially contaminated equipment in TH. The information includes: contamination category, type of material, as well as the expected quantity, according to the DP [36].

Table 1.12.1-7 Classification of the materials

Material	Symbol		Description
Carbon steel	CST	CST01	Wall Thickness > 30 mm
		CST02	Wall Thickness 10 ÷ 30 mm
		CST03	Wall Thickness < 10 mm
Stainless steel	SST	SST01	Wall Thickness > 30 mm
		SST02	Wall Thickness 7 ÷ 30 mm
		SST03	Wall Thickness < 7 mm
Electric motors	ELM	ELM01	Largest diameter < 400 mm
		ELM02	Largest diameter > 400 mm
Cables	CAB	CAB01	Cable insulation (plastic, rubber)
		CAB02	Largest diameter < 8 mm
		CAB03	Largest diameter > 8 mm
Insulation	ISW	-	
Non-ferrous metals	NFM	NFM01	The metal can be melted
		NFM02	The metal cannot be melted
Electrical parts	ELP	-	
Concrete	CON	-	
Other material	OTH	OTH01	Other material (other than OTH02 and OTH03)
		OTH02	Liquids
		OTH03	Construction materials and soil
Components	COM	-	

Table 1.12.1-8 Summary of the type and quantities of materials in TH, according to [36]

Element	Category I Material not subject to restrictions		Category II Potentially contaminated material	
	Type of material	Mass, t	Type of material	Mass, t

Element	Category I Material not subject to restrictions		Category II Potentially contaminated material	
Pipes				
Pipelines Pipelines	CST02 CST03	62 425	CST02 CST03	485 103
Turbine pipelines			CST	1280
Various elements	CST	219	CST	265
Valves				
Motor and manual valves	CST SST NFM ELM	177	CST SST NFM ELM	248
Reverse valves	CST SST NFM	12	CST SST NFM	23
Regulating valves	CST SST NFM ELM	0.5	CST SST NFM ELM	31
Reduction installations			CST SST NFM ELM	74
Small valves			CST SST NFM ELM	10
Disc armature	CST SST NFM ELM	58		
Pumps				
Pumps	CST SST NFM ELM	15	CST SST NFM ELM	320
Tanks				
Tanks	CST	110	CST	400

Table 1.12.1-9 Description of the potentially contaminated materials in TH, according to [36]

Element	Type of material	Mass, t	Description of the material	Notes
Steam turbines				
Turbine components				
Cage of high pressure (HP) part	CST	38	Carbon steel: 25Л МСт.3	
Shaft of HP part	SST	10	Alloyed steel 34ХН9М 34ХМ	1 element
Rotor blades of HP part	SST		Alloyed steel 1Х13 1Х11 МФ	Max. mass of 1 element: 12 kg
Cage of low pressure (LP) part	CST		Carbon steel: 25L МСт.3	
Shaft of LP part	SST	27	Alloyed steel 34ХН9М 34ХМ	7 welded elements; Max. mass of 1 element: 7.5 t
Rotor blades of LP part	SST		Alloyed steel 1Х13 1Х11 МФ	Max. mass of 1 element: 12 kg
Diaphragms of LP part	CST		Cast iron	
Seals	SST NFM		Alloyed steel 1Х11 МФ ЭИ-802	
Bearing cages	CST		Cast iron	
Stop valves	CST		Carbon steel 25Л	
Regulating valves	CST		Carbon steel 25Л	
Total mass		1347 t		
Condensers				
Main condensers	CST NFM	1446 900	Carbon steel Brass Л-68	8 pcs; Total mass for both units

Element	Type of material	Mass, t	Description of the material	Notes
Technological condenser	CST NFM	11.47 t	Carbon steel Brass Л-70-1	1 piece
Regenerative burners				
LPB		392 t		20 pcs; total mass of LPB
HPB		1128 t		12 pcs; total mass of HPB
Ejectors				
Main ejectors	CST NFM	31.2	Cast iron Carbon steel Brass Л-68	12 pcs; total mass
Starting ejectors	CST	1.1	Carbon steel	16 pcs; total mass
Ejectors		46.3		4 pcs; total mass
Seals ejectors		16.1		total mass
Coolers				
LPB 1 drainage coolant	CST	10.7	Carbon steel	4 pcs; total mass
LPB 3 drainage coolant	CST	29.6	Carbon steel	4 pcs; total mass
Cooler of steam to the deaerator	CST	14.2	Carbon steel	4 pcs; total mass
Technological condenser cooler	CST NFM	11.5	Carbon steel Brass Л-70-1	1 pcs; total mass
Cooler		2.7		1 pcs; total mass

Element	Type of material	Mass, t	Description of the material	Notes
Cooler of samples		5		100 pcs; total mass
Other heat exchangers				
Separator - steam super-heaters		913.6		8 pcs; total mass
	Category 1 Material not subject to restrictions			
	Type of material	**Mass, t**	**Material**	**Notes**
Heat exchangers				
Boilers		27.2		4 pcs; total mass
Oil coolers		39.4		16 pcs; total mass

As the data from the above tables show, radioactively contaminated equipment from Category 3 in TH is not expected.

Fig. 1.12.1-1 shows the flow diagram of the control of 1st and 2nd category wastes level of activity and their release as conventional waste and fig.1.12.1-2 shows the flow diagram of treatment of the streams of 2nd and 3rd category materials and their release as conventional waste. The Invesmnent Proposal aims that maximum amount of RAW and RAM, including the potentially contaminated materials to be decontaminated in accordance with the diagrams below and to be treated as conventional waste. One of the advantages of the investment proposal implementation idea is that the already decontaminated waste will be in their major part utilized by recycling.

This statement is confirmed by the experience from the decommissioning of similar nuclear facilities, as it is the case with the experience from Greifswald NPP decommissioning [50].

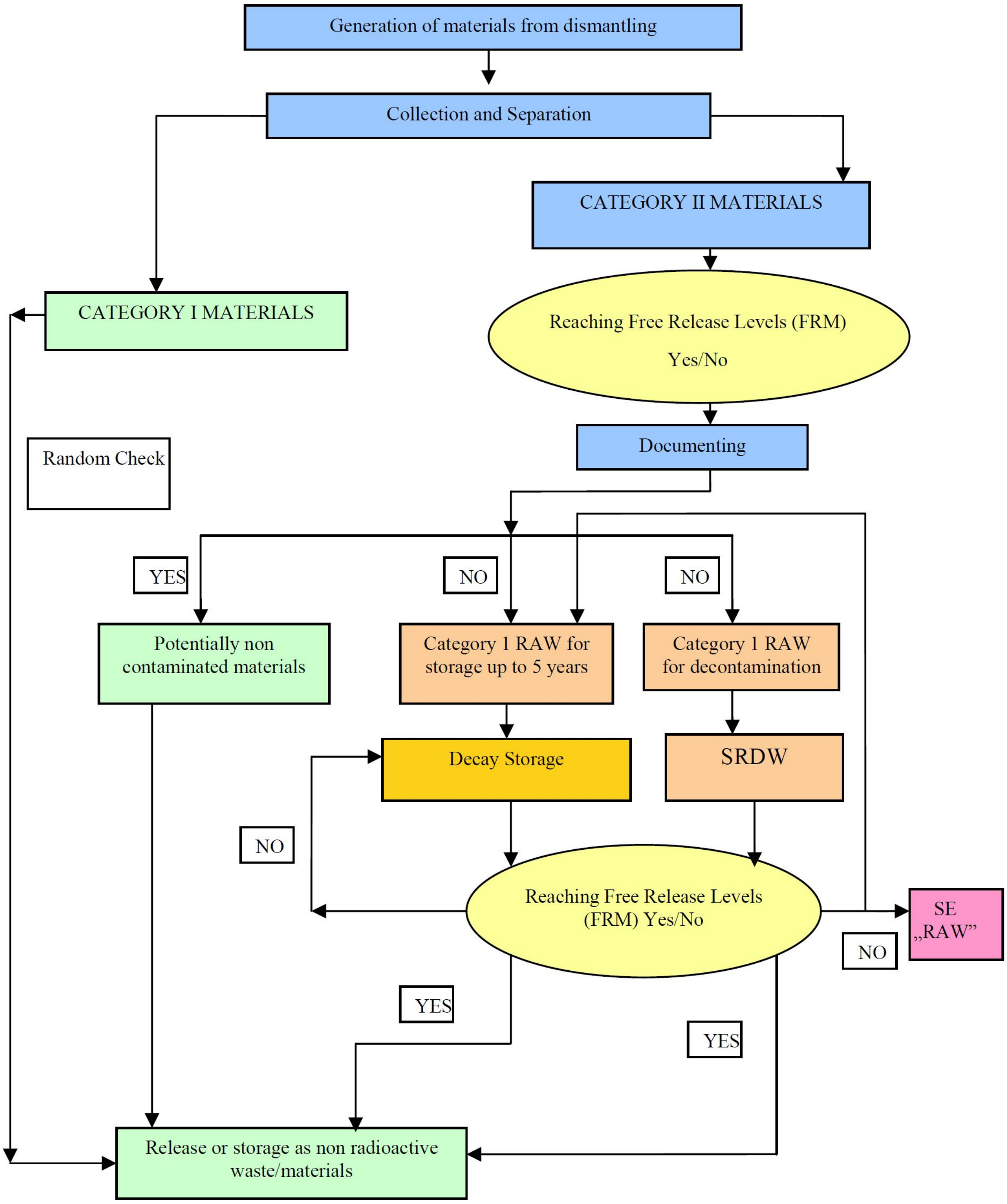

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Fig.1.12.1-1 Flow diagram of category 1 and 2 materials streams and their release as non-radioactive (conventional) wastes

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

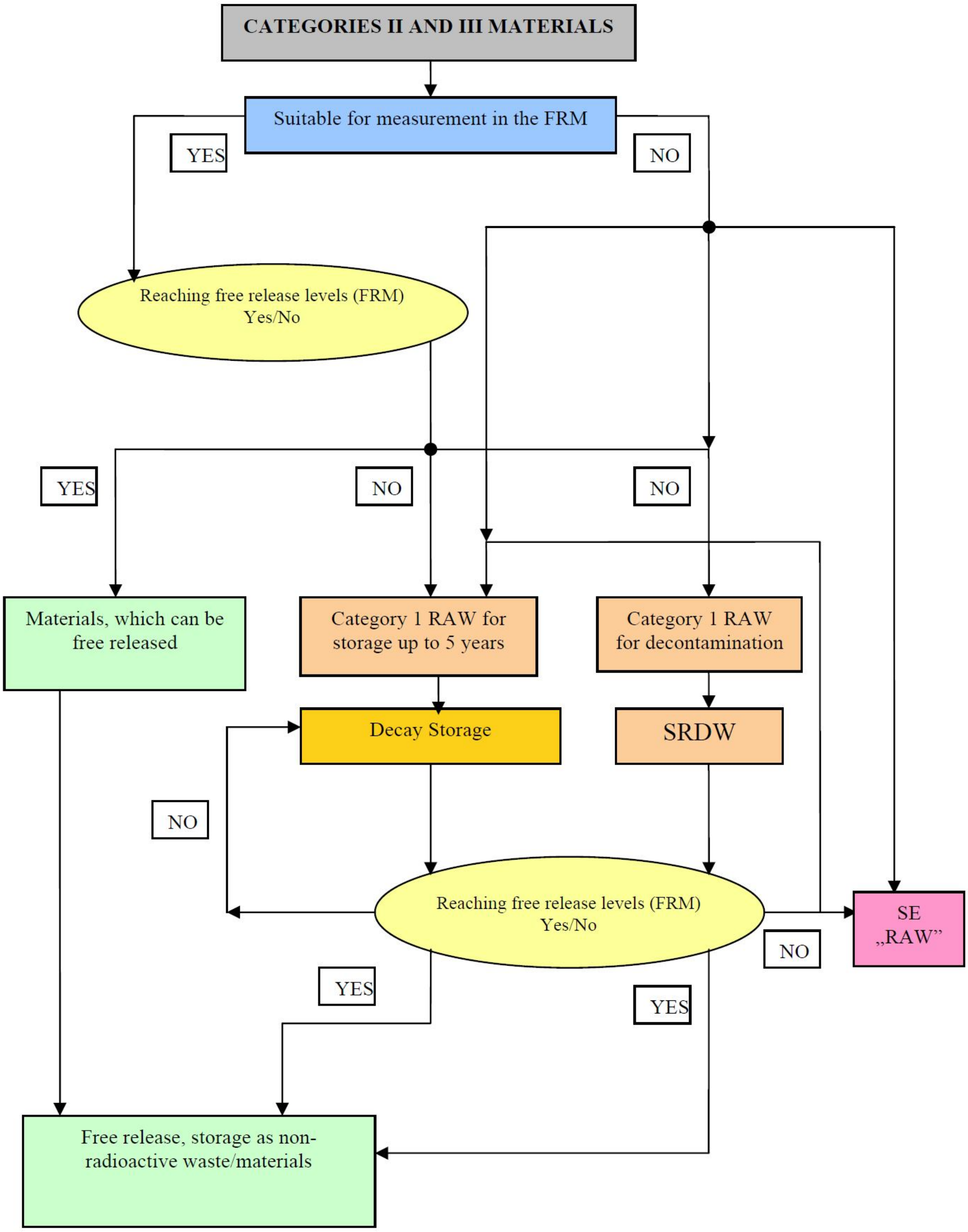

Fig.1.12.1-2 Flow diagram of category 2 and 3 materials streams and their release as non-radioactive (conventional) wastes

As it can be seen from both flow diagrams, the materials not achieving the free release levels are treated as RAW in compliance with the adopted practices in Kozloduy NPP.

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.12.2 Non-radioactive waste

The adopted Environmental Management Policy of Kozloduy NPP shall be extended for the stage of decommissioning of KNPP Units 1-4 with as main goal the safe management, minimization and utilization of the conventional wastes.

Pre-decommissioning Stage (PDS)

During the implementation of decommissioning preparatory activities, upon release from regulatory control and decontamination, considerable solid non-radioactive waste is expected to be generated from dismantling of process equipment and some auxiliary activities (industrial (operational) and hazardous waste) and demolition of ancillary concrete structures (construction waste). Table 1.12.2-1 shows the characteristics and classification of the waste generated during the PDS.

Table 1.12.2-1 Characteristics and classification of the waste generated during the PDS

Description	Code under Regulation No.3/2004 on waste classification	Amount, t/y
Industrial (operational) waste		
End-of life components, other than those mentioned in 16 02 15	16 02 16	4
Inorganic wastes other than those mentioned in 16 03 03	16 03 04	2
Organic wastes other than those mentioned in 16 03 05	16 03 06	1
Ferrous metal	19 12 02	1300
Non-ferrous metal	19 12 03	414
Hazardous substances		
Packaging containing residues of or contaminated by hazardous substances	15 01 10*	1
Insulation materials containing asbestos	17 06 01*	20
Construction materials containing asbestos	17 06 05*	70
Fluorescent tubes and other mercury-containing waste	20 01 21*	5000 pcs./y
Batteries and accumulators included in 16 06 01, 16 06 02 or 16 06 03 and unsorted batteries and accumulators containing these batteries	20 01 33*	7
Construction waste		
Concrete	17 01 01	75
insulation materials other than those mentioned in 17 06 01 and 17 06 03	17 06 04	80
mixed construction and demolition wastes other than those mentioned in 17 09 01, 17 09 02 and 17 09 03	17 09 04	5

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20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
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Municipal waste		
Mixed municipal waste	20 03 01	18
Batteries and accumulators other than those mentioned in 20 01 33	20 01 34	3

During the first stage of the decommissioning the dismantling of the equipment wich is not needed in the further decommissioning stages and is not contaminated will be done outside SE. Dismantling of the *Turbine hall* will be done after the removal of the inflammable (oils from the transformers and turbine generators) and hazardous materials (asbestos) and dismantling of the heat insulation in the Turbine hall of Units 1-4 and organization of decay storage facilities.

Dismantling includes the unit transformers, generators, condensate and air-condensate tract of the TG of Units 1-4, feed up pipelines and facilities of Units 1 and 2, turbines and auxiliary equipment, condensers and turbines of Units 1-4, indoor civil structures [36]. After assessment of possible contamination in accordance with the methodology presented in [39] it is expect that 1145.1 t of materials from the dismantling at Units 1 and 2 can be referred to category 1 – unrestricted material. According to the preliminary assessments given in [36] for Units 1 and 2 it is expected that 1/11 of the total amount od the dismantled materials in TH can be classified as category 1.

The total volume of the basic equipment dismantled in Turbine Hall concerning Units 1 and 2 (steam turbines, condensers, re-heaters (LPR, HPR), ejectors, chillers, other heat exchangers) is 6306.5 t. For the four units subject to decommissioning the volume will be doubled.

From the dismantling of piping, valves, pumps and tanks are expected to be isolated more basic, useful materials like motors, electrical parts, and cables with total estimated mass of 3239 t.

Important aim of the dismantling is to reach maximal re-use and recycling of the dismantled materials, especially of the metals and it is assessed as positive impact. Items of equipment which are definitely recycling are: turbines, transformers, diesel generators, pumps, motors, cables.

In spite that a part of the equipment could be re-used or be used as spare parts, more realistic is that the biggest part of the equipment will be sold as scrap material**.** Dismantling should be directed towards complete disassembly, sorting and decontamination of the metals, if needed, with following re-sale and free release as *scrap,* depending on the market situation.

Another area of high importance for recycling utilization is the retrieval of the non-ferrous metals from spent cables. There are some technologies for recycling of copper wire used for extracting of the copper from polluted insulation and dust. Restored copper wire could be recycled and used again in order to reduce the scope for disposal as well as aiming to re-use this natural product. This technology is currently applied in a number of projects for decommissioning of nuclear facilities worldwide. Technology is elaborated in Germany and is successfully applied in Europe for restoration of many tons of copper wire, discharge from control and second use as well. Treated cable is split in pure copper, lightly polluted insulation and polluted dust. Polluted dust generated by the grinding is filtered at several stages in order to prevent the air pollution. Process of copper recycling is made under a negative pressure at the moment when the wire is supplied before the cutting machine to its output from the separator. Processed air is filtered through the ventilation system with HEPA filters [7].

The management of the construction waste generated during decommissioning of KNPP Units 1 to 4 shall be in compliance with the Regulation on the construction waste management and use of recycled construction waste (prom. SG 89/134.11.2012), and in line with MoEW recommendation (letter No OVOS-289/09.01.2013)

The hazardous waste will be temporary stored on the industrial site at especially dedicated areas (Site 3 – fig. 4.1.11.1-1) and after collecting of certain amounts they will be handed over for further treatment to specialized companies holding a permit under art. 35 of the WMA. For exemple the management of the discarded fluorescent tubes and mercury-containing tubes, is planned to be temporary stored on the RCMIW in concrete container until their handing over to the company autorized for further treatment. The capacity of one container is for about 2000 tubes.

During the preparatory activities for decommissioning the asbestos from Units 1-4 will be removed and disposed of. The quantities and characteristics of radioactive and non-radioactive asbestos, as well as the disposal locations are given in the Plan for decommissioning [36]. According to the schedule for decommissioning [36], the preparation for SE of Units 1-2 begins with the coming into effect of the permit for decommissioning. The QA program for Stage 1 from the decommissioning of KNPP Units 1-4 includes, first of all, the removal of asbestos, combustible materials and conventional waste. Regardless of the fact that the permit for decommissioning has not been issued yet, in order to reduce the potential hazards from the facilities, the activities related to the removal of these materials from Units 1-2 have already started.

After the decommissioning of Units 1-4 in condition “E” the materials containing asbestos, that are used for thermal insulation, are in “safe” condition (under a tin, cement plaster or at some places of restricted access).

In compliance with the requirements of the Ordinance for prevention of the environment pollution with asbestos [40] and Ordinance for protection against risks for work with asbestos [41] updated detailed inventory is elaborated of the places in the RB and TH, where a heat insulation containing asbestos materials is available as well as estimation of the quantities of hazardous materials. It is foreseen to remove the heat insulation and other asbestos materials from the carrying structures such as roof panels, inside/outside linings of walls and lining panels, pipelines and reservoirs etc. during the operation license. It is foreseen to remove the insulation (asbestos) before the dismantling of the pipelines and in this way to prevent the possibility for breathing in or swallow asbestos dust during the cutting and fragmentation.

It is estimated that the total quantity of the heat insulation of asbestos rope for insulation of the pipelines in Units 1 and 2 is 247.6 kg. Total quantity of the heat insulation materials containing asbestos is of thickness of 30 to 120 mm in the pipelines and facilities in the turbine hall Secondary circuit of Units 1 and 2 is estimated to 89391 kg. After the elimination of 17700 kg of turbines (TG-2 and TG-4)

EWN Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

before 2001, the approximate section is 72 t. Analogical asbestos quantity will be removed from the facilities of the TH in the Secondary circuit of Units 3 and 4.

In compliance with program P.NECT – 18AB for safe dismantling in March 2009 another 11825 kg of heat insulation materials containing asbestos were removed in the TH, Secondary circuit of Units 1 and 2, which are placed in polypropylene flexible containers after being marked and described.

Different methods are accepted for elimination of the asbestos depending on the asbestos type. They could vary from the movement of the material before elimination of the asbestos (in case of non-covered synthetic fibres and asbestos rope or uncovered corrugated asbestos cloth) until construction of entirely ventilated closed room, where the staff should wear respirators (covered with tents heat-insulation slab/cement cover slabs/wall panels/fire protection slabs/insulation pipes).

After an execution of control of the radioactive contamination of the heat insulation the contaminated bags are placed in a closed metal barrel for decay storage in BPF-1 (Boron Preparation Facility) in the restricted area. Not-contaminated bags are placed in the decay storage facility in the TH Secondary circuit, elevation 00.00 m to 1 HC. The activities are performed according to decision N 1/08.05.2008 of Regional inspection for protection and control of the public health, Vratsa.

Transportation diagram is prepared for the containers with waste, storage of the containers with the waste. With decision 05-ДО-72-01 from 12.06.2008 the storage of insulation and construction materials containing asbestos is arranged in RCMIW – 90 t/year of each type until 31.12.2010 which is amended and remains valid until 31.12.2015.

The municipal waste will be as currently collected in two type of containers (2 m^3 and 4 m^3) located next to the main buildings. For the municipal waste collected in a mixed way the disposal will be to the dedicated for this purpose regulated depot, namely:

- Wastes from the controlled area are transported to the RCMIW
- Wastes from the site outside the CA – to the regional landfill – Oriahovo town.

Stage 1 and Stage 2 of the Decommissioning)
About 80 % of the conventional waste generated in Decommissioning stages 1 and 2 is utilizable waste.This waste will be temporary stored on the industrial site at especially dedicated areas (Site 3 – fig. 4.1.11.1-1) and after collecting of the defined amounts they shall be handed over, under the respective contract, for further treatment to specialized companies holding a permit for waste activities, issued by the competent authority under art. 35 of the WMA.
During both decommissioning stages is expected the generation of the waste given in table 1.12.2-2 with its classification according Regulation 3/2004. The estimate of the waste state and treatment methods is aligned to the provision of the national legislation related to the management of the different types of waste. From the dismantling activities important volume of conventional waste is generated from buildings, equipment, systems and their components. The expected quantities are given based on an expert assessment. In the present supplemented EIA report is respected the MoEW recommendation (letter ref No.OVOS-289/09.01.2013) and a

classification of waste is made according the Regulation on waste classification subject to art.3 of the Waste Management Act.

Table 1.12.2-2 Characteristics and classification of the waste generated during Decommissioning Stages 1 and 2

Description	Code under Regulation No.3/2004 on waste classification	Amount, t/y
Industrial (operational) waste		
End-of life components, other than the specified in code 16 02 09 - 16 02 13	16 02 14	5
End-of life components, other than the specified in code 16 02 15	16 02 16	8
Inorganic wastes other than those mentioned in 16 03 03	16 03 04	20
Organic wastes other than those mentioned in 16 03 05	16 03 06	30
Alkaline batteries (except 16 06 03)	16 06 04	5
Ferrous metal	19 12 02	1300
Non-ferrous metal	19 12 03	414
Hazardous substances		
Wastes not otherwise specified (spent lubricants and greases)	07 06 99*	1
Mineral based non-chlorinated hydraulic oils	13 01 10*	30
Non-chlorinated engine, lubricating and gear oils, mineral based	13 02 05*	50
Packaging containing residues of or contaminated by hazardous substances	15 01 10*	1
Absorbents, filter materials, wiping cloths and protective clothing contaminated by hazardous substances	15 02 02*	1
Interceptor (collector) sludges	13 05 03*	1200
Discarded equipment containing hazardous components (3), other than those mentioned in 16 02 09 to 16 02 12	16 02 13*	15
Inorganic waste containing hazardous substances	16 03 03*	20
Organic wastes containing hazardous substances	16 03 05*	20
Lead batteries	16 06 01*	40
Ni – Cd batteries	16 06 02*	10

Construction waste		
Concrete	17 01 01	100
Insulation materials other than those mentioned in 17 06 01 and 17 06 03	17 06 04	50
Mixed construction and demolition wastes other than those mentioned in 17 09 01, 17 09 02 and 17 09 03	17 09 04	100
Municipal waste		
Mixed municipal waste	20 03 01	20
Bulky waste	20 03 07	10
Discarded electrical and electronic equipment other than those mentioned in 20 01 21, 20 01 23 and 20 01 35	20 01 36	3

Waste treatment shall be conducted in compliance with the WMA and the subsequent regulatory documents on its application and in accordance with MoEW recommendation (letter ref No.OVOS-289/09.01.2013).

Subject to segregate collection and treatment are:

- the waste which due to its specific characteristics and/or the regulatory requirements is defined as hazardous waste;
- the operational (industrial) waste, the hazardous and recoverable industrial (operational) waste, temporary stored on especially dedicated areas on the plant site, which will be subject to further sale or handing over to external licensed organization for further treatment.

In connection with the SE Preparation and the SE Operation the waste storage sites shall meet the requirements of the relevant regulations under art.13 (1) and art.43 (1) of MoEW (MoEW recommendation - letter ref No.OVOS-289/09.01.2013)

Also, it is necessary to create a data base for the conventional waste movement as soon as possible, to solve problems faced with the handing over to licensed subcontractors of the waste for segregate collection, as well as to solve the problem with the stale chemical substances and mixtures.

Waste transportation and treatment shall be compliant with the requirements of the Regulation on the requirements for treatment and transportation of industrial (operational) and hazardous waste.

Industrial (operational) waste: The waste from the dismantling shall be sorted out on site and shall be properly directed to the disposal sites. For the different waste streams separate places for unloading shall be dedicated. Special sites for scrap storage shall be identified. The ferrous and non-ferrous metals can be sold or sorted out and prepared for sale at the site. The same is valid for other waste, suitable for utilization by recycling.

Hazardous waste: The sites dedicated for temporary storage shall be adequately designated in line with the Regulation on the requirements for treatment and transportation of industrial (operational) and hazardous waste. For instance, the good practice to collect the discarded fluorescent and mercury-containing tubes to be temporary stored in especially designed concrete container for temporary storage (i.e. according the Regulation on the requirement for marcketing of fluorescent and other mercury-containing tubes and for treatment and transportation of discarded fluorescent and other mercury-containing tubes). Spent oils from equipment will be collected in drums and after that handed over to external companies holding of a Permit under art.35 of WMA for utilization. The accumulators shall be collected in a segregate way and handed over for utilization of recycling by external companies holding of a Permit under art.35 of WMA etc.

Construction waste: The waste from demolition shall be segregatelly collected and transported and their mixture is unacceptable as per MoEW recommendation - letter ref No.OVOS-289/09.01.2013. The management of the construction waste generated during decommissioning of KNPP Units 1-4 shall be compliant with the provisions of the Regulation on the construction waste management and use of recycled construction waste (prom. SG 89/134.11.2012), and in line with MoEW recommendation (letter No OVOS-289/09.01.2013).

Municipal waste collected in a mixed way will be disposed to the dedicated depot (landfill) under contract by licenzed company holder of the respective permit for the corresponding waste activities according art.35 of WMA. The municipal waste collected in a mixed way will be disposed of to the, for this purpose dedicated and regulated depot, namely:

- Wastes from the controlled area are transported to the RCMIW
- Wastes from the site outside the CA – to the regional landfill – Oriahovo town.

The waste generated from the activities at this decommissioning stage shall be handed over based on written contract to operators holding the relevant document under art.35 of WMA and and in line with MoEW recommendation (letter No OVOS-289/09.01.2013).

The management of the conventional waste during the stages of decommissioning of Units 1-4 has to be compliant to the new Waste management Act which is in force from 13 July 2012.

Stage Close down and Land Reclamation

This stage is quite remote in the future and given the specificity and location of Units 1-4 most probably the site will be used for new production capacities and auxiliary equipment and when it is not applicable a biological reclamation project shall be implemented.

The biological reclamation will generate waste from packings of fertilizers as well as municipal waste from the workers household activities. Their quantities will be insignificant and will be managed following the recognized practices applied during the previous stages.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.13 Generated energetic pollutants - quantitative and qualitative assessment

The main pollutants generated during decommissioning of Units KNPP 1-4 are the following harmful physical factors:

- Ionizing radiation;
- Thermal releases;
- Noise;
- Vibrations;
- Other factors.

Ionizing radiation

Estimation of the Collective Effective Dose (CED) from dismantling of the KNPP Units 1 to 4

Based on the comparison of the CED during the EWN experience during the decommissioning, the CED can be estimated for the dismantling of the KNPP Units 1 – 4, including the activities in the SRDW and the Sites for temporary safe storage of radioactive and non-radioactive materials generated by the decommissioning activities of Units 1-4 at Kozloduy Nuclear Power Plant (Decay Storage Sites):

- CED: 200 mSv average annual value during the decommissioning period for 350 exposed persons.

This CED summarizes all activities during the dismantling period: the dismantling of the equipment; the treatment, transport and storage of waste from dismantling and operational waste from dismantling in the Units 1 – 4, including the SRDW and the Decay Storage; post operation activities and maintenance of all needed facilities; radiation protection and last, but not least security.

The shares of summarized CED for the dismantling of the activated components, the dismantling of the whole equipment of one unit and the treatment and storage of RAW can be estimated according to the EWN experience as follows:

- Share of summarized CED for the dismantling of activated components: 300 mSv,
- Share of summarized CED for the dismantling of one unit (without waste treatment): 500 mSv,
- Share of summarized CED for the treatment and storage of RAW: 300 mSv.

These CED values may be interpreted as upper limit, because additionally the high level of the safety culture at KNPP has to be taken into account and furthermore the consequent realization of the ALARA principles.

During the operation period the CED for Greifswald NPP was around 10 Sv/year (average for the period 1986 – 1989).

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

In the post-operation period (1991 – 1995) the CED decreases from 0.56 Sv/year (1991) to 0.23 Sv/year (1995). During the on-going dismantling stage the average annual CED (1996 – 2011) was between 0.26 Sv/year (1998) and 0.06 Sv/year (2011).

Based on the EWN experience during the dismantling stage the average annual CED (1996 – 2008) was between 0.12 Sv (2001) and 0.26 Sv/year (1998). For this stage the radiation exposure for the average annual CED can be divided depending on the activities in the following approximate range:

- Dismantling, including treatment of waste from dismantling between 30 and 50 %
- After operation and all other activities (above 3. – 6.) between 20 to 30 %
- Control around 10 to 20 %
- Radiation protection less than 10 %
- Refuelling (in CASTOR containers) around 5 to 10 % (1999 – 2006)
- Conditioning of the operational waste around 5 to 10 %

The annual CED according to the EWN experience [50] is comparable to other decommissioning projects for NPP in Germany (Würgassen and Stade). The data show that the radiation impact during several stages of the decommissioning and dismantling is considerably lower than during operation [10].

The total CED for dismantling of all facilities and parts of the plant (without dismantling of the building structures and the active components) for one unit is < 0.5 Sv. The total CED for the remote dismantling of the active components (components from the active zone without the reactor vessel) for one unit is < 0.1 Sv. The main part of the radiation impact is due to the preparatory activities. The total CED for all activities with all activated components of Units 1-5 until their storage in TSA was 270 mSv.

The total CED for the entire transition period and for the decommissioning period is expected to be lower than the CED for one year of operation.

Thermal releases

The necessary quantity of water during normal operation of KNPP at average annual power of 2500 – 3000 MW is between 110 and 140 m^3/s, which is 2 % from the average annual flow of the Danube, according to [82]. The difference in the temperatures of the cold and the hot channel during KNPP operation at full power, when 180 m^3/s of water are used, reaches 11.6 °C. The water of the Danube is heated by 0.37 °C after the inflow of water from the hot channel.

It is calculated that after the final shutdown of Units 1 and 2**,** the KNPP site global thermal releases will be reduced by about 24 %. After the shutdown of Units 3 and 4 the total heat discharges will be reduced additionally (total for Units 1 to 4 – around 50 %), which makes the heat impact on the river water twice lower than when the units were in operation, i.e. it is negligible and limited only within the area around the hot channel. Small residual heat emissions will be mainly caused by cooling of the spent fuel and the operation of the evaporators of the water cleaning system.

Noise

Activities with the highest level of noise are demolitions of building structures (not foreseen at present) by operation of hydraulic chisels and operation of excavators and the ramming of pilots for the foundation of new buildings (e.g. for the projects SRDW and Heat Generation Plant).

Table 1.13-1 Noise levels resulting from the performed activities according to EWN experience [50]

Activity	Sound power level L_{WA} [dB (A)]	Peak sound power level L_{WA} [dB (A)]
Ramming of pilots	126	140
Operation of hydraulic chisels	105	125
Operation of excavator	100.5	125

These impacts are short term impacts.

Vibrations

As per EWN experience there are no data about vibrations levels from the demolition and ramming of pilots activities.

Vibrations by operation of the SRDW

According to the Technical Specification for the SRDW areas for cutting operations are planned. The selection of the necessary equipment is the task of the Contractor. In case of usageof a scrap shear, essential vibrations are not expected when this facility is mounted on leaf spring packs (EWN experience with the scrap shear MARS).

Other harmful physical factors

Workers, involved in the dismantling facilities will be exposed to:

- Dust and metal aerosols;
- Asbestos;
- IF and UV radiation.

Passive exposition of metal aerosols could cause so called “zinc fever” taking a course as severe pneumonic reaction. Specific toxic effect depends on the type of the metal aerosols.

During the dismantling of the insulation equipment some dust will be generated containing asbestos particles. It is clear that the asbestos is a proven carcinogen for the lungs and pleura. All individual protection measures will be applied (wearing of anti-dust masks during the demolition of the insulations and waste collection).

During the installation and construction works some working groups will be exposed to infrared and ultraviolet radiation (welding).

EWN Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

1.14 Risk of accidents

During execution of various decommissioning activities for Units 1-4 certain accidents are possible. In this case the limiting accidents in terms of risk for the personnel and the population are considered.

1.14.0 Dispersion calculation methodology of KNPP radionuclide releases and field of contaminations in the ground athmosphere layer

The fields of contamination are defined using the methodology by the concentrations at ground level and by depositions, based on the models GAS_E (for gases) and AER_E (for aerosols – dust), elaborated and implemented in NIMH at Bulgarian Academy of Sciences. The models are designated for use by means of a PC and comprise a sophisticated (detailed) version of representation of the so-called Gaussian plume. It is being described in a most general way using the Gaussian plume formula proposed by EPA-USA, introduced in the “Methodology for the calculation of the height of the discharging facilities, the dispersion and the expected concentrations of polluting substances in the ground atmosphere layer” implemented in 1998 by the MoEW, MRDPW and MH:

$$C(x,y,0)=\frac{Q}{2\pi\sigma_y\sigma_z U}\exp\left[-\frac{1}{2}\left(\frac{y}{\sigma_y}\right)^2\right]\exp\left[-\frac{1}{2}\left(\frac{H+\delta H}{\sigma_z}\right)^2\right],$$

where C - the concentration of a pollutant at ground level (z=0), generated by a single spot source at constant meteorological and emission conditions (as such can be considered the conditions within 1 hour);

Q - source capacity;

x, y – spot coordinates;

U - wind speed;

H – stack (chimney) height;

δH – thermodynamic lift force;

σx, σy - dispersion functions, depending on atmospheric stability.

The main characteristics of the GAS_E and AER_E models are defined as follows:

- *Stability.* It is defined in accordance with the meteorological conditions based on the Pasquil classification in 6 classes - A, B, ..., F. Class A corresponds to strong instability, D - to neutral stratification, and F - to strong stability. For better precision intermediate classes AB, BC, ..., EF are also introduced. The determination of the stability classes is made using tables, charts or formulas based on various meteorological parameters, most often measured close to ground level. In this case, when using the models for the estimation of pollutants from NPP Kozloduy the stability class is determined with measuring devices and is included into the complete set of meteorological parameters, including wind, temperature, precipitations.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

- *Dispersion functions σ_x and σ_y.* In the models the Briggs formulas are used. They are derived from abundant experimental material and are accommodated to account for urban and rural conditions.
- *Thermodynamic lift.* Proposed by Briggs formulas are used for the lift of the plume over the stack opening as a result of the inertial and Archimedes forces. The temperature at stack level is determined by the ground level temperature, using vertical temperature gradients, typical for the various stability classes.
- *Wind speed at the height of the stack opening*. It is determined by the ground level wind and by the proposed from EPA exponential factors, depending on the stability class.
- *Effect of the plume meandering.* A correction is considered for the time of concentration sampling and is implemented in the dispersion functions.
- *Height of the mixing layer.* In the models is provided the possibility to implement such information.
- *Calm weather conditions.* They are treated by averaging the plume in all directions, formed by wind speed of 0.5 m/s.
- *Height above sea level.* It is introduced optionally in the model for each point of the field.

The input information consists of 4 parts:

- Modeling area parameters - rectangular area around the polluting source, oriented by **x**-coordinate in East direction and by **y**-coordinate in north direction. The number of points, where the concentration and the deposition are calculated, depends on the area dimensions and on the preferred step in the space.
- File with meteorological parameters – height of the mast and type of the wind gauge (anemometer), date and hour, stability class, wind direction and speed, temperature at ground level, precipitation intensity, height of mixing layer. The duration of the period of averaging and sampling is also put in.
- Source parameters - number of sources, X and Y- coordinates, stack height, elevation of its base above sea level (optional), initial horizontal and vertical dispersion (optional), diameter of opening, temperature of discharged (released) gases, discharge velocity or mass flow rate. The initial dispersions are introduced, in order to enable the spot (point) source to approximate area and volume sources.
- Parameters of the modeled pollutants – gas or aerosol. In case of a gaseous pollutant is introduced velocity of dry/wet deposition and speed of chemical transformation (period of radioactive half-life). By the aerosols is also taken into account the gravitational deposition. A change in the spectra of the aerosols as a result of the heavier particles-closer to the source is considered by means of parameterization of the specific processes by the aerosol processes, proposed by M.Galperin and by the implementation of an average factor to reduce the aerosol mass. The wet deposition (washing) is considered by the precipitation intensity.

The deposition of the pollutants from the atmosphere (surface contamination of open areas with harmful substances, deposited on the ground surface) is determined with a numerical calculation considering the following processes of cleaning the atmosphere from pollutants:

a) flow in direction to the surface as a result of the diffusion adhesion (dry precipitation), determined by the formula

$$DD(x,y) = V_d C(x,y)$$

where Vd - deposition speed;

C - concentration at point (x, y).

b) washing by precipitations (wet deposition)

$$WD(x,y) = Q(x,y)HT(x,y)[1-\exp(-\gamma Rt)]$$

where Q(x,y) - the mass of pollutant in the plume in the coss-sectional area through point (x,y);

HT - mass distribution through cross-sectional area;

γ - washing factor (for gas or aerosol);

R - precipitation intensity, mm/h.

c) by the aerosols is considered also another deposition flow, caused by gravitational precipitation

$$GD(x,y) = Q(x,y)HT(x,y)[1-\exp(-Xt)]$$

where X - average speed for atmosphere cleaning as a result of the gravitational precipitation, depending on the particle size, s^{-1}.

From the total and averaged deposition flow for the period [g/m2s]

$$FD = DD + WD + GD$$

is determined the daily average deposition by integrating for the respective period.

In case of radioactive pollutants also the total decay for the period is determined as:

$$\int_0^T FDdt = \int_0^T Fe^{-C_t t}dt = \frac{F}{c}(1-e^{-C_t T})$$

The source, or the quantity of radioactive substances, discharged in the air, is usually estimated with the following main quantities: initial activity, present in the zone of the accident (Ai); the atmosphere released fraction (ARF); the inhaled (respiration) fraction (RF) and the transmittance factor (LPF).

$$A = \sum_i^n A_i.ARF_i.RF_i.LPF$$

With:

Ai - the initial quantity of radioactive substances, subjected to risk during the accident, accessible for physical effects.

ARF - factor, used for estimation of the quantity of radioactive substances, released in the air as aerosol, and thus being accessible to transportation due to physical influences by specific accident.

RF - fraction of transmittable though the air (airborne) radionuclides, as particles, that can be transmitted through the air and inhaled into the human respiratory system and in general is assumed to contain particles with aerodynamic equivalent diameter (AED) of 10-µm and less.

LPF - fraction of radionuclides in aerosol form, passing through some capturing environment, deposition or a filtration mechanism.

LPF is not appropriate for the HEPA filters of the ventilation system of AB-2 for the estimation of the source, that will be used for the calculation of the doses of radiation exposure of the personnel (staff).

Methodology for calculation of personnel and population exposure in emergency cases

Input data

The initial amounts of radioactive substances, which are causing a risk for the personnel and the population in case of progress of the accident.

Model for the calculation of the exposure dose

The inhalation dose, which corresponds to the discharged radioactivity, for the population and for the personnel regarding every isotope is calculated.

The aerial path is of major importance for the non-reactor nuclear installations. This position is also supported by the publication NUREG-1140, “A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees”, according to which “for all materials of greatest interest for the fuel cycle and for the issuing of licenses for use of other radioactive materials, the dose from the inhalation pathway dominates in the overall dose”.

For emergency conditions, which are causing discharge (release) of radionuclides in the air, to the inhalation dose must be added also the doses, caused by radiation from the soil as well by radiation from the clouds, in order to obtain the total individual dose. Only the inhalation doses are calculated, since the doses from the other radiation sources are usually lower by orders of magnitude than the inhalation doses. The dose at a certain point is calculated as the sum of the results, obtained for the inhalation doses for all isotopes.

For a given isotope, the inhalation dose is calculated using the atmospheric dispersion factor (X/Q), the dose conversion factors (DCF), and the volume of inhaled air per unit time. The dose at a given place is calculated as the sum of the results, obtained for all isotopes.

Effective dose of the personnel (staff)

The inhalation dose for a given isotope or organ is calculated in the following way:

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

$$D_{inh} = \sum \frac{A_i}{V}.BR.T_e.DCF_{inh}$$

The meaning of the parameters in these equations is the following:

A = released activity of a given isotope (Bq).
V = room (premise) volume (m^3)
T_e = time for sojourn in the room (s)
DCFinh = dose conversion factor of inhaled dose of a given isotope (Sv/Bq) for a staff member.
BR = breathing rate (volume of the inhaled air per unit time for a staff member) (m^3/s).

The computer code for performing the dose calculations of the staff is MICROSHIELD 6.20.

Models to calculate the population exposure dose

According the SAR [58], by the calculation of the radiation exposure of the population during the decommissioning, the following models are used:

For a given isotope and organ, the inhaled doses are calculated as follows:

Effective population dose

$$D_{inh} = A_i.\frac{X}{Q}.BR.DCF_{inh}$$

The meaning of the parameters in these equations is the following:

Ai = released (discharged) activity of a given isotope (Bq);

X/Q = atmospheric dispersion factor (s/m^3), calculated according, 9.3.4.1

BR = breathing rate - volume of the inhaled air by a member of the population (m^3/sec);

DCF_{inh} = conversion factor for inhaled dose for a given isotope (Sv/Bq) for population members of the group of adults (>17 years).

The doses are calculated for the closest population and for the critical distance, where the deposition of particles is bigger due to the dispersion by the wind. In accordance with this and considering the atmospheric dispersion, the doses are calculated in the PAZ of NPP „Kozloduy" (2.2 km from AB-2) and up to 5.5 km.

The dose conversion factors (DCF) are taken from the recommendations of ICRP (International Commition on Radiological Protection):

- PUBLICATION 68 OF ICRP: Dose factors for aspiration of radionuclides by the personnel.
- PUBLICATION 72 OF ICRP: Aspiration doses of radionuclides depending on the age of the individuals of the population, part 5, combination of dose factors from digestion and inhalation.

These are the newest, internationally recognized values, adopted in Directive 96/29 of Euratom. These factors are included in tables B and D of this Directive (Ref. [58]).

The inhalation dose factors (Sv/Bq) for the population are presented for six age groups (0-1 year; 1-2 years; 2-7 years; 7-12 years; 12-17 years; over 17 years) based on the radioactivity of inhaled particles with a median aerodynamic diameter 1 µm.

In order to determine which group is critical, it is necessary to analyze the combination of the inhaled air volume (BR - Breathing Rate) per unit time and the dose conversion factor DCF for each group. Higher values of the product of DCF and BR for the nuclides are of greater practical importance in the Plasma melthing facility (PMF (^{60}Co and ^{137}Cs) correspond to the group of the adults. Therefore the doses factors of the inhalation dose of adults are used, and this age group is considered to be critical for the estimation of the inhalation doses. The dose factors of the ICRP for the inhalation dose of the personnel are given as activity median aerodynamic diameter (AMAD) 1 µm to 5 µm. In the calculations are used dose factors for conditions of slow solubility (type S) and AMAD, equal to 5 µm, as prescribed by ICRP in case of lack of specific information.

Other input data

Assessment data on the inhaled air per unit time (BR) according the Regulation on basic radiation protection norms, considering, that the annually inhaled air volume by an adult population member of age over 17 years is 8100 m^3 and by personnel member is 2400 m^3:

- BR population =2.57E-4 m^3/s

- BR personnel = 3.92 E-4 m^3/s

- It is assumed, that the room volume BK301 is 23700 m^3 (Ref. [58]).

Atmospheric dispersion model

The atmospheric dispersion factors (X/Qs) are calculated according the criteria, defined in NUREG/CR-5164 (Ref.[58]). The calculations of the X/Q factors are based on the theory, that the released (discharged) into the atmosphere substances have a normal (Gaussian) distribution around the central line of the plume. A straight-line trajectory between the discharge (release) point and all points, for which the X/Q values are calculated, is assumed.

After being released (discharged) in the atmosphere, the radioactive gases and aerosols will follow the prevailing winds and will be dispersed, due to the cumulative effects of the atmospheric turbulence. Estimates of dispersion are made using an expression, based on the Gaussian plume model.

$$\frac{X}{Q} = \frac{1}{\overline{U}.2\pi.\sigma_y.\sigma_z} \exp\left(\frac{-y^2}{2\sigma_y^2}\right) . \left[\exp\left(\frac{(z+h)^2}{2\sigma_z^2}\right) + \exp\left(\frac{(z-h)^2}{2\sigma_z^2}\right)\right] \quad (1)$$

This expression is used in the calculations, related to accidents, taking into account that for cases in which the ventilation system is out of service, the equation is a derivation of that one, considering release (discharge) at ground level through the building penetrations and the openings in the walls.

Discharges through the vent pipe

Considering equation [1] and accounting that the diffusion is in the predominating wind direction and that the critical population member is at ground level, the equation that should be used for relative concentration in the central line of the plume to be discharged through the vent tube, is as follows:

$$\frac{X}{Q} = \frac{1}{\overline{U}.\pi.\sigma_y.\sigma_z} \exp\left(\frac{-(h)^2}{2\sigma_z^2}\right) \quad (2)$$

where:

X /Q = atmospheric dispersion factor (sec/m^3).

$\overline{U}$ = wind speed, considering the conditions at the discharge height in m/s,

h = effective height of the vent pipe in m. h = hs - ht;

hs = the initial plume height (in this case - the vent pipe height) above power station level, in m

ht = maximum ground height above power station level (assumed is ht = 0).

σy = lateral spread of plume (m). Is is a function of the atmospheric stability and its distance is calculated with the following expression (Ref. [28]):

$$\sigma_y = \exp\left(sy_0 + sy_1 Ln(x) + sy_2 Ln(x)^2\right) \quad (3)$$

σz = vertical spread of the plume (m). It is a function of the atmospheric stability and the distance, and is calculated with the following expression (Ref. [28]):

$$\sigma_z = \exp\left(sz_0 + sz_1 Ln(x) + sz_2 Ln(x)^2\right) \quad (4)$$

According Ref. [58], the empirical factors for the estimation of the lateral and vertical spread of the plume are the following:

Discharges at ground level

The dispersion factors (X/Q) are calculated using the following expression, which is applicable for discharges through building openings, when the height at the discharge point is 2.5 times smaller than the height of the neighboring solid structures.

The expression is derived from equation [1], considering, that the diffusion is at the predominant wind direction and that this is a discharge at ground level.

$$\frac{X}{Q} = \frac{1}{\overline{U}.\pi.\sigma_y.\sigma_z} \quad [5]$$

The parameters in these equations are the following:

X /Q = atmospheric dispersion factor (s/m3).

$\overline{U}$ = wind speed at 10 meters height (m/s).

$\sigma_y, \sigma_z =$ lateral and vertical spread of the flow (m) according the definition for discharge through the vent tube.

The values of the X/Q factor are calculated for two distances: the distances to the boundary of the NPP Site Boundary (SB), and to the outer boundary of the Precautionary Action Zone (PAZ). The distance to the SB and to the outer PAZ boundary for the calculation of X/Q will be defined as the minimum distance from the vent pipe to each sector.

The minimum distance from AB-2 to the precautionary action zone is 2200 m, based on the choice of Kozloduy as a settlement center, it is determined, that the PAZ is at 2.2 km from AB–2 (Table 2.3 of SAR of NPP Kozloduy).

When the discharge is through the vent pipe, the maximum concentration at ground level can occur outside the site boundary or outside the distance to the outer PAZ boundary and depends on the vent pipe height.

Hence, the X/Q factors are calculated for different distances beyond the site boundaries, in order to determine the distance to the maximum ground level concentration. The critical distance for the releases (discharges) from the vent pipe is 5500 m.

After that the doses for the critical distance and for the distance to the outer boundary of the precautionary action zone are evaluated.

For releases at ground level the atmospheric dispersion is calculated for the minimum distance from the closest point of the building to the SB for each sector and for the distance to the outer boundary of the precautionary actions zone (in western direction).

Brief description of the computer codes applied in the accident modes analyses

<u>Method for evaluation of the dose intake as a result of external exposure – Microshield 6.20</u>

The main purpose of Microshield 6.20 code is the calculation of the protection in case of gamma radioactivity and assessment of the dose burden. It is used for designing of shieldings, evaluation of the source term, minimizing the exposure of people etc. The description of the program is presented in [58].

The main characteristics of the program are:

- 16 geometries that include 10 standard protections.
- A library with data (radionuclides, attenuation, build-up and dose transformations) containing standard data from RSICC, ANS and ICRP.
- Simultaneous calculation of the build-up and the non collimated results;
- A possibility to be designed and kept to the 12th embedded materials down to 8 materials for a random case;
- Creation, keeping and displacement of sources between the various cases, such as nuclides or energies, as well as concentrations or as a whole. Some

methods for accumulation of photons, including definition of a user's method;

- Calculation of the decay of a source with taking into account the generation of the daughter fission products;
- Usage of 25 energy groups (energy range 15 keV to 10 MeV).
- Calculation of the residual heat removal
- A possibility for definition of 6 different points for recording the doses in all geometries, incorporated in the code;
- A possibility to work with several cases at one and the same time. Several etalon tasks are presented in [5], which purpose is to prove the adequate performance of the program. The etalon tasks are defined by:
- – American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants (ANSI/ANS 6.6.1-1979).
- - American Nuclear Society;
- – ESIS (1981). Specification for Gamma Ray Shielding Benchmark Applicable to a Nuclear Radioactive Waste Facility. European Shielding Information Service.

The obtained results are compared with the results, received by various computer codes (KAPV; ANISN; QAD; DOT; COHORT II; QADMOD; SKREEN). The comparison shows a good coincidence of the results.

<u>Method for the evaluation of the dose intake, caused by gamma exposure – VISIPLAN code</u>

VISIPLAN 3D ALARA [58] is the code for evaluation of the dose intake caused by gamma radiation and it is also used for planning of activities related to radioactive sources causing gamma exposure.

VISIPLAN 3D ALARA Code allows:

- − To map the dose rate iso-lines;
- − To get the individual doses, related to definite activities, i.e. as a function of the walk routes and the time of stay of the personnel at definite places;
- − To get the respective collective doses,
- − To compare the individual and the collective doses, related to various scenarios , i.e. to various operational procedures;
- − To perform the calculations for delectability, related to various radioactive sources, dependent for instance on decontamination, installation of shielding etc.

The mathematical model of the VISIPLAN is based on point-kernel calculations, which take into account the increase of the flux during the flux penetration during various materials. The photon flux at a point for dose calculations is located at a distance "r" from the point source and it is evaluated by the following formula:

$$\Phi = \frac{S.B.e^{-x}}{4.\pi.r^2}$$

where

Φ - Flux, photons m-2s-1

S - Power of the source, photons/s

B - Accumulation factor, -

r - Distance from the source, considered to be a point, m

"x" is an infinite value, which is calculated by the formula $x = \sum \mu_i \tau_i$, were μ_i is the attenuation coefficient during the "i" material, while τ_i is the distance passed through the "i" material.

The transition to effective doses is based on the application of the dose transition factors described in ICRP 51 (1) for bodies having rotational forms.

The volumetric sources are treated as a superposition of point sources and is calculated with the equation:

$$\Phi = \int_V \frac{S.B.e^{-x}}{4.\pi.r^2} dV$$

In this case S is the power of the source in an unit volume [photons. $cm^{-3}.s^{-1}$]

$$\Phi = \sum_{i=1}^{N_S} \frac{S_{tot}}{N_S} \cdot \frac{B.e^{-x}}{4.\pi.r_i^2}$$

where

Ns is the extract amount whilst S_{tot} is the total power of the volumetric source.

The size of extract can be randomly set by the user.

Various calculations are done for the validation of the algorithms incorporated in VISIPLAN 3D ALARA. All they are based on comparisons of the received results with reference data from ANSI/ANS 6.6.1 (1979) "Calculations and Measurements of direct and scattered gamma radiation from LWR Nuclear Power Plants".

A comparison was also done with Task №1 of ESIS (1981),"Specification for gamma ray shielding benchmark applicable to a nuclear radwaste facility", Newsletter №37, European Shielding Information Service (ESIS) [58].

The validation includes and compares the results with other programs of similar application.

Method for the assessment of the dose burden during the dismantling and the fragmentation of the dismantled equipment

The effective dose of the workers performing the dismantling activities and the fragmentation of the dismantled equipment will be defined as a sum of the doses from external exposure, inhaling and ingestion. A method from the Argonne National

Laboratory (ANL) [58] was used. By the use of this method, the dose burden of the workers is evaluated as a sum of the doses from external and from internal exposure. The internal exposure is caused by inhaling and by ingestion. The doses are calculated by the following formulas:

Doses from external exposure

EDEi = Ci x T x Σsurfaces (D∞ x FGi), (1)

where:

EDE_i – received effective equivalent dose from i nuclide;

C_i – concentration of the I nuclide in the treated metal (Bq/cm^2 in case of surface contamination and Bq/g in case of volume contamination), accepted as = 1

T – exposure time for the respective surface [h];

D_∞ - dose factor for infinite plat source;

FGi – geometric factor for: r

Inhalation doses:

In case of surface contamination of the treated metal:

CEDEi = Ci x ER x IR x F x ED x DCFi x As x W/(V x k) (2)

In case of volume contamination of the treated metal:

CEDEi = Ci x F x IR x AD x W x ED x DCFi (3)

where:

$CEDE_i$ – effective equivalent dose received from ingestion of the “i" nuclide [Sv];

C_i – concentration of the I nuclide in the treated metal (Bq/cm^2 in case of surface contamination and Bq/g in case of volume contamination), accepted as = 1

ER – intensity of the emission (accepted as 1.10^{-6} 1/h);

IR – inhaled volume (accepted as 1.2 m^3/ h);

F – decay factor (dimensionless);

ED – exposure time [h];

DSF_i – conversion factor for doses caused by inhaling of the nuclide i [Sv/Bq]

As – source surface area [cm^2];

W – dust share in the air, caused by a contaminated source (infinite, accepted as 1);

V- volume of the room (in cm^3)

k – air recirculation frequency (accepted as 1/hour).

AD - Dust Concentration in the air (in g/m3);

Ingestion doses:

CEDEi = Ci x IG x F x ED x DCFi x W (4)

where:

$CEDE_i$ – effective equivalent dose received from ingestion of the “i" nuclide [Sv];

C_i – concentration of the “i” nuclide in the treated metal (Bq/cm^2 in case of surface contamination and Bq/g in case of volume contamination), accepted as = 1

Method for assessment of the dose intake by the population – COSYMA code

The COSYMA code is used for the evaluation of the dose intake by individuals who are at the site inside and outside the boundaries of the Radiation protection Area. It is developed by the Forschungszentrum, Karlsruhe in Germany and the Council of the National Radiation Protection of UK as a part of the project of the European Commission named MARIA (Methods for Assessment of the Radiation Impacts by Accident) [58].

The disperse model, used in the Code is a model where the radioactive plume is divided into segments. It allows to take into account at each hour the speed and the direction of the wind, the category of the atmospheric stability and the effect of the quantity of the precipitations on the disperse material.

COSYMA uses horizontal and vertical parameters of the scattering, which are a function of the atmospheric stability and there are two groups of values of the parameters – broken and not broken terrene.

The doses from the external gamma radiation, received from the materials in the radioactive plume are calculated by the multiplication of the integral concentration in the according to a preliminary calculated coefficient, where the form and the size of the plume is taken into account.

The doses from the external gamma radiation, received by the fall-out materials on the ground are calculated by multiplication of the quantity of the fall-out materials and the dose for a unit of the fall out material.

The dose received from the radioactive plume by inhaling is calculated on the grounds of the integral concentration of the radioactivity in the air, the rate of inhaling and the dose of a unit absorbed activity. The average values for the dose intake by inhaling is based on the inhalation model of ICRP and includes the last versions of the bio-kinetic models recommended in ICRP Publications №№ 56, 67 and 69.

COSYMA is a verified and validated [58] code and it is widely used in Europe (France, Germany, UK, Slovakia etc.) Asia and Australia.

1.14.1 Results from the analyses of the limiting initiating events during decommissioning of KNPP Units 1-2

Following the results from the analyses of various initiating events during decommissioning of KNPP Units 1-2, presented in SAR [58], **HEPA filter rupture is considered as a limiting accident**.

Description of the scenario

The rupture of a HEPA filter may be a result of too high pressure drop over the filter. The pressure drop over the filters is a monitored parameter, a signal for replacement of the filter is generated when $\Delta p \geq 1500$ Pa.

It is assumed that due to some reason the filter was not replaces (signal failure, operator error)

Initial and limiting conditions:

For the analysis of this scenario it is assumed that:

- - the rupture occurs at the end of the filter's service life, when the aerosols load is maximum;
- - all the activity, contained in the filter is released to the environment.

The analysis is performed with COSYMA code

Activity inventory on the HEPA filter

In a conservative approach, the activity inventory on the ruptured filter is assumed to be equal to that measured on the corresponding exhausted filter after its replacement during normal power operation.

Such an inventory will exceed, likely by at least one or two orders of magnitude, the activity that may accumulate during the SE operation, due to the higher specific activity of the filtered air streams during NPP power operation, as a result of circuits/equipment leakages, of aerosols generation during the refuelling outages due, for example, to maintenance and repair activities in contaminated environments (SGs collectors, handling of thermal insulation materials for welding inspection, decontamination/cleaning activities, etc.)

The total β - γ activity, measured on exhausted Petrianov filters during normal plant operation was found to vary between 0.4 to 7.4 MBq/filter (main isotopes: ^{54}Mn, ^{60}Co, ^{134}Cs and ^{137}Cs). The flammable Petrianov filters are no longer used at KNPP and have been replaced by non-flammable HEPA filters. These new filters have not been replaced since they have been put in service. Taking into account that operational characteristics of the former Petrianov and of the new HEPA filters are practically the same (efficiency, flow rate, Δp), the maximum activity measured on the exhausted former filters can be considered to define the source term for the considered accident (10 MBq).

The calculations are performed with COSYMA code which verification and validation is a fact [58] and it is widely in use in Europe (France. Germany, UK, Slovak Republic etc,), Asia and Australia.

Nuclide vector

It must be taken into account that, during the SE operation, the contribution of the short half life nuclides (^{54}Mn, ^{60}Co, ^{134}Cs) to the global activity inventory will progressively decrease in function of the time elapsed since the final shut-down of the reactor. On the other hand, as it can be seen from Table 1.14.1-1, ^{137}Cs is the nuclide characterised by the highest effective dose contamination conversion factor for adults. It is thus conservatively assumed that the activity inventory of the ruptured HEPA filter amounts to 10 MBq of ^{137}Cs.

The inventories of the radiologically significant nuclides which are not measured routinely (^{90}Sr, TRU isotopes) are derived by the application of the scaling factors of the primary coolant, taking into account a decay period of 40 years. Table 1.14.1-1 shows the activity inventory of the significant nuclides in the filter.

Table 1.14.1-1 Activity inventory of the radiologically significant nuclides in the filter.

Nuclide	Activity inventory in HEPA filter, Bq
^{137}Cs	**1.10+7**
^{90}Sr	**2.10+5**
^{238}Pu	**1.78.10+3**
^{239}Pu	**2.5.10+3**
^{240}Pu	**1.61.10+3**
^{241}Pu	**3.10+4**
^{241}Am	**7.14.10+3**
^{244}Cm	**5.10+2**

Atmospheric conditions

The calculations are done for the cases with or without rain during the discharge period. The intensity of the precipitations is accepted to be 6.52 mm/h, which complies with the average annual value of the precipitations in the region of Kozloduy NPP during the spring-summer season, specified in [58].

Table 1.14.1-2 gives wind velocity in reference of various categories of atmospheric stability

Table 1.14.1-2 Wind velocity in reference of various categories of atmospheric stability

Category of stability	**A**	**B**	**C**	**D**	**E**	**F**
Wind velocity at 10m average height	**0.9**	**1.3**	**1.7**	**2.0**	**1.2**	**0.4**

The results of the analyses performed based on various initiating events during decommissioning of units 1 and 2 are presented in SAR [58]. A HEPA filter rupture is considered as limiting accident and the results are summarized in the table below.

Table 1.14.1-3 Effective dose by inhalation to an individual of the critical group of the population resulting from the rupture of a HEPA filter

Nuclide i	**H_{eff},i (mSv)**
^{137}Cs	5.2 10-6
^{90}Sr	4.3 10-7
^{238}Pu	2.6 10-6
^{239}Pu	4.0 10-6
^{240}Pu	2.6 10-6
^{241}Pu	1.2 10-6
^{241}Am	9.2 10-6
^{244}Cm	3.8 10-7
H_{eff} (mSv)	**2.56 10-5**

Results show that the effective dose to an individual of the critical group of the population (adult) under the most conservative development of the relevant accidents

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

(taking into account the wind rose for the area of Kozloduy and the corresponding meteorological data respectively) amounts to 2.56 10-5 mSv and is much lower than the limits of acceptability, determined in the Regulation for radiation protection BNRP [16]. According to BNRP [16], the annual individual effective dose from internal and external exposure of the population in case of accidents is ≤ 5 mSv during the first year after a design basis accident (at the border of the radiation protection area and outside it).

Table 1.14.1-4 Individual Effective Dose [Sv] for the first 24 hours after the release, in case of rain during the release of the nuclides into the atmosphere

Distance km	Category of atmospheric stability A	Category of atmospheric stability B	Category of atmospheric stability C	Category of atmospheric stability D	Category of atmospheric stability E	Category of atmospheric stability F
0.1	2.48E-11	1.17E-11	0	0	0	0
0.2	1.38E-11	8.28E-12	5.18E-12	0	0	0
0.3	9.36E-12	5.62E-12	3.54E-12	2.15E-12	2.15E-12	0
0.4	6.95E-12	4.2-E-12	2.64E-12	1.61E-12	1.61.E-12	0
0.5	5.46E-12	3.36E-12	2.10E-12	1.28E-12	1.26E-12	3.68E-12
0.6	4.44E-12	2.78E-12	1.75E-12	1.06E-12	1.06E-12	3.10E-12
0.7	3.70E-12	2.36E-12	1.50E-12	9.07E-13	9.07E-13	2.60E-12
0.8	3.14E-12	2.04E-12	1.31E-12	7.93E-13	7.93E-13	2.22E-12
1.0	2.35E-12	1.58E-12	1.04E-12	6.34E-13	6.34E-13	1.70E-12

Results show that the maximum individual effective dose 24-hours upon the activity releaseto is at 100 m of the release source and amounts **2.84E-8mSv**, This dose is in reference of rainy weather and in good wheather the values.will be far below.

Table 1.14.1-5 Individual Effective Dose [Sv] for the first year after the release, in case of raining during the release of the nuclides into the atmosphere

Distance km	Category of atmospheric stability A	Category of atmospheric stability B	Category of atmospheric stability C	Category of atmospheric stability D	Category of atmospheric stability E	Category of atmospheric stability F
0.1	7.73E-09	3.18E-09	0	0	0	0
0.2	3.70E-09	2.25E-09	1.41E-09	0	0	0
0.3	2.48E-09	1.52E-09	9.60E-10	5.84E-10	5.84E-10	0
0.4	1.81E-09	1.12-E-09	7.13E-10	4.36E-10	4.36.E-10	0
0.5	1.41E-09	8.85E-10	5.65E-10	3.46E-10	3.46E-10	1.00E-09
0.6	1.14E-09	7.26E-10	4.67E-10	2.87E-10	2.87E-10	8.42E-10
0.7	9.49E-10	6.13E-10	3.97E-10	2.45E-10	2.45E-10	7.05E-10
0.8	8.04E-10	5.27E-10	3.44E-10	2.13E-10	2.13E-10	6.03E-10
1.0	6.03E-10	4.07E-10	3.70E-10	1.68E-10	1.68E-10	4.61E-10

Results show that the maximum individual effective dose in the most concervative accident scenario (given the wind rose of Kozloduy and respectively the meteorologic data) amounts to 7.73.10-6 mSv at a distance of 100 m from the source, i.e. far below the acceptance limits, defined in Regulation on the Basic Norms for Radiation Protection Norms (BNRP) [16].

1.14.2 Results from the analyses of the limiting initiating events during decommissioning of KNPP Units 3-4

Description of the methods and models used for the assessment

The selection method for analysis of the radiological consequences from a certain initiating event is defined by the nature of the event and the expected routes of exposure. Specialized computer codes, which take into account both the radiation discharge source and the spread of the contamination into the environment, are used in order to evaluate the dose intake by the population.

The methods used for the assessments of the radiological consequences from the implementation of the decommissioning activities both in normal and in emergency conditions, shall provide a possibility for the evaluation of the dose intake by the personnel during external and internal exposure caused by inhaling and ingestion. They have to be capable to take into account the state of the source (solid, liquid, gaseous), the impact of the environment (for example the dust concentration), the duration of the work, the use of protective equipment and other factors that impact the exposure.

The evaluation methods for the dose intake of the population shall contain appropriate models for the spread of the plume as well as they shall take into account the various possibilities for exposure (external and internal).

For the safety assessment during the implementation of decommissioning activities the following computer codes and models are used:

(1) Evaluation of the dose intake by the personnel caused by external exposure **(Microshield 6.20 Code)**

(2) Evaluation of the dose intake, caused by gamma exposure, the planning of the activities, related to radioactive sources causing gamma exposure (VISIPLAN 3D ALARA Code)

3) Models for the dose intake evaluation is a result of internal exposure in case of spills of radioactive liquids (internal exposure caused by inhaling and ingestion)

(4) Models for evaluation of the dose intake of the personnel during dismantling and fragmentation activities at the dismantled equipment (external exposure and internal exposure through inhaling and ingestion)

Description of the models and data for their verification and validation are presented in section 1.14.0.

COSYMA code is used for the evaluation of the dose intake of the population, caused by the discharge of radioactive noble gases through the ventilation stack. A brief description of the code is presented in section 1.14.0.

1.14.2.1 Analysis of the envelope initiating events

This section summarizes the analyses of the various initiating events.

Leak of liquid from a container with spent filters during SFP decontamination

Description of the scenario

The liquid leak from inside a container of spent filters from SFP decontamination is postulated as an initiating event.

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

The scenario is an envelope one concerning events of water spills from SFP, because the specific activity of the water in the container is significantly higher than this of the SFP water, while the quantity of the water (13 l) is much bigger than this, which could spill outside the SFP when an object drops inside it.

Filters used for SFP decontamination are collected underwater in a special container, which is steady fixed in the SFP under a water layer, which depth is not less than 1 m. After the loading of the container, the filters are transported underwater in a specially shielded transport container. The container is closed by a shielded cap, lifted by a crane, drained and transferred outside the SFP for transportation. Four filters are placed in the transport container. Prior to the transportation, the container is drained above the SFP by a special draining line on which an isolating valve is installed.

The operations for placing and fixing the cap, as well as for opening and closing of the draining line are done manually by personnel, who are trained for the purpose. It is assumed that due to an operator's error, the container is transported outside the SFP prior to be completely drained and with not closed draining valve.

The calculation of the received dose from external exposure is done by Microshield code, by the method that is specified in Section 6.4 and described in Attachment 6-1.

<u>Initial and limiting conditions</u>

The analysis is done upon the assumption for the specific activity of the sludge, which is typical for SFPs 1 and 2. After the date for Units 3 and 4 will be received, the assessment will be updated.

- It is accepted that the specific activity of the sludge at the bottom of the SFP is **3E+6 Bq/g** and that it is due mainly to ^{60}Co;

- Volume of the water in the container: It is assumed that the container was transported without to be drained, i.e. the volume of the water in it is maximal and equal to 13 l (the volume was calculated on the grounds of drawings of the container for transportation of filter cartridges, presented in [58]).

- Specific activity of the filter cartridge: In [58] is specified that the filter cartridge can undertake a maximum quantity of 1 kg sludge from the SFP and from here it comes out that the maximal activity in one filter cartridge is **3E+9 Bq**, mainly due to ^{60}C.

- One container can store 4 filter cartridges, and it becomes clear that the maximal activity of the filters in the container is equal **to 1.2E+10 Bq**;

- For calculation purposes we assume conservatively that the activity of the filter cartridge is evenly distributed in the water inside the container. The specific activity of the water is equal to **9.23E+5 Bq/g**.

- In order to define a more precise geometry of the source after water leak from the container, it is assumed that floor where the leak stays is not drained. The source is defined as a cylinder having a 2 m radius and 1 mm thickness;

- The dose from the external exposure is calculated for four points: In the centre and at the periphery of the surface and in the same two points at 1.5 m height from the surface.

Main Assumptions

The analysis of the event for evaluation of the dose intake from internal exposure (ingestion or inhaling) is done under the following assumptions:

- DCFi - dose transformation coefficient in case of inhaling of ^{60}Co- is 3.4E-09 Sv/Bq according to [58]

- BR – the inhaled volume is equal to 1.2 m^3/h as per [58];

- ARR – evaporation rate is equal to 4E-7/h according to section 3.2.4.5 of [58];

- IR – ingestion rate of the dust is equal to 2.08E-3 g/h, recommended in [58];

- T- time of exposure – it is accepted the dose rate to be defined for 1 hour;

- DU – dust concentration is equal to 6.25E-4 g/m^3, defined in [58].

Main results

The main calculated occupational dose is 5.634 mSv/h. Radiological consequences outside the Reactor building are not expected. No consequences for the population are expected.

Hanging on of a load above the SFP due to loss of power supply of the crane

Description of the scenario

The loss of power supply of the crane during the removal of a rack from the SFP is postulated as an initiating event.

According to the procedure for racks decontamination, the activities for their washing are performed underwater, while they are transportated after the measurements. The analysis is done under the assumption that as a result of an operator's error, the rack was picked-up prior to be decontaminated.

Input Data

During removal of racks from the SFP of Units 3 and 4 , the recorded dose rate at the surface of the rack was 3 mSv/h.

Main Results

Upon the postulated assumptions, the annual admissible occupational dose could be exceeded if the personnel stays for about 16 hours close to the rack.

The dose rate is a controlled parameter. The activity is performed in the Reactor Hall, which is a semi-serviced premise. In case of loss of power supply and exceeding the dose rate, the workers have to leave the premise. The further actions are according to the emergency instructions.

After the power supply is resumed, the operator's error can be eliminated for very short time as the improperly lifted up rack will be lowered down for underwater decontamination.

Rupture of aerosol filter

Description of the scenario

Aerosol filters of the type HEPA-23cl made of non-combustible filtration material, manufactured by the German company Binzer are used in the ventilation system B-2. The purpose of the filters is to purify the sucked air from solid or liquid aerosol, which particulates diameter is of the order of 0.1 μm. The level of particulates retaining that have greater penetration capability through the filtration material (it is conditionally accepted that the particulates are spherical and have 0.15÷0.20 μm diameters) is not less than 99,97 % numerical.

The rupture of an aerosol filter can be caused by a too big pressure drop through the filter. The pressure drop through the filter is a controlled parameter and at $\Delta p \geq 1500$ Pa an alarm signal is generated to replace the filter. It is assumed that for some reason, the filter has not been changed (false signalling, operator's error).

For this scenario analysis, it is assumed that:

- The rupture happens at the end of the operational life time of the filter, when the aerosol load is maximal.

- The whole activity contained on the filter is discharged into the environment.

The analysis is done by the use of the COSYMA code under the following assumptions:

Input data and assumptions for the analysis

Under a conservative approach, the activity of the ruptured filter is assumed to be equal to the one that is measured in the relevant spent filter after its replacement during the operation at power.

Such a quantity, most probably will exceed by one or two orders the activity, which could be accumulated during the SE Operation, because of the higher specific activity of the flow of the filtrated air when NPP runs at power, when leaks are possible from the systems/equipment, creation of aerosols during the shut-down for refuelling, due for instance to maintenance activities and repairs in radioactively contaminated environment (SG collectors, handling of insulation materials during the inspections of the welds, decontamination/cleaning activities etc.).

The total β - γ activity, measured in the spent Petrianov filters during the normal operation of WWER-440 units, varies from 0.4 to 7.4 MBq (main isotopes ^{54}Mn, ^{60}Co, ^{134}Cs and ^{137}Cs).

Because of a lack of data for the activity of the filters used in the period after the final shut-down of the Unit (changes were not done), it is anticipated that the maximal activity measured in the previous spent filters defines the radioactive discharge for the concerned accident. Considering the indefiniteness, 10 MBq are conservatively assumed.

The quantities of the radiologically significant radionuclides (^{90}Sr, TRU isotopes), which are not measured on regular basis are received after using the scaling factors for the Primary Circuit coolant, taking into account the 40 years half-life].

Table 1.14.2.1-1 presents the activity inventory of the radiologically significant nuclides.

Table 1.14.2.1-1 Activity inventory of the radiologically significant nuclides

Nuclide	**Activity inventory in HEPA filter, Bq**
^{137}Cs	1.10+7
^{90}Sr	2.10+5
^{238}Pu	1.78.10+3
^{239}Pu	2.5.10+3
^{240}Pu	1.61.10+3
^{241}Pu	3.10+4
^{241}Am	7.14.10+3
^{244}Cm	5.10+2

The calculations are done for the cases with or without rain during the discharge period. Theintensity of the precipitations is accepted to be 6.52 mm/h, which complies with the averageannual value of the precipitations in the region of Kozloduy NPP during the spring-summerseason, specified in [58].

The recommendations from [58] done by the German regulator for the wind velocity and for the various categories of atmospheric stability are presented in table 1.14.2.1-2.

Table 1.14.2.1-2 Wind velocity in reference of various categories of atmospheric stability

Category of stability	**A**	**B**	**C**	**D**	**E**	**F**
Wind velocity at 10m average height	**0.9**	**1.3**	**1.7**	**2.0**	**1.2**	**0.4**

Results of the analysis

The results of the analyses performed based on various initiating events during decommissioning of Units 1 and 2 are presented in SAR [58]. As limiting accident is considered HEPA filter rupture and the results are summarized in the table bellows.

Drop of aerosol filter during its replacement

The initiating event is defined in case a spent aerosol filter is dropped during its change and caused by an operator's error.

Description of the scenario:

The spent filter of HEPA-23 type is made of nonflammable, manufactured by the German company Binzer, which are 610 x 610 x 292 mm size cartridges. The weight of the new cartridge is 18 kg.

If a filtrating cartridge drops, a damage of the structure is possible and release of a large amount of dust.

Input data and assumptions for the analysis

It is assumed that the dust concentration around the filter is increased twice above the admissible value for mixed dust according to [58] and reaches **1.25E-3 g/m3** .

The activity accumulated in the filter is accepted as a maximal one, according to the data presented in Table 1.14.2.1-3 is assumed that when it is dropping, a half of the content of the filter is spilt.

The analysis of the event for evaluation of the received dose from external exposure is done by **Microshield** code. The source is defined as a point source. The dose from external exposure is calculated for 50cm distance from the source under the following assumptions:

- BR – the inhaled volume is equal to 1.2 m^3/h (recommended in [58]);
- ARR – it is assumed that all the activity released from the filter is spread in the air and is possible to be inhaled, i.e. ARR=1;
- IR – ingestion rate of the dust is equal to 2.08E-3 g/h, recommended in [58];
- T- time of exposure – it is accepted the dose rate to be defined for 1 hour.
- DU – dust concentration is equal to 1.25E-3 g/m^3. It is assumed that the dust concentration around the filter exceeds twice the admissible limit for dust mixture, as per [58].

The dose conversion factors DCFi in case of inhaling and in case of ingestion are according to ICRP-51 [58] and these are presented in Table 1.14.2.1-3.

Table 1.14.2.1-3 Dose conversion factors

Isotope	Conversion factor, Sv/Bq	
	Inhalation	Ingestion
^{137}Cs	9.7E-9	1.3 E-8
^{90}Sr	3.6E-8	3.6 E-8
^{238}Pu	4.6E-5	2.3 E-7
^{239}Pu	5.0E-5	2.5 E-7
^{240}Pu	5.0E-5	2.5 E-7
^{241}Pu	9.0E-7	4.8 E-7
^{241}Am	4.2E-5	2.7 E-7
^{244}Cm	2.7E-5	1.2 E-7

<u>Results from the analysis</u>

The resultant dose intake of the personnel is evaluable as a sum of the doses from external and internal exposure:

- Dose intake as a result of ingestion 146 mSv/h
- Dose intake as a result of inhaling 0.557 mSv/h
- Dose intake as a result of external exposure 1.81E-03 mSv/h

The total dose burden is evaluated to 0.705 mSv/h.

This means that one worker without protective equipment can work up to 70 hours by the time of cleaning the place of the accident prior to exceed the criteria, defined in the national legislation.

In order to avoid the unnecessary exposure during incidents related to handling of spent filters is enough to follow the safety instructions and the operations shall be done by the use of personal protective equipment.

1.14.2.2 Summary of the results

The summarized results from the analyses of various initiating events during decommissioning of KNPP Units 3-4, presented in SAR [58], are shown in table 1.14.2.2-1

Table 1.14.2.2-1Results from the analyses of the limiting initiating events during decommissioning of KNPP Units 3-4

Initiating event	Activity	Main results
Splash of fluid from the container with spent filters during their transportation from SFSP	SFSP decontamination	The maximum calculated dose for the personnel is 5.634 mSv/h. Radiological consequences outside of the reactor hall are not expected. There are no consequences for the population.
Dropping of a load during change of filters from SFSP due to crane electrical power breakdown	SFSP decontamination	The expected maximum dose on the surface of the racks is 3 mSv/h. The time of stay near the racks is limited to several minutes. Radiological consequences outside of the reactor hall are not expected. There are no consequences for the population.
Rupture of aerosol filter with release of activity into the environment	Operation of ventilation system B-2	This event is related with release of activity outside of the units. CED for member of the population is 7.73E-9 Sv/year at 100m from the source 1 6.03E-10 Sv at 1 km from the source.
Mistake during replacement of an aerosol	Operation of ventilation system B-2	Occupational dose for the personnel 0.705 mSv/h. Time for cleanup must

filter leading to falling-down of a cassette filter and spilling out of part of its contents		not exceed 1 hour. Radiological consequences are limited in the place of the accident. There are no consequences for the population.

Conclusion

The impact of the processes and activities of decommissioning of KNPP Units 1-4 on the health of the population will be insignificant, because the hazardous and radioactive materials will not leave the territory of the site of the Units (as during planned shutdown of the units in the period of their normal operation). The main impact from NPP on the population results from the nuclear fuel during normal operation of the plant. Before start of the decommissioning activities for the Units the NF is removed from the units and is transported to the SNF storage sites. Therefore, in this case measures for mitigating of the impact on the health of the population are not necessary.

The results from the performed analyses show that even under the most conservative development of the relevant accidents (taking into account the wind rose for the area of NPP Kozloduy and the corresponding meteorological data respectively [53]) the dose rate for the personnel and the population is much lower than the limiting values, determined in the Regulation for radiation protection BNRP [16].

Therefore, the impact of KNPP Units 1-4 decommissioning activities on the population will be insignificant even in case of a possible serious accident.

It can be concluded that the dose rate for the population in the 30-km area of KNPP, including the Romanian territories, in cases of limiting accidents during the IP implementation will be many times lower than the requirements of the International documents ICPR 103 and of the BNRP.

1.15 Measures for prevention and reaction in cases of incidents and unexpected events

The decommissioning activities are planned based on the results from the analysis of the accidents in the current Emergency plan on the KNPP territory [132]. The emergency plan provides technical and human resources, as well as instructions and procedures, for actions in case of emergencies occurred during operation as well as during decommissioning according to [12] and [27]. The emergency plan defines the organization of the activation of KNPP emergency structures; implementation of urgent measures to protect the personnel, population and environment, as well as measures for protection of the facilities on KNPP site in case of an accident or other critical events; and the interaction with the executive authorities.

The manner of informing and communicating with the Romanian authorities, as well as the cooperation in relation with the necessary measures should be improved.

Nevertheless, based on the results from the performed analyses of the limiting initial events during decommissioning of KNPP Units 1-4 and in order to minimize the dose rate for the personnel and the population the following recommendations can be made:

- Providing of ventilation for limiting the accumulation of radioactive gases and aerosols and for maintaining of a lower dust concentration.
- Provision of Personal Protective Equipment;
- Increase of the number of personnel for performing certain activities in order to reduce the time of stay near radioactive equipment
- Preparation of exact and clear instructions for each activity, related to radioactive materials;
- Use of modern and reliable systems for control of the radiation conditions in the rooms and buildings.

The assessment that was done is subject to updating in case of definition of new activities and processes during decommissioning.

1.16 Monitoring

This section presents the KNPP existing in-house radiation and non-radiation monitoring of the environment, as well as the non-radiation monitoring of SE RAW and the proposals for additional monitoring of the environment during execution of the decommissioning activities for KNPP Units 1-4.

1.16.1 Non-radiation monitoring of the environment

The current non-radiation in-house monitoring at Kozloduy NPP and SE RAW will continue during the execution of the decommissioning activities on Units 1-4 of Kozloduy NPP. Additional non-radiation monitoring related to some of the environmental components is proposed to be performed during decommissioning.

Air monitoring

At KNPP non-radiation monitoring of the emissions into air is currently not performed, only greenhouse gases (GHG) (CO_2 emissions) released from KNPP Diesel Generation Stations (DGS) are monitored. This monitoring is performed according to a Monitoring Plan which is inseparable part of the GHG Emissions Permit No143/2008. Additional conventional monitoring of the air emissions related to the decommissioning process is not planned. Additional monitoring measure could be necessary for the new facilities such as the Plasma Melting Facility and the Heat Generation Plant and this will be estimated in the scope of the EIA concerning these new projects.

Harmful physical factors monitoring

Currently at KNPP non-ionizing radiation, vibration and noise are not monitored. Concerning exposure to non-radiation harmful physical factors, the use of effective means of collective and personal protection should be monitored, as this will minimize the adverse effect on health.

Non-ionizing radiation

During the decommissioning activities non-ionizing radiation is not expected and therefore monitoring activities are not foreseen.

Vibrations

Single measurement should be executed in case of activities with possible vibration e.g. erection of new buildings and reconstruction of existing building.

Noise

Single measurement of the present state and later during decommissioning according to MoEW procedure should be executed if required by the authority.

Soil monitoring

At KNPP non-radiation monitoring of soils is not currently performed.

It should be considered whether the current soil's radiation monitoring network could be used for the determination of additional non-radiological parameters of the soil samples (e.g. acidity, content of heavy metal, ion exchange capacity, and pH analysis). The same parameters shall be monitored again one year after the completion of the decommissioning and if there is no identification of proven impact this monitoring can be repeated every 5 years or if judged opportune – every 10 years.

Water monitoring

In Kozloduy NPP a Program for non-radiation in-house monitoring [127] was established and is functioning. In the scope of this Program are monitored the quantitative and qualitative characteristics of the:

- Waste water from operation discharged by CH-1 and CH-2 into Danube river;
- Waste water (from operation and sanitary sewage water) and rainfall water, discharged in the Main drainage channel (MDC) of Kozloduy drainage system.

The scope and frequency of the monitoring are stated in the permits issued to KNPP as follows:

- Permit for use of water site for discharge of waste water in surface water body N 03120003/15.12.2007 (discharge of HC 1 and HC 2 into Danube river);
- Permit for use of water site for discharge of waste water in surface water body N 1375/20.04.2007 (discharge in Main drainage channel) with amendment and extension of validity according Decision No 216/25.02.2010/25.02.2010.

Non-radiation in-house monitoring is performed also for the groundwater at KNPP site and the adjacent territory and at the site of the Repository for conventional municipal and industrial waste (RCMIW), (Program Id.UOS.PM.013) [127]. This program covers also waste water at RCMIW site [159].

The scope and frequency of the monitoring are stated in Decision 05-DO-72-00 dated 24.01.2006 on waste management activities (amended by Decision 05-DO-72-01 dated 24.01.2006). The monitoring carried out according to the above mentioned programs is performed by authorized laboratories.

The following additional activities are recommended concerning water monitoring:

- Update of the Program for non-radiation in-house monitoring with relevant adaptation to the decommissioning activities including incorporation of new monitoring stations in the existing monitoring network corresponding to the new activities locations;
- Leakage control within the areas of collection and storage of wastes generated during dismantling works.
- After completion of the planned additional non-radiation monitoring of KNPP waters, additional monitoring of surface and ground water is not necessary.

The planned construction of the Site for conventional waste during decommissioning will required additional water monitoring according to the national legislation requirements. Requirements for additional water monitoring should be part of project implementation permission.

Waste monitoring

Non-radioactive (conventional) waste management procedure is in compliance with the Non-radioactive (conventional) waste management program [151]. Concerning the activities related to Waste management, KNPP holds a permit, issued by RIoEW – Vratsa. As indicated above, the non-radiation monitoring of waste water and groundwater, related to conventional waste management, is performed according the Program for non-radiation in-house monitoring Id.UOS.PM.013 [127].

The following additional activities related to the waste monitoring are recommended in compliance with the Non-radioactive (conventional) waste management program [151]:

- Monitoring of the non-radioactive (conventional) waste in compliance with the Action plan to the Non-radioactive (conventional) waste management program [151] appropriately updated and adapted to the decommissioning activities;
- Spent oils monitoring according the Program for collection, temporary storage and transportation of spent oils and their ultimate disposal;
- Monitoring, inventory and record of the quantity and location of asbestos containing heat insulation and construction materials, according to the internal procedure for safe dismantling;
- Continuous control, inventory and record of the quantity and location of hazardous waste as contaminated synthetic fibers, mercury contained in some tools and instruments, oil/gas, plumb coating and other hazardous chemical substances and materials related to the storage batteries premises etc.
- Continuous control, inventory and record of the quantity and location of dismantling wastes in view of stimulation for their recycling and reuse;
- Non-radiation monitoring of the temporary storage areas for containers of waste generated as a result of equipment dismantling;
- Monitoring of the accumulated chemical substances and chemicals as well as of waste packing of discarded chemical substances and chemicals.

Flora monitoring

At KNPP non-radiation monitoring of the flora is not currently performed. It is recommended to use the results from sampling analysis within the ongoing radiation monitoring analysis during the implementation of the Investment Proposal for decommissioning of Units 1 to 4 at Kozloduy NPP, for definition of some non-radiological parameters such as heavy metal contents in plant samples from the 3-km area.

Fauna monitoring

At KNPP non-radiation monitoring of the flora is not currently performed. During KNPP Units 1 to 4 decommissioning non-radiation monitoring of the fauna is not planned.

Protected Territories and Protected Areas monitoring

During KNPP Units 1 to 4 decommissioning non-radiation monitoring of the Protected Territories located at different distance form KNPP site where Units 1 to 4 are located is not planned. The reason for this is justified by the following:

- The considerable remoteness of the Protected Territories from the KNPP site.
- The presence of data from performed surveys on the environmental pollution which shows that the environmental radiological status of the habitats is not influenced by the KNPP operation;
- With regard to the adopted technology for decommissioning of Units 1 to 4 and provided that the regulatory requirements on waste management are met as well as the permissible emissions levels are not exceeded, potential impact on this component is not expected.

Protected areas

Permanent environmental monitoring of natural habitats, subject of conservation in the Protected Area and of the plant species is not envisaged. However, for conservation of the Protected Area the prescriptions given in the Compatibility Assessment Report should be observed.

1.16.2 Radiation monitoring of the environment (radioecological monitoring

The current radiation monitoring at Kozloduy NPP, Program UB.MOS.PM.262/03 [150] will continue to be applied during KNPP Units 1-4 decommissioning activities implementation. Description of the existing radiation monitoring is included in section 1.1 of this chapter. Additional monitoring related to some of the environmental components is proposed to be performed during decommissioning.

Air radiation monitoring

KNPP possesses well engineered air radiation monitoring systems on site and in the 100-km area. This system has a capacity to provides reliable information on the atmospheric particulates and aerosols in the region, during the construction and operation of the new facilities, during the decommissioning process of the Units 1 to 4 and afterwards.

Additional air radiation monitoring is not planned and only an extension and optimization of the existing system for control can be foreseen under Project 10. Independently of this consideration and depending of the adopted building technics, the construction of the new Size Reduction and Decontamination Workshop (SRDW), as well of the Decay Storage Site for Transitional Waste (DSS) may require additional radiation monitoring.

The necessity of additional air radiation monitoring during construction and operation of the Plasma Melting Facility in AB-2 will be detailed within the scope of the respective EIA on this facility.

Harmful physical factors radiation monitoring

Kozloduy NPP had established and performs regular radiation and radiological in-house monitoring on-site and in the special statutory areas. It is regulated by the relevant programs, instructions, methodologies, procedures, rules.

The monitored parameters of the ionizing radiation concerning the environment are the radiation γ-background and in particular the γ- radiation.

In the Decommissioning Plan [36] and in compliance with the Regulation on safety during Decommissioning of Nuclear Facilities [12] as well as in compliance with the provisions of the Radiation Protection Concept containing Programs for Radiation Monitoring covering, as currently, measurements of the γ- radiation and aiming to achieve its reduction, monitoring of gaseous and liquid releases and monitoring of RAW and contaminated equipment treatment is foreseen.

For the control of the γ- radiation and the γ-background radiation on the industrial site, programs and instruction were set up, ensuring the non-exceeding of the regulatory thresholds for radiation and dose burden and dispersion of radiation pollutants.

The update of these programs and instruction is necessary in the course of the different decommissioning stages and shall guarantee at a minimum the current site status in terms of radiation and in the best case – decrease of the site radiation impact (if the site is considered as compact source of radiation and contamination) on the environment.

For this purpose and in consideration of the listed documents related to the monitoring, in line with the currently effective rules and thresholds, new ones shall be formulated and set up with regard to the :

- New facilities for RAW treatment and storage and the adjacent areas;
- New facilities for treatment of activated and irradiated equipment and the adjacent areas;
- New facilities for treatment of spent nuclear fuel and the adjacent areas;
- New facilities, corridors and techniques for transportation of RAW, spent fuel, activated and irradiated equipment etc.

If the regulatory limits in force were modified, it is recommended to keep or reduce when applicable the controlled and allowable threshold for γ-radiation equivalent dose rate in the respective areas on the site.

The monitoring of the environmental γ-background radiation beyond the industrial site (special-statutory areas) is carried out in compliance with the Environmental Radiation Monitoring Program at Kozloduy NPP, NoUB.MOC.PM.262/03 [150], approved by the competent authorities. The results from the measurements from the past three years give the basis for the forecast of the

Energy Institute JSC
20, Fr. Joliot – Courie Str.1113 Sofia, Bulgaria Tel:(+359 2) 963 45 76, Fax: (+359 2)963 40 38,
E- mail: office@eninbg.com

environmental impact (on the γ-radiation background) caused by the γ -radiation generated at NPP site during decommissioning. In Romania, the National Environmental Radioactivity Surveillance Network (NERSN) ensures the radioactivity monitoring of the influence area of Kozloduy NPP – Bulgaria, conducting under two Radiation monitoring programs monitoring of the radiation gamma background, For this purpose 4 laboratories are located and work in the 100-km area of the considered territory. They are called Surveillance Stations for Radioactivity Monitoring (SSRM): SSRM Bechet, SSRM Craiova, SSRM Drobeta Turnu Severin and SSRM Zimnicea, and also 13 automatic air gamma dose rate monitoring stations in Dolj county, in Mehedinti county and in Teleorman county. To reach some traceability and precision in the definition of the modification of the γ-radiation impact and its operational management, it is not opportune to introduce modification during decommissioning in the currently operating program and schemes for monitroing, except if supplements are needed in the cases when more precision or modification in the surveillance scope is requested.

For the post decommissioning period the radiation monitoring plan for the γ-radiation and the γ-radiation background at NPP site and in the environment shall endure an update in line with the new situation, transboundary aspects and the statements of the EIA decision concerning :

- Location of the γ-radiation sources on site;
- Capability of the biological protection for γ-radiation reduction;
- Permissible emissions level concerning presence of radioactive substances in environment.

Soil radiation monitoring

The current soil radiation monitoring shall continue to be performed during the decommissioning activities implementation, according the already adopted system related to the number of sampling points and indicators to be analyzed. Independently of this and based on the conservative principle in establishment of sampling network and when applicable, incorporation of new sampling points for the purposes of soils radiation monitoring, related to the location of the decommissioning activities, should be adopted. For instance, if there is occurrence of leakages of radioactive substances an additional monitoring could be required.

In case of proven absence of soils contamination the monitoring can continue to be performed with the adopted frequency.

Water radiation monitoring

A Program for radiation in-house monitoring [127] is established and is operating at KNPP for the control of the groundwater.

The current water radiation monitoring shall continue to be performed during implementation of the decommissioning activities in accordance with the current system in place. If the event of a leakage or effluent infiltration in waste storage and decontamination areas, additional monitoring may be required.

Waste radiation monitoring

A Program for radiation in-house monitoring [150] and a Complex program for RAW management at Kozloduy NPP [162] are established and function at KNPP.

The current radiation monitoring related to waste and the associated facilities and storage sites shall continue to be performed during implementation of the decommissioning activities in accordance with the current system in place and the following additional activities related to waste are recommended in consideration of the waste radiation monitoring:

- Radiation monitoring of the temporary storage area for waste containers from dismantling
- Continue the Radiation monitoring of the RCMIW site.

Flora radiation monitoring

The current flora radiation monitoring is sufficient in consideration of the decommissioning activities implementation and additional monitoring on flora is not foreseen.

Fauna radiation monitoring

The current fauna radiation monitoring is sufficient in consideration of the decommissioning activities implementation and additional monitoring on fauna is not foreseen.

Protected Territories and Protected Areas radiation monitoring

Radiological impact on Protected Territories and Protected Areas is not expected and for this reason it is not foreseen to provide radiation monitoring of the protected territories located at different distance from KNPP site, where are located Units 1-4.

Personnel and population health radiation monitoring

Radioactive aerosols are a real risk of internal exposure in the course of dismantling activities; therefore, the strict individual dosimetric control and compliance with regulatory requirements must continue.

The current occupational and public health radiation monitoring is sufficient in consideration of the decommissioning activities implementation and additional monitoring is not foreseen to be done.