4. PROPOSED PROJECT

Khmelnitsky Unit 2 (Figure 4.1) is a VVER-1000/320 type reactor with one turbo generator of 1000 MWe. The unit had a design elaborated in accordance with general rule OPB-73 and norms/standards which were in force at that time. During construction, the requirements of new standards, OPB-82 and OPB-88 were also taken into account. The design of the plant did not entirely comply with safety standards that were current in June 1996 and therefore Goskomatom and Khmelnitsky NPP decided to introduce a modernisation programme to upgrade the plant (Section 8).

The acronym WWER stands for Water Water Energetic Reactors, which are pressurised water reactors designed in the former Soviet Union. The reactor core is composed of fuel rods made of low enriched uranium dioxide, enclosed in zirconium cladding, in a hexagonal geometry. The steam generators are horizontal types. The VVER-1000 is the latest design of VVER's. It has four loops and four steam generators. The electrical output is 1000 MWe in a single turbine-generator set. It has a large dry containment. The emergency core cooling system is composed of three 100% trains, each one of them containing hydro-accumulators, and high pressure and low pressure injection pumps, ensuring high active high pressure and low pressure safety injection. The passive core flooding system consists of four hydro-accumulators.

This Section summarises the main elements of the proposed project as they relate to either occupational safety or environmental impacts.

4.1 The main heat transfer cycles

A typical steam-turbine power station, whether it is fuelled by nuclear or fossil fuel incorporates three principal heat transfer cycles.

- Heat derived from the fuel is used to boil water to produce high pressure steam. In an NPP, the plant that performs this function is called the 'nuclear steam supply system'.
- The high pressure steam is used to drive turbines, each of which drives an alternator that produces electricity. This plant is called the 'power conversion system'.
- The remaining energy in the steam is rejected through a 'cooling water system'.

4.2 Nuclear steam supply system

The nuclear steam supply system consists of the reactor, the reactor coolant system, and a number of auxiliary and safety systems (Figures 4.1 and 4.2).

ng pond

4 –Make up deaerator

5 –Sprinkler pump

7 –Boron injection pump

8 –Boron injection pump

9 –Diesel generator

13 –Sump-tank of emergency boron supply

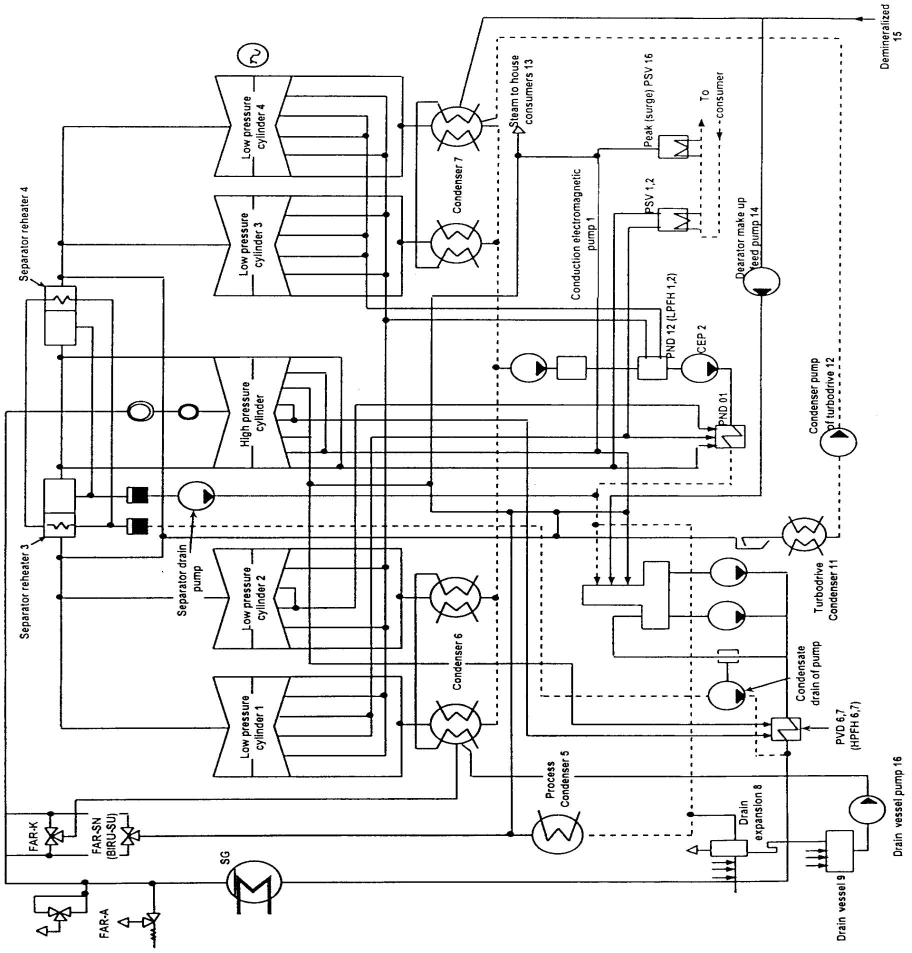
14 –Reactor coolant pump

15 –Steam generator

16 –ECCS passive system tank

rinkler extension piece

Figure 4.2
Schematic of secondary circuit



4.2.1 The reactor

The main features of the VVER are as follows.

- Heat is generated by the process of nuclear fission within the uranium dioxide fuel. Fuel elements are constructed by assembling pins (diameter 9.1 mm, height 3.56 m) of low enriched UO_2 sintered pellets in metallic cladding tubes. These fuel elements are placed in the core of the reactor vessel.
- The neutron moderator is light demineralized water with boron. Boron is incorporated to allow slow reactivity variation control, burn-up compensation and prevention of xenon poisoning.
- The reactor coolant is the same light water which passes through the core and removes the heat generated in the fuel pins from the cladding surface. The average linear power is about 166 W/cm.
- Control of the nuclear chain reaction is achieved by the movement of neutron-absorbing control rods (in boron carbide) and by varying the concentration of boric acid in the reactor coolant.

4.2.2 Reactor coolant system

The reactor vessel which houses the core is a steel pressure vessel. After removing heat from the fuel, the coolant enters the 4 main coolant loops. Each coolant loop includes a steam generator, a primary pump and the main circulating pipework (of diameter 850 mm).

The heated primary coolant enters the steam generator heat exchanging pipes. These pipes are surrounded by water from the lower pressure secondary heat transfer, which is itself heated and boils to steam. The primary coolant then returns to the core via the main coolant pump.

4.2.3 Controlling the power

To ensure this main safety function, means are provided to prevent unacceptable reactivity transients and to shut down the reactor as required. Anticipated occurrences are prevented from leading to accident conditions.

The reactivity control is performed by changing the boron concentration in the coolant and by actuating the reactor control and protection system which is used for both normal operation and emergency scram.

- The reactor protection actions are of three types:

  - reactor scram: there are rapid cut-off of the control rod power supply and dropping of the control rods by gravity in less than 4 seconds. Disappearance of the scram signals does not stop the dropping of the control rods;

  - emergency protection 1: in this case, there is reduction of the reactor power by gradual lowering of the control rods at a rate of 20 mm per second. This action ceases when the emergency protection 1 signal disappears;

- emergency protection 2: in this case, there is stabilisation of the reactor power by inhibiting or blocking control rod rise commands, lowering alone remaining possible. This action ceases when the related signal disappears.

- The role of the emergency protection 1 and 2 signals is to counter trends in certain parameters by setting thresholds slightly lower than those resulting in a scram in order to limit the causes of scram activation.

- Protection system output signals controlling scrams and protecting actions are generated by two out of three voting in the measurement channels in two redundant control tains.

- The scram signals are quite commensurate to those of western PWRs:

  - neutron flux criteria
  - primary coolant system criteria
  - steam generator criteria
  - electrical power supply criteria
  - external criteria

The most important system for controlling power during transients and accidents is the control and protection system. Furthermore, the chemical and volume control system and emergency core cooling system are activated respectively, depending on the transient and accident scenario to achieve shutdown conditions.

4.2.4 Chemical and volume control system

Chemical and volume control of the primary coolant is maintaining by bleeding flow from the reactor coolant system and processing it to provide continuous coolant quality. This is accomplished by reducing the corrosion and fission product content, and by injecting chemicals such as boric acid and corrosion inhibitors. Although this is not primarily a safety system, its role is important for maintaining safety and it is able to provide some compensation for minor unplanned leakage from the primary circuit.

4.2.5 Residual heat removal system

Residual heat removal systems are required to provide cooling of the fuel and primary coolant following reactor shutdown. Even though the nuclear chain reaction may be terminated, residual heat production continues due to radioactive decay of fission products. At low pressures, heat removal via the primary steam generators cannot be achieved readily so a separate system is used.

4.2.6 Emergency core cooling system

An emergency core cooling system (ECCS) (Figure 4.3) is provided on the VVER-1000 reactor to protect against the potential consequences of a loss of coolant accident (LOCA). Although such events are judged to have a very low probability of occurrence, the system is designed to replace sufficient of the lost primary coolant so as to prevent fuel temperature limits from being exceeded. This ECCS comprises the high pressure emergency core cooling system (HP ECCS), the low pressure emergency core cooling system (LP ECCS) and the system of water collectors which is a passive part of the whole ECCS system.

The HP ECCS is designed for insertion of borated water in the primary circuit under accidental conditions. It comprises emergency storage tanks of high concentration boron solution, an HP pump for boron insertion, pipework and valves. The three channels of that system are connected to the 'cold lines' of the primary coolant circuit.

The LP ECCS is designed for residual heat removal from the core. This system is composed of three independent channels, each of which includes an LP pump, a sump tank located in the confinement, and heat exchanger pipework and valves. Two channels are connected to the borated water supply lines of the water collectors system, ensuring supply of borated water to the upper and lower chambers of the reactor, the third channel is connected to 'hot' and 'cold' lines of the primary coolant loops.

The system of water collectors comprises four pressurised accumulation tanks, containing borated water, connected separate inlet nozzles across non-return valves to the reactor pressure vessel (to upper and lower chambers of the reactor). In an emergency event, should the primary circuit pressure fall below 5.9 MPa, borated water will flow without the need for any external energy supply from this passive low-pressure injection system into the reactor core.

4.2.7 Containment

The reactor building (see Figure 4.4) is provided with a prestressed concrete containment with a steel liner housing all equipment associated with the primary coolant circuit. This hermetic zone is designed to withstand an overpressure of 0.5 Mpa and temperatures (150°C for 24 hours) under accident conditions. Leak tightness has to be ensured i.e. the maximum leak rate shall not exceed 0.1% per day of the volume of air within the containment at maximum design basis accident pressure. A spray system is installed within the containment to provide means for relieving pressure and temperature rises following a loss of coolant accident (LOCA). The containment isolation system prevents any radioactive release into the environment by isolating all systems which penetrate the containment and which are not necessary to control the accident.

Figure 4.3
Schematic of emergency core cooling system

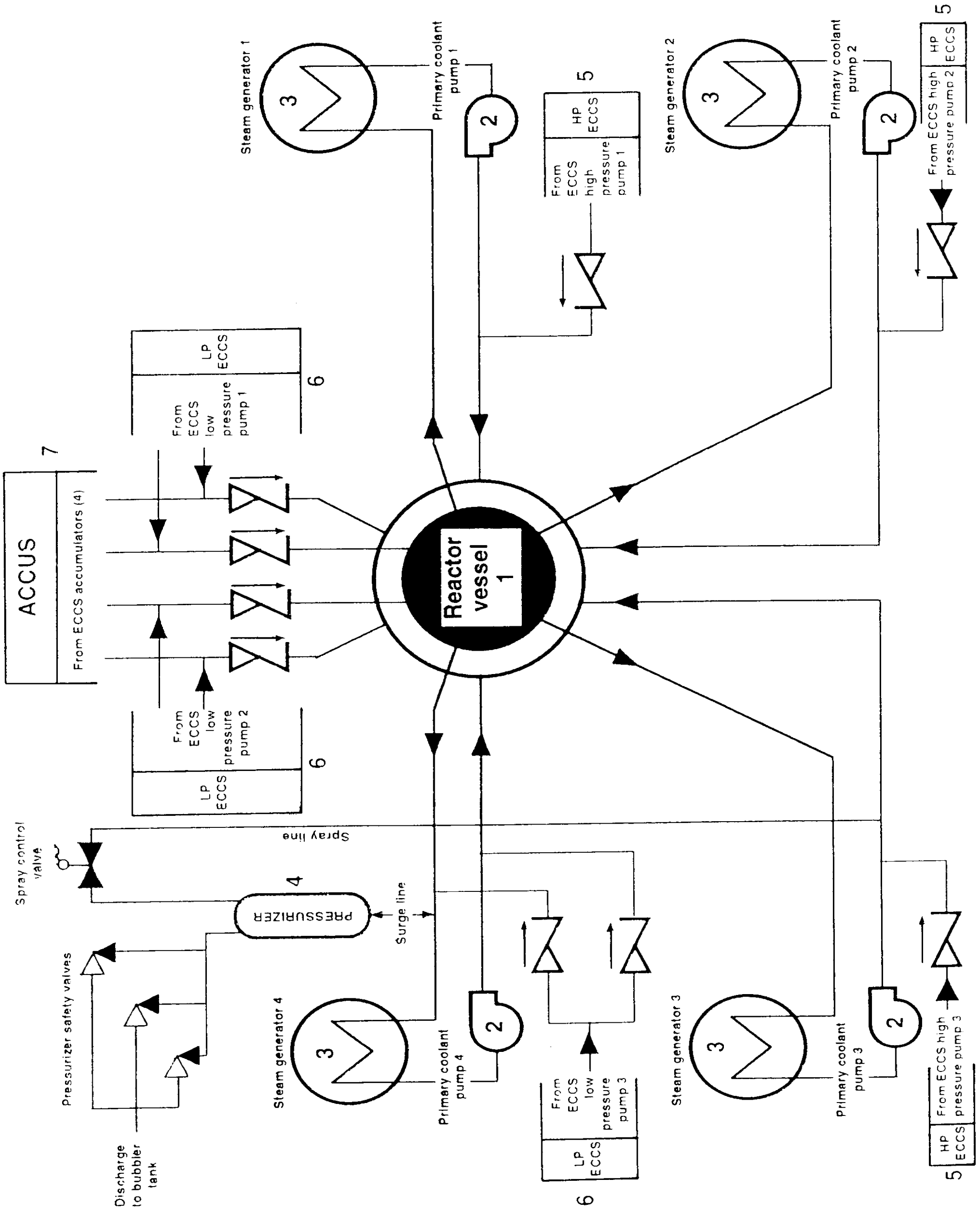
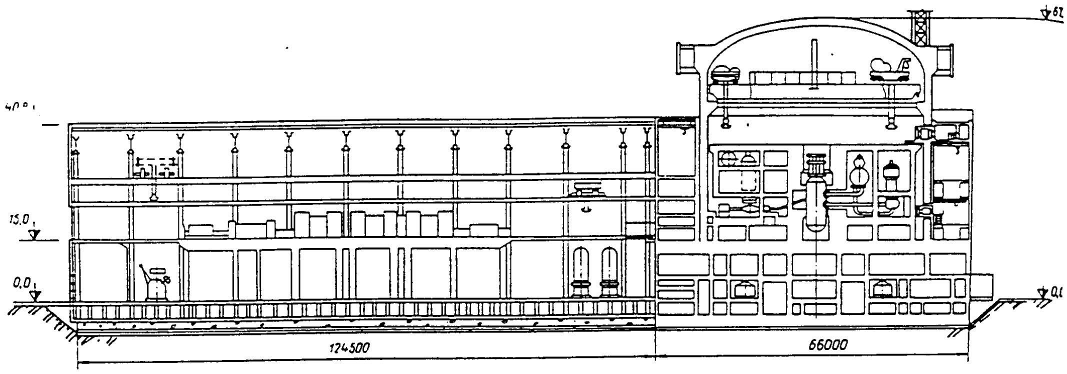
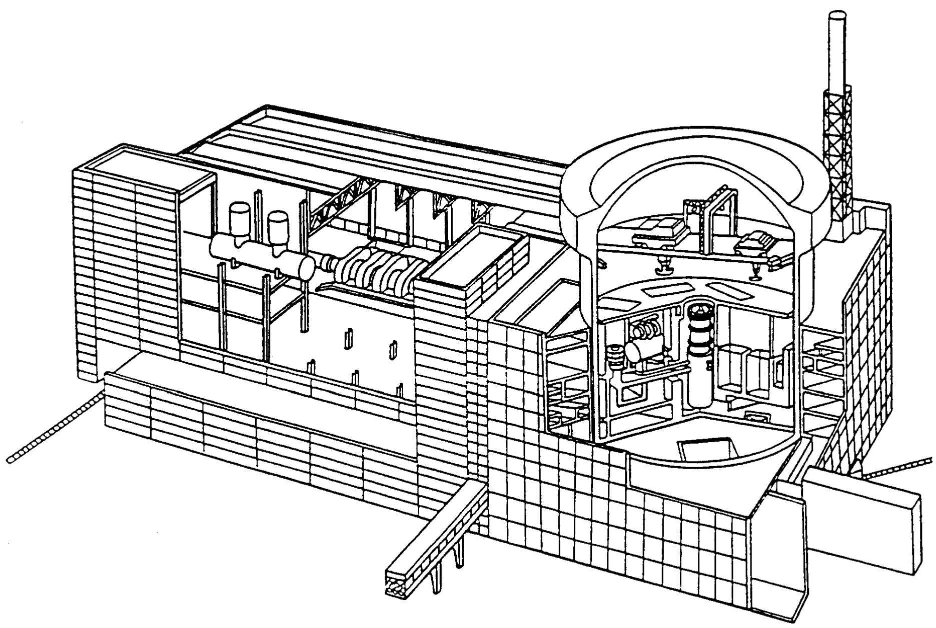


Figure 4.4
Reactor building





4.3 Power conversion system

The power conversion system consists of various water and steam systems and one steam turbine. Demineralised water (secondary system water) is pumped from the turbine condensers to the four steam generators, where it passes over tubes containing reactor coolant water (from the primary circuit). Heat transferred through the walls of the tubes causes the secondary system water to boil, producing steam at pressure of about 6.4 MPa. This steam is then collected in a common main steam header and passes via pipelines into the turbine, where it gives up approximately one-third of its acquired energy in rotating the turbines and the connected electrical generator. The electrical power generated is approximately 1,000 MW. The steam is then condensed in the turbine condenser by passing over tubes containing circulating water (from the main cooling system described below) to which it gives up the remaining two-thirds of its acquired heat energy.

4.4 Cooling water systems

The general diagram of service water supply is given in Figure 4.5.

In compliance with the subdivision of water consumers into three technological groups, the cooling equipment of K2 incorporates three cooling systems, namely:

- the main cooling system, this system being isolated from the cooling system of unit 1;
- the cooling system for 'group A consumers', i.e. critical users due to safety considerations; and
- the cooling system for 'group B consumers', i.e. non-critical users.

4.4.1 Main cooling system

The main cooling system is based on the circulation principle. Cooling is carried out in a reservoir (# 1 on Figure 4.6). The system is designed for the circulation of 966.18 million m^3/s year.

The system shares the intake and blow out pipes with the group B consumer cooling system (Section 4.4.3).

The designed water consumption for the main cooling system and group B consumer cooling system is 16 million m^3 /year (# 4 on Figure 4.6).

Figure 4.5
Diagram of service water supply for K1 and K2

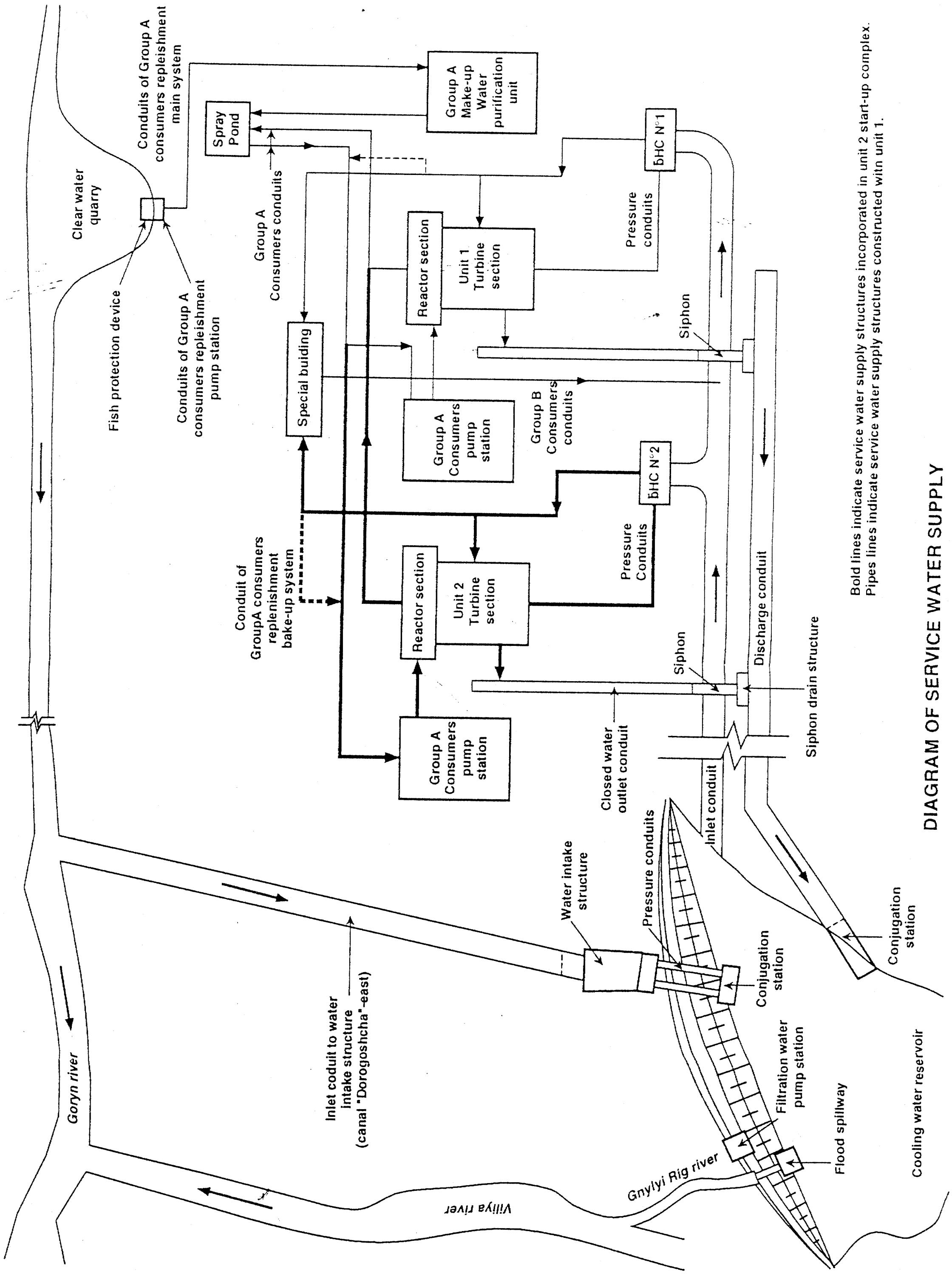
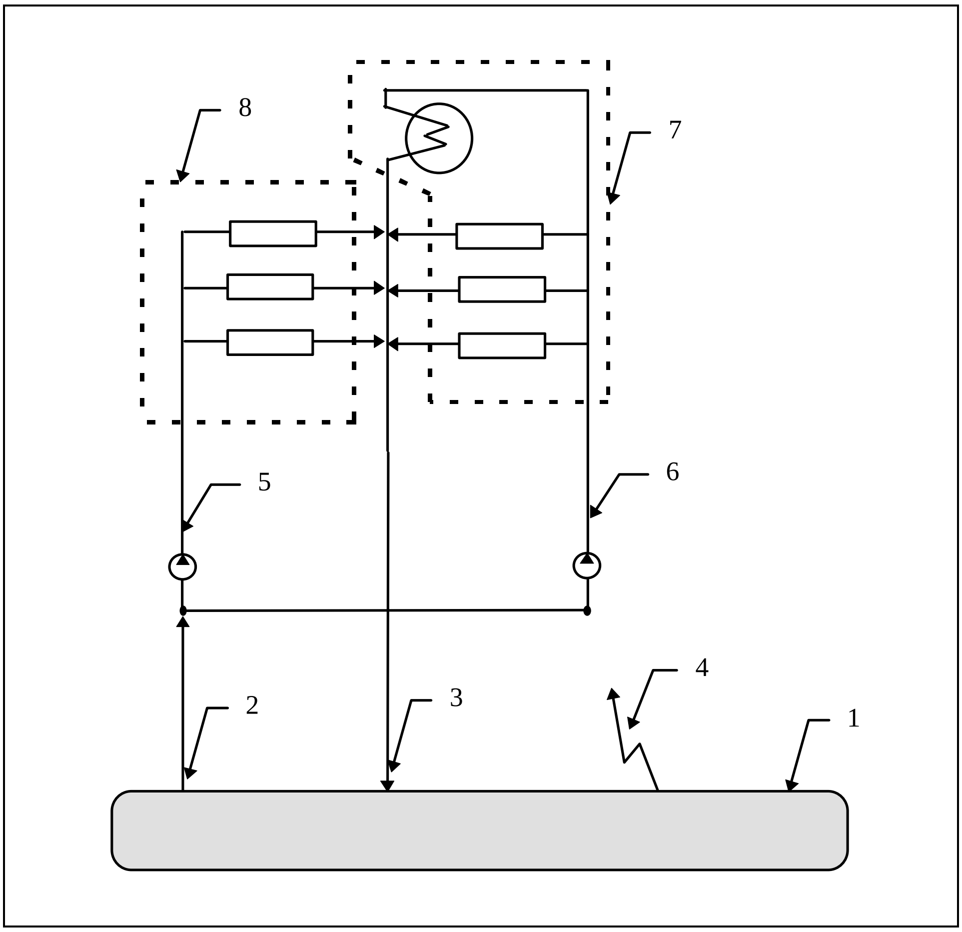


Figure 4.6
Main cooling system and Group B consumers cooling system



Legend:

1. Reservoir
2. Water intake from the reservoir
   (Main cooling system + Group B consumers cooling system)
3. Water bleed-off to the reservoir
4. Losses (mainly vapour)
5. Incoming flow for group B consumers cooling system
6. Incoming flow for main cooling system
7. Main cooling system consumers
8. Group B consumers

4.4.2 Cooling system for group A consumers

The design circulation of cooling water in the system is 53 million m^3/year (Figure 4.7). Water cooling is carried out in sprayer ponds (# 2 on Figure 4.7). The design consumption of cooling water in the system is 1.2 million m^3/y for one unit (# 4 on Figure 4.7).

The design rate of maximal water intake from the Goryn river for the NPP cooling systems' make-up is 1.2 million m^3/y for K2 (# 3 on Figure 4.7). The other source of replenishment water is water from the water treatment facility for the fenced-off area, providing 0.01 million m^3 /y (# 6 on Figure 4.7). This water is provided from artesian wells.

4.4.3 Cooling system for group B consumers

Based on the circulation principle, the cooling system for group B consumers shares the intake and blow down pipes with the main cooling system (Figure 4.6).

The system is designed for the circulation of 40 million m^3/y (# 5 on Figure 4.6). The designed water consumption for main cooling system and group B consumer cooling system is 16 million m^3 /y (# 4 on Figure 4.6).

4.4.4 Water balance

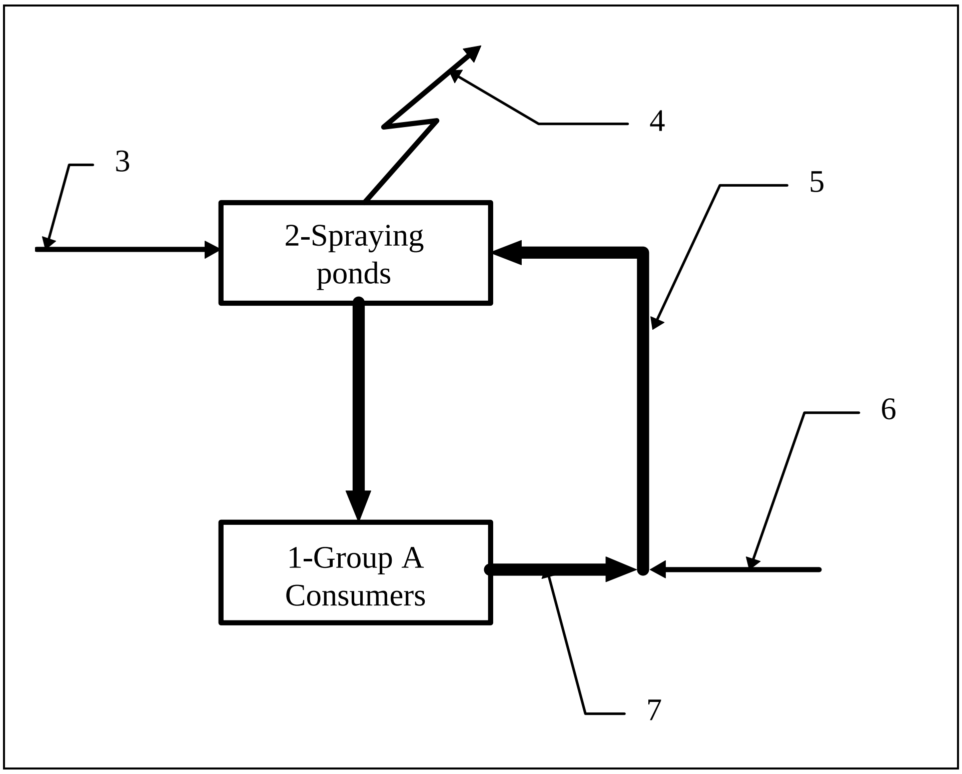
Data provided for water balance at KNPP (Figure 4.8a) were not entirely self-consistent since, for the existing unit, they indicated more water leaving the system than entering it. Revised data (Figure 4.8b) indicate a higher abstraction from the Goryn (23.86 million m^3 compared with 18.26 million m^3) and a smaller return to the Goryn (10.44 million m^3 compared with 11.38 million m^3). The revised data do not indicate the fraction of total requirements which will be obtained from artesian sources.

The revised data also include:

- 30.69 million m^3/y for irreversible loss of water;
- 11.86 million m^3/y for inherent evaporation; and
- 0.24 million m^3/y for output of water and steam.

Actual utilisation of water on the NPP site will be subject to a specific investigation and abstraction will be subject to regulatory control (Section 9).

Figure 4.7
Group A consumers cooling system



Legend:

1 Group A consumers

2 Spraying ponds

3 Water replenishment (from Goryn river): 1.20 x 10^6 m^3/y

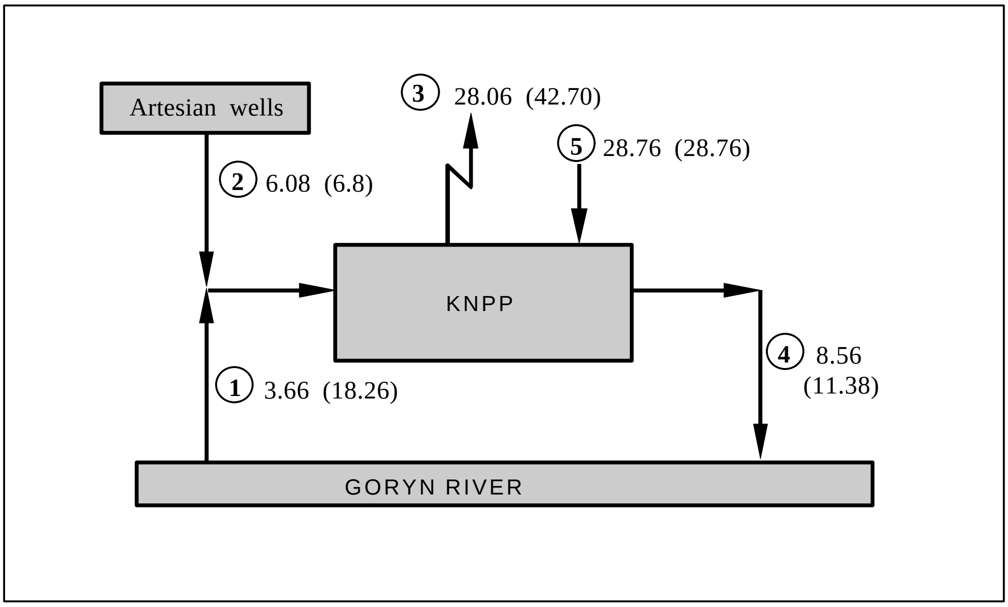
4 Losses: 1.21 x 10^6 m^3/y

5 Design water circulation flow: 52.9 x 10^6 m^3/y

6 Water from fenced off area water treatment facility: 10000 m^3/y

7 52.9 x 10^6 m^3/y

Figure 4.8a
Water utilisation at Khmelnitsky NPP –initial data

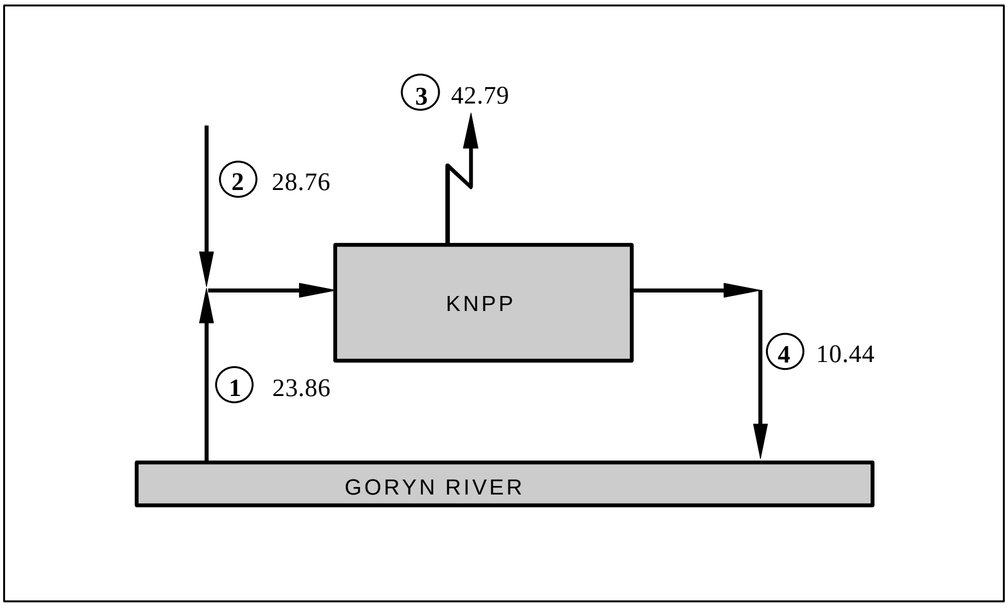


Legend:

Values without brackets are given for KNPP unit 1; values in brackets are for units 1 and 2.

1. Water intake from Goryn river (10^6 m^3/y)
2. Water intake from artesian wells (10^6 m^3/y)
3. Water losses (mainly steam) (10^6 m^3/y)
4. Water drain to Goryn river (10^6 m^3/y)
5. Water entering the reservoir (Gnilyi Rig + rain water) (10^6 m^3/y)

Figure 4.8b
Water utilisation at Khmelnitsky NPP – revised data [4.2]



Legend:

Values given are for units 1 and 2; water intake from Goryn river (10^6 m^3/y)

1. Water intake from Goryn river (10^6 m^3/y)
2. Influx and precipitation (10^6 m^3/y)
3. Water consumption (10^6 m^3/y)
4. Filtration (10.38) plus disposal (0.06) (10^6 m^3/y)

4.5 Principal discharge sources

Operation of the NPP is cyclical. The reactor is designed to be run continuously for a period and then shut down annually, for one or two months, for routine maintenance, shuffling of fuel and partial refuelling.

4.5.1 Reactor operation

Under normal operation, any leakage from, or partial failure of, the fuel cladding will lead to small amounts of fission products being released into the primary circuit. These quantities may be by the fission of any 'tramp' uranium that might be present on the exterior surface of the fuel pins from contamination during fabrication (Section 4.7.1). Tritium, produced in the fuel by fission, can be released through the cladding by diffusion and through any pin holes or defects. The amounts released depend on the design and quality of the fuel pins.

Small amounts of radioactive material may also be formed within the primary coolant as a result of neutron activation of fuel tubes, primary circuit and structural material surfaces. Corrosion and erosion processes tend to release activation products from such materials into the primary coolant circuit. Tritium, generated from activation of boric acid in the primary coolant, is a particularly significant activation product. In addition, activation processes in the air surrounding the reactor pressure vessel produce small quantities of gaseous radioactive species including tritiated water vapour and noble gases.

A number of separate radioactive discharges from the reactor can be identified, concerned principally with chemical and volume control of the primary circuit coolant. Dissolved fission and activation products are removed from the coolant by an ion exchange process, which produces contaminated resins. The periodic removal and replacement of such resins generates both solid and liquid wastes. Periodically, some coolant is also discharged from the primary circuit in order to remove tritium, so that the activity concentration is maintained below a defined maximum operating limit. This discharge from the primary circuit also gives rise to a liquid waste stream.

Gases that build up in the primary circuit during operation must be removed. This results in a gaseous waste stream. Atmospheric releases may also derive from the ventilation of fugitive emissions of primary circuit coolant through minor leakage. Such releases will typically comprise tritiated water vapour, noble gases, aerosols and other vapours.

Estimates of the quantities of radioactivity present in the primary circuit coolant and the various waste streams have been made as part of the design basis of the reactor, using conservative assumptions. These estimates, together with consideration of the potential health impacts of any radioactive releases, form a general basis for establishing operational limits in respect of emissions and waste management requirements. Information on discharges arising from normal operation, based on operating experience of other VVER-1000 type reactors, illustrates that reactor operation can readily meet such discharge limits. Indeed, in practice, plant performance against operational limits is routinely monitored by the regulatory authorities and can lead to a progressive reduction in target emission levels.

4.5.2 Power conversion and waste heat rejection operation

The principal waste derived from turbogenerator operations is low grade heat, which is extracted from the secondary circuit at the main turbine condenser.

Any nuclear power plant is a source of a large amount of heat. Approximately two-thirds of the heat energy generated by the reactors cannot be used for the generation of electric power and is released into the environment.

In the NPP cooling systems, the waste heat in the capacitors and heat exchangers is transferred to the circulation cooling water which, through the end absorbers (the heat sink for the main cooling system and for the cooling system of the group B consumers; the spraying ponds for the cooling system of the Group A consumers) withdraws the waste heat to the atmosphere.

The heat is mainly transferred to the environment through the evaporative cooling of water in the end absorbers and partially through convective heat transfer to ambient air in the end absorbers.

Water evaporation in the circulation water cooling systems leads to accumulation of salts carried there by the make-up water. Technological limitations regarding the salt content of the cooling water require blowdowns of the cooling systems in order to maintain the salt regime within admissible levels.

No provision has been made for a constant blowdown for the Khmelnitsky NPP heat sink. The design makes provision for blowdown of the heat sink both through non-recoverable water leaks from the heat sink, and through periodic dumping discharge of water through the spillway during spring floods. Additional pumping of water from the Goryn river is used to make up for the water losses.

Dumping of water is admissible following application by the NPP management and upon receipt of a permit for the blowdown to be carried out from the water protection bodies, based on results of chemical analysis of the water in the heat sink.

4.5.3 Refuelling and maintenance

At annual shut down, the cooling systems are depressurised, the primary circuit pressure vessel head removed, and one third of the fuel assemblies removed and transferred to a storage pond adjacent to the pressure vessel. The remaining two thirds are then rearranged to maintain optimum power densities and new fuel is inserted in the core. Typically, therefore, after the initial start-up period, each fuel assembly will remain in the reactor for three years.

In addition to the spent fuel, refuelling operations may give rise to active liquid effluents and atmospheric discharges that are of a similar nature to those derived from the primary circuit coolant during normal operation.

Repair and maintenance activities undertaken during shut-down also give rise to various contaminated solid wastes, caused by contact with activation products or by contact with contamination from the reactor primary circuit. Certain components, activated by neutron irradiation, may also be replaced, giving rise to solid wastes.

4.6 Water supply, treatment, and liquid effluent disposal

4.6.1 Process water abstraction

4.6.1.1 Service water supply

Water supply is indicated on Figure 4.8.

The artificial water reservoir, which is the heat sink for the NPP, was built in the flood lands on the left bank of the Goryn (Section 3). This pond is of the channel type for the Gnilyi Rig river and the inflow type for the Goryn.

In compliance with the terms of the industrial use of water by the NPP, water intake from the Goryn into the reservoir is allowed exclusively within the non-vegetation period, from October through May. At any other time, replenishment of the pond by Goryn water is prohibited, and the NPP gets water from the reservoir at the expense of its volume capacity.

The principal water management characteristics of the inflow reservoir are as follows.

- Normal water level (NL) - 203.00 m (Baltic system).
- Dead storage level (DL) - 198.00 m (Baltic system).
- Water table area at NL - 20.0 km^2.
- Water table area at DL - 11.5 km^2.
- Full capacity at NL - 120.0 million m^3.
- Capacity at DL - 39.0 million m^3.
- Average depth at NL - 6.0 m.
- Average depth at DL - 3.4 m.

The reservoir is furnished with a shaft-type automated emergency flood water header with a polygonal overflow side. The header was designed for transit discharge of emergency volumes of rainfall floods of the Gnilyi Rig river with a 0.01 % probability and the boost-up of the water level to the 203.7 m mark.

The automated emergency flood water header is equipped with a bottom aperture with a plug gate, which allows for drawdown of water during regular blow-downs.

An exposed drainage conduit runs along the lower slope of the ground dam. This conduit intercepts the filtering flow from the reservoir and forwards the filtered water to the drainage facility which pumps water back to the reservoir.

Water to fill and replenish the reservoir is taken from the Goryn river by the make-up water pump station. The pump station is located in the tail race of the dam. River water reaches the pump station through a cut-through canal to the 'Doroghoshcha-Skhidna' cut which serves as a water intake conduit. An antechamber is installed upstream from the pump station. The premises of the pump station are integrated with the water intake. The subterranean facility of the pump station is 60 m long and 12 m wide. The pump station is equipped with the following machinery

- five pumps: Q = 18,000 m^3/h, H = 15 m;
- two pumps:Q = 4,000 m^3/h, H = 16.5 m; and
- two pumps: Q = 1,800 m^3/h, H = 16.0 m.

The type and number of pumps in operation at any time is determined automatically by the flow rates of the Goryn river and by the respective water levels. The maximal rate of supply of water by the pump station is 30 m^3/sec.

Facilities for the replenishment of cooling systems used by group A consumers were designed and built separately. Such facilities include a pump station on the Goryn river as well as head pressure water conduits, with a design flow rate of 0.3 m^3/sec. Water intake from the Goryn river to replenish the cooling systems group A consumers is allowed throughout the year.

Since other intakes of water from the Goryn river are allowed only during the non-vegetation period the minimal flow rates of the summer and autumn low water level period do not restrict the service water supply system of the NPP. The intake of Goryn water to replenish the reservoir is greatest at the time of spring floods. For this reason the design flow rate for the replenishment pump station was set at 30 m^3/sec, and the control capacity of the reservoir at 80 million m^3.

The average design volume of water intake from the Goryn river for reservoir replenishment, chemical treatment of water and make-up of crucial consumers following commissioning of K2 will be as follows:

- pumping water to replenish the reservoir – 15 million m^3/y; and
- pump feeding of water for demineralisation and make-up of crucial consumers' facilities 3.3 million m^3/y.

Moreover, the reservoir accumulates the full volume of the Gnilyi Rig river outflow, the respective design capacity being 19.1 million m^3/y.

4.6.1.2 Drinking water

Drinking water and water for other domestic use at the NPP is provided from artesian wells.

The artesian water intake is located northeast of the NPP along a forest tract. The intake is the source of drinking water for the city of Netishin, the NPP and the industrial zone. The total capacity of the water intake is up to 18,000 m^3/d. Current permits (UkrYuzh-943) allow for abstraction of 15,000 m^3/d. The design water consumption for the NPP site is 720 m^3/d.

The wells are 240-300 m deep. The water line is spanned by crag rocks. The static water level in the wells is 25-40 m from the water surface. Such water meets the requirements of GOST 'drinking water' [5.1], with the only exception being the increased content of Fe (1.59 mg/l, against a standard of 0.3 mg/1).

In order for the water to comply with the above GOST requirements it goes through additional purification at a deferrization unit before it reaches the consumers.

4.6.1.3 Water for fire safety

Water for the safety and fire extinguishing needs of the site is taken from the water feed conduit of the NPP service water supply system.

Fire pumps with a capacity of 600 m^3/h and a head pressure of 100 m water column are installed at the modular pump stations Nos. 1 and 2. Each pump station is equipped with two working pump assemblies and two reserves. Two fire extinguishing pumps are able to attain the water consumption level required by fire probability design calculations.

4.6.2 <u>Water treatment</u>

Chemical treatment of water is required to replace losses of steam and service water in the second circuit as well as for initial filling of the primary circuit of the NPP. The overall system of water treatment is shown in Figure 4.9.

The source water for chemical treatment is that of the Goryn river, special heaters bring its temperature to 30^{o}C.

Pre-treatment of chemically demineralized make-up water involves the following procedure: calcification and coagulation in clarifying agents, clarification treatment on mechanical filters and two-stage H-OH ion enrichment with interim decarbonization. The initial filling of the first circuit is carried out by pretreated chemically demineralized water which has been subjected to a third stage of demineralisation in internal regeneration combined mode filters involving both mechanical filters and an ion exchange process.

The anionic unit of the demineralisation set is designed in accordance with the principle of modular activation of different ion filters ('chains'). The set usually has three working units of filters and one reserve unit not loaded with the ion. The unit of filters consists of the preactivated and the main cation first stage filter, the anionic first stage filter, the decarbonizer, the receptacle for partly demineralized water, the H-cation second stage filter and the anionic second stage filter.

Wastes arising from water treatment include sludge waters from the clarifying agents, flushouts from the mechanical filters, and the regeneration and washout waters of the ion exchange systems.

Figure 4.9
Principles of chemical water treatment

****Figure needs correction – Goryn river****

Legend:

1. Goryn river
2. heater
3. treatment by clarifying agents
4. mechanical filters
5. ionic treatment
6. neutralization
7. setting tank
8. flush-out water from mechanical filters
9. settled water
10. lime 60%
11. underburnt lime
12. lime sludge burial
13. softened water for group A consumers and make-up of heat supply neywork
14. regeneration and washout water from mechanical filters
15. demineralized water for NPP cycle make-up
16. neutralized regeneration and wash-out water, delivered to water service supply system
17. lime purification

The sludge waters of the clarifying agents are fed to the sludge discharge strainer. The flushouts from the mechanical filters are collected in a special flask to be sent on to the chemical treatment clarifying agents. Regeneration and washout waters are streamed into the neutralisation tanks and then to the blowdown water line of the service water supply.

The capacity of the water demineralisation system is 145 tons/h, its maximal capacity being 180 tons/h.

4.6.3 Effluent disposal

4.6.3.1 Radioactive effluent treatment

The NPP liquid radioactive waste (LRW) is formed as a result of the operation of the special water purification units (SPU). The treatment scheme for LRW is as shown in Figure 4.10.

The LRW includes:

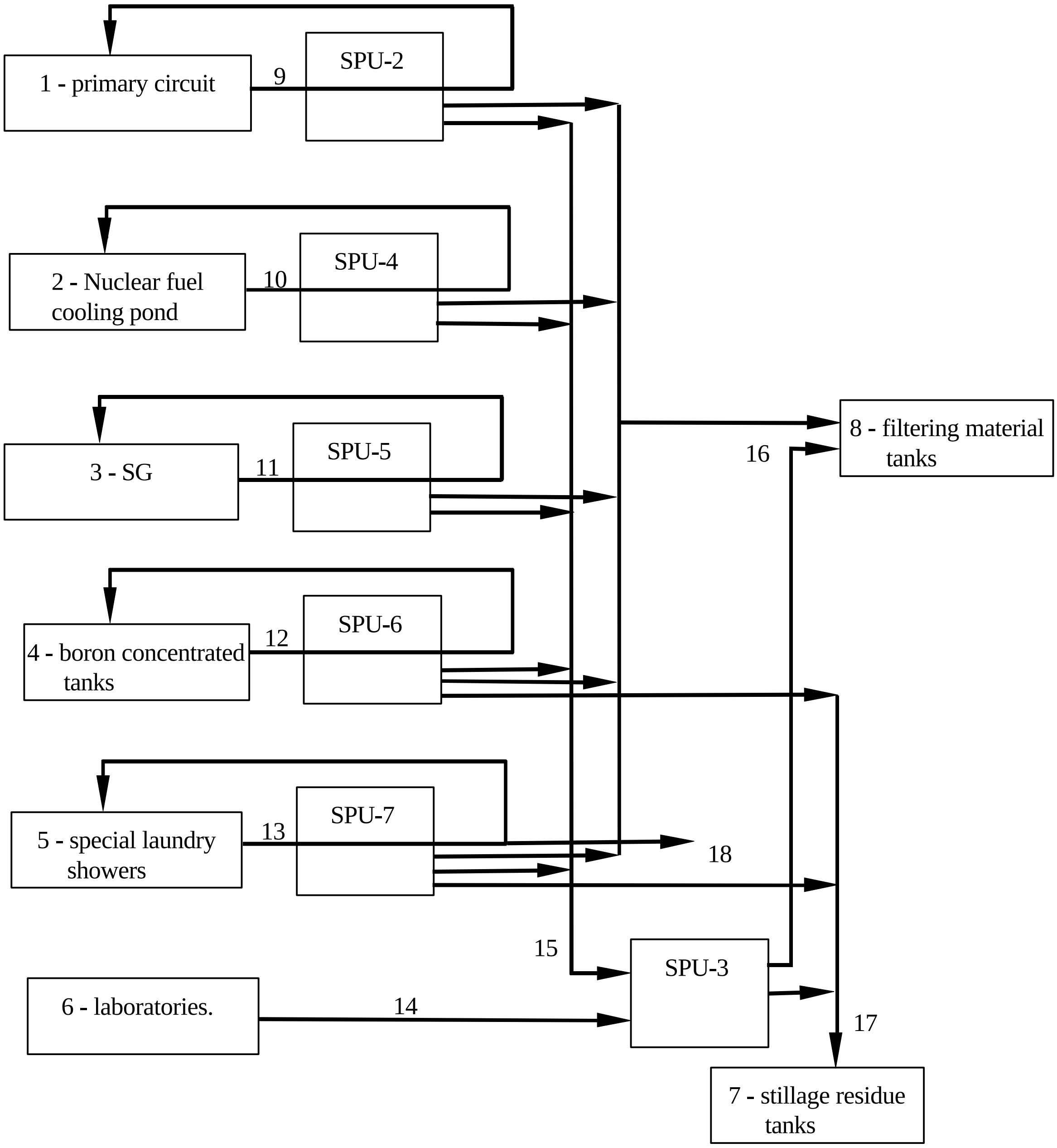
- spent ion exchange resins of the SPU;
- high-concentration salt solutions (the stillage residue) resulting from processing of the drain gully water, waste water from special laundries and shower rooms at the evaporators; and
- sludge from the settling tank and the sump tank of the drain gully water.

In addition to LRW from regeneration, washing-off, loosening, and hydraulic unloading of the special water purification filters, the following comes into the drain gully water system for treatment:

- uncontrolled leakages of the primary circuit during the period of preventive maintenance work - up to 556 cm^3/s for 10 days from one power unit, in case of nominal power operation - 56 cm^3/s;
- waste water from the laboratories: 69 cm^3/s from each laboratory;
- waters from decontamination of rooms/compartments in the nominal power operation mode: 1 cm^3/s for each power unit;
- in the mode of repair and refueling: 93 to 116 cm^3/s for each power unit for 50 days;
- waters from decontamination of detachable equipment (small parts) in the nominal power operation mode: 28 cm^3/s; in the mode of repair and refueling: 168 cm^3/s for 50 days;
- sampling: 8.3 cm^3/s for each sampling point twice a day but no more than 139 cm^3/s (total for all sampling points); and
- unaccounted-for and emergency leakages: 20 % of the overall drains.

With respect to the latter, it has subsequently been stated that an allowance of 20% has been included to take account of the capacity of a number of SPU-3 steam devices for treating drain gully water.

Figure 4.10
LRW treatment scheme



Legend:

1	primary circuit	10	water from cooling pond
2	nuclear fuel cooling pond	11	SG leakage
3	steam generator	12	boron concentrated water
4	boron water supply system	13	special sewage
5	special laundry	14	special sewage
6	laboratories	15	floor drain water, filters regeneration water
7	stillage residue tanks	16	spent ion exchange resins
8	filtering material tanks	17	stillage residue
9	organized leakage	18	imbalance water from SPU-7
SPU	Special Water Purification Unit (see text)		

The formed LRW go to the interim storage unit of the special-purpose building. This includes the filtering materials tanks and the stillage residue tanks (#7 & #8 on Figure 4.10).

In addition to the interim storage unit, there are the tanks designed for receiving stillage residue, which are located in the expanded part of the special-purpose building.

There is no discharge of LRW into the environment. To avoid any spread of radioactivity in case of the loss of integrity of the radwaste-containing tanks, the rooms where the tanks are installed are coated with corrosion-resistant steel up to the level of possible flooding of the room. The rooms include pits with humidity indicators.

4.6.3.2 Effluent from process water treatment

Clarified waste water (# 16 on Figure 4.9) is discharged into the cooling water reservoir.

Sludge from water treatment clarifiers is removed into the sludge accumulation tank (# 17 on Figure 4.9) with the return of settled water to the chemical water treatment system (# 9 on Figure 4.9).

The flow rate of sludge-bearing water is 20 m^3/h. The flow rate of clarified water is 90 m^3/h.

The sludge accumulation tank has been designed to allow for 10 years of storage.

4.6.3.3 Oily water treatment

Sources of waste water contaminated with oil products include:

- in the main building: the oil systems of the turbines, generators, feed pumps, fans, drains of the stuffing box seals of pumps, spillages of oil when repairing the oil systems and equipment;
- in the auxiliary rooms: the drains of the stuffing box seats of pumps, compressors, fans, spillages of oil and fuel oil when repairing the oil systems and equipment;
- at the sites intended for the installation of transformers;
- emergency oil drainages; and
- at the oil and fuel oil facilities: drainage of the floors of the fuel oil pump room, the storm and melt waters from around the outdoor storage facilities.

Waste water contaminated with oil products is fed, under pressure, from the site to the integrated treatment facilities which are a part of the chemical water treatment system. Oil and fuel containing waters are collected and neutralised in a receiving tank, settled in the settling tank and then purified in mechanical and charcoal filters. Treated water is drained to the discharge circulation conduit while the trapped oil products are burnt at the PRC facility.

The capacity of the treatment facilities is 50 m^3/h.

4.6.3.4 Domestic and other non-radioactive effluents

Sanitary sewerage system of the 'non-fenced off' area

Sanitary/household waste waters from toilets, washrooms, shower rooms, cafeteria, laboratories and the laundry room of the 'non-fenced off' area of the NPP site are pumped to the municipal treatment facilities located 500 m away from Unit 1 beyond the site territory. The flow rate of the sanitary sewage (from two power units) is 540 m^3/d.

The design capacity of the treatment facilities is 20,000 m^3/d. Treated sewage from the municipal treatment facilities is supplied to the cooling water pond.

Sludge from the treatment facilities is used for the production of agricultural fertilisers.

Sanitary sewerage system of the 'fenced-off' area

Sanitary sewage from the WC pans of the reactor building, the washrooms and shower rooms of the special-purpose building of the 'fenced-off' (controlled access) areas, as well as waste water from the third rinsing at the special laundry of the NPP site, are pumped to the treatment facilities of the 'fenced-off' area, which are located on the NPP site territory. The flow rate of sanitary sewage (from two power units) is 148 m^3/d.

The design capacity of the treatment facilities is 400 m^3/d. Treated sewage from the treatment facilities is supplied to the spraying ponds of the service water supply system of the NPP group A consumers.

Sludge, once dried, is delivered for disposal at the solid radioactive waste repository/storage facility.

Surface water drainage

Storm and drain water is provided by gravity-flow mode to the inlet channel (conduit) of the NPP service water supply system.

The flow rate of the storm drain water (from two power units) is 930 m^3/d.

4.7 Sources of emissions to atmosphere

Activity sources within a nuclear reactor are generally classified as one of the following:

- fission products;
- corrosion products; and
- activation products and actinides.

4.7.1 Fission products

Fission products are formed by means of nuclear fission within the uranium dioxide fuel. They comprise nearly 200 radioisotopes of some 40 different chemical elements (atomic numbers 30-66) with diverse chemical and physical properties. Some are gases (e.g. the noble gases krypton and xenon), others are quite volatile at reactor temperature (such as caesium and iodine), and some are refractory metals (such as the lanthanides). A considerable proportion of the fission product inventory is too short-lived to be of any environmental significance; these radionuclides decay rapidly before they are able to reach the environment in any significant quantity.

A series of barriers prevent release of fission products into the primary coolant. These include the fuel matrix itself, which serves to contain the majority of non-volatile fission products under normal operation. The volatile fission products tend to migrate through the fuel matrix and accumulate at the grain boundaries and its gaps within the fuel pin. The fuel cladding is designed to contain the volatile fission products within the fuel pin and to prevent contact between fuel and primary coolant.

Only fission product gases and the more volatile elements escape from the fuel matrix in any significant quantities and accumulate in the fuel pin gaps. The fuel cladding normally contains these radionuclides; however, some pins may develop small cladding defects as a result of mechanical or thermal stresses, corrosion or other causes. This can result in escape of some of the more volatile fission products into the primary coolant. Gross failure of cladding is also possible, but activity monitoring of the primary coolant ensures that these failed pins are detected and can be removed from the reactor.

'Tramp' uranium (trace amounts of uranium present on the exterior surface of the fuel pins remaining from fuel manufacture) will also be a source of fission products released into the primary coolant. This is because such uranium is liable to undergo fission in the same way as the uranium inside the fuel pins but with no barrier between it and the primary coolant.

4.7.2 Activation and corrosion products

The neutron flux in the reactor core results in activation of stable isotopes in various materials, including those found in the fuel cladding, structural components, coolant water and dissolved ions, dissolved air, and air in gaps in the reactor shaft. The active isotopes may be created by simple neutron capture or by secondary processes, such as neutron capture followed by α-decay. Some activation products are very short-lived and are not significant for environmental impact assessment or waste management; however, they may have implications for shielding in the reactor hall as a result of their penetrating γ-rays (Section 4).

Activation of materials used for structural components and fuel cladding, and the subsequent corrosion or erosion of these materials, can lead to the presence of radionuclides in the coolant in either soluble or particulate form. However, the chemical properties of the primary coolant are controlled to minimise corrosion. This entails routine monitoring of a wide range of parameters, including pH, conductivity, transparency, boric acid concentration, potassium and sodium concentrations, dissolved gases, fluoride and chloride concentrations and oil content.

Activation of boron dissolved in the primary coolant leads to the formation of tritium. Although of low radiotoxicity, tritium is potentially significant from a radiological point of view because of its chemical behaviour as hydrogen, which means that it is readily formed into water molecules where it is chemically indistinguishable and therefore extremely difficult to separate from 'normal water'. Tritium, in the form of tritiated water, is highly mobile in the environment and in living tissue. Its half-life is approximately 12 years, it will therefore always be present in the coolant in quantities determined by the power history of the reactor and the coolant replenishment cycle. This is the most significant source of tritium because it gives rise to tritium in the coolant; other sources, such as its formation as a ternary fission product or by activation of boron carbide used in the control rods, require failure in the cladding of the fuel rod or control rod respectively in order for tritium to reach the primary circuit.

Instead of undergoing nuclear fission, uranium in the reactor fuel may absorb neutrons to form actinides such as plutonium; these can reach the coolant by slowly leaching from exposed fuel if a breach in the cladding occurs. Tramp uranium can also be a source of actinides in the coolant.

The most radiologically important radionuclide arising from neutron activation of air in the pressure vessel shaft is Ar-41, produced by activation of Ar-40. This is fed via the hermetic zone ventilation system through iodine and aerosol filters to the stack. Decay tanks at the gas cleaning station are used to allow the Ar-41 (half-life 1.8 hours) to decay before final discharge.

N-16 is produced by nuclear reactions on the oxygen of the primary coolant water. Although it has a short half-life (seven seconds) it is transported around the reactor coolant circuit to the heat exchangers. It is unlikely to represent a health hazard unless released under accident conditions. However it does produce very high energy gamma rays which are very penetrating and which require adequate shielding.

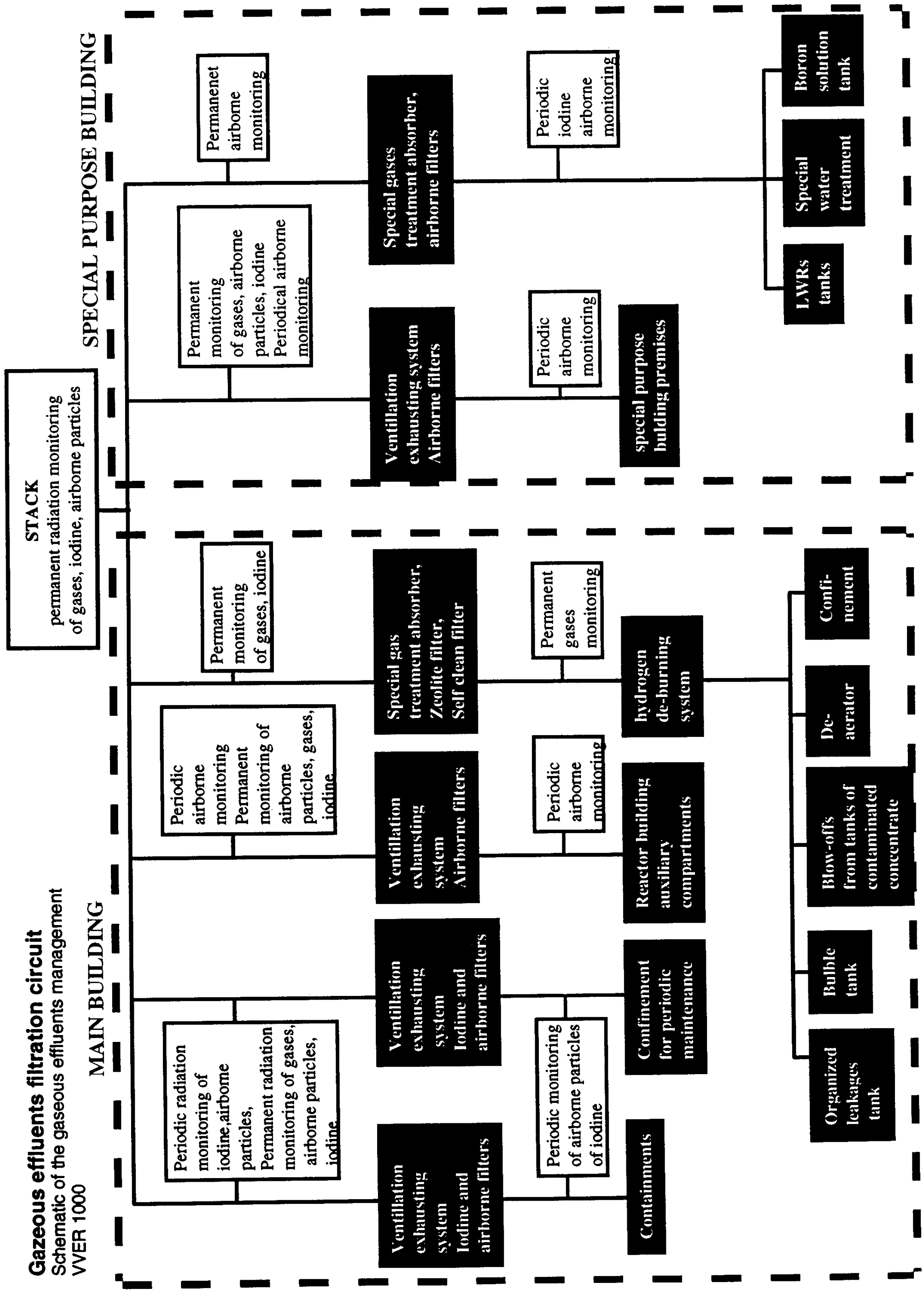
4.7.3 Treatment and monitoring

Gaseous radioactive emissions result from the release of radioactive gases and aerosols from the liquid radioactive media. Sources, treatment and monitoring are summarised in Figure 4.11.

Sources from normal operation of the NPP include:

- leakages of the primary circuit coolant;
- the fuel cooling pond;
- sanitary and household sewages; and
- equipment blowoffs.

Figure 4.11
Airborne emissions treatment and monitoring



In the case of an accident the main source is:

- through the evaporation of spilled primary circuit coolant.

Gaseous emissions are arbitrarily divided into three groups:

- radioactive noble gases;
- aerosols; and
- iodine isotopes.

The normative level of gaseous emissions is ensured by the following engineering approaches and solutions:

- selection of equipment and the process flow diagram;
- availability of normal operation systems and emergency systems;
- special measures for working with radioactive media; and
- an air purification and removal scheme supported by the following measures:

  - the air to be removed, which contains radioactive isotopes, is subjected to purification while passing through aerosol and iodine filters;
  - purification/cleaning of technological blowoffs via filters-absorbers, where decay of the large part of radioactive isotopes of xenon and krypton takes place;
  - discharges of air from rooms of the 'fenced-off' area of the instrument section and the special-purpose building in a controlled manner through the 100 m high ventilation stack which ensures the required dilution of emitted radionuclides in atmospheric air below admissible concentrations;
  - organisation of the sanitary protection zone with a radius of 3 km; and
  - organisation of continuous dosimetric monitoring of emissions into the atmosphere, monitoring of the level of contamination of atmospheric air, soil, vegetation and water of the open water reservoirs/basins.

Further information on controls on atmospheric discharges is provided in Section 6.2.4.

4.7.4 Non-radioactive emissions

Possible sources of non-radioactive emissions include:

- the boiler house;
- diesel generators;
- vehicles;
- 'smelting works';
- battery houses; and
- issues associated with installation and construction.

The main non-radioactive emissions to atmosphere during reactor operations are water vapour and water droplets from the spray ponds and ventilation stacks. Impacts of heat release are assessed in Section 6.

There may also be small fugitive emissions of cleaning solvents, such as degreasers, from various site locations during normal operations. However, these are unlikely to be significant as the quantities used on site can be expected to be quite small and high pressure hot water cleaning will be used for much of the mechanical equipment.

4.8 Solid waste management

Three principal types of solid waste are identified:

- low and intermediate level operational radioactive wastes;
- non-active process water treatment sludges; and
- general non-active wastes.

Spent fuel management is addressed separately in Section 4.9.

4.8.1 Solid radioactive wastes

Normal operation and maintenance works result in solid radioactive waste (additional contributions can arise from abnormal events). According to "Sanitation Rules on NPP Design and Operation" [4.8], solid waste is considered to be radioactive if it exceeds any one of the following three criteria.

1. The gamma dose rate at a distance of 0.1 m from its surface exceeds 10^{-4} mSv/hr.
2. The volumetric activity exceeds for a beta emitter 7.4 10^{4} Bq/kg, or for an alpha emitter 7.4 10^{3} Bq/kg.
3. The surface contamination exceeds for a beta emitter 500 particles/cm^2. min., or for an alpha emitter 5 particles/cm^2. min.

Radioactive waste is classified into one of three groups:

I	-	Low
II	-	Average
III	-	High

according to dose rate, volumetric activity and surface contamination as set out in Table 4.1.

Table 4.1
Radioactive waste classification

Group of waste	Equivalent dose rate at 0.1 m from the surface (mSv/hr)	Volumetric activity (Bq/kg)		Surface contamination (particles/cm^2.min)	
		Beta Emitters	Alpha emitters	Beta emitters	Alpha emitters
I – Low	10^{-4} - 0.3	7.4 10^{4} - 3.7 10^{6}	7.4 10^{3} - 3.7 10^{5}	5 10^{2}-10^{4}	5-10^{3}
II– Average	0.3 - 10	3.7 10^{6} - 3.7 10^{9}	3.7 10^{5} - 3.7 10^{8}	10^{4} -10^{7}	10^{3} - 10^{6}
III – High	>10	3.7 10^{9}	>3.7 10^{8}	>10^{7}	>10^{6}

Group I radwaste includes: cleaning and insulating material, specialised use uniforms, footwear, individual radiation protection means, flexible PVC, construction waste, implements and tools.

Group II radwaste includes: pipework, reinforcement, parts of pumps and drives of control and protection systems, filters of ventilation systems, waste metal, heat insulation material, detachable detectors.

Group III radwaste includes: intermediate hoses, scram control/shim assembly tops, ionisation chambers with communication lines, heat and energy release detectors with communication lines.

Group I radwaste is collected by hand into bags in the reactor unit and then brought from the shield building to Facility A 111/2, where locally-made casks of 1 m^3 each are kept.

Similarly, Group II radwaste is gathered into bags wherever it is found.

According to activity levels, Group II solid waste is stored temporarily in the compartments of the building used for handling and processing radioactive wastes.

Group I and II radwaste is stored in the concrete bays of the storage facility, the capacity of which has been calculated on the basis of the following criteria:

- term of storage: 10 years;
- possibility of further removal and reburial;
- storage of flammable and nonflammable waste in plastic bags; and
- storage of specialised ventilation filters without prior processing.

Owing to lack of necessary equipment, radwaste has not undergone preliminary processing or been sorted into flammable and nonflammable categories. The distribution of pressable and non pressable radioactive wastes is given in Table 4.2.

Group III radwaste is stored in the reactor unit's burials, the capacity of which has been determined to suffice for 30 years of the unit's operation.

As of 31 December 1996, the quantities of waste given in Table 4.3 had been accumulated. The total quantity of radioactive waste in the radwaste building was 1,606 m^3. At the end of 1996, 100% of the cells designed for Group I wastes were full and Group I wastes were being disposed of in cells designed for Group II wastes.

The radionuclide composition and collective activity of solid radioactive waste is not calculated because of a lack of methodology.

The project for the Group I radioactive waste handling facility complex includes equipment for:

- low-active solid radwaste sorting with prior processing;
- pressing; and
- incineration (the system includes NUKEM emission control equipment to monitor chemical and radiological releases).

The project does not envisage the processing of:

- radiation sources;
- scrap metal in the form of disassembled equipment and pipework; and
- Group II and III radwaste.

Table 4.2
Distribution of pressable and non-pressable radioactive wastes

Pressable	**%**	**Non-pressable**	**%**
Clothes	5	Woodwork	42
Rubber	5	Filters, filter frames	5
Flexible PVC	5	Pipework	10
Heat insulation	40	Glass	8
Paper	25.4	Tools	15
Filters	2		
Sorbents	8		
Other (mixed)	9.6	Other	20
Total	100	**Total**	100

Table 4.3
Wastes accumulated in the solid radioactive waste storage facility as of 31 December 1996 [4.2]

Waste category	**Capacity of cells (m^3)**	**Arising in quarter 4 1996 (m^3)**	**Total amount of waste (m^3)**	**Percent of capacity**
Group I	1082	44.4	1553	100
Group II	2988	None	48.4	17
Group III	381	None	3.95	1
Total:		44.4	1606	

4.8.2 Processing of liquid radioactive waste

Waste water from the reactor building and the special-purpose building, is used in the NPP cycle after being treated at corresponding special water treatment installations (see treatment units 'SPUs' on Figure 4.10)

The sole exception is provided by the imbalance water of the SPU-7 special water treatment facility, which may be drained into the spraying pond, provided the content of radioactive substances does not exceed admissible values according to the radiation safety standards. For K1 in 1996 and 1997, 9145 and 9005 m^3 of water were released from the SPU-7 facility. No standards have been obtained for such discharges. Emergency limits for the spraying pond are set as follows.

I-131	74	Bq/l
Cs-137	555	Bq/l
Cs-134	320	Bq/l
Sr-90	15	Bq/l

These limits have been designed taking into account dilution in the spraying ponds [4.2].

The amount of the imbalance water is determined by the flow rate of the shower water during the maximum quantity shift and constitutes 25 m^3/day for K2.

Provision is made for the following way of reprocessing of the liquid radioactive waste (LRW): temporary storage in the interim facility tanks to allow decay of short lived isotopes, with subsequent solidification at the available installations following temporary storage in interim facility tanks to allow for decay of short-lived isotopes. LRW is supplied to the solidification system.

Originally, it was proposed that provision should be made for using bitumenization as the solidification system, however, due to major design shortcomings and taking account of the hazardous nature of the process due to the possibility of fire, bitumenization was rejected.

In connection with the above, the UGU-1-500 high-degree evaporation installation is used for solidification of the stillage residue. This installation is designed for high-degree evaporation of the stillage residue with the resultant salt product packed into special containers and delivered for storage in a special building.

At present, in cooperation with 'NUKEM' work is in progress to develop a project for an integrated facility intended for the processing of radioactive waste. This would include installations for concentration, cementing and combustion of liquid radioactive waste.

4.8.3 Solidified liquid radioactive wastes

A deep evaporation plant (UGU-l-500) has been constructed for solidification of liquid radioactive waste. Characteristics of the salt fusion cake arising from this process, are as follows:

- main chemical constituents: pH 12; H_3BO_3; K^+; NH_3; NO^-_3: Na^+; Cl^-; SO_4^{2-};
- density: 1.93 g/cm^3;

- isotope composition: Cs-137 - 62%; Cs-134 - 35%; Co-60 - 1.4%, Mn-54, Co-58, Nb-95, Ag-110, Sb-124 < 1 %; and
- activity of salt fusion cake: $3.7\ 10^7$ Bq/l.

Throughout the time that UGU-l-500 has been in operation, 1120 full casks with a storage life of 3 years and 258 full containers with a storage life of 15 years have been accumulated.

The existing design excludes the possibility of any LRW release to the environment under normal work conditions. In an emergency the potential release is brought to a minimum.

The plant is equipped with ventilation and radiation control systems.

Casks filled with salt fusion cake are transported to the special radioactive waste handling building for temporary storage. A module-type storage facility for salt fusion cake containers is under construction.

As of 1 January 1996, the following LRW arisings were listed:

- casks accumulated: 1378
- total volume: 275.5 m^3
- total activity: $6.41\ 10^{12}$ Bq
- isotope composition:
  - Cs-137 approx. $2\ 10^{11}$ Bq/kg
  - Cs-134 approx. $1.5\ 10^{11}$ Bq/kg
  - Co-60 approx. 10^{11} Bq/kg

The chemical composition was as follows.

Cl (g/kg)	**Na (g/kg)**	**FeO (g/kg)**	**Ca (g/kg)**	**NO_3 (g/kg)**	**HBO (g/kg)**	**Humidity (%)**	**Density (g/cm^3)**
2.3	160	0.5	0.6	99	523	6-20	1.93

The status of the facility for storing LRW arising after processing of spent water and spent ion-exchange resins is summarised in Table 4.4 as of 31 December 1996 [4.7].

Table 4.4
Status of the facility for storing LWR as of 31 December 1996 [4.2]

Vessel type	Vessel No.	Capacity (m^3)	Content (m^3)	Boric acid Content (g/l)	Radioactivity content (Bq/l)	Salt concentration (g/l)
For fillers	OTW10 BO1	100	30	135	$7.0\ 10^6$	307
For fillers	OTW10 B02	100	90	15	$9.3\ 10^5$	30
For remains	OTW20 BO1	200	171	140	$1.4\ 10^7$	270
For remains	OTW20 OB02	200	135	126	$3.6\ 10^7$	270
Reserve volume	OTW30 BO1	200	184	83	$3.7\ 10^6$	240

4.8.4 Non-radioactive wastes

The proposed project will have an impact on wastes arising at the nearby town of Netishin. Other sources of non-radioactive waste are discussed in Section 6.6.1.

4.8.4.1 Solid household waste (SHW)

The expected amount of SHW taking into account development of the city is 11,600 t/yr.

The approximate composition of SHW (%) is as following:

- paper, carton - 30-38
- wood - 1-2.5
- bones - 0.5-2
- stone - 1-3
- textile - 3.5-4.5
- metal - 2-3
- plastic - 1.5-2
- leather, rubber - 1-5
- food waste - 30-39
- other - 0.5-1

The chemical composition of SHW has not been analysed but it is largely organic material.

4.8.4.2 Sewage sludges (SWS)

The quantity of SWS (dry weight) produced by the city's household sewage purification facilities amounts to:

- 1994: 932 t

- 1995: 813.8 t

The percentage composition of major chemical elements is as follows:

Ma - 0.14; Mg - 0.42; Ca - 5.98; Al - 1.0; Fe - 7.39; Si - 1.38; Cl - 0.1.

The bulk content of major nutrients in SWS is as follows:

- N (total) - 3.0
- P (total) - 2.97
- K (bulk) - 0.6

The bulk content of microelements in alluvia of the city of Netishin is as follows (mg/kg dry weight):

V - 188; Cr - 194; Ni - 352; Zn - 2377; As - 1.3; Bi - 19.6; Rb -16.8; Cu - 38.4.

The purification facilities have a site for composting sewage water sludges. Currently, the potential farming uses of biohumus obtained as a result of processing the organic part of SHW and SWS by a vermiculture is being investigated.

4.9 Spent fuel handling system

Spent fuel (SF) includes:

- the spent fuel assemblies (SFA);
- the absorber rods of the control and protection system (CPSAR); and
- the shim rods (SR).

4.9.1 Unloading spent fuel from reactor

Scheduled refuelling takes place once a year during a 3-year fuel lifetime, that is to say, each time about 1/3 of the reactor core (54-55 SFA, or approximately 25 t of uranium dioxide) is replaced. The procedure for the refuelling, rearrangement and replacement of spent fuel assemblies is determined by the reactor core loading chart and such information on fuel depletion as is obtained in the process of reactor operation.

In a scheduled or emergency refuelling of all the reactor core, 163 fuel assemblies (about 75 t of uranium dioxide) are unloaded from the reactor.

The SF is unloaded from the reactor by a remote-controlled refuelling machine under a layer of bioprotective water, without the presence of personnel in the refuelling area.

Prior to loading SFA into the spent fuel pool (SFP) for storage, fuel rod cladding is checked for leaktightness using an incorrect fuel assembly detection system (IFADS). Depending on the results of sample testing, the checked assemblies are transported from the IFADS compartments either to cells or sealed containers on the SFP rack.

Sealed containers with incorrect irradiated fuel assemblies are stored in the SFP up to the end of the design operation life of the power unit.

Refuelling technology envisages the use of combined technical means and organisational activities ensuring nuclear and radiation safety, and preventing the release of radioactive products and ionising radiation over and above the levels established by safety norms.

In transporting and refuelling, subcriticality is ensured, chiefly, by the design of the subsystem equipment.

Residual heat releases from SFA are removed by the SFP cool-down system.

4.9.2 Spent nuclear fuel storage (SNFS) in the cooling pond

After defuelling, spent fuel is stored in the reactor's SFP for at least three years. SF needs to be stored to allow activity to decay and residual heat releases to drop to admissible values such that it can be transported, and such that irradiated CPSAR and SR can be put into temporary storage.

The SFP is equipped with high-density fuel storage racks (HDFSR) designed by Skoda, the main structural element being casing made of corrosion-proof borate steel.

The total capacity of the SFP with HDFSR is sufficient for SF storage based on the possibility of parallel storage and cooling of SF for no less than 3 years plus accommodation of all the reactor core defuelled on a scheduled basis or in an emergency.

A maximum of 704 assemblies (approx. 320 t of uranium dioxide) can be stored on SFP racks at a time.

The SNFS subsystem in the SFP is comprised of the following equipment (taking into account equipment having an influence on the subsystem's operation):

- construction elements of SFP;
- HDFSR;
- SFP cool-down system;
- ventilation extraction system over the SFP;
- mechanical part (baskets) of the IFADS;
- sealed containers for damaged fuel assemblies;
- refuelling machine; and
- SFP water locks.

The SFP is located in a watertight part of the hall within the limits of the steam generator compartment between two of the circuit's circulation loops. It is connected to the upper part of the concrete vault by a refuelling channel designed to transport one fuel assembly at a time.

The top elevation (36.9 m) is determined by reactor design and the high mark of the protective water level on the active part of an SFA while the latter is transported down the refuelling channel. If it becomes necessary to maintain the upper level of water in the SFP (while defuelling the reactor or the SFP), the refuelling channel can be sealed by a water lock.

The floor elevation of the pool (20.7 m) was chosen to keep the protective mass of water (covering the tops of SFA - 2,800 mm -as they lie in storage) lower than the threshold of the refuelling channel and the partition between SFP bays, which makes it possible to store SFA in one bay while repair work is carried out in another.

To remove residual heat from SFA and to ensure fuel handling, the SFP is filled with a 16 g/kg solution of boric acid. In each SFP bay, water circulates through the cool-down system and the SFP water purification system connected with it. The design envisages a three-channel CP heat exchange system, all channels being functionally independent and physically isolated.

The characteristics of the equipment make it possible to ensure residual heat removal from SFA during fuel storage, scheduled refuelling and complete defuelling of the reactor core into the SFP.

To improve the radiation situation in the central hall, air (active vapours rising during SNF handling) is drawn off from the surface of the water level 36.2 m through exhaust windows in opposite walls of the SFP and into the reactor unit special ventilation system where it is treated as stated in Section 4.7.3.

4.9.3 Shipping spent fuel assemblies from the power unit

It is intended that SNF that has been held in the SFP for no less than three years will be shipped in special (TK-13-type) casks from the reactor unit to a reprocessing plant (principal design scheme) or to the SNFS facility (when it has been built) for intermediate storage for at least five years and subsequent transportation beyond the NPP**.**

4.9.4 Failed fuel

As described in Section 4.9.1, before defuelling, fuel assemblies are monitored for tightness and failed assemblies are moved to closed storage channels in the cooling ponds. Occasionally, fuel assemblies will fail during normal operation. However, operating experience at other VVER-1000 plants suggests that this failure rate is very low indeed; the average number of failed assemblies is one per refuelling period, the actual number for one cycle varying from 0 to 3.

4.10 Spent fuel transport and storage

4.10.1 Spent fuel transportation to reprocessing plant

The procedure for unloading SFA from the SFP includes the following operations: installation of an empty cask by a polar crane in the reactor unit into the refuelling compartment of the SFP, the loading of 12 leaktight SFA that have been aged no less than three years from the SFP racks into the cask, cask sealing, control of density and operating parameters, and subsequent transportation of a cask by the polar crane from the 36.2 m elevation mark of the central hall of the reactor unit, using the main and reserve elevation system of the crane, into the cask carriage standing in the transport corridor of the reactor unit.

Five or 6 casks (60-72 SFA) can be loaded by this means one by one and then put on a specially arranged holding track in the NPP territory, allowing a cask carriage train to be used, and then prepared for transportation from the NPP territory as part of a larger specialised train.

The fact that casks and cask carriages can be taken out of the reactor unit only after their external surfaces have been decontaminated to meet effective standards and specifications rules out the spread of radioactive substances outside the NPP territory.

The level of ionising radiation from the TK-13 cask when loaded with 12 VVER-1000 SFA aged three years in the SFP and with a maximum burnup of 50 GW/day/t U does not exceed the value of 0.1 mSv/hr at 1 m distance set by IAEA for such transport packages [4.3]. To comply with MKRZ recommendations, series #6, the allowable maximum dose rate for TK-13 containers is 1.2 mSv/hr (or for special conditions) up to 10 mSv/hr at any point on the surface.

4.10.2 Spent nuclear fuel storage

After the SNFS facility has been put into operation, SF will be transported from the reactor unit in TK-13 casks for intermediate storage (at least five years) in the facility prior to further shipping beyond the NPP boundaries.

The procedure for loading and transporting a cask filled with SF from the reactor unit to the SNFS facility is similar to the one described above i.e. containers will be taken one by one to the storage facility on the NPP's own cask carriage.

The SNFS will be located within the limits of the sanitary protection zone of the Khmelnitsky NPP. Its capacity will be 30 TK-13 containers. It will include the following buildings and structures:

- cask storage building;
- cask maintenance and control section;
- service railroad; and
- building for service personnel providing service, auxiliary (showers, etc.) and repair shops.

The SNFS will have all the conditions for proper cask arrangement to ensure compliance with safety standards and prevent insolation, as well as to ensure ease of handling and maintenance operations.

If necessary, the design of the SNFS allows for a future increase in its storage capacity.

In both normal operation and in a design-basis emergency, residual heat will be removed from the casks by a passive cooling system using natural convection of ambient air.

Elevation and transportation of loads will be performed in the SNFS by a general-purpose industrial gantry crane.

The SNFS will have railroad and automobile road links to inter-site transport communications, and will be equipped with radiation monitoring systems and engineering means for physical protection. It will also be equipped with an automatic fire alarm and working and emergency lighting systems.

The design levels of radiation inside and outside the SNFS under normal operation, calculated by type and characteristics of radiation from casks and a design capacity of 30 casks, ensure compliance with radiation safety norms.

4.10.3 Status of spent fuel storage capacity in cooling pond

When K2 is in operation, it will generate an average of 54 spent fuel assemblies at each refuelling campaign (every year), representing a weight of 25 tons. Taking into account the fact that it is stated that the cooling pond will be able to house 704 SFA (Section 4.9.2), this implies that there will be no problem of storage of spent fuel during the first 10 years of operation.

After this period, spent fuel assemblies will have to be removed from the cooling pond according to their period of storage and after a minimum period of storage of three years.

In 1998 spent fuel generated by Ukrainian NPP's was being shipped to Russia for reprocessing (Section 4.10.1), (1164 were shipped in 1995 for example). Due to financial aspects of this operation, it is envisaged that a temporary long term storage facility will be constructed at each Ukrainian VVER NPP. The concept consists of special dry storage containers (Section 4.10.2).

Another concept concerns a central nation-wide storage facility in the Chornobyl exclusion zone. TACIS projects are currently in preparation or in progress concerning this concept and facilities.

4.11 Decommissioning and dismantling

Whilst there are currently no specific Ukrainian regulations concerning decommissioning and no detailed plans for dismantling NPP's, this situation is not much different from that prevailing in much of Europe, especially as concerns plants whose construction started before 1980. According to regulations in force and in preparation in Ukraine, Goskomatom will be responsible for all decommissioning plans for Ukrainian VVER's.

An agreed package of regulatory documents dealing with the subject of decommissioning is currently in preparation. Various aspects of decommissioning are considered in existing laws as follows.

- The law on Nuclear Energy Use and Radiation Safety of 1995 [4.4] notes that decommissioning should be taken into account during design and construction of nuclear facilities and that plans for decommissioning should be subject to State expertise and approval by the Council of Ministers.
- The law on Radioactive Waste Management of 1995 [4.5] takes into account wastes arising from decommissioning.

Additionally, aspects of decommissioning requirements are taken into account in existing regulatory documents such as:

- 'sanitary rules on NPP design and operation'
- 'main statements on NPP safety'; and
- 'rules of radiation safety at NPP operation'.

It is also a requirement that, prior to commissioning of VVER reactors, the operator shall have demonstrated during design an assessment of different strategies for decommissioning.

There are presently two documents which are being elaborated. The first 'general requirements on NPPs decommissioning safety and research reactors', defines the purposes of decommissioning, the order of decommissioning, the stage of decommissioning, the planning of decommissioning, and general requirements for maintaining radiation safety, radioactive waste management, quality assurance and documentation. The second, 'conception of NPP decommissioning in Ukraine' will formulate the strategy and general solutions to decommissioning.

Ultimately, all the KNPP including Unit 2 will need to be decommissioned and dismantled. This will result in production of significant volumes of both non-radioactive and radioactive wastes depending on the extent of decommissioning carried out. Iourmanov and Zimin [4.6] for example, have estimated that the arising of waste from dismantling a VVER-1000 would amount to 12,000 t of concrete, 900 t of metallic constructions, and 6,000 t of equipment. The State programme on radioactive waste management [4.7] has determined that a base element of the national radioactive waste disposal system, including wastes arising from decommissioning, will be the central enterprise for processing radioactive waste. This organisation is intended to state its operations in the year 2000. Additionally a feasibility report has determined that it would be appropriate for an enterprise handling solid radioactive waste to be located within the Chornobyl zone. Clearly, decommissioning of the Chornobyl NPP will itself provide valuable experience and facilities appropriate to other reactors such as those at Khmelnitsky.

4.12 Environmental management

All operating units within the overall management structure of the Khmelnitsky NPP share responsibility for the environmental impacts of site activities. The different operating units are responsible for ensuring that their environmental performance meets the defined targets for the site as a whole (e.g. emission limits or discharge authorisation requirements).

The Environment Department is responsible for liaison with regulatory authorities. The Dosimetry Department is responsible for dosimetric measurement, including the environment dosimetry monitoring program on and off site.

Further information on environmental management is given in Sections 5 and 6.

4.13 References

4.1 GOST 2874-82. Drinking water. Hygienic standards and quality monitoring.
4.2 Information for updating EIAs (Parts 5 and 6). SSEC CSER, November 1997.
4.3 IAEA. Regulation for safe transport of radioactive material. 85/90/96 Safety Series No 6. IAEA, Vienna.
4.4 Law on nuclear energy use and radiation safety. N 39/95, 8 February 1995.
4.5 Law on radioactive waste management. N255/95, 30 June 1995.
4.6 Iuormanov, V. and Zimin, V. Characterisation of Russia's WWER including any decontamination to reduce radioactive inventory. In: Planning and Management of Decommissioning WWER NPP's Workshop. 25-29 November 1996, Hungary.
4.7 State programme on radioactive waste management (upto 2005). Cabinet of Ministers of Ukraine, decree N. 480, 29 April 1996.
4.8 Sanitation rules on NPP design and operation (SPAS-88)