4.9. Radiological Impact

4.9.1. Radiation Sources

The main radiation sources located in different installations of Cernavoda NPP-U3, respectively U4 are: reactor core, primary heat transport system, moderator circuit, fuelling machine, spent fuel bays, active circuit purification systems, radioactive waste storage systems, etc. (Ref. 4.9-6, 4.9-7).

The releases from the systems transporting radioactive fluids may represent air and surface contamination sources in the areas accessible for operating and maintenance personnel.

Tables 4.9.1-1 and 4.9.1-2 summarize the radiation sources from Reactor Building and Service Building, respectively; the tables contain information regarding the components of the systems transporting radioactive fluids, the type of radiation emitted and its origin, as well as the type of radionuclide that are present.

4.9.1.1. Reactor Core

The reactor core is a strong radiation source for alpha, beta, gamma and neutrons emitted by the radioactive nuclides produced in the fission process or capture reactions.

Most of the alpha and beta particles are absorbed in the core.

The neutrons resulted from the fission process lead to new fission processes or radiative capture.

Normally, the fission products are retained by the sheath of the fuel element, and only a small fraction can reach the coolant and can be deposited on any surface, resulting in a new radiation source.

The activation products arising in the coolant or in any other process fluid can be transported from a region with intense neutron flux (inside the core or close to the core) to a region with low neutron flux (outside the core).

The activation products could arise also in the structural components of the reactor and the reactivity control system.

Therefore, during reactor operation, the main types of radiation taken into consideration in the shielding calculations are neutrons and gamma radiation.

After reactor shutdown, the activation products are an important radiation source in the reactor and each component removed from the reactor must be considered radioactive.

At the reactor ends, in the fuelling machine area, the following radiation sources contribute to the dose rate after shutdown:

- Radionuclide produced from the reactions of neutrons with Co-59 and Mn-55 nuclei present in steel (end shield activation);
- The radioactive fission products arising inside the fuel elements from the core;
- The fission reactions due to the delayed neutrons and photoneutrons in the core (low power source);
- The activated corrosion products and the radioactive fission products deposited in the end fittings and the feeders.

To allow the personnel to work in the proximity of the reactor core, it is necessary to provide shields around the reactor core (primary shield) in order to reduce the radiation doses to acceptable values. Even in the zones where access is allowed only after reactor shutdown, it is necessary to reduce the radiation level due to the fission products from the fuel and due to activation of the structural materials.

The biological radiation shields have also the function of protecting the public against the radiation sources from the plant and to stop the degradation of the structural materials of the plant due to nuclear radiation.

4.9.1.2. Primary Heat Transport System

The activity of the fluid in the primary heat transport system is given by the fission products and activation products (including tritium).

a) Fission products

The primary heat transport circuit is the main source of radioactive materials as it contains the fuel; during fuel irradiation, fission products are formed in increasing quantities until the equilibrium state is achieved.

These fission products are retained within the fuel elements unless there is a sheath failure, in which case some of the fission products are released to the coolant.

At the end of the first year of normal reactor operation, irradiated fuel bundles are discharged from the reactor and replaced by new fuel bundles (continuous refueling). Thus an equilibrium inventory of fission products is essentially created in the primary circuit. The value of the inventory is determined primarily by the bundle irradiation average time.

Generally, if there are no sheath failures, there are essentially no fission products in the coolant. The only fission products present result from the "trace" quantities of uranium found in the materials of the components of the system. For example there is an average of about 1 ppm of uranium in the zirconium alloys used for pressure and calandria tubes, in CANDU reactors.

From CANDU-6 experience, it has resulted that only about 0.1% from fuel bundle sheaths may have fabrication defects. These defects result in the escape of only a small fraction of the total inventory of the fission products from the defected element in a bundle.

The defective bundles are removed on power from the reactor core, hence normally very low fission product inventory is present in the primary circuit. The released fission products, mainly the more volatile ones, enter the primary coolant and are distributed on the surfaces of the primary circuit and purification system, or remain in the primary coolant. Thus, the majority of the escaped fission products will decay within the boundaries of the primary circuit.

The estimated fission product activity concentrations within the primary circuit of CANDU-6 reactors are presented in Table 4.9.1-3. The half-life for each radionuclide is indicated in the brackets, since its presence in the radioactive waste resulted from primary circuit depends on the activity concentration as well as its half-life.

b) Activation products

Metals in contact with the coolant in the heat transport system corrode easily. Some of the corrosion products are dissolving or become suspended in the coolant and so may pass through the neutron flux in the reactor core. The neutron activation of some corrosion products occurs during their transit time through the reactor core. Subsequently, these activated corrosion products are likely to redeposit in other portions of the system. Since there is a continuous exchange of dissolved and suspended materials between the coolant and the surfaces of the primary heat transport system, there is a gradual build-up of radioactive material on the system surfaces outside the active zone, producing radiation fields. Some of the radioactive materials may escape from the primary circuit, becoming a secondary source of radiation. By selection of system components to the continuing PHT purification by filtration and ion exchange, to minimize the quantities of activated corrosion products in the system are minimized.

The estimated activity concentrations of the activation products within the primary circuit at CANDU-6 reactors are presented in Table 4.9.1-3.

The coolant itself and any impurity in the coolant can be activated with neutrons becoming radioactive.

The main radioactivity induced in coolant during reactor operation is given by the radionuclide N-16. Others important radionuclide are O-19 and F-17. These activation products have relatively short lifetimes: 7.1s for N-16, 26.3s for O-19 and 64.5s for F-17.

Shielding of the PHTS main equipment (feeders, headers, piping, steam generators, etc) is mainly dictated by the high gamma radiation levels from N-16 sources.

c) Tritium

Tritium is also produced in the coolant mainly by activation reaction of deuterium with the thermal neutron flux in reactor core.

The tritium activity in the primary coolant is affected by the capacity factor of the plant and by the operation period.

The estimated tritium activity concentration in the primary coolant of CANDU-6 reactors is presented in Table 4.9.1-3.

d) Carbon-14

Another radioactive isotope produced in primary coolant heavy water is Carbon-14, which is also produced by neutron activation reactions (mainly of O-17 isotope).

C-14 produced in the primary coolant is mainly in a bicarbonated form and it is easily removed by the ion-exchange resins used in the primary coolant purification system.

The estimated Carbon-14 concentration activity in the primary coolant of CANDU-6 reactors is presented in Table 4.9.1-3.

Generally, heavy water is an expensive fluid and conservation of heavy water has priority during the design as well as during the operation of CANDU plants. Thus primary coolant leakages are collected and/or recovered by the D_2O vapour recovery and active drainage systems. Therefore, although the leakage will carry both fission and activation products, the heavy water management systems serve as a means of control within the plant and as barriers to the release of radioactivity to the environment.

4.9.1.3. Primary Coolant Purification Circuit

For estimation of ion exchange resins activity in the primary coolant purification system, the fission and corrosion products are taken into consideration.

The estimated values of the concentration activity on primary circuit ion exchange resins, for each radionuclide, are presented in Table 4.9.1-4, for CANDU-6 reactors.

4.9.1.4. Main Moderator Circuit

The moderator system is the other major source of radioactive materials. However, the system operates at relatively low temperature and pressure and so has a much lower heavy water leakage rate than the primary circuit.

Heavy water in moderator system is in interaction with neutron fluxes inside the reactor core, and activation processes take place, which determine the intrinsic moderator activity.

On the other hand, the corrosion of the surfaces inside the core (already activated or which can be subsequently activated) and the corrosion products penetration in the moderator, lead to another component of the moderator activity.

Stainless steel and zirconium alloy are the principal materials used for the system components and thus the concentrations of corrosion products are generally lower than in the primary circuit.

The main contribution to the radioactivity induced in the moderator fluid during reactor operation is radionuclide N-16, O-19 and F-17 are also produced.

Concerning tritium activity, the following are mentioned: as the exposure period to neutron flux of the moderator heavy water is longer than the exposure of the primary coolant heavy water and, on the other hand, the thermal neutron fluxes in the moderator are higher than those in the coolant, the activation of the moderator heavy water leads to a higher concentration of tritium. In spite of this difference in tritium levels, the primary heat transport system makes a one third contribution to the emission of tritium, due to its relatively large leakage rate. Thus, the primary heat transport system has a significant contribution to the operation dose. It is essential to keep the primary system tritium concentration to a level as low as possible, and hence the requirement that the possibility of transferring moderator heavy water into the primary circuit must be minimized. To this end, there are separate heavy water collection systems, purification and separation of the moderator and coolant heavy water, being processed by the upgrading tower. There is an enclosure around the moderator pumps, valves and heat exchangers, which is maintained at a lower pressure than the rest of containment and is provided with a vapour recovery dryer.

Refering to Carbon-14, over 95% of its total activity is produced in the moderator heavy water by the activation of O-17. The most part of Carbon-14 produced in the moderator heavy water is transformed into bicarbonate, which then is removed by the purification system. More than 98% of the Carbon-14 produced in the moderator system is removed by the moderator purification ion exchangers.

The estimated radionuclide activity concentrations in the moderator system of CANDU-6 reactors are presented in Table 4.9.1-5.

4.9.1.5. Moderator Cover Gas System

The moderator cover gas system provides an inert gas covering the surface of the moderator in the calandria and overpressure relief ducts.

High purity helium is used for transporting the D_2 and O_2 resulted from radiolysis in the moderator through a recirculation loop designed to recombine them into D_2O, thus avoiding dangerous concentrations of potentially explosive D_2.

The helium used in the cover gas system has a purity of 99.99% He, and an argon impurity of not more than 1ppm. This prevents corrosion of the moderator system components and minimizes argon activation to Ar-41 in the system.

The D_2 concentration is maintained normally at less than 2% by recombination; exceptional increases of up to 4% are tolerated but lead to automatic action to further reduce D_2 concentration, to eliminate any possibility of a deuterium explosion.

Air ingress into the cover gas system (for example from outages with opening of the calandria, some residual air remains after purging) would increase nitrogen, oxygen and argon concentrations in the cover gas, so it is strictly controlled by design and closely monitored. Nitrogen is responsible for radiolytic production of nitric acid which encourages higher rates of radiolysis of D_2O and also consumes ion exchange resins.

In summary, the activity in the moderator cover gas system is partially due to Ar-41, the activation product of the reaction $^{40}Ar(n, \gamma)\ ^{41}Ar$; Ar-41 emits about 1.3 MeV (99%) gamma radiation and has a half-life of 1.83 hours.

The main source of activity in this system is the tritium, whose concentration in the cover gas is the same as in the moderator itself.

Carbon-14 is also produced in the moderator heavy water, mainly by the activation reaction of O-17. In the moderator cover gas system 99% of total C-14 is present as CO_2 while the organic compounds (methan-CH_4) is negligible. Carbon dioxide is removed by ion exchange purification resins.

4.9.1.6. Moderator Purification Circuit

The purification system is a closed circuit, which recirculates the moderator D_2O from the discharge header of the two moderator pumps through the filter and ion exchanger columns to the suction header of the moderator pumps.

The activity of the ion exchanger resins and filters in the system is due to the radionuclide N-16 and the principal corrosion products in the heavy water: Co-60, Mn-56 and Fe-59.

4.9.1.7. Liquid Zone Control System

The system consists of a light water subsystem and a helium subsystem.

The liquid zone control system is part of the reactor regulating system and a fraction of the system water is in the reactor during normal operation. The neutron flux activates the water and corrosion products, producing tritium and activation products.

A source of radiation is given by the radionuclide N-16 and O-19, which result as activation products in the light water passing through the reactor core.

Another source of radiation in the water subsystem is given by the steel corrosion products which lead to the production of Mn-56, Fe-59, Co-60 by activation, and the sodium impurities leading to Na-24 production. These activation products are retained on the ion exchanger resins in the circuit.

As the water subsystem contains light water, tritium concentrations are much smaller than in the primary coolant or moderator. At the helium subsystem, the ingress of air into this subsystem can lead to Ar-41 production.

The estimated activity concentrations of the radioactive products in the liquid zone control system of CANDU-6 reactors are given in Table 4.9.1-6.

This activity may escape with any system leakage and add to the activity entering the radioactive liquid waste management system; from the CANDU-6 operating experience it has resulted that the system does not contribute significantly to the total activity of liquid radioactivity wastes.

4.9.1.8. End Shield Cooling System

The shield cooling system recirculates the light water from the calandria vault, which is irradiated by reactor residual neutron flux.

The water contains activated products and a very low amount of tritium.

The origins of the radiation sources that are present in the system water are:

a) Induced radioactivity, due to neutron activation of the oxygen from the water (the main radionuclides are N-16 and O-19);

b) Activated corrosion products: the water from the system corrodes the component surfaces and the corrosion products being transported are activated in the neutron flux, inside the end shields and the calandria vault;

c) Activated impurities: although the water entering the circuit is demineralized, it still contains a small fraction of impurities, which is activated; also, ^{41}Ar, produced by the activation of ^{40}Ar dissolved in the water, can be present.

This radioactivity may escape with leakage of shield cooling water and reaches the radioactive liquid waste management system. From CANDU reactor operating experience it has resulted that the shield cooling system does not contribute significantly to the total activity of the liquid waste system.

The estimated specific activity at equilibrium of the isotopes in the shield cooling system for CANDU-6 reactors is presented in Table 4.9.1-7.

4.9.1.9. D_2O Supply System

The tanks of the system are provided to store the reactor initial heavy water load. They also contain the primary coolant and moderator heavy water due to the leakages resulted during the plant operation.

For the evaluation of the source term the most unfavorable situation has been assumed, when the tanks are filled with D_2O from the primary circuit.

Therefore, the source activity is the same as the activity of the D_2O cleanup system supply tanks (see Table 4.9.1-8).

4.9.1.10. D_2O Cleanup System

Generally, the heavy water recovered from leakage, spills and deuteration - dedeuteration system ion exchange resins must be cleaned up before being upgraded.

The cleanup system is designed to remove particulate, dissolved and organic ionic impurities from recovered heavy water.

The water to be cleaned comes especially from the primary coolant and moderator collection systems, D_2O vapour recovery system and moderator and primary coolant deuteration and dedeuteration systems.

The estimated values of radionuclide activity concentration in the components of the D_2O cleanup system for CANDU-6 reactors are presented in Table 4.9.1-8.

4.9.1.11. D_2O Vapour Recovery System

The D_2O vapour recovery system recovers the heavy water leaks from the equipment located in the Reactor Building. During the processing of heavy water vapours, the tritium and other airborne contaminants or noble gases are absorbed in the dryer desiccant beds.

The presence of some radioactive substances in air (iodine, noble gases, tritium, particulates) due to leakages from process systems containing radioactive fluids is anticipated in some areas in the Reactor Building and in the Service Building.

The concentrations of radionuclides in air in the heavy water vapour recovery areas in Reactor Building are presented in Tables 4.9.1-9 ÷ 4.9.1-12 for the following working areas:

- accessible areas in the Reactor Building (Table 4.9.1-9);
- steam generators room (Table 4.9.1-10);
- F/M vault and moderator room (Table 4.9.1-11);
- moderator equipment enclosure (Table 4.9.1-12).

4.9.1.12. Annulus Gas System

The annulus gas acts as a thermal barrier between the pressure tube and the calandria tube. The annular space is filled with CO_2 which produces an inert atmosphere. CO_2 was selected for its low argon and nitrogen impurity levels, ensuring that there is little Ar-41 production in the annulus gas. Another radioisotope produced in the annulus gas system in a small concentration of activity, is C-14.

The estimated activity concentration of Ar-41 in the annulus gas system, based on the existing data from CANDU-6 plants with CO_2 as annulus gas is 25 kBq/cm^3.

4.9.1.13. Fuel Handling Systems

The new fuel and spent fuel handling systems are both potential sources of radioactive materials. In the first case (new fuel handling), the source is considered negligible. New fuel contains small quantities of uranium daughter products but since the fuel is enclosed in a sheath, only very little radioactive material can escape from sheath, the total annual escape being negligible.

The spent fuel handling system discharges fuel bundles containing large inventories of fission products.

Generally, fuelling operations do not result in any significant release of radioactive material because the fuel sheath is intact.

CANDU-6 operating experience shows that about five bundles with fuel sheath defects are handled per year (0.1% failure rate); in this case, taking into account the

noble gases released during a fuel handling sequence, the annual activity released from the defective fuel bundles is comparable with that corresponding to the intact fuel.

The activity of C-14 produced within the fuel elements is about 3.3% of the total C-14 production; considering the small rate of fuel sheath defects, the activity of C-14 produced in the fuel which could be released in the primary system is negligible compared with the coolant activity.

4.9.1.14. Moderator and Primary Coolant Deuteration- Dedeuteration Systems

The moderator and primary coolant deuteration-dedeuteration systems present a risk for contamination with tritium, because the water in these systems is strongly tritiated.

The tritium activity of the water collected by the radioactive liquid waste system is approximately 10 TBq/year from the primary heat transfer deuteration-dedeuteration system and ~ 60 TBq/year from the moderator deuteration-dedeuteration system. Tritium as well as liquid and gaseous waste releases are monitored.

As concerns the gamma radiation level, this rises in the operating zones during the transfer of the spent resin from the ion exchanger to the dedeuteration tank and from there to the spent resin storage tank.

4.9.1.15. Spent Fuel Bays

The spent fuel bays constitute external irradiation sources, as well as contamination sources. The fuel bundles extracted from the reactor and temporary stored inside the bays are gamma radiation sources, whose intensity depends of the fuel cooling time.

On the other hand, there is another potential radiation source at the bays, consisting of the fission products that can be released from failed fuel bundles in the reception bay and failed fuel bay (the presence of failed fuel bundles inside the main storage bay is not expected).

Although the purification system removes most of the solid fission products, some of them remain inside the bay water. In addition, part of the gaseous fission products

are dissolved in the water and part of them are in the air above the bay (the rest is driven through the ventilation system).

The activity concentrations of the radionuclides inside the water of the main spent fuel bay estimated for CANDU-6 reactors are presented in Table 4.9.1-13.

4.9.1.16. Spent Fuel Bay Cooling and Purification System

The spent fuel bay stores the fuel bundles discharged from the core. The bay water provides both shielding and cooling of the bundles.

Fission products from the defective spent fuel bundles and radioactive corrosion products released with impurities from the fuel element surface may enter the water of the spent fuel bays.

Also, the transfer of radioactive materials in the bay water with the small quantity of heavy water carried over with the fuel bundle should be mentioned.

All these contribute to the inventory of radionuclides in the bay water (see Table 4.9.1-13).

The specific activity accumulated in the ion exchange resins is due to fission products; the estimated values for CANDU-6 reactors are:

- I-131 (8.05 days) 1.11 E+6 kBq/kg
- Cs-134 (2.05 years) 1.58 E+6 kBq/kg
- Cs-137 (30 years) 5.71 E+6 kBq/kg

CANDU-6 operating experience demonstrated that the radioactivity accumulated in the pumps and heat exchangers is negligible.

4.9.1.17. Spent Resin Storage Vaults

The ion exchange spent resins are transferred from the various process systems (e.g. primary circuit purification, D_2O clean-up, spent fuel bays, end shield cooling system, liquid zone control system, etc.), via the resin transfer system, and temporary stored into the storage concrete vaults (room S-020 from Service Building).

The spent resins activity is mainly due to radionuclides Cs-137, Cs-134 and Co-60. The activity of all resin source systems is presented in Table 4.9.1-14. The values indicated in the table are estimated based on CANDU-6 operating experience.

It has been assumed that the activities in the resins from the liquid zone control system and fueling machine auxiliary system are negligible.

4.9.1.18. Radioactive Liquid Waste Management System

The radioactive liquid waste management system is designed for the collection, storage, treatment (when necessary) and removing of any radioactive liquid waste produced in the plant. The main components of the system are the liquid waste storage concrete tanks located in the basement of the Service Building and filter-ion exchanger assembly.

It is conservatively assumed that the tanks contain liquid wastes with activity of 370 kBq/l. This assumption is conservative because decontamination operation is performed if the liquid wastes activity is higher than 5 kBq/l.

The estimated specific activity in the tanks and the filter-ion exchanger assembly for CANDU-6 reactors is presented in Table 4.9.1-15.

4.9.1.19. Ventilation Systems

The plant ventilation systems which are sources of radioactivity are: the Reactor Building (RB) Ventilation System and the Spent Fuel Bay Ventilation System. The Service Building (SB) Ventilation System, located inside and outside of the Service Building, does not contain and release radioactivity in the environment during normal operation.

1) The Reactor Building Ventilation System located in the RB and SB transports contaminated fluid during reactor normal operation.

The contamination removal is controlled through the RB Ventilation System and D_2O Vapour Recovery System together with nuclear zoning procedural control, buffer zones, change rooms, protective clothing (plastic suits, etc), etc.

The Reactor Building Ventilation System maintains the sub-atmospheric pressure in the Reactor Building to prevent the radioactive releases through the airlock or the access doors. A differential pressure is also maintained to direct all air circulations from the low contamination areas to the high contamination areas.

The air is filtered to retain radioactive particles before discharging to the ventilation stack. Therefore the filters become themselves sources of radiation and contamination, and then radioactive solid wastes.

The controlled discharging into the environment of the air recovered from RB rooms is performed through the plant exhaust stack, after a proper filtration and monitoring.

Radionuclides, as particulates (e.g. Cr-51) and iodine isotopes, remain in filters and filter cartridges of the Reactor Building Ventilation System.

In order to reduce heavy water downgrading and for improvement of the D_2O vapor recovery efficiency and tritium collection, additional dryers are installed on the inlet air filter in the RB Ventilation System. These will reduce the amount of light water entered in the RB and the quantity of light water collected by the existent vapor recovery dryers.

Therefore, the isotopic content of the condensed D_2O vapors is increased and the tritium released to environment decreases.

2) The Spent Fuel Bay Ventilation System is located both in the Reactor Building and in the Service Building; it transports contaminated fluid and radioactivity outside the containment during normal operation.

In order to retain the radioactive materials of the air coming from the spent fuel transfer and storage bays area, the filter unit ensures filter parameters higher than 99.99% for the radioactive elemental iodine, 99.9% for the radioactive methyl iodide and 99.97% for the radioactive airborne.

4.9.1.20. Steam Generators and Feedwater Systems

During normal plant operation there is not primary to secondary circuit coolant leakage; the only radioactive material which is transferred (by diffusion) to the secondary circuit is tritium.

Any steam generator tube failure results in the transfer of some of the coolant together with the associated radionuclides from the primary circuit to the steam generator secondary side. This results in the releases of radionuclides to the environment, but the permissible limits are not exceeded. The occurrences of such failures require the reactor shutdown in order to locate and repair the failed tubes.

4.9.1.21. Decontamination Centre

The decontamination centre of Cernavoda NPP U3, respectively U4 is designed for centralized routine decontamination of the reusable materials and equipment, such as fueling machine components, small metallic components (fittings), plastic protective clothing, breathing devices, D_2O drums, flasks, etc.

The decontamination equipment is arranged to allow larger objects to be handled using special procedures.

The equipment allows remote handling of the components for complete decontamination of the fixed or non-fixed contamination before maintenance and/or repairs.

Fume-hoods, benches, sinks are provided for any required manual cleaning; they are located in the low radioactivity areas of the decontamination centre.

The materials and equipment entering the decontamination centre are monitored to check radioactivity levels. The cleaned (washed) articles are monitored before being in service.

The cleaning solutions and wash water became contaminated are removed by the decontamination centre drains, connected to the radioactive liquid waste management system. The aims of each method of decontamination are:

- to minimize radiation fields by maximizing the removal of radioactive material;
- to minimize the cost of the decontamination and the consequential radiation exposure.

The decontamination methods depend on the level of equipment contamination and type of contamination, non-fixed or fixed. Should some equipment be shipped off-

site, non-fixed contamination is not allowed and fixed contamination should meet the permissible limits for radioactive materials transfer.

CANDU practices used for heavy and small equipment decontamination are presented below:

a) Heavy Equipment

- A plastic material sorting area is set up, with controlled access;
- Depending on the level and type of contamination, the decontamination means are the use of vacuum, sandpaper, grinding, solvents and chemicals;
- - Ventilation tents with a filter train to retain contaminants may need to be installed (depending on contamination level). After utilization, the spent filters become solid radioactive waste;
- The liquids from the decontamination are transferred to the radioactive liquid waste management system.

b) Small Equipment

- Decontamination practices may include combinations of these, a vacuum cleaning, grinding, chemicals, etc.

Since the equipment brought to the centre are generally dry, there is a very small quantity of heavy water, and hence tritium, on its surface, so that the liquid and gaseous effluents from the decontamination centre will contain very little tritium.

The D_2O leakage is minimized from economic reasons also; practically, the leakage higher than 1% D_2O in H_2O imposes liquid waste recovery to be upgraded. Generally, the radionuclides adhering to the surface of the equipment are the long-lived fission and activation products.

During the normal operation of the plant, the high activity wastes should be separated from the low activity ones; liquid wastes activity should be measured prior to be transferred to the radioactive liquid waste management system. In this way, solutions with high activity may be segregated, treated and/or stored.

The small volumes produced in the decontamination centre are much easier to be treated than those that are diluted into a large volume, such as the volume of one of tanks in the radioactive liquid waste management system.

Table 4.9.1-1. Radiation Sources in Reactor Building

No.	Radiation source	Radiation type and origin	Radionuclide type
1.	Reactor core	- fission gamma radiation - fission neutrons - capture gamma radiation - gamma and beta radiation from fission products	Fission products of ^{235}U $^{60}Co.$,^{56}Mn, ^{59}Fe
2.	Primary Heat Transport System (BSI 33100) -steam generators -headers -pumps -pipes -pipes	- gamma and beta radiation from fission products - gamma and beta radiation from activation - intrinsic - corrosion - photoneutrons	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs ^{16}N, ^{14}C, T ^{24}Na, ^{59}Fe, ^{60}Co, ^{65}Zn, ^{58}Co
3.	Moderator Main Circuit (BSI 32110) - heat exchangers - pipes - pump	- gamma and beta radiation from activation products - intrinsic - impurities - corrosion - photoneutrons	 ^{16}N, ^{14}C, T ^{41}Ar, ^{14}C ^{24}Na, ^{51}Cr, ^{56}Mn, ^{59}Fe, ^{60}Co, ^{65}Zn
4.	Fuelling Machine System (BSI 35200)	- gamma and beta radiation from fission products	Fission products of ^{235}U
5.	Liquid Zone Control System (BSI 34810) a) Water subsystem - H_2O return lines - H_2O return header - delay tank - ion exchangers - heat exchangers - H_2O pumps	- gamma and beta radiation from activation products - intrinsic - impurities - corrosion	 ^{16}N, ^{19}O, T ^{41}Ar ^{24}Na, ^{51}Cr, ^{56}Mn, ^{59}Fe, ^{60}Co

Table 4.9.1-1. Radiation Sources in Reactor Building (continued)

No.	Radiation source	Radiation type and origin	Radionuclide type
6.	PHT Purification System (BSI 33350 - ion exchanger - filters - cooler - heat exchanger	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{131}I, ^{137}Cs, ^{140}La, ^{140}Ba ^{99}Mo, ^{134}Cs, ^{133}I T ^{59}Fe, ^{60}Co, ^{58}Co
7.	D_2O Feed and Bleed, Pressure and Inventory Control System (BSI 33300) - pressurizer - D_2O feed pumps - heat exchanger	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion - photoneutrons	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs ^{16}N, ^{19}O, T ^{24}Na, ^{58}Co, ^{59}Fe, ^{60}Co, ^{65}Zn
8.	Coolant Storage, Transfer and Recovery System (BSI 33330 - storage tank - recovery tank - pumps	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs T ^{24}Na, ^{58}Co, ^{59}Fe, ^{60}Co, ^{65}Zn

Table 4.9.1-1. Radiation Sources in Reactor Building (continued)

No.	Radiation source	Radiation type and origin	Radionuclide type
9.	D_2O Collection System (BSI 33810) - pumps - tank - heat exchanger - condenser	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs T ^{24}Na, ^{58}Co, ^{59}Fe, ^{60}Co, ^{65}Zn
10.	Annulus gas system (BSI 34980) - CO_2 supply lines - CO_2 bottles	- gamma and beta radiation from activation products	^{41}Ar, ^{14}C
11.	D_2O Moderator Collection System (BSI 32510) - tank - pump	- gamma and beta radiation from activation products - intrinsic - corrosion	 T ^{24}Na, ^{51}Cr, ^{56}Mn, ^{59}Fe, ^{60}Co, ^{65}Zn
12.	Moderator Cover Gas System (32310) - Helium supply lines - Helium bottles	- gamma and beta radiation from activation products	^{41}Ar, T
13.	PHT Deuteration and Dedeuteration System (BSI 33360) - dedeuteration tank - resin transfer pump	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{131}I, ^{137}Cs, ^{140}La, ^{140}Ba ^{99}Mo, ^{134}Cs, ^{133}I T ^{59}Fe, ^{60}Co, ^{58}Co

Table 4.9.1-1. Radiation Sources in Reactor Building (continued)

No.	Radiation source	Radiation type and origin	Radionuclide type
14.	PHT Pump Gland Seal System (BSI 33340) - filters	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion - photoneutrons	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe^{141}Ce, ^{144}Ce, ^{144}Cs, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La ^{16}N, ^{14}C, T ^{24}Na, ^{59}Fe, ^{60}Co, ^{65}Zn, ^{58}Co
15.	Shutdown Cooling System (BSI 33410) -heat exchangers -pumps	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion - photoneutrons	^{85}Kr, ^{85m}Kr, ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba,^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs T ^{24}N, ^{59}Fe, ^{60}Co, ^{65}Zn, ^{58}Co

Table 4.9.1-2. Radiation Sources in Service Building

No.	Radiation source	Radiation type and origin	Radionuclide type
1.	Moderator Purification System (BSI 32210) - ion exchangers - filter - heat exchanger - pipes	- gamma and beta radiation from activation products - intrinsic - corrosion	 ^{16}N, T ^{58}Co, ^{59}Fe, ^{60}Co
2.	Moderator Deuteration and Dedeuteration System (BSI 32220) - dedeuteration tank	- gamma and beta radiation from activation products - intrinsic - corrosion	 ^{16}N, T ^{58}Co, ^{59}Fe, ^{60}Co
3.	Spent Fuel Bays (34400) - reception bay - main storage bay - failed fuel bay	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{95}Nb, ^{95}Zr, ^{131}I, ^{133}I, ^{124}Sb, ^{134}Cs, ^{137}Cs T ^{56}Mn, ^{60}Co
4.	Spent Fuel Bay Cooling and Purification System (BSI 34410) - ion exchanger - heat exchangers - pumps - filters	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{131}I, ^{134}Cs, ^{137}Cs T ^{56}Mn, ^{60}Co
5.	End Shield Cooling System (BSI 34110) -pumps -heat exchangers -ion exchanger delay tank (in Reactor Building, room R-111) - buffer tank	gamma and beta radiation from activation products - intrinsic - impurities - corrosion	 ^{16}N, ^{19}O, T ^{41}Ar, ^{24}Na ^{56}Mn, ^{64}Cu, ^{59}Fe, ^{60}Co, ^{65}Zn

Table 4.9.1-2. Radiation Sources in Service Building (continued)

No.	Radiation source	Radiation type and origin	Radionuclide type
6.	D_2O Cleanup System (BSI 38410) -supply tanks -ion exchangers -filters - downgraded D_2O storage tank	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{99}Mo, ^{131}I, ^{133}I, ^{134}Cs, ^{137}Cs, ^{140}La, ^{140}Ba T ^{58}Co, ^{59}Fe, ^{60}Co
7.	D_2O Supply System (BSI 38110) - D2O storage tanks	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{99}Mo, ^{131}I, ^{133}I, ^{134}Cs, ^{137}Cs, ^{140}La, ^{140}Ba T ^{58}Co, ^{59}Fe, ^{60}Co
8.	Liquid Radioactive Waste Handling System (BSI 79210) - tanks - filter/ion exchanger unit - pumps	- gamma and beta radiation from fission products - gamma and beta radiation from activation products - intrinsic - corrosion	^{99}Mo, ^{131}I, ^{134}Cs, ^{137}Cs, ^{144}Ce T ^{51}Cr, ^{60}Co
9.	Spent Resin Storage System (BSI 79140) - storage bays	- gamma and beta radiation from fission products - gamma and beta radiation from activated corrosion products	^{134}Cs, ^{137}Cs ^{60}Co
10.	D_2O Vapour Recovery System (BSI 38310) - dryers - tanks - pumps	- gamma and beta radiation from fission products - beta radiation from activation products	^{85}Kr, ^{85m}Kr ^{88}Kr, ^{95}Zr, ^{95}Nb, ^{99}Mo, ^{131}I, ^{133}I, ^{134}I, ^{135}I, ^{133m}Xe, ^{133}Xe, ^{135}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{144}Cs T

Table 4.9.1-3. Radioactive Product Activity Concentrations in the Primary Heat Transport System – Estimated Values for CANDU-6 Reactors

Radioactive Product (half-life)		**Activity Concentration (Bq/kg)**
Fission Products:		
Krypton – 85	(10.76 y)	3.30 E+5
Krypton – 85 m	(4.56 h)	1.32 E+5
Krypton – 88	(2.84 h)	3.36 E+5
Zirconium – 95	(64.03 d)	4.40 E+4
Niobium – 95	(34.98 d)	5.20 E+4
Molybdenum – 99	(66.7 h)	3.00 E+4
Iodine – 131	(8.05 d)	5.73 E+5
Iodine – 133	(20.05 h)	1.66 E+6
Iodine – 134	(52.6 m)	5.69 E+6
Iodine – 135	(6.61 h)	2.56 E+6
Xenon – 133 m	(2.19 d)	8.70 E+4
Xenon – 133	(5.25 d)	6.89 E+4
Xenon – 135	(9.09 h)	3.25 E+6
Cesium – 137	(30y)	1.90 E+4
Barium-140	(12.8 d)	1.10 E+4
Lantan – 140	(40.3 h)	
Cerium – 141	(32.5 d)	1.50 E+3
Cerium – 144	(284.9 d)	2.20 E+3
Activation Products		
Sodium – 24	(15 h)	9.80 E+4
Iron – 59	(45 d)	3.51 E+5
Cobalt – 60	(5.28 y)	1.30 E+4
Zinc – 65	(244 d)	1.10 E+3
Carbon-14	(5730 y)	1.50 E+3
Nitrogen -16	(7.245 s)	2.35 E+9
Tritium	(12.33 y)	
After 7.5 operating years		3.70 E+10
After 15 operating years		6.70 E+10
After 30 operating years		9.20 E+10

Table 4.9.1-4. Activity Concentration of Ion Exchange Resins in the Primary Coolant Purification System - Estimated Values for CANDU-6 Reactors

Radionuclide (half-life)	Activity Concentration (Bq/cm^3)
Co – 60 (5.28 y)	5.86 E+4
Fe – 59 (45 d)	3.40 E+4
Co – 58 (71 d)	1.91 E+4
I – 131 (8.05 d)	3.57 E+7
Cs – 137 (30 y)	4.71 E+6
La – 140 (40.3 h)	7.60 E+4
Ba – 140 (12.8 d)	5.29 E+5
Mo – 99 (66.7 h)	8.39 E+5
Cs – 134 (2.05 y)	2.31 E+6
I – 133 (20.05 h)	9.68 E+6

Table 4.9.1-5. Radioactive Product Activity Concentrations in the Moderator System - Estimated Values for CANDU-6 Reactors

Radioactive Products (half-life)	Activity Concentration (Bq/kg)
Activation Products	
Sodium - 24 (15 h)	2.30 E+4
Argon - 41 (1.83 h)	1.80 E+5
Chromium - 51 (27.7 d)	5.90 E+4
Manganese– 56 (2.58 h)	1.70 E+6
Iron – 59 (45 d)	3.00 E+4
Cobalt - 60 (5.28 y)	9.60 E+3
Zinc – 65 (244 d)	4.80 E+4
Nitrogen – 16 (7.245 s)	3.13 E+9
Tritium (12.33 y)	
After 7.5 operating years	7.40 E+11
After 15 operating years	1.40 E+12
After 30 operating years	2.00 E+12
Carbon – 14 (5730 y)	3.70 E+4

Table 4.9.1-6. Radioactive Product Activity Concentrations for The Liquid Zone Control System – Estimated Values for CANDU-6 Reactors

Radioactive Products (half-life)	Activity Concentration (Bq/kg)
Activation Products:	
Sodium – 24 (15 h)	9.30 E+3
Chromium – 51 (27.7 d)	1.90 E+4
Manganese – 56 (2.58 h)	1.50 E+5
Iron – 59 (45 d)	1.90 E+3
Cobalt – 60 (5.28 y)	1.10 E+3
Argon – 41 (1.83 h)	3.55 E+3 (Bq/cm^3)
Nitrogen– 16 (7.245 s)	2.34 E+7 (Bq/cm^3)
Tritium (12.33 y)	
At the end of plant life	1.80 E+6

Table 4.9.1-7. Radionuclide Activity Concentrations in The End Shield Cooling System – Estimated Values for CANDU-6 Reactors

Radionuclide (half-life)	Activity Concentration (Bq/kg)		
	Cooling Water	Pumps /heat exchanger	Ion Exchanger
N-16 (7.245 s)	3.54 E+5	1.30 E+5	-
O-19 (27.15 y)	1.64 E+5	1.25 E+5	-
Mn-56 (2.58 h)	3.27 E+2	3.27 E+2	1.67 E+5
Cu-64 (12.8 h)	7.29 E+3	7.29 E+3	1.86 E+7
Fe-59 (45 d)	2.21 E+0	2.21 E+0	3.60 E+5
Co-60 (5.28 y)	1.37 E+0	1.37 E+0	4.01 E+5
Zn-65 (244 d)	3.40 E+2	3.40 E+2	8.93 E+7
Ar-41 (1.83 h)	7.29 E+4	7.29 E+4	-
Na-24 (15 h)	3.16 E+3	3.16 E+3	9.38 E+6

Table 4.9.1-8. Radionuclide Activity Concentrations in the D_2O Clean-up System – Estimated Values for CANDU-6 Reactors

Radionuclide (half-life)		Activity Concentration (kBq/kg)	
		Tanks	Ion Exchangers
Co-60	(5.28 y)	4.63 E-1	2.32 E-1
Fe-59	(45 d)	3.70 E-1	1.85 E-1
Co-58	(71 d)	1.85 E-1	9.30 E-2
I-131	(8.05 d)	1.11 E+3	5.50 E+2
I-133	(20.05 h)	2.78 E+3	1.39 E+3
Cs-137	(30 y)	3.70 E+1	1.85 E+1
Cs-134	(2.05 y)	1.85 E+1	9.30 E+0
La-140	(40.3 h)	1.11 E+1	5.5 E+0
Ba-140	(12.8 d)	1.11 E+1	5.50 E+0
Mo-99	(66.7 h)	7.40 E+1	3.70 E+1

Table 4.9.1-9. Radionuclide Activity Concentrations in Air in Accessible Areas of the Reactor Building-Estimated Values for CANDU-6 Reactors

No.	Radionuclide (half-life)		Activity Concentration in Air (Bq/m^3)
1	Kr-83m	(1.83 h)	2.64 E+1
2	Kr-85m	(4.56 h)	3.34 E+2
3	Kr-85	(10.76 y)	2.99 E+3
4	Kr-87	(1.26 h)	6.34 E+1
5	Kr-88	(2.84 h)	3.92 E+2
6	Kr-89	(3.17 m)	2.19 E-1
7	Xe-131m	(12 d)	1.95 E+3
8	Xe-133m	(2.19 d)	1.84 E+3
9	Xe-133	(5.25 d)	1.62 E+5
10	Xe-135m	(15.7 m)	1.18 E+0
11	Xe-135	(9.09 h)	2.15 E+2
12	Xe-137	(3.83 m)	4.15 E-1
13	Xe-138	(14.13 m)	5.76 E+0
14	I-131	(8.05 d)	1.42 E+1
15	I-133	(20.05 h)	2.17 E+1
16	I-130	(12.36 h)	0
17	I-132	(2.28 h)	2.23 E+0
18	I-134	(52.6 m)	1.02 E+0
19	I-135	(6.61 h)	9.79 E+0
20	H-3	(12.33 y)	1.98 E+7

Table 4.9.1-10. Radionuclide Activity Concentrations in Air in Steam Generator Room - Estimated Values for CANDU-6 Reactors

No.	Radionuclide (half-life)		Activity Concentration in Air (Bq/m^3)
1	Kr-83m	(1.83 h)	7.05 E-1
2	Kr-85m	(4.56 h)	9.81 E+0
3	Kr-85	(10.76 y)	2.31 E+2
4	Kr-87	(1.26 h)	1.65 E+0
5	Kr-88	(2.84 h)	1.09 E+1
6	Kr-89	(3.17 m)	5.42 E-3
7	Xe-131m	(12 d)	1.36 E+2
8	Xe-133m	(2.19 d)	9.81 E+1
9	Xe-133	(5.25 d)	1.03 E+4
10	Xe-135m	(15.7 m)	2.96 E-2
11	Xe-135	(9.09 h)	7.17 E+0
12	Xe-137	(3.83 m)	1.03 E-2
13	Xe-138	(14.13 m)	1.44 E-1
14	I-131	(8.05 d)	9.46 E-1
15	I-133	(20.05 h)	8.95 E-1
16	I-130	(12.36 h)	0
17	I-132	(2.28 h)	6.05 E-2
18	I-134	(52.6 m)	2.61 E-2
19	I-135	(6.61 h)	3.07 E-1
20	H-3	(12.33 y)	1.52 E+7
21	Ar-41	(1.83 h)	0

Table 4.9.1-11. Radionuclide Activity Concentrations in Air in the F/M Vault and Moderator Room - Estimated Values for CANDU-6 Reactors

No.	Radionuclide (half-life)		Activity Concentration in Air (Bq/m^3)
1	Kr-83m	(1.83 h)	2.01 E+2
2	Kr-85m	(4.56 h)	2.22 E+3
3	Kr-85	(10.76 y)	1.19 E+4
4	Kr-87	(1.26 h)	5.01 E+2
5	Kr-88	(2.84 h)	2.82 E+3
6	Kr-89	(3.17 m)	1.92 E+0
7	Xe-131m	(12 d)	7.93 E+3
8	Xe-133m	(2.19 d)	8.14 E+3
9	Xe-133	(5.25 d)	6.75 E+5
10	Xe-135m	(15.7 m)	1.02 E+1
11	Xe-135	(9.09 h)	1.25 E+3
12	Xe-137	(3.83 m)	3.62 E+0
13	Xe-138	(14.13 m)	4.95 E+1
14	I-131	(8.05 d)	5.78 E+1
15	I-133	(20.05 h)	1.08 E+2
16	I-130	(12.36 h)	0
17	I-132	(2.28 h)	1.65 E+1
18	I-134	(52.6 m)	8.29 E+0
19	I-135	(6.61 h)	6.05 E+1
20	H-3	(12.33 y)	4.16 E+7
21	Ar-41	(1.83 h)	1.87 E+5

Table 4.9.1-12. Radionuclide Activity Concentrations in Moderator Equipment Enclosure - Estimated Values for CANDU-6 Reactors

No.	Radionuclide (half-life)	Activity Concentration in Air (Bq/m^3)
1	Kr-83m (1.83 h)	2.65 E+1
2	Kr-85m (4.56 h)	6.01 E+2
3	Kr-85 (10.76 y)	1.19 E+4
4	Kr-87 (1.26 h)	4.79 E+1
5	Kr-88 (2.84 h)	5.30 E+2
6	Kr-89 (3.17 m)	8.40 E-3
7	Xe-131m (12 d)	7.59 E+3
8	Xe-133m (2.19 d)	6.65 E+3
9	Xe-133 (5.25 d)	6.16 E+5
10	Xe-135m (15.7 m)	2.15 E-1
11	Xe-135 (9.09 h)	5.35 E+2
12	Xe-137 (3.83 m)	1.91 E-2
13	Xe-138 (14.13 m)	9.53 E-1
14	I-131 (8.05 d)	5.43 E+1
15	I-133 (20.05 h)	6.86 E+1
16	I-130 (12.36 h)	0
17	I-132 (2.28 h)	2.62 E+0
18	I-134 (52.6 m)	5.61 E-1
19	I-135 (6.61 h)	2.14 E+1
20	H-3 (12.33 y)	9.07 E+7
21	Ar-41 (1.83 h)	2.47 E+4

Table 4.9.1-13. Radioactive Product Activity Concentrations in The Spent Fuel Bay Water – Estimated Values for CANDU-6 Reactors

Radioactive Product (half-life)		Activity Concentration (Bq/kg)
Fission Products		
Niobium – 95	(34.98 d)	3.30 E+3
Zirconium – 95	(64.03 d)	3.00 E+3
Iodine – 131	(8.05 d)	1.80 E+4
Iodine – 133	(20.05 h)	3.70 E+3
Antimony – 124	(60 d)	1.50 E+2
Cesium – 134	(2.05 y)	3.70 E+3
Cesium – 137	(30 y)	1.50 E+4
Activation Products		
Tritium	(12.33 y)	1.90 E+6
Manganese – 56	(2.58 h)	7.40 E+2
Cobalt – 60	(5.28 y)	1.10 E+4

Table 4.9.1-14. Spent Resin Activity Concentrations – Estimated Values for CANDU-6 Reactors

System	Activity Concentration (Bq/cm^3)		
	Cs-137	Cs-134	Co-60
Spent Fuel Bay	5.70 E+6	1.58 E+6	-
End Shield Cooling	-	-	4.01 E+2
D_2O Clean-up	7.78 E+3	3.34 E+3	9.26 E+1
Liquid Radioactive Wastes	7.62 E+4	3.54 E+4	1.51 E+4
Primary Heat Transport	4.77 E+6	2.32 E+6	5.86 E+4
Moderator	-	-	2.67 E+2

Table 4.9.1-15. Isotope Activity Concentrations in the Radioactive Liquid Waste Management System - Estimated Values for CANDU-6 Reactors

Radionuclide (half-life)		Activity Concentrations (kBq/kg)	
		Tanks	Filter-Ion Exchanger
I-131	(8.05 d)	2.05 E+3	1.51 E+5
Cs-137	(30 y)	6.14 E+2	3.54 E+4
Cs-134	(2.05 y)	2.89 E+2	7.62 E+4
Ce-144	(284.9 d)	2.07 E+2	2.53 E+4
Cr-51	(27.7d)	2.07 E+2	2.15 E+4
Mo-99	(66.7 h)	2.07 E+2	9.64 E+3
Co-60	(5.28 y)	1.22 E+2	1.51 E+4

4.9.2. Derived Emission Limits

4.9.2.1. Methodology for Calculation of Derived Emission Limits (DEL)

The methodology used in calculating the DELs for Cernavoda NPP is as given by Canadian Standard Association CAN/CSA – N288.1-M87 (Ref. 4.9-3).

The actual process of calculating the DELs is broken into 6 steps (Ref. 4.9-6, 4.9-7):

1) Identify the critical groups and exposure pathways, preferably from site specific surveys. Exposure pathways may be different for adults and infants. Note that there may be more than one critical group and different radionuclides may be different critical groups.

2) Develop appropriate expressions relating the release rates to the dose rates to an individual.

3) Select appropriate values for the transfer parameters that are relevant to the critical group under consideration.

4) Calculate the DELs based on the whole body dose limit. The calculations must be done separately for adults and infants.

5) For those radionuclides for which organ or skin dose may be limiting, separate DEL calculations should be performed.

6) For each radionuclide select the smallest of the calculated DELs.

The critical groups considered for Cernavoda NPP are the following:

a) airborne emissions;

- the residents (both infants and adults) of Cernavoda town;

b) liquid emissions.

- Cernavoda town, and for drinking water, Constanta city since its population is supplied with drinking water from the Danube – Black Sea Canal. In the

event that liquid effluents are released directly to the Danube River through the alternate effluent discharge tunnel, the critical group is considered to be the population of Seimenii Mici. This critical group will be adequately protected since the DEL calculated based on the critical groups described above is more restrictive.

The pathways considered for airborne releases are:

- Atmosphere → Vegetated Soil → Dose (External);
- Atmosphere → Crops → Dose (Ingestion);
- Atmosphere → Vegetal Soil → Crops → Dose (Ingestion);
- Atmosphere → Vegetal Soil → Forage → Animal Produce → Dose (Ingestion);
- Atmosphere →Forage → Animal Produce → Dose (Ingestion);
- Atmosphere → Animal Produce → Dose (Ingestion);
- Atmosphere → Surface Water → Aquatic Animals → Dose (Ingestion);
- Atmosphere → Dose (Inhalation);
- Atmosphere → Dose (Immersion).

The pathways considered for liquid releases include:

a) The pathways for the residents of Cernavoda town

- Water → Vegetated Soil → Dose (External);
- Water → Vegetated Soil → Crops → Dose (Ingestion);
- Water → Forage → Animal Produce → Dose (Ingestion);
- Water → Vegetated Soil → Forage → Dose (Ingestion);
- Water → Crops → Dose (Ingestion);
- Water → Animal Produce → Dose (Ingestion).

b) The pathway considered for residence of Constanta city

Water → Dose (Ingestion)

The DELs calculated for Cernavoda NPP are conservative; the actual dose that would result from a specified release rate would be less than that calculated. This occur because the following assumption are made:

- The transfer parameters along each step of the food chain are generally conservative values selected from the open literature;
- The radionuclides that are ingested or inhaled are assumed to be in their most restrictive chemical form;
- Foodstuff consumed by the local population is assumed to be grown on the site boundary.

Dose limitation requirements will be met, if during operation

$$\sum_{j=1}^{N} \frac{Q_j}{DEL_j} < 1.0$$

where Q_j is the annual quantity emitted of the j radionuclide in the group of N radionuclides which irradiate a given organ, and DEL_j, is the corresponding emission limit for that particular radionuclide when considering the organ in question.

In this case no individual from the population shall receive a radiation dose larger than 1 mSv/year.

The DELs for gaseous and liquid releases (established for U1) are presented in Tables 4.9.2-1 and 4.3.9-2 (Ref. 4.9-4). The Unit 3, respectively Unit 4 being similar to Unit 1, the DEL values will be the same.

4.9.2.2. Administrative Limits for Release of Both Liquid and Gaseous Effluents

At Cernavoda NPP Unit 1, as it was previously presented, DEL-s has been calculated for a dose limit of 1 mSv/year. In order to obtain one more safety measure, those who operate the CANDU nuclear-power plants, will restrict the releases of radioactive material in the environment, by imposing administrative limits.

For Cernavoda NPP, this limit has been established at the equivalent of effluents releases, which could lead to a dose for the persons in the critical group of maximum 50 μSv/year, representing 5 % from the dose limit of 1 mSv/year.

This limit will be maintained, whether the radioactive emissions in the gaseous and liquid effluents, will be controlled so that:

$$\sum_{J=1}^{N} \frac{Qj}{DELj} \leq 0.05$$

where Q_j is the annual quantity emitted of the j radionuclide in the group of N radionuclides which irradiate a given organ, and DEL_j, is the corresponding emission limit for that particular radionuclide when considering the organ in question.

For a safety operation of Cernavoda NPP, the administrative limits represent a warning level; exceeding these limits represents a symptom of a bad operation and the moment of starting the process of identifying the causes. On the other hand, maintenance of emissions at a level of 5 % of DEL leads to maintenance of some doses under the limit allowed by the norms.

Conforming to the Fundamental Norms for Radiological Safety (Ref. 4.9-2), during the licensing process of a nuclear facility, CNCAN shall define a dose constraint for this facility, value which must not be exceeded during normal operating conditions. CNCAN have defined, in operating license nr. SNN U1-5/2003, the dose constraint for Cernavoda NPP Unit 1, namely 0.1 mSv/year.

Since the operating target at Unit 1 (administrative dose limit) represents 50% of the dose constraint recommended by CNCAN, a good margin is ensured during Cernavoda NPP operation. This option will be also respected during Unit 3 and Unit 4 exploitation.

Table 4.9.2-1. Derived Emission Limits for Airborne Releases at U1

Radionuclide / Radionuclide Group	DEL (GBq/week)	Critical Group*	Limiting Organ
H–3 (oxide)	1.01E+06	1	WB
C-14 (gaseous)	2.11E+03	2	WB
I-131 (mfp)**	6.62E+00	2	WB
Noble Gases (GBq·Mev/week)	4.15E+05	1.2	WB
Particulates***	1.33E+00	1	WB
Cr-51	2.69E+04	1	WB
Mn-54	2.96E+02	1	WB
Fe-59	1.08E+02	1	WB
Co-58	9.54E+00	1	WB
Co-60	8.84E+00	1	WB
Zn-65	3.20E+01	1	WB
Sr-89	6.01E+01	2	WB
Sr-90+****	2.77E+00	2	WB
Zr-95+	3.04E+02	1	WB
Nb-95	6.29E+01	2	WB
Ru-103	6.47E+02	1	WB
Ru-106+	4.99E+01	1	WB
Sb-124	1.75E+02	1	WB
Sb-125	1.94E+02	1	WB
Cs-134	6.65E+00	1	WB
Cs-137+	1.33E+00	1	WB
Ba-140+	5.88E+02	1	WB
Ce-141	6.90E+02	1	WB
Ce-144+	6.43E+01	1	WB
Eu-152	2.68E+01	1	WB
Eu-154	1.95E+01	1	WB
Gd-153	1.09E+03	1	WB

* Critical group 1 – adult Cernavoda town
2 – infant (0-1 year) Cernavoda town

** mfp indicates the fact that conservatively, it is assumed that for the receiver, I-131 is found in a balanced mixture with the other radioactive isotopes which are fission products of iodine in a ratio of I-131:I-133:I-134:I-135 = 1.00:1.45:2.00:2.04:1.81

*** for unidentified particles DEL is considered for the most restrictive radionuclide

**** "+" after a isotope indicates the fact that the conversion factor of the dose includes the dose from the radionuclide

Table 4.9.2-2. Liquid DELs for Cernavoda NPP U1

Radionuclide / Radionuclide Group	DEL (GBq/month)	Critical Group*	Limiting Organ
H–3 (oxide)	6.06E+06	4	WB
C-14 (soluble carbonates)	2.45E+05	4	WB
Gross Beta / Gamma Activity**	1.83E+02	2	WB
Cr-51	2.26E+06	1	WB
Mn-54	1.01E+04	1	WB
Fe-59	1.92E+04	1	WB
Co-58	2.55E+03	1	WB
Co-60	5.72E+02	1	WB
Zn-65	5.75E+03	2	WB
Sr-89	1.98E+04	2	WB
Sr-90+***	1.83E+02	2	WB
Zr-95+	1.53E+04	1	WB
Nb-95	1.27E+04	1	WB
Ru-103	9.03E+04	1	WB
Ru-106+	1.33E+04	1	WB
Sb-124	1.89E+04	1	WB
Sb-125	6.09E+03	1	WB
I-131(mfp)****	2.90E+03	4	WB
Cs-134	1.47E+03	1	WB
CS-137+	4.63E+02	1	WB
Ba-140+	5.99E+04	1	WB
Ce-141	2.08E+05	1	WB
Ce-144+	2.36E+04	1	WB
Eu-152	5.79E+02	1	WB
Eu-154	4.37E+02	1	WB
Gd-153	7.64E+04	1	WB

* Critical group 1 – adult Cernavoda town
2 – infant (0-1 year) Cernavoda town
3 - adult Constanta town
3 – infant (0-1 year) Constanta town

** For Gross Beta/Gamma the DEL for the most restrictive radionuclide is used (i.e. Sr-90+)

*** "+" after a isotope indicates the fact that the conversion factor of the dose includes the dose from the radionuclide

**** mfp indicates the fact that conservatively, it is assumed that for the receiver, I-131 is found in a balanced mixture with the other radioactive isotopes which are fission products of iodine in a ratio of I-131:I-133:I-134:I-135 = 1.00:1.45:2.00:2.04:1.81

4.9.3. The Impact on Both the Waters and Air

As per the Fundamental Norms of Radiological Safety, releasing into environment of both the liquid and gaseous radioactive effluents can be performed only by meeting the derived emission limits approved by CNCAN during the licensing process (Ref. 4.9-6, 4.9-7).

Tables 4.9.2-1 and 4.9.2-2 present the Derived Emission Limits for both gaseous and liquid emissions for Cernavoda NPP Unit 1. As Unit 3, respectively Unit 4 design is similarly with Unit 1, the DEL values will be the same.

A comparison between the estimated emissions and the emissions reported at Cernavoda NPP Unit 1, as well as the derived emission limits are presented in Chapters 4.1.14 and 4.2.3. From this comparisons it is clearly stated that the emissions for CANDU 6 plants and those reported at Unit 1 are under these DEL values. More than that, at Cernavoda NPP the administrative limits, representing 5 % of DEL will be applied.

Under the operating conditions of the four units, although the quantity of radioactive effluents will be four times higher, both the gaseous emissions and the liquid ones will be under the derived emission limits.

Intermediate Spent Fuel Storage (DICA) is situated on NPP site. Under DICA normal operation, there are no emissions of radioactive effluents into the air or water (Ref. 4.9-5).

4.9.4. Radiation Doses for Population

Estimation of the dose for a member of the critical group from Cernavoda NPP was made (Ref. 4.9-6, 4.9-7) using the methodology in Ref. 4.9.3.

In Table 4.9.4-1 the estimated doses for a unit have been presented, being calculated on the basis of emissions mediated from CANDU 6 type plants; the doses reported at Cernavoda NPP Unit 1 in period 1997-2003 are also presented (Ref. 4.9-1).

In table is shown that both the estimated doses and the doses reported at Unit 1 are smaller than the dose limit established by Norms, 1 mSv/year, respectively:

- total annual dose, due to the liquid radioactive effluents recorded at U1, represents a maximum of 3.5 % (in 2003) from the administrative limit of 50 μSv/year, which is the operating target for a nuclear power unit located on Cernavoda site; comparing with the dose constraint value recommended by CNCAN for Cernavoda NPP Unit 1, the maximum annual dose due to liquid effluents (recorded in 2003) represents only about 1.75 %;
- total annual dose, due to the gaseous radioactive effluents recorded at U1, represents a maximum of 13 % (in 2002) from the administrative limit of 50 μSv/year; comparing with the dose constraint value recommended by CNCAN for Cernavoda NPP Unit 1, the maximum annual dose due to gaseous effluents (recorded in 2002) represents only about 6.6 %. Taking into account that the internal dose due to natural background on Cernavoda site is about 1.55 mSv/y, it results that the contribution of gaseous effluent emissions from nuclear units to the total internal dose is insignificant.

As Unit 3 and Unit 4 are similar with Unit 1 and Unit 2, the maximum dose received by a member of the critical group at operation of the four units will be of 0.2 mSv/year (4x50 μSv/year), a value being situated under the dose limit for population (1mSv/year).

The contribution of the Intermediate Spent Fuel Storage (DICA) to the commitment dose of an individual living in the exclusion zone of the site is insignificant considering that its value is 6.5×10^{-4} mSv/year (Ref. 4.9-5).

Table 4.9.4-1. Estimated and reported doses for a unit for a member of critical group around Cernavoda NPP (μSv/an**)**

Emission Type	Estimated values for CANDU 6 (μSv/a)	Values reported at Cernavoda NPP Unit 1 (μSv/an)						
		1997	1998	1999	2000	2001	2002	2003
Liquid emission	0.77	0.61	1.16	0.33	0.81	1.09	1.52	1.75
Gaseous emission	19.1	5.00	4.42	4.17	6.41	6.18	6.58	4.36
Total dose	19.7	5.61	5.58	4.50	7.22	7.27	8.10	6.11

4.9.5. Effects of other Nuclear Objectives from Cernavoda NPP Platform on the Constructors at Unit 3, respectively Unit 4

Both the design and operation of NPP Unit 1 and Unit 2 provide maintaining of radiation doses outside nuclear buildings under the limit value allowed for public by the regulations in effect (Ref. 4.9-6, 4.9-7).

Unit 3 is located at about 400 m from DICA platform. At this distance the gamma dose rate of 2 x 10^{-3} μ Sv/h was estimated (Ref. 4.9-5). This value is lower than the dose limit imposed by CNCAN for DICA (100 μSv/year, 1x10^{-2} μSv/h, respectively). Thus, the constructors of Unit 3 will not be affected by DICA Cernavoda.

Unit 4 is located at about 250 m from DICA platform. At this distance the gamma dose rate of 1 x 10^{-2}μ Sv/h was estimated (Ref. 4.9-5). This value is equal with the dose limit imposed by CNCAN for DICA (100 μSv/year, 1x10^{-2} μSv/h, respectively). Thus, the constructors of Unit 4 will not be affected by DICA Cernavoda.

It is noted that the dose evaluation for DICA has been performed under conservative assumptions, for 27 storage modules loaded with cooled fuel along 6 years. It is expected that the real values of dose rates at different distances from DICA will be lower than the estimated ones.

4.9.6. Radiation Doses for Operating Personnel

In the followings, an estimation of both individual and collective doses for plant personnel is presented, taking into consideration both the external irradiation exposure and internal contamination (Ref. 4.9-6, 4.9-7).

In Table 4.9.6-1 the values of the total individual doses are presented (the sum between the inside and outside doses) received by the operating personnel during maintenance and plant operating works, obtained further to operating experience of CANDU type stations (based upon the average value obtained from 6 plants), as well as the registered values corresponding to 1997÷2003 period at Cernavoda NPP Unit 1 (Ref. 4.9-1). The individual average doses are also presented on activity groups of the personnel.

Tables 4.9.6-2 (a, b, c, d) present the distribution of radiation doses on working groups, from Cernavoda NPP Unit 1, during 1999 – 2003 (Ref. 4.9-1).

Table 4.9.6-3 presents the collective exposure doses of the operating personnel on groups of workers, obtained on the basis of the average value from 6 plants of CANDU-type, in the years 1987 and 1998 and from Cernavoda NPP U1 in the period 1997 ÷ 2003 (Ref. 4.9-1).

In the already presented tables, it is stated that the doses for the professionally exposed personnel are much smaller than the limits established by norms (Ref. 4.9-2).

Table 4.9.6-1. Individual doses of annual occupational exposure by work groups at CANDU plants (average values)

Plant	Individual dose [mSv]						Plant average
	I & C Personnel	Mechanicals	Operators	Maintenance personnel	Other personnel categories	Temporary personnel	
1987							
CANDU (average)	1.65	4.9	2.63	1.62	1.38	2.7	2.49
1988							
CANDU (average)	2.9	5.8	4.0	1.7	1.4	2.7	2.61
Cernavoda U1*							
1997	0.86	0.88	0.58	0.70	1.86	- **	0.976
1998	0.57	0.91	0.53	0.52	0.95		0.696
1999	0.60	1.81	0.64	1.82	1.28		1.23
2000	0.71	1.54	1.02	1.48	1.14		1.25
2001	0.61	2.35	1.10	1.05	1.36		1.26
2002	0.59	1.35	1.08	1.03	0.61		1.12
2003	0.53	1.89	1.04	1.45	0.93		1.26

*) The values are averaged taking into consideration personnel having a reference dose only
**) Temporary personnel was taken into consideration on each category of operators

Table 4.9.6-2 a. Radiations Dose Distribution by Work Group at Cernavoda NPP U1 from 27 Dec. 1999 - 31 Dec. 2000

Work Group	Number of Persons Monitored	Work Group Size on Qtr End	Number Exposed > 0.2 mSv Effective Dose (Whole Body)	External Effective Dose (Whole Body) (man-mSv)	Internal Effective Dose (Whole Body) (man-mSv)	Equivalent Dose (Skin) (man-mSv)	Total Effective Dose (Whole Body) (man-mSv)	Station Total (%)	Maximum Individual Effective Dose (Whole Body) (mSv)	Average Effective Dose (WB)/Exposed Worker (mSv)
EI & C Maintenance	174	133	34	20.80	3.36	24.76	24.16	5.2	2.85	0.71
Fuel handling	62	41	27	28.1	15.69	44.7	43.79	9.4	3.9	1.62
Mech Maint.	177	132	53	54.34	27.39	80.29	81.73	17.5	6.85	1.54
Operations	200	186	61	19.22	42.76	62.42	61.98	13.3	3.66	1.02
Service Maint.	304	220	88	121.66	8.87	128.97	130.53	28.0	5.00	1.48
Other Groups	802	643	109	111.27	12.74	121.97	124.01	26.6	6.6	1.14
Station Total	1719	1355	372	355.39	110.81	463.1	466.2	100.0	6.85	1.25

Table 4.9.6-2 b. Radiations Dose Distribution by Work Group at Cernavoda NPP U1 from 01 Jan. 2001 to 31 Dec. **2001**

Work Group	Number of Persons Monitored	Work Group Size on Qtr End	Number Exposed > 0.2 mSv Effective Dose (Whole Body)	External Effective Dose (Whole Body) (man-mSv)	Internal Effective Dose (Whole Body) (man-mSv)	Equivalent Dose (Skin) (man-mSv)	Total Effective Dose (Whole Body) (man-mSv)	Station Total (%)	Maximum Individual Effective Dose (Whole Body) (mSv)	Average Effective Dose (WB)/Exposed Worker (mSv)
El & C Maintenance	200	145	75	34.32	11.71	46.69	46.03	8.0	5.02	0.61
Fuel handling	53	38	34	35.78	21.56	60.02	57.34	10.0	5.14	1.67
Mech. Maint.	160	114	48	89.19	23.67	113.46	112.86	19.6	7.25	2.35
Operations	203	190	62	38.60	29.33	69.68	67.93	11.8	5.58	1.10
Service Maint.	341	229	107	87.95	24.88	122.43	112.82	19.6	6.06	1.05
Other Groups	895	689	131	147.6	30.27	181.26	177.88	31.0	7.94	1.36
Station Total	1852	1405	457	433.44	141.42	593.54	574.86	100.0	7.94	1.26

Table 4.9.6-2 c. Radiations Dose Distribution by Work Group at Cernavoda NPP U1 from 31 Dec. 2001 to 29 Dec. 2002

Work Group	Number of Persons Monitored	Work Group Size on Qtr End	Number Exposed > 0.2 mSv Effective Dose (Whole Body)	External Effective Dose (Whole Body) (man-mSv)	Internal Effective Dose (Whole Body) (man-mSv)	Equivalent Dose (Skin) (man-mSv)	Total Effective Dose (Whole Body) (man-mSv)	Station Total (%)	Maximum Individual Effective Dose (Whole Body) (mSv)	Average Effective Dose (WB)/Exposed Worker (mSv)
EI & C Maintenance	196	176	59	27.02	8.02	35.27	35.04	6.4	4.32	0.59
Fuel handling	65	50	43	91.31	37.84	132.87	129.16	23.5	9.23	3.00
Mech Maint	163	133	89	66.81	53.40	122.73	120.21	21.8	5.96	1.35
Operations	245	241	75	42.17	38.86	81.64	81.03	14.7	6.41	1.08
Service Maint	340	251	111	68.77	45.48	114.81	114.25	20.8	3.85	1.03
Other Groups	1069	850	116	47.96	22.83	72.54	70.79	12.8	4.51	0.61
Station Total	2078	1701	493	344.04	206.43	559.58	550.48	100.0	9.23	1.12

Table 4.9.6-2 d. Radiations Dose Distribution by Work Group at Cernavoda NPP U1 from 30 Dec. 2002 to 28 Dec. 2003

Work Group	Number of Persons Monitored	Work Group Size on Qtr End	Number Exposed > 0.2 mSv Effective Dose (Whole Body)	External Effective Dose (Whole Body) (man-mSv)	Internal Effective Dose (Whole Body) (man-mSv)	Equivalent Dose (Skin) (man-mSv)	Total Effective Dose (Whole Body) (man-mSv)	Station Total (%)	Maximum Individual Effective Dose (Whole Body) (mSv)	Average Effective Dose (WB)/Exposed Worker (mSv)
El & C Maintenance	233	195	61	17.22	15.15	34.25	32.37	4.0	3.62	0.53
Fuel handling	71	59	40	56.81	28.26	89.06	85.08	10.4	9.68	2.13
Mech Maint.	189	127	99	127.04	60.53	189.99	187.57	22.9	9.66	1.89
Operations	287	278	84	43.87	43.90	91.17	87.76	10.7	5.01	1.04
Service Maint.	324	243	165	142.12	96.96	244.36	239.08	29.2	7.71	1.45
Other Groups	1125	837	200	133.21	53.22	192.74	186.42	22.8	7.68	0.93
Station Total	2229	1736	649	520.27	298.02	841.57	818.25	100.0	9.68	1.26

Table 4.9.6-3. Annual Occupational Radiation Exposure by Work Groups at CANDU Plants

	Collective Dose [man · mSv]					
Plant	Personnel I&C	Mechanical Personnel	Operators	Service Maintenance Personnel	Other Station Staff	Attached Personnel
1987 CANDU (average)	57.5 (6.5%)	294.33 (33.31%)	165.66 (18.75%)	49.4 (4.65%)	118.33 (13.4%)	206.5 (23.37%)
1988 CANDU (average)	71.5 (1.13%)	407 (6.458%)	241.16 (3.826%)	47 (0.745%)	126.66 (2%)	164.33 (2.6%)
Cernavoda U1						
1997	20.6	30.7	27.2	37.8	169.8	- *
1998	24.94	37.3	26.0	29.43	140.1	
1999	29.79	74.2	31.4	152.84	168.33	
2000	24.16	81.73	61.98	130.53	167.80	
2001	46.03	112.86	67.93	112.82	235.22	
2002	35.04	120.21	81.03	114.25	199.95	
2003	32.37	187.57	87.76	239.08	271.50	

*) Temporary personnel was taken into consideration on each category of operators

4.9.7. Engineered Provisions for Radiation Protection of the Personnel and the Public

The plant design for Cernavoda NPP U3, respectively U4 has in view the radiation protection of the site personnel, members of the public and environment both in normal operation and accident conditions (Ref. 4.9-6, 4.9-7).

The main provisions taken by design in order to protect personnel and public are presented below.

4.9.7.1. Provisions for Personnel Protection

4.9.7.1.1. Provisions for Personnel Protection in Normal Operation

During normal operation, the provisions taken by design for radiation protection of the site personnel refer to:

a) minimizing the radioactive sources, by the following means:

- reducing cobalt impurities in stainless steel;
- providing chemical control of primary and moderator systems to prevent extensive build-up of activated corrosion products;
- surface treatment of components and pipes;
- minimizing of D_2O leakage from technological circuits;
- D_2O leakage collection from primary circuit and moderator system, by using separate collection systems;
- D_2O vapor recovery system, to retain the heavy water and tritium vapors released to the room atmosphere from PHTS, moderator, steam generator and fueling machine associated areas;
- purification system for all the water systems in connection with the reactor core;
- spent fuel bay purification system.

b) reduction of the amount of liquid impurities, by the following technical solutions:

- purification system, to maintain the purity of the liquid and to minimize corrosion of components;
- using demineralized water;
- cleaning up the heavy water recovered from leakages;
- managing tritiated heavy water make-up, so that D_2O with the lowest tritium level is added to the primary coolant system.

c) keeping activity levels in liquid and gaseous processes low, by the following means:

- provisions for coolant and moderator purification, in order to reduce their activity, by removing the activated impurities and corrosion products;
- early detection of the failed fuel followed by prompt removal of defective fuel, to reduce the fission product contamination in the primary coolant system;
- providing a D_2O vapor recovery system in rooms with a probability of heavy water leakage, to collect D_2O vapors by a system of dryers.

d) provisions for radiation shielding, as follows:

- primary shield, which attenuates radiation from the reactor;
- secondary shield, which attenuates radiation from the primary coolant;
- auxiliary shield, which attenuates radiation from auxiliary systems, as moderator, fueling machine, etc;
- special shield, not categorized as above, located also in Reactor Building;
- shields in Service Building;
- supplementary shields, in the penetration areas;
- temporary shields, to protect the workers in some special conditions.

e) classification of the areas inside the plant depending on the radiation hazard existing in those areas:

- accessible zones, during normal operation (free access or conditioned access);
- inaccessible zones; these zones become accessible only after reactor shutdown (conditioned access, depending on the time after shutdown).

f) provisions for contamination protection:

- use of high capacity ventilation systems which provide purging of ventilated areas and reduce local airborne hazards;
- establishing ventilation flows from areas which are more often frequented by workers towards areas of higher potential contamination;
- reducing airborne tritium levels by using D_2O vapor recovery system and passing the contaminated air through dryers;
- use of filtration units before the contaminated air from ventilation systems is discharged into the environment through the plant exhaust stack;
- use of breathing air equipment, to minimize the internal dose uptake of workers;
- use of decontaminable surface finishes, to prevent the fixed contamination and to make easy decontamination;
- minimizing leakage from radioactive liquid systems, in order to ensure radiation protection concerning internal radiation hazard;
- provision of enclosed drain systems, to reduce the hazard levels due to leaks;
- nuclear zoning (area contamination classification), to protect workers and to keep the contamination under control.

Nuclear zoning for Cernavoda NPP-U3, respectively U4 consists in dividing the plant area in three radiological zones, depending on potential radiation level.

The definition of radiological zones is:

Zone 3 is a clean area, without radioactive sources, except as approved, with a very low probability of cross contamination from adjacent areas and with a general radiation dose rate less than 0.5 μSv/h.

Zone 2 is a controlled area, without radioactive sources, except as approved, normally free of contamination, but is subject to infrequent cross-contamination due to the movement of personnel and equipment. This area contains no radioactive system and has a general radiation dose rate less than 10 μSv/h.

Zone 1 is a controlled area and contains systems and equipment which may be significant sources of contamination or radiation exposure.

There are provisions for clear demarcation between zones. Access to and from radiological zones is controlled in accordance with approved procedures.

g) provisions for hazard warnings: there are provided signs to warn personnel about radiation hazard;

h) providing the access control system, ensuring an adequate access control by 3 subsystems using locked doors and interlocks;

i) provisions for a good maintenance:

- portable shielding, to reduce the dose rate from equipment which is being worked on or is present in the area;
- shielded cabinets for workers performing duties at reactor face;
- keeping the equipment or vessels full of water to provide shielding during maintenance activities when appropriate;
- equipment for lifting/handling permanently installed where appropriate;
- decreasing of the time of presence in radiation field by:
    - selecting high quality equipment with low failure frequency, which requires a minimum activity for maintenance and repair;
    - providing facilities for equipment maintenance and repair (suitable tools, platforms or walkways for easy access to equipment);

- ensuring the station and equipment layout so that allows easy access to equipment for easy of maintenance and repair operations, free corridors as large as possible allowing easy access of personnel and quick transport of system components inside of plant to decontamination and repair workshops;

- provisions for special remote/automated inspection equipment;

j) provisions for fixed and portable radiation monitoring equipment, contamination monitoring equipment, airborne activity monitoring equipment and liquid process monitoring;

k) provisions for personnel facilities, as washing/showering facilities, changing rooms, protective clothing laundry, decontamination facility.

4.9.7.1.2. Provisions for Personnel Protection in Accident Conditions

Most of the engineered provisions for radiation protection of the operator in normal operation also contribute to the radiation protection in accident conditions. The special measures provided for personnel protection during accident situations refer to:

- safe evacuation routes of the operating personnel;
- ensuring that the required post-accident operator actions can be carried out safely.

As concerns the evacuation of operators from the plant, the main requirement are following:

- provisions of early warning: this is achieved via the area monitoring system, and the alarm raised by the main control room;
- provisions of ventilated access routes: this is achieved via the ventilation system provided for operator protection in normal operation with air flows from clean to potentially contaminated zones;
- provisions for shielded routes.

As concerns the post-accident actions necessary to be carried out to maintain the plant in a safe state, there is the provision for ensuring that the radiation doses received by operators remain within limits accepted by current norms.

4.9.7.2. Provisions for Public Protection

4.9.7.2.1. Provisions for Public Protection in Normal Operation

The following elements ensure a corresponding radiation protection of population and environment:

a) Source Control

This item concerns the measures and activities necessary for radioactive material confinement. Multiple barriers are provided to prevent the radioactive material releases from sources.

b) Effluent Control

This item covers the measures and activities necessary to control the radioactivity releases into environment to comply with specified limits.

c) Effluent Monitoring

This item concerns the measures and activities necessary for measurements of effluent releases in emission points.

d) Environment Monitoring

This item covers the measures and activities necessary for measurements of environmental radioactivity levels, to assess the radiological impact over public health and environment, due to effluent releases from the plant. In case of significant or greater releases, supplementary programs are implemented for accurate dose assessment.

4.9.7.2.2. Provisions for Public Protection in Accident Conditions

Many of the engineered provisions for radiation protection of the public in normal operation also contribute to the radiation protection in accident conditions. The special measures provided for public protection during accident situations refer to:

- ensuring containment isolation to terminate releases;
- reducing activity releases.

The first requirement for public protection is to ensure that releases are terminated. Containment isolation is provided by diverse means including isolation based on measurements of Reactor Building airborne activity.

The second requirement is to ensure that where releases take place, these are reduced by an appropriate filtration and by ensuring that the volatility of radionuclides (in particular iodine) is reduced.

The Reactor Building ventilation system provides particulate and iodine filtration to reduce discharges from the Reactor Building. Given that venting to release containment pressure may only be required following a LOCA, it is essential to ensure that the iodine filters do not degrade due to moisture loading. Reactor Building depressurisation following LOCA is performed through the D_2O vapour recovery system, in order to protect against the risk of degrading the iodine filters due to moisture loading.

The release of iodine from the water within the containment is reduced by ensuring an appropriate water chemistry.

4.9.8. Transboundary Effects

In normal operation the Cernavoda NPP releases small quantities of radioactive materials to the environment. These releases are closely monitored to ensure compliance with regulatory limits. In practice, the yearly operating target is sub-divided into weekly or monthly targets to better control NPP releases. These limits are well below 5% of the derived emission limit (DEL), which ties directly to the 1 mSv/year public dose limit. Conservative modeling show that the highest dose, from four units, to a member of the critical group would be 0.2 mSv/year (0.05 mSv/year each from Units 1, 2, 3 and 4). Although these limits are focused on public radiation protection, they are so low that they have been used to assure environmental protection as well.

Taking into account the reasons mentioned above and related information provided by this assessment, it results that the operation activities of Unit 1, 2, 3 and 4 NPPs should have no significant effects in the next neighborhood of the plant (the critical groop are located at about 2 km distance from the Cernavoda NPP) being greatly under both the Romanian and CE norms. As much as the distance to the plant increases, these effects are diminishing, while at distances of about 30-40 km, these effects have no practical significance.

Given the distance between Cernavoda Site and other countries (the nearest country, at around 40 Km, is Bulgaria), this study have concluded that no significant adverse transboundary effects should occur.

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