2. TECHNOLOGICAL PROCESSES

2.1. Activities during the Construction Phase

2.1.1. General Arrangement

Characteristic for Cernavoda NPP Units is the location in a row of Unit 1 - Unit 4 reactors at 160 m center-to-center and the fact that each of the 5 units is independent (Ref. 2-34, 2-36).

The center of Unit 5 reactor is located at 44.00 m misalignment from the center-to-center line that is connecting the centers of the first 4 reactors towards CDMN, aiming to locate Unit 5 on a foundation ground evidencing better siting conditions.

Each unit is a functionally independent assembly consisting of a nuclear part (NSP) with its auxiliaries and a classic part (BOP) with its auxiliaries.

Within Cernavoda NPP - Unit 5 enclosure there is an area that is technologically servicing the units and it is including items common to the 5 units (see the List of items common to the 5 units - GA Dwg, Sc 1:2000, code: U3/U4 - 08230-6024-CU/PG-6025-2-GA-0, Rew.1). The location (sitting) area for these items is called "Unit 0 ".

The activity of each independent unit is correlated with the activity developed in the common items positioned in "Unit 0".

The layout concept for all the units was so elaborated to functionally combine the two areas: "Unit 0" and the specific area of each unit.

All the buildings, warehouses and communication routes are arranged to form a optimum technological and architectural assembly. They are all located in safe conditions, separated from each other by minimum technological and protection distances that correspond to the technological and traffic flows in order to provide optimum construction and operation. The Layout Drawing is arranging and planning all the buildings, installations and networks and their inter-relationships.

The location of the Plant items within the “Unit 0” Layout was solved on Unit 1 finalization.

The items specific to each unit were located so to define a “module” area that is repeated in each unit.

Unit 3

The location of Unit 3 items followed the same pattern like with Unit 1 and Unit 2.

The enclosure of Unit 3 has the following configuration:

- to the Intake Duct the enclosure is confined by Unit 3 Pump Station platform boundary;
- to Unit 2 Pump Station, the Unit 3 enclosure boundary is represented by the limit of the building segment associated to Unit 2 Pump Station;
- to Unit 4 Pump Station, the Unit 3 enclosure boundary is represented by the limit of the building segment associated to Unit 3 Pump Station;
- to the access road in front of the reactors, the Unit 3 enclosure boundary is considered sideways to this road, towards Unit 3;
- the land area in front of Unit 4, positioned across the access road in front of the Unit 1 - Unit 4 reactors up to the NPP secondary access road;
- to Unit 2, the Unit 3 enclosure boundary follows the route of the road between U2 and U3 on the side towards U2 up to the 110 kV Station fence boundary;
- to Unit 4 , the U3 enclosure boundary follows the route of the road between U3 and U4 on the side towards U3, up to Valea Cismelei.

Unit 4

Location of U 4 items is the same like for U1, U2 and U3.

Unit 4 enclosure has the following configuration:

- to the Intake Duct the enclosure is confined by U4 Pump Station platform boundary;
- to U3 Pump Station, U4 enclosure boundary is represented by the limit of the building segment associated to the Pump Station for U3;
- to U5 Pump Station, U4 enclosure boundary is represented by the limit of the building segment associated to Pump Station for U4;
- to the access road in front of the reactors, U4 enclosure boundary is considered sideways to this road, towards Unit 4;
- to Unit 3, U4 enclosure boundary follows the route of the road between U3 and U4 along the U3 side of the road up to the NPP enclosure boundary towards Valea Cismelei;
- to U 5, U4 enclosure boundary follows the route of the road between U4 and U5 along U4 side of the road up to NPP enclosure fence boundary towards Valea Cismelei;
- to Valea Cismelei, U4 enclosure boundary is represented by NPP enclosure fence.

The El. +/- 0.00 of the buildings is the same for all the items positioned inside NPP enclosure and it is El. + 16.30 mBSL.

The ground elevation within Unit 3 and Unit 4 areas is El. +16.00 mBSL, the same with the elevation of Unit 5 platform.

The cold water supply source for the 5 units cooling water systems is the Danube River water via the Danube - Black Sea Channel (DBSC) - Race 1, towards the Derivation Channel and the Water Intake (item 140) and next to the Intake Duct (item 141) and towards the Distribution Bay (item 047).

DBSC – Race 1, the Derivation Channel, the Intake Duct, the Distribution Bay, are all common to all NPP units.

The cooling water required to NPP units is taken from the Distribution Bay via a block building that is housing all the Screen Houses, all the Pump Stations and all the

Electric Stations of the units. The block building is divided into segments – one for each NPP unit. Each segment includes items specific to each unit.

So, the segment dedicated to a unit includes the screen houses (for the circulation water and raw service water - items 048, 049), the pump stations (for the circulation water and raw service water - items 050, 051) and the electric station of the screen house (item 058).

All these items are providing the cleaning and pumping of water required for the operation of the circulation water system and raw service water system.

The circulation water system is distributing the filtered raw water to the Turbine Hall (item 020) for the condenser cooling.

The raw service water system is distributing the filtered raw water to the Service Building (item 002) and the Turbine Building (item 020).

The discharge of the warm water (item 054) resulted from the condenser cooling and the other chillers in the Unit, is made via the associated hydrotechnical buildings and it can be directed (item 056 – Syphonating Bay and redirectioning) to DBSC race 2 or to the Danube (see Layout Dwg. Sc. 1:5000 U3/U4-08230-6024-CU/PG-6024-1-GA-1, Rew. 1).

Both the cold water intake system and the warm water discharge system are dedicated to each unit.

The Distribution Bay is also collecting the emergency water and fire water. The Pump Station building for the Emergency Water Supply (item 004) is a building common to U3, U4 and U5. The emergency water intake (item 010) is also common to U3, U4 and U5. The water pipe lines between the building common to U3-U4-U5 end the consumers as well as the electric cable routes for EWS are ensured to each unit.

The fire water intake, its Pump Station building (item 093), the associated tanks (item 094) are all common to all the NPP units and they are positioned in “Unit 0” area. The fire water pipe routes (AI) are particular to each of the units.

Item associated to U3, i.e. NSP and BOP items, are listed in “U3 specific items” presented on U3 enclosure General Arrangement Drawing, Sc 1:2000, code: U3/U4 - 08230-6024-CU/PG-6025-2-GA-0, Rew.1.

Items associated to U4, i.e. NSP and BOP items, are listed in “U4 specific items” presented on U4 enclosure General Arrangement Drawing, Sc 1:2000, code: U3/U4 - 08230-6024-CU/PG-6025-2-GA-0, Rew.1.

Around the nuclear area of each unit, a shield (item 011) vertically penetrating down to the marl waterproofing layer and aiming to provide the control of the underground water beneath of the plant.

The cable channels associated to the electric network systems are both underground and aboveground, around Unit 3 and Unit 4.

The General Arrangement Drawing, Sc 1:2000, code: U3/U4 - 08230-6024-CU/PG-6025-2-GA-0, Rew.1, also shows the railways. Unit 3 and Unit 4 are not directly railway – connected from the existing railway, i.e. Saligny – Cernavoda Town railway.

The Transformer area is located in front of the Turbine Hall and it includes the Self Service Transformer Block (item 030), Power Discharge Transformer Block (item 031) and the related railways (item 033) required to handle the transformers for inspection, maintenance and repair.

The roads and vehicle platforms inside Unit 3 and Unit 4 enclosure are so sized to provide the circulation of vehicles carrying the equipment necessary to the Plant systems and of Fire Brigade intervention teams vehicles.

The road in front of Unit 3 and Unit 4 is a segment of the access road in front of Unit 1 to Unit 4 reactors. This road is providing the safe transport of the spent fuel by trailer from the Spent Fuel Bay of each Unit to the Spent Fuel Intermediate Storage Facility (DICA). The width of the road is 8.00 m.

The road between the Units is 6.00 m wide and it is used for transport during the construction- installation work periods as well as during the operation of the Units.

The curvature radius of the roads are calculated in function of the size of vehicles traveling inside the NPP enclosure during the construction-installation works and during operation.

Each of the Unit 3 and Unit 4 enclosure shall be individually fenced by wire- mash fences with metal pillars. The solution was adopted for fencing both Unit 1 and the Spent Fuel Intermediate Storage (DICA).

The vehicle access into Unit 3 and Unit 4 enclosure is made from Medgidia Street via the Secondary Access Road to NPP and along the road in front of Unit 1 - Unit 4 reactors.

The surface and juridical status of the land for Unit 3 and Unit 4 are the followings:

- the land used for Unit 3 is part of Cernavoda NPP enclosure and it was expropriated along with the land associated to Cernavoda NPP – Unit 5;
- the land is owned by SNN-SA as proprietary according to the Certificate of Land Propriety Right Assignment, Series MO3, no. 5415, issued by the Ministry of Industry and Resources on April, 25, 2000.

Out of the projects presented within the above – mentioned design, the common projects for Units U1 and U4 are presented in list 1. The projects corresponding to units 3 and 4 investment are specified in list 2, while the projects included in the services provided by SNN are presented in list 3 (Ref. 2-36).

List 1

COMMON OBJECTS FOR UNITS U1 ÷ U4

IN THE ENCLOSURE

025 Common Sevice System Transformers

035 High Voltage Station (110 kV) for Common Service Power Supply

036 Transformers for Common Service High Voltage Power Supply Station

037 Transformers for Common Service Power Supply

038* Common Services Channels and Cable Ducts

047 Distribution Pool

054a Hot Water Pipe & Ducts from Condensers (CCW + RSW) to DBSC-Race2

054b Hot Water Pipe & Ducts from Condensers (CCW + RSW) to Danube

056 Siphonating and Switching Pool (Pump Station and Electric Station)

057* Service Water Network

065 Auxiliary Boiler Station

066 Liquid and Oil Fuel Discharge Ramp

067 Liquid and Oil Fuel Pump Station – Stage I

068 Liquid and Oil Fuel Pump Station – Stage II

069 Motor Oil and Light Fuel Warehouse

071 Transformer and Turbine Oil Warehouse

072 Lubricator Warehouse

073 Emergency Diesel Station for Common Services

075 Pipe Support (from Unit 0)

076* Process Piping for Liquid and Oil Fuel

077 Hydrogen Production Station

079 Oxygen and Acetylene Warehouse

080 Equipment Warehouse

082 Maintenance Workshop and Warehouse

084 Motor Oil Separator

086 Motor Oil Supply for Common Service Diesel Group

090 Drinking Water Pump Station

091 Drinking Water Tanks

092* Drinking Water Distribution Pipe Network

096 Fire Cabinet and Ambulance Station

097 Pump Stations, stage I and stage II

098* Sewage Water Pipe System including the Pump Station

125 Spent Fuel Interim Storage Facility

130 Intermediate Storage Facility of Radioactive Wastes

OUT OF ENCLOSURE

131 Meteorological Tower

132 Housing for Meteo Tower Electronic Equipment

133 Environmental Radiological Monitoring Laboratory

134 Electric Connection to Meteo Tower

140 Water Intake

141 Cooling Water Admission Canal

142a Hot Water Discharge Canal in Danube DBSC Race 2

142b Hot Water Discharge Canal in Danube in Danube

143 Hot Water – Cold Water mixture canal

146* Telex, phones, weak current connections

150 Special structures and hydrotechnical works for site protection

152 Main Access Road

153 Secondary Access Road

154 Guard House

157 Outer Equipment Warehouse (Seiru Area)

165 Guard House for NPP Security Personnel (Saligny)

166 Suspended Power Line for Power Supply (20kV)

172 NPP Personnel Training Center

173 Commissioning Administrative Ward

261 Hydro Power Station (12 MW) on Cernavoda NPP Hot Water Discharge Duct Caption:

* Networks

List 2

COMPLETE LIST OF THE OBJECTS CORRESPONDING FOR UNITS 3 AND 4 INVESTMENT

NUCLEAR PART (NSP)

001 Reactor Building CR, (R/B)

001a R/B Air Conditioning Equipment Building

002 Service Building CSAN, (S/B)

002a F/M Extension Building

002b SFB Extension Building

003 D_2O Upgrading Tower and Ventilation Stack

004 Emergency Water Supply Building

005 EPS/SCR Building

006 HPECC Building

007 NSP- BOP Interface Building (K-L)

008* NSP Electric Networks

009* EWS Electric Networks

010 EWS Water Intake and Pipe

011 Water Table Drainage Shielding Structure

012 Hydrogen Storage

015 Deck

BALANCE OF PLANT (BOP)

020 Turbine Building SM (T/B)

021 Deareator Building

022 Electrical Services Area

023 Standby Diesel Generators

026 Chiller Building

030 Unit Service Transformers

031 Main Output Transformers

032* Common Service Pipes and Cable Ducts

033 Railway for Transformers

034 Reley Cabinet

039* Drainage Systems for Cable Network Ducts

048 Screen House for Circulating Water

049 Screen House for Technical Water

050 Circulating Water Pump House

051 Technical Water Pump House

052* Cold Water Pipe & Ducts from Condenser (CCW)

053* Cold Water Pipe & Ducts from Turbine (RSW)

054 Hot Water Pipe & Ducts from Condensers (CCW + RSW)

055 Warm Service Water Ducts and pipes in the enclosure

058 Screen House Power Station

070 Diesel Oil Storage

085 Diesel Oil Supply Station

092* Drinking Water Distribution Pipe Network

095* Fire Water Distribution Pipe Network

098* Sewage Water Pipe System including the Pump Station

101 Inside Rainfall Drainage System

102 Inside Low Current and Telecommunication Networks

106 Gate Buildings

107 Roads and Platforms

108 Inside Railways

109 Enclosure Fence

110 Physical Protection

* Networks

List 3

OBJECTS INCLUDED IN THE SERVICES PROVIDED BY SNN FOR UNITS 3 AND 4

IN THE ENCLOSURE

024 Water Treatment Plant

025 Common Service System Transformers

054a Hot Water Pipe & Ducts from Condensers (CCW + RSW) to DBSC Race 2 (Partial)

054b Hot Water Pipe & Ducts from Condensers (CCW + RSW) to Danube (Partial)

062* Raw Water Supply System

063* Raw Water Supply Systems

065 Auxiliary Boiler Station

066 Liquid and Oil Fuel Discharge Ramp

067 Liquid and Oil Fuel Pump Station – Stage I

068 Liquid and Oil Fuel Pump Station – Stage II

069 Motor Oil and Light Fuel Warehouse

071 Transformer and Turbine Oil Warehouse

072 Lubricator Warehouse

073 Emergency Diesel Station for Common Services

077 Hydrogen Production Station

078 Technical Gases Warehouse (H2, CO_2, He, N)

079 Oxygen and Acetylene Warehouse

080 Equipment Warehouse

081 Equipment Storage Platform

082 Maintenance Workshop and Warehouse

084 Motor Oil Separator

086 Motor Oil Supply for Common Service Diesel Group

090 Drinking Water Pump Station

091 Drinking Water Tanks

093 Fire Water Pump Station

094 Fire Water Tanks

095* Fire Water Distribution Pipe Network

096 Fire Cabinet and Ambulance Station

OUT OF ENCLOSURE

131 Meteorological Tower

132 Housing for Meteo Tower Electronic Equipment

133 Environmental Radiological Monitoring Laboratory

134 Electric Connection to Meteo Tower

142a Hot Water Discharge Canal to DBSC Race 2 (Partial)

142b Hot Water Discharge Canal to Danube (Partial)

* Networks

2.1.2. Completion of the Construction of the NPP Units 3 and 4

The Cernavoda NPP enclosure is located on the platform of a former limestone quarry (Ilie Barza Quarry) situated at about 3 km south-east from Cernavoda Town and at 1.5 km away from the first lock of the Danube - Black Sea Canal.

Initially, the quarry had elevations ranging between + 9 m and + 10 mBSL and it was bordered in the north-west by a limestone massif covered by loess and having the elevations + 35 mBSL and + 45 mBSL.

The arrangement of the Cernavoda NPP platform required a large volume of excavations in limestone and surface ground works in loess.

The excavated limestone was used both for the arrangement works on the NPP platform and the arrangement of the temporary buildings and facilities site.

The excess of limestone was stored on the right bank of Valea Cismelei. The loess resulted from the platform arrangement was used for the reclamation of some agricultural land (i.e. the Ramadan area), an area equivalent to the land occupied by Cernavoda NPP.

The above described land arrangement was made for the 5 units planned to be built between 1979 -1985. The works developed with the construction of the main buildings (i. e. Reactor Building, Service Building, Turbine Hall and Hydro Pump House).

Until 1989, the common systems (water, sewage management, firewater, underground cable ducts, hydro ducts) for Units 1 to 5 had been constructed to various completion stages.

It is estimated that the civil work finalization degree for the Unit 3 nuclear part and classical part (except the hydro works) for Unit 3 is about 52 % and 35 % for Unit 4. It is estimated that the hydro works, water supply and sewage networks are finalized 49 % as regards their construction civil part and 2 % for the equipment installation, the balance of 51 % and 98 %, respectively, representing the mean of the work percentage to be finalized for various project Unit 3.

Therewith, for Unit 4 it is estimated that the hydro works, water supply and sewage networks are finalized 30 % as regards their construction civil part and 0 % for the equipment installation, the balance of 70 % and 100 %, respectively, representing the mean of the work percentage to be finalized for various project Unit 4.

The works required for the Units 3 and 4 finalization consist of (Ref. 2-36):

a) Construction:

- excavations within the unit enclosure;
- floor, masonry, building internal structure and equipment foundation repairing works;
- concrete pouring for new building elements;
- steel structure installation works;
- painting, coating, epoxy liner application at various building elements;
- architectural works.

b) Equipment and pipe installation works:

- equipment anchoring to foundations (for equipment not yet anchored);
- pipe support installation by welding to the embedded plates in the structures or by anchoring;
- installation of various pipe supporting metal structures inside;
- installation of pipes and check and adjusting valves by welding;
- finalization of pipe installation and pipe connections to outer networks.

c) Electrical and I & C works:

- installation of cable trays and routes;
- local I & C installation and connection for remote control;
- installation of electrical distribution and control panels;
- installation of process computers, main and secondary control works;

- connection to the national power grid from the electric generator bus bars to the 400 KV transformer station.

d) Clearing and flushing of the process systems and their hydraulic testing.

2.1.3. Site Temporary Buildings and Facilities

The Cernavoda NPP constructors have developed their own temporary buildings and facilities function of the volume of typical works and technologies available at that time so that in 1989, the area was large, including pre-assembling spaces, machining and repair workshops, warehouses and storage, all of them oversized. Note that, at that time, Cernavoda NPP platform was the place where construction works for all the 5 units developed. That is the main reasons for which the temporary buildings and facilities had large spaces.

The area is encompassed between the NPP enclosure, the intake duct, Medgidia Street and the rainfall discharge duct on Valea Cismelei.

After 1990, the works have been gradually ceased for the other NPP units so that the volume of works is that for one unit only, so it is smaller.

For example, the concrete mixing plant is covering a very large area on the left bank of Valea Cismelei and it is outfitted with a sorting station, underground bunkers and aggregate warehouses. This concrete mixing plant was initially sized to fabricate and deliver about 400 - 500 m^3/day concrete and it is still operational. The needs of the site, in case that U3 and U4 works go on, shall not exceed some tens of m^3 of concrete a day, a volume which can be provided by a smaller sized concrete mixing plant.

Most of these oversized temporary buildings and facilities (i.e. warehouses, workshops, storage, etc.) were built long time ago and a large part of them were demolished. Note that they had been generally designed for about 10 year life-span.

The temporary buildings and facilities built after 1990 are generally belonging to some new private enterprises or companies established on the basis of the previous existing companies.

These constructors, generally private companies, came while a small volume of works were developing on the site, and they have developed their own temporary buildings and facilities corresponding to this volume of works.

These temporary buildings and facilities have the following general characteristics:

- they cover a small piece of land and have a high degree of occupancy;
- the units associated to these temporary buildings and facilities are generally including only the actually necessary things, namely:
    - offices;
    - penthouses for tools and construction personnel;
    - warehouses with low volume;
    - halls and workshops specially sized for a low volume of works;
- trimmed aspect if compared to the large site arrangement and the buildings and facilities are well maintained;
- careful storage of materials, tools and equipment within the temporary buildings and facilities area;
- satisfactory cleanness inside the site arrangement.

Another category of buildings and facilities designed for a good development of the construction-installation works on the Cernavoda NPP site is represented by the Owner's storage.

In order to fulfill the Cernavoda NPP Project, a central main warehouse belonging to the Owner, capable to provide the storage of materials and equipment to be purchased, has been designed from the beginning and constructed.

In the warehouses and storage associated to the constructor's temporary buildings and facilities, only the materials directly purchased by the constructor and the equipment that require pre-assembling or verifications, are stored.

The central main warehouse belonging to S.N. Nuclearelectrica S.A., located in the SEIRU area, covers about 5 ha land area and is still housing equipment and materials purchased for all the 5 units of Cernavoda NPP.

The warehouse includes outer storage areas and roofed sheds for keeping the materials and equipment that must not be subject to bad weather conditions.

Materials and equipment inside the warehouse are permanently maintained.

The physical condition of the warehouses is satisfactory and there are spaces, platforms and adequate access roads.

The warehouse has a road connection and a railroad connection to the local railway as well as capacities for on-ramp unloading.

Besides this central warehouse, storage areas have been arranged after 1995, in the Turbine Hall and NSP - Service Building associated to Units 3 and 4 such as extended platforms for pipe storage.

These spaces are mainly housing the materials and components associated to U2 and some materials for Unit 1 maintenance works. In the phase of completion of the Units 3 and 4 construction, the materials must be removed from these spaces.

Moreover, various workshops of the constructors (fabrication of tube and isolation, blasting, painting, etc.) have been built in Unit 4 – Turbine Hall.

All the platforms and temporary buildings and facilities, constructed before or after 1990, are provided with access roads, now in a satisfactory condition.

Big Site facilities are including railway connection to the local railway in the area.

2.2. Technological Processes (During the Unit 3 and Unit 4 Operation)

2.2.1. Nuclear Safety Principles and General Design Criteria

The safety philosophy of the CANDU6 NPP is based on three main safety principles, namely (Ref. 2-36):

i - defense-in-depth;

ii - ALARA;

iii - grouping and separation.

2.2.1.1. Defense-in-depth Principle

The most important goals of the nuclear unit are the attendance of a high level of performance and the assurance of safe operation. These targets are accomplished by means of multiple physical barriers against the radioactivity release to environment and by application of the defense-in-depth principle whose requirements are as follows:

- high standards in the design process, commissioning and operation have to be applied;
- high quality materials and equipment whose performances were established by testing or analyses should be selected;
- application and strictly compliance with the quality requirements in the design, construction, installation, operation and decommissioning phases should be mandatory;
- attendance of a high level of reliability using redundant systems and components as well as by compliance with the diversity concept both in design and fabrication processes;
- a well defined set of operating policies and principles should be approved and adopted which intend to avoid the reactor operation in potentially unsafe conditions.

Five barriers are included in the CANDU PHWR-600 project (four physical and one administrative) by which defense-in-depth is provided, namely:

(1) - fuel matrix;

(2) - fuel element sheath;

(3) - reactor coolant pressure boundary;

(4) - containment system;

(5) - exclusion area.

These five barriers along with the associated systems that assure their integrity are shown in Figures 2.2.1-1, 2.2.1-2, 2.2.1-3, 2.2.1-4 and 2.2.1-5. The systems are structured according to their level of importance from the nuclear safety and process point of view.

The defense-in-depth principle requires a specific approach. Although the best quality components are selected, it is recognized that the equipment may deteriorate and hence the measures/provisions in order to avoid their failures should be implemented. As a consequence, the main requirements imposed on design and safety analyses are to anticipate the potentially failure modes and failures themselves, and shall demonstrate that the nuclear unit is adequately protected.

2.2.1.2. ALARA Principle

The nuclear safety objective is to protect the operating personnel, the public and the surrounding environment against the radiological risk. In order to accomplish this, adequate protective means/measures/provisions should be established and maintained. This objective is also the fundamental criterion which the nuclear unit safety concept is based on. According to the specific risk associated to nuclear activities, the radiation protection objectives have to be realized such that:

- during normal and abnormal operating conditions the personnel and public radiation exposure shall be maintained below the allowable limits and As Low As Reasonably Achievable (ALARA);
- radiation exposure due to accident conditions should be minimized.

As far as the accident conditions are concerned, the following nuclear safety targets should be accomplished:

- generally, the accident conditions have to be avoided;
- the radiological consequences have to be minimized for all postulated event sequences, even for those with a very low probability of occurrence;
- due to the radiation exposure preventive and mitigating measures, the accidents associated with serious consequences should be extremely improbable.

The radiation protection optimization applies wherever the radiation exposure may be controlled using protective measures.

The actual radiation protection strategy is the result of a historical process that reflects the desire to obtain the most efficient radiation protection for public and environment, along with the use of radioactive material on large scale. In this respect, the ALARA principle is the fundamental and decisive factor by which the radiation protection is optimized maintaining the radiation exposure as low as practical attainable, below the allowable dose limits and justified from both the economic and social point of view. This principle derives from the three fundamental criteria associated to systems for radiation dose limiting, that are adopted, settled and identified (ICRP, 1977) as follows:

- no activity with radiation exposure should be developed unless a net positive benefit is produced;
- all irradiation should be maintained as low as practical achievable taking into account both economical and social factors (ALARA);
- the total dose equivalent received by an individual should not exceed the corresponding application dose limits.

2.2.1.3. Grouping and Separation Principle

2.2.1.3.1. Separation

The systems separation is one of the most important safety principle adopted in the design of the nuclear unit. This results from the single/dual failure criterion application as the fundamental concept of the CANDU NPP project. The separation principle applies not only between safety related systems and other systems, but also between safety related systems themselves.

2.2.1.3.1.1. Single/dual Failure Criterion

The plant systems are divided into two groups namely: process systems (required to power production) or systems with essential process functions, and safety systems (required to protect the public and environment) or safety related systems. The single failure is always associated with a process system failure; the dual failure is defined as a process system failure coincident with a special safety system failure or unavailability.

From the dual failure criterion were the Design Basis Events derived, such as a LOCA type event (single failure) coincident with ECCS unavailability (dual failure). This approach resulted in four robust safety special systems provided in the nuclear unit design. The special safety systems are:

- Shutdown System No. 1 (SDS#1);
- Shutdown System No. 2 (SDS#2);
- Emergency Core Cooling System (ECCS);
- Containment System (Isolation, Dousing, Local Air Coolers).

According to the single/dual failure criterion, the following concepts and fundamental requirements associated to special safety systems resulted:

- systems and logical devices physical independence concept;
- definition with maximum accuracy of reliability requirements;
- testing requirements;
- redundancy requirements;
- diversity concept;
- fail-safe concept.

These concepts and requirements are shortly described as follows.

2.2.1.3.1.1.1. Independence

In order to prevent that a special safety systems failure or unavailability affect another special safety system, it is required that independence between each other to be maintained, both physically and between logical devices. This means that there is no special safety system equipment in common for two or more systems (e.g., pump, valve, detection device, etc.). However, the electrical power supplies are common provided that the redundant sources are available (e.g., three set of batteries with separated supply buses, or two Diesel generators with separated supply buses, etc.).

2.2.1.3.1.1.2. Reliability

It is required that the special safety systems reliability to be very high in order to reduce the dual failure occurrence frequency. The unavailability target for each special safety system was settled at 10^{-3} year/year, value that is verified by reliability analyses.

2.2.1.3.1.1.3. Testing

It was developed a periodically testing program applied to any system components in order to demonstrate that the unavailability target is accomplished and this value is maintained during the nuclear unit operation.

2.2.1.3.1.1.4. Redundancy

The special safety systems are provided with duplicated or triplicated components in order to maintain a high level of reliability and testability during operation. For instance, special safety systems trip logic as well as the active logic of other safety related systems, are triplicated: there are three protective channels that process the same information. If two channels receive an actuation signal then the special safety system will trip (2-out-of-3 logic). Any failure that affects one protective channel will trip that channel, which means a safe state for it.

2.2.1.3.1.1.5. Diversity

Wherever is possible, the special safety systems components that satisfy the same function were supplied from different manufacturers and have a different design such that the common mode failure being avoided. For example, the dousing system consists of six dousing headers, each of them having an isolation valve in normal closed position. Three of these valves are supplied from a manufacturer being pneumatically actuated, and the other three are supplied from another manufacturer having electrically actuation devices.

2.2.1.3.1.1.6. Fail Safe

All special safety systems are designed such that any mechanical failure or loss of electrical power supplies, or protective logic channels failure associated to their components, should allow the reactor to reach a safe state or should assure an alternative mean to accomplish the corresponding function. For example, the valve that controls the neutron liquid poison injection (SDS#2) is designed such that it opens forced by a spring and the poison solution required to reduce the core reactivity is injected, should the instrument air supply is lost.

2.2.1.3.1.2. Special Safety Systems Acting Devices

Passive components are used for special safety systems in order to fulfill the allotted functions. SDS#1 is actuated by means of gravity and compressed springs that force the shut-off rods to be inserted into the core (normally, they are parked above the reactor core). SDS#2 uses as actuator the compressed helium stored in a tank. When the gas is expanded the gadolinium nitrate solution stored in injection tank is injected in moderator volume from calandria vessel.

The only active elements are the clutch devices that sustain the shut off rods in withdrawn position (SDS#1), and the quick acting valves that initiate the helium injection (SDS#2). The clutch devices (SDS#1) are normally powered so that on loss of power supply the shut off rods are dropped into the core providing the system fail safe state. The quick acting valves (SDS#2) use compressed air to maintain their closed position. On loss of instrument air the valves are very fast opened by mean of compressed springs and the poison solution will be injected into the core providing the system fail safe state.

The containment and emergency core cooling systems are not passive systems, the fulfillment of their associated functions necessitating power, instrument air and cooling water supplies. Normally, these services are provided by the plant process systems. However, there are provided back-up supplies required to sustain the safety systems operation, should the normal supplies become unavailable. The back-up supplies are known as safety support systems and include the emergency power supply system (EPS), the emergency water supply system (EWS) and backed-up instrument air supplies provided by local compressed air tanks.

2.2.1.3.2. Grouping

2.2.1.3.2.1. Conceptual Basis

Despite the fact that the special safety systems are well designed and a high level of reliability is assured according to the previous design safety criteria, there are some events that might alter the associated safety functions. This category of events includes common cause or external events and man induced events (earthquakes, fires, missiles, etc.). The main characteristic of these events is that they may affect large areas of the plant in the same time.

Concerning with this aspect, the plant design adopted the two group separation principle for safety related systems, Group 1 and Group 2, in order to provide an adequate protection against such events. The design intent is to assure the accomplishment of the essential safety functions, even if a common cause event should produce damages on large areas of the plant.

2.2.1.3.2.2. Key Safety Functions

The key or essential safety functions are those functions required to place the nuclear unit in a safe steady state following an accident and to maintain it for an indefinite period of time. These functions are as follows:

- to maintain the reactivity control, equivalent with the reactor shutdown and to maintain it in safe shutdown state;
- to cool the nuclear fuel, equivalent with the residual heat removal;
- to confine the radioactive materials, equivalent with the mitigation of the radioactive releases to environment;
- to monitor the plant status, equivalent with the plant control and to maintain the plant monitoring capabilities;
- to reduce the radiological consequences.

The four special safety systems are divided into two groups as follows:

Group 1	Group 2
Shut Down System No.1, SDS#1	Shut Down System No.2, SDS#2
Emergency Core Cooling System, ECCS	Containment System

For any postulated event, including LOCA, each group of systems, acting alone, shall limit the off site radiological consequences at the reference level corresponding to a dual failure. The containment system and ECCS comply with the design requirements and objectives regarding to the accomplishment of key safety functions, although these are from different groups of systems. Thus, should a LOCA type event occur and the ECCS is unavailable, the nuclear fuel shall be seriously damaged but the containment system will mitigate the radioactive material releases to environment. Reciprocally, a LOCA type event occurrence with the ECCS available shall not generate a systematic fuel failure, such that the radioactive materials inventory that potentially may be released to environment is limited, should the containment system fails. For both scenarios the off site dose levels comply with the allowable reference limits.

The Group 1 of systems is provided with back-up support systems that include normal power and cooling water supplies, while for Group 2 the safety support systems (EWS, EPS, local instrument air tanks) are allotted.

All the buildings and structures pertaining to Group 2 of systems, including Secondary Control Area (SCA), are seismically qualified to Design Basis Earthquake (DBE). The SCA is equipped with the required instrumentation necessary to plant stabilization, should an earthquake will make the Main Control Room (MCR) unavailable or will generate inadequate operating condition of it. The SCA may be used also when other events occur but their consequences have similar effects to those induced by an earthquake (e.g., steam line break outside containment, fires, toxic gases releases, sabotages, terrorist strikes, etc.).

2.2.1.3.2.3. Location and Cable Trays

The Groups 1 and 2 are physically separated by distance or barriers in order to assure that, in case of a postulated event occurrence, the systems from both Groups will not be jeopardized. The location of buildings and structures is realized such that there is no straight pathway or cable free trays between the redundant elements of two groups above the soil level. For instance, MCR belongs to Group 1 and it is located at the third level of Service Building (S/B). The cable trays pass through Reactor Building (R/B) wall at one side and follow a certain path up to control equipment room. Secondary Control Area pertains to Group 2 and is located in the east side of R/B, away from MCR and at lower floor level. The cable trays associated to Group 2 pass through R/B wall at a different side from those of Group 1.

2.2.1.4. General Safety Design Criteria

A set of principles and criteria essential for nuclear safety is applied in the design, construction and operation of the nuclear unit. One of the most important nuclear safety requirements is to maintain the personnel and public radiation exposure ALARA during all operating and postulated accident conditions. The design process includes aspects regarding plant safe operation by establishing a coherent set of operating requirements and restrictions based on the general safety design criteria. Shortly, the safety design criteria applicable to a nuclear power plant include the issues described below.

2.2.1.4.1. Ergonomic Principles

Based on the interest towards the accomplishment of nuclear safety requirements, the environment and working areas in which the operating personnel has to develop his activity are designed complying with the ergonomic principles.

2.2.1.4.2. Radiation Protection

In order to fulfill the settled nuclear safety objectives, measures that assure the compliance of these goals concerning the radiation protection of the operating personnel, public and the surrounding environment are provided.

2.2.1.4.3. Nuclear Safety Functions

During the design process it is essential that the nuclear safety be treated as an inseparable element in order to obtain the desired safety level. Consequently, the following general requirements shall be accomplished by the plant project:

- to provide means to shut the reactor down in a safe manner and to maintain it in safe shutdown state from any operating regime including during and following postulated accident conditions;
- to provide means for residual heat removal from reactor core following shutdown, including postulated accident conditions;
- to provide means in order to reduce the radioactive materials potential releases and to assure that the possible released radioactivity will be lower than operating condition limits and below the maximum allowable limits in accident conditions.

2.2.1.4.4. Plant Safety Features

The basic issue of the defense-in-depth principle should be reflected in the plant safety features. Consequently, the nuclear unit should be designed and operated such that the consequences of any postulated initiating event or sequence of initiating events must be as closed as much technical by feasible to one of the following scenarios:

- a postulated initiating event will not produce any significant effect that might affect the nuclear safety, or it shall involve only a plant condition change towards a safe state due to its intrinsic characteristics;
- following a postulated initiating event the plant will be in a safe state by the action of those systems that are continuously operating in this state, in order to control the event;
- following a postulated initiating event the plant will be in a safe state by the action of those systems that are initiated in order to react against the event occurrence.

2.2.1.4.5. Design Basis

The design basis includes the specification of plant capabilities to withstand against a determined domain of operating and accident conditions, complying with the radiation protection requirements. Consequently, the design basis includes all specifications regarding normal operating condition, postulated initiating events, design standards and regulations and site characteristics.

2.2.1.4.5.1. Normal Operating Condition

The design studies contain details about the nuclear unit safe operation in normal conditions, as well as a set of operating requirements and restrictions, including the followings:

- restriction of process variables and of other important parameters;
- special safety systems trip setpoints;
- plant maintenance, testing and inspection requirements provided in order to confirm that structures, systems and components are operating according to design assumptions.

These requirements and restrictions are setting up the fundamentals for safe operating limits within which the nuclear unit operation can be authorized.

2.2.1.4.5.2. Postulated Initiating Events

The nuclear unit project considers a set of postulated initiating events corresponding to the case when all the protective groups of systems shall be challenged, as well as the provisions of any necessary measures with the aim to comply with the nuclear safety objectives.

2.2.1.4.5.3. Design Standards and Regulations

The structures, systems and components design uses acceptance criteria in terms of standards or design technical rules. These regulations are clearly set by the Authority based on the applicable standards and technical norms already adopted, or based on those used worldwide (e.g., ASME code).

2.2.1.4.5.4. Site Characteristics

An important factor for the plant design basis development is represented by different interactions between the nuclear unit and environment factors such as demographic, meteorological, hydrological, geological and seismological characteristics and also the effects of external human activities nearby. It should be also considered the possible difficulties concerning the site external services on which the nuclear unit safety and public protection may be dependent.

2.2.1.4.6. Accident Conditions

The design basis for normal operating conditions, initiating events and postulated accident conditions should confirm, within a very high level of confidence, that the reactor core shall not be significantly damaged and the radioactivity releases will be below the prescribed limits for operating conditions and lower than the allowable limits in accident conditions. Some event sequences may lead to a significant reactor core degradation being known as severe accidents. From the nuclear safety point of view, it is cautious to consider also the representative and dominant severe accidents, at least in a limited manner (e.g. emergency plan preparedness).

2.2.1.4.7. Quality Assurance Requirements

All safety functions allotted to the structures, systems and components are clearly defined and classified according to their importance for the plant nuclear safety. A special attention is paid to all aspects of quality assurance (structures, systems and components design, materials selection, technical specifications, fabrication, construction, installation, operating procedures, maintenance, testing, high qualification operating personnel selection), in order to assure a high reliability in operation and to fulfill the nuclear safety functions.

2.2.1.4.8. Testing, Maintenance, Repair, Inspection and Surveillance Requirements

The safety related structures, systems and components are conceived and realized such that these can be calibrated, tested, maintained, repaired and inspected or surveyed to demonstrate their capacity to accomplish the allotted functions during the plant life time.

2.2.1.4.9. Systems and Components Reliability

It is necessary to attain and maintain a high reliability for the safety related structures, systems and components in order to accomplish their intended functions. As long as there can not be defined universal quantitative objectives to different reliability requirements for each group of protective systems, it seems natural that the first protective systems group to be analyzed in detail. The principles and design characteristics associated to a high level of reliability for the safety related structures, systems and components are summarized as follows.

2.2.1.4.9.1. Redundancy

Redundancy, that means the use of a number of equipment greater than the minimum required to accomplish the specific safety functions, represents an important design principle applied to improve the safety related systems reliability and to comply with the single failure criterion for special safety systems. The redundancy allows for a component failure or unavailability to be tolerated without compromising the associated safety function.

2.2.1.4.9.2. Single Failure Criterion

A system or component comply with the single failure criterion if it can fulfill its intended function in spite of a random single failure or defect that would affect a component or sub-assembly. The single failure criterion should be applied to each safety group included in the plant project. A safety group is an assembly of systems or components that execute all necessary actions, should a postulated initiating event occur, such that the limits specified in design basis are not exceeded for this event. Non-compliance of single failure criterion requirements may be justified whenever:

- the occurrence probability of initiating events is extremely low;
- the consequences of an initiating event occurrence are very unlikely;
- certain components are out of service during limited period of time in order to maintain, repair or test them.

2.2.1.4.9.3. Diversity

The system reliability increase may be attained also applying the diversity principle to reduce the frequency of potential common cause failure. The diversity applies to redundant systems or components that accomplish the same safety function, with specific characteristics for different systems and components. These characteristics may be different operating principles, different physical parameters, different operating conditions or supplied from different manufacturers.

2.2.1.4.9.4. Independence

Applying the independence principles results in the systems reliability increase. These principles are as follows:

- to maintain the independence between redundant components of a system;
- to maintain the independence between redundant components of a system and the effects of postulated initiating events;
- to maintain an appropriate degree of independence between systems or components pertaining to different safety categories;
- to maintain the independence between the safety related and non safety related components.

The independence is effective to the systems level by functional isolation and physical separation.

2.2.1.4.9.5. Fail Safe

The fail safe principle should be applied to safety related systems and components, whenever this is required. The application of this principle will place the nuclear unit in a safe state without corrective actions initiation, should a system or component failure occur.

2.2.1.4.9.6. Auxiliaries Services

During the plant life time provisions should be made in order to assure the necessary auxiliaries services required to maintain the plant in safe state. These services consist of power supply systems, cooling water systems, compressed air systems, process gases, lubricants, etc. The auxiliaries systems pertaining to an equipment from a safety related system are integral parts of it. The reliability, redundancy, diversity, independence and the isolation of these systems to test their functional capabilities, should be reported to the global reliability of the assisted system.

2.2.1.4.9.7. Unavailability

Concerning the nuclear unit and safety related systems operational reliability, consideration is given to components unavailability as well as to the provided impact of maintenance, testing and repair activities on every safety related system.

2.2.1.4.10. Ultimate Heat Sink

The plant safety related structures, systems and components are provided with specialized systems required to transfer the residual heat towards the ultimate heat sink. This function should be accomplished during all the operating conditions and the postulated accident conditions.

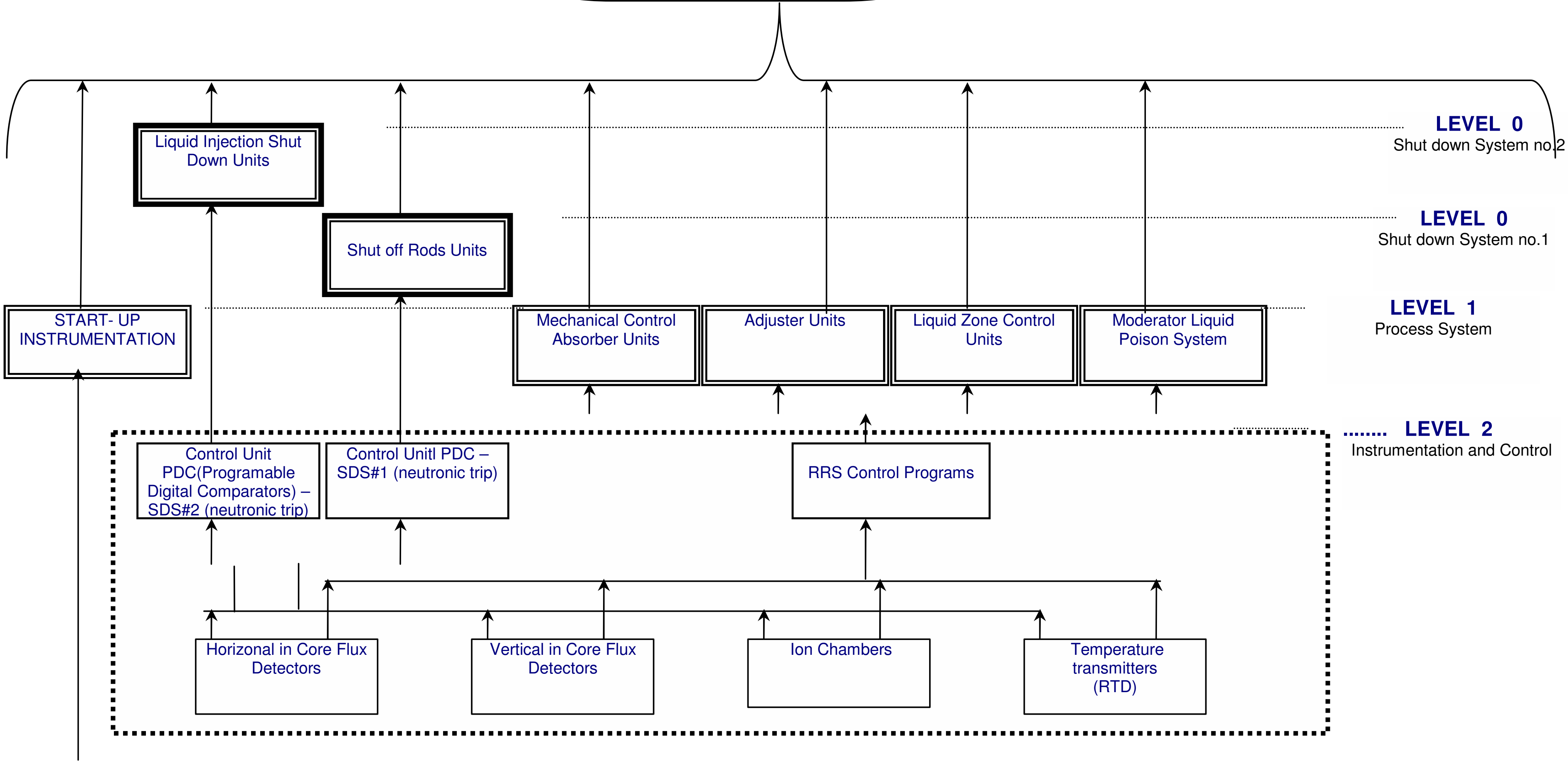
2.2.1.5. CANDU Design Requirements

As specified in Licensing Basis Documents for Cernavoda NPP Unit 3 and Unit 4 (Ref. 2-37, 2-38), the safety requirements for the reactor containment that apply to CANDU design are provided in CNSC (Canadian Nuclear Safety Commission) regulatory document R-7, *Requirements for Containment Systems for CANDU Nuclear Power Plants (February 1991),* which have been endorsed in the corresponding Romanian Norm CNCAN NSN-12 (Ref. 2-39).

The code effective date for Cernavoda NPP Unit 3 and Unit 4 project is April 1, 2005 with the exception of the codes related to the existing Civil structures, which were built to the codes at the time of Cernavoda 1. This is the case of Units 3 and 4 containments. The Civil structure Code Effective Dates will be the same as those used for Cernavoda 1, with some assessments to determine if the existing structures meet the April 1, 2005 basic requirements. If required, improvements to the existing civil structures will be implemented as long as the risk-cost benefit principles are followed.

The design, construction, commissioning and operation of the plant will be based on the Nuclear Standards as listed in Licensing Basis Documents for Cernavoda NPP Unit 3 and Unit 4 (Ref. 2-37, 2-38). The design documentations clearly identify the codes and standards being applied and describe the extent and limitations, if any, of their application.

Figure 2.2.1-1. Barrier 1
Nuclear Fuel



2.2.2. Process Systems Description

Cernavoda NPP Unit 3 will produce electric power based on nuclear power generation by successive transformation of the fission energy into thermal energy in the nuclear reactor, of the thermal energy into mechanical energy in the steam turbine and of the mechanical energy in electric energy in the electric generator. The plant turbine generator output power is 720 MWe, produced by converting the steam generated using the energy developed in a CANDU-PHWR-600 nuclear reactor type. The same characteristics of the technological process are valid for Unit 4 as well.

This type of reactor which both are common to design of nuclear units, is employing heavy water both as moderator and coolant, in two separate systems. The fuel is natural uranium in form of uranium dioxide syntherized pellets shielded in zircaloy pencils, as fuel elements, and assembled in fuel bundles which contain 37 fuel elements each. The ceramic pellets have the property to confine the fission products within themselves. The reactor fuel loading and unloading is continuous, bi-directional and with the reactor at power. The reactor is provided with a heat transport system with two independent loops which transfer the heat generated by the fuel during the controlled fission reaction, to four steam generators which contain light water. The saturated steam from the steam generators is expanding into the turbine developing mechanical work and afterwards, passing through the condenser, the steam is cooled by the light water taken from the Danube River via an open intake duct connected to Race 1 of the Danube - Black Sea Canal (DBSC).

The Unit 3 and Unit 4 structures, systems and components will be designed, manufactured, installed and operated according to the content and requirements of the plant project Current Licensing Basis (CLB) approved. The reference project of Units 3 and 4 is based on Cernavoda NPP Unit 2 "*as commissioned*" amended by a set of changes in order to improve the technical, economical and nuclear safety performances. The main Unit 3 and Unit 4 design changes versus Unit 2 project are as follows (Ref. 2-36):

a) *Modifications that are required as a consequence of equipment replacement or improvement due to its obsolescence*

Many equipment or components should be replaced due to obsolescence. But not all of them impact on cost or nuclear safety. The replacement of plant digital computer control, DCC, with distributed control system, DCS, technology, should affect the plant configuration and will have the significance of an improvement versus Unit 2 project.

b) *Modifications imposed by the licensing and nuclear safety additional requirements*

These changes cover the improvements and additions versus reference plant (Unit 2) due to the implementation of new licensing and safety requirements applicable from April 2005. In this concerne, there are those that ensure an improved nuclear safety by conformance with Canadian regulatory documents (e.g., CNSC R-7, regarding the containment isolation), by qualification of some systems and components for severe accident conditions (e.g., calandria vault relief devices, ESCS), by qualification of some process systems as ultimate heat sink in accident conditions (e.g., main moderator D_2O supply), or by the mitigation of radiological risk to public (e.g., main steam isolation valves).

c) *Modifications that are required to implement the latest design codes and standards editions*

Cernavoda Unit 2 NPP used as a code effective date for licensing basis documentation the 1989 edition or later. However, AECL used the documentation with code edition dated 1992, and addenda of 1994. For Unit 3 and Unit 4, the code edition of the licensing basis documentation was established to be 2005, corresponding to an effective code edition dated 2004, excepts for the code related to civil structure (code edition alike Unit 1). For this reason, the civil structure documentation code edition effective date will remain the same as for Unit1, with some assessments in order to confirm that the structures meet the 2005 edition basic requirements. The changes contain the codes and standards applicable to Unit 2 project, but updated on the latest available version. The impact of these updated code edition on the plant project will be assessed a later date, but will be finalized prior to the formal definition of the code edition effective date agreed by Authority.

d) Modifications imposed by operating experience feed-back and by improvement of project economical indicators

These changes arrised as a result of the extensive review program carried out on feedback from various CANDU-600 NPP, regarding the operating experience. It is considered that this kind of modifications may alleviate the operational problems encountered up to date, and their character becomes mandatory. Such of modifications are as follows: replacement of nuclear diaphragm valves with equivalent ball valves, changes of the elevation for some equipment and system connected to reactor, changes of some components associated to steam generators, complete separation of PHTS feed pumps recirculation lines, provisions for the second auxiliary boiler feedwater pump, improvement of the main control room, etc.

Upon the recent experience in the U2 licensing process, it resulted that significant modification have not occurred and neither such modifications have altered the hypotheses and conclusions of the environmental impact analyses in any way.

Certainly, the considered modifications result in an increase of the nuclear performance of the unit and the reaching of a high nuclear safety level according to the European Standards (see Table No 2.2.2-1).

Figure 2.2.2-1 shows the basic diagram of a CANDU type NPP.

Table 2.2.2-1 presents the design characteristics of a power unit fitted with a CANDU 600 type reactor.

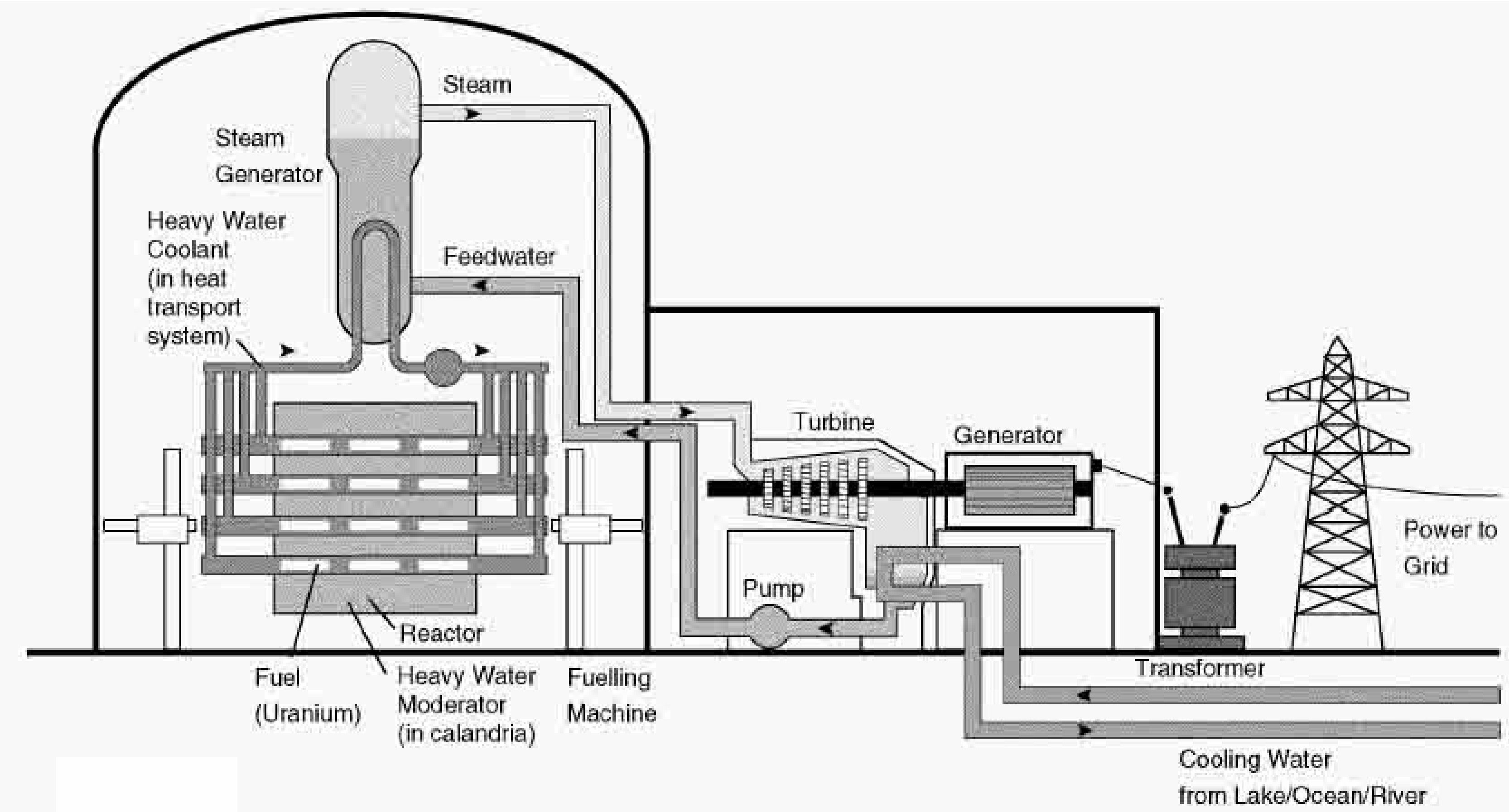

Steam
Steam Generator
Heavy Water Coolant (in heat transport system)
Feedwater
Turbine
Generator
Power to Grid
Pump
Reactor
Fuel (Uranium)
Heavy Water Moderator (in calandria)
Fuelling Machine
Transformer
Cooling Water from Lake/Ocean/River

Table 2.2.2-1. Design Characteristics of Cernavoda NPP for One Unit

Number of Nuclear Rectors	1
Electrical Power to Grid	720 MWe *)
Fission Power	2180 MWt
- Nuclear Reactor	
Type	PHWR
Moderator and Reflector	D_2O
Coolant	Pressurized D_2O
Fuel	Natural UO_2
Refueling	On-power, Bi-directional in Adjacent Fuel Channels
Refueling Direction	Coolant Flow Direction
Fuel Cycle	Non-recycle
Number of Fuel Channels	380
- Fuel Channel	
Fuel Channel Length (including end fittings)	10.8 m
Reactor Core Coolant Flow	7.7 Mg/s
Reactor Inlet / Outlet Header Temperature	266 / 312 °C
Reactor Inlet / Outlet Header Pressure	11.04 / 10.3 MPa
- Pressure Tube	
Material	Zirconium - 2.5% Niobium Alloy
Inside Diameter	103 mm
Maximum Channel Power	6.5 MW
Reactor Outlet Header Quality	3%
- Calandria Assembly	
Total Length /Inlet Length - Calandria Assembly	7.82 / 5.94 m
Number of Calandria Tubes / Material	380 / Zr-2
Inside Diameter (minimum) Outside Diameter (maximum)	129 mm 132.3 mm
- Primary Heat Transport System (PHTS)	
Primary Coolant Inventory	133.18 Mg
Reactor Core Coolant Flow / Maximum Fuel Channel Coolant Flow	7.7 Mg/s / 24 kg/s
Number of Steam Generators (SG)	4
Steam Generators Type	Vertical U-Tube with Integral Steam Drum and Pre-heater
Primary / Secondary SG Design Pressure	10.03 MPa / 5.07 MPa
Primary / Secondary SG Design Temperature	318°C / 266 °C
Pressure at SG Inlet / Outlet Nozzle	9.9 MPa(a) / 9.6 MPa(a)
Temperature at SG Inlet / Outlet Nozzle	309°C / 266°C
Quality in SG Inlet / Outlet Plenum	4.4 %
Heat Transferred to Secondary System	2064 MWt

Table 2.2.2-1. Design Characteristics of Cernavoda NPP for One Unit (continued)

Feedwater Flow	958 kg/s
Feedwater Inlet Temperature (at Full Power)	187℃
Blowdown Minimum / Maximum Continuous Flow	1 kg/s / 3 kg/s
Primary Heat Transport Pumps	4
Primary Heat Transport Pumps Type	Vertical, Centrifugal, Single Suction, Double Discharge
Steam Generator nominal flow	1044 kg/s
- Main Steam System	
Number of Steam Pipes	4
Design Temperature / Pressure	266℃ / 5.07 MPa
Number / Type of Main Steam Safety Valves (MSSV)	16 / Spring and Air Operated Valves
MSSV Relief Setpoints	5.0 ÷ 5.14 MPa
MSSV Capacity	1.200 Mg/s
Discharge	Atmosphere
Discharged Flow	Saturated Steam
- Nuclear Fuel	
Material	UO_2 Natural sinter and compacted Pellets
Structure	Fuel Bundle with 37 Fuel Pins
Fuel Weight: - UO_2 - U	 96 Mg 84.5 Mg
Number of Fuel Bundles per Reactor Core	4560
Number of Fuel Bundles per Fuel Channel	12
Fuel Bundle Weight UO_2 / U	21.3 kg / 18.8 kg
- D_2O Management	
System Inventory: - Moderator and Associated Systems - PHTS and Associated Systems - Fuel Handling System TOTAL	 267.6 Mg 184.8 Mg 4.5 Mg 456.9 Mg
D_2O irrecoverable losses	14.3 Kg/day
- Other parameters	
Plant Lifetime	30 years **)
Plant Power Availability Factor	> 88 %
Plant Internal Services Average Power	7 % Gross Power
Nominal Frequency	50 Hz

Note:

*) The 720 MW(e) rated power implies the provision of a new turbine in the Units 3 and 4 design, with implicit modifications in the thermal cycle, which represents an improvement of the design. The modification is feasible and it will be analyzed in detail in the next phases.

**) Plant lifetime extension is envisaged

Unit 3 similarly Unit 4 is a complex objective which includes an assembly of buildings, installations and systems aimed to accomplish the technological process and the public protection function in case of a process failure. Besides these objects, there are their auxiliaries and some common buildings and facilities for several nuclear units. When locating the buildings inside of the Units 3 and 4 sites, the nuclear safety requirements, minimum allowed distances, protection distances corresponding to the process and circulation fluxes have been considered and optimum construction and operating conditions were provided.

The general arrangement of Cernavoda NPP buildings & structures is shown in Drawing No. U3/U4-08230-6024-CU/PG-6025-2-GA-0, Rev.1. The main buildings and structures for Units 3 and 4 of the Cernavoda NPP are (Ref. 2-36):

- Nuclear Steam Plant (NSP) buildings;
- BOP buildings and structures;
- Common systems on the NPP platform.

2.2.2.1. NPP Buildings and Structures

2.2.2.1.1. Nuclear Steam Plant (NSP)

NSP main buildings are:

- Reactor Building;
- Service Building;
- Emergency Power Supply and Secondary Control Area Buildings (EPS/SCA);
- High - Pressure Emergency Core Cooling System Building (HP - ECCS);
- D_2O Upgrading Tower and Ventilation Stack.

2.2.2.1.1.1. Reactor Building (R/B)

The Reactor Building (R/B) process function is to support the nuclear systems and equipment and protect them against natural phenomena and external events. Besides, the R/B is shielding the external environment against radiation, maintaining the radioactive material leakage rate below the allowable limits (0.5 % of the containment atmosphere volume/ day at the 124 kPa design pressure).

The envelope of R/B (containment) represents the 4th protection barrier against the radioactive releases to the environment. The Reactor Building structure is designed to an internal pressure of 124 kPa, a value which is higher than the value corresponding to the Design Basis Accidents (DBA) that imply radioactive release, namely, large LOCA coincident with dousing system unavailability. The containment maintains its structural integrity in case of main steam line break (MSLB) inside containment coincident with partial loss of dousing, this sequence being credited with the highest pressure transient in R/B.

This building houses the nuclear reactor and its control and auxiliary systems, the primary heat transport system, moderator system and part of the special safety systems.

The R/B is made up of the following main pre-stressed concrete structures: base slab and containment. The containment is made up of a cylindrical perimetral wall. The containment inside separation is made by internal reinforced concrete structures.

On the upper part of the R/B there is a tank made up from a spherical reinforced concrete dome parallel to the R/B dome and surrounded by a ring beam with the inner surface covered by a fiber glass reinforced epoxy liner. The tank is storing the required water inventory necessary in accident conditions for dousing system, emergency core cooling system, as well as the make up water for the steam generators secondary side, should a total loss of feed water supply event occurs.

In order to minimize the radioactive releases to the environment, the R/B inside surface is covered by a synthetic based resin seal coating.

The R/B is seismically qualified at DBE (Design Basis Earthquake).

2.2.2.1.1.2. Service Building (S/B)

The S/B is a multifunctional structural assembly located adjacent to the R/B, irregular in shape and a differentiated loading condition, determined by the functional and size requirements of each area.

Service Building is made up of a cast-in-place reinforced concrete infrastructure and a seismically braced steel superstructure closed by thermal isolation coated steel panels.

This building houses: the main control room (MCR), D_2O moderator purification system, ECCS Low Pressure heat exchangers and pumps, spent fuel bay cooling system, R/B ventilation system, electric equipment and I&C associated to Cernavoda NPP-Unit 3 and similarly Unit 4, emergency planning room, spent fuel reception bay and spent fuel storage bay, liquid radioactive spent fuel and spent resins, heavy water management structures, warehouses, workshops, change-rooms, decontamination center and labs.

S/B is seismically qualified at DBE so that it maintains its structural integrity, it prevents the failure of the inside seismically qualified systems if a seismic event occurs and it provides personnel access for the operation or maintenance of such seismically qualified systems.

The connection between the S/B and the R/B internals is made via the equipment and personnel airlocks.

2.2.2.1.1.3. Emergency Power Supply and Secondary Control Area Buildings (EPS - SCA)

EPS - SCA is located in the vicinity of the Reactor Building.

The Building houses the emergency power supply system consisting of two Diesel generator sets and their auxiliary systems, as well as the equipment of SCA which is aimed to provide the main nuclear safety functions accomplishment, should the Main Control Room becomes unavailable. The two areas, EPS and SCA, are separated through a reinforced concrete wall.

The building has a reinforced concrete infrastructure and a steel superstructure. In the SCA area, there is a building basement. The building is seismically qualified at DBE.

2.2.2.1.1.4. High Pressure Emergency Core Cooling System (HP - ECCS)

The building houses the equipment consisting of 2 vertical tanks with light water and a horizontal tank filled with gas, along with their auxiliaries.

HP-ECCS building is located near the R/B and is made up of a reinforced concrete basement infrastructure and a steel frame superstructure; the building closures are made of corrugated sheet, thermal insulated light panels.

The building is seismically designed and qualified at DBE.

2.2.2.1.1.5. D_2O Upgrading Tower and Ventilation Stack

D_2O-Upgrading Tower includes the D_2O upgrading systems, the ventilation stack and the related mechanical and electric equipment. The building is provided with a steel superstructure and facilities for inspections and controls, as well as I & C locating devices to monitor the parameters of the gases exhausted via the stack.

The ventilation stack is provided with pipe connections for the air exhaust ducts from S/B and D_2O-Tower, as well as for the radioactive gaseous effluent monitoring. The pollutants are retained through special fillers located on the systems connected to the stack.

The releases trough the stack are monitored by continuous sampling in order to measure the concentrations of radioactive iodine, noble gases and particulate. Tritium is monitored by periodic sampling and laboratory analyses. When certain radioactivity limits in the air exhausted via the stack are exceeded, this is signaled by an alarm system to the control room.

2.2.2.1.2. Balance of Plant (BOP)

The main buildings and structures in BOP of the Units 3 and 4, consist in:

- Integrated Buildings (Turbine Hall, Degasser Building and Electric Bay);
- Cooling Water Pump House;
- Repair workshop;
- Nondestructive Examination Building.

2.2.2.1.2.1. Main Integrated Building (Turbine Hall, De-aerator Building, Electric Bay)

The Integrated Building, separated from S/B by a gap building, includes the Turbine Hall, the De-aerator Building and the Electric Bay and it is made of a cast in plan reinforced concrete infrastructure and a steel superstructure.

The Turbine Hall houses the turbine, the electrical generator, the condenser and their associated auxiliary systems (compressed air, feedwater to the steam generators, etc).

The de-aerator building and electric bay accommodate the de-aerator, the auxiliary equipment and the electric power distribution equipment.

The gap building (K - L area) is separating the BOP (Turbine Hall) from NSP (Service Building). This building is seismically qualified to DBE so that in case of an earthquake it maintains its structural integrity and S/B structures and seismically qualified systems are not damaged.

2.2.2.1.2.2. Cooling Water Pump House

The Cooling Water Pump House accommodates the turbine condenser cooling water pumps, the raw service water pumps, the fire water pumps and their associated auxiliaries.

Cooling Water Pump House is made up of cast-in-place reinforced concrete infrastructure and concrete prefabricated plate covered steel superstructure. It is provided with some common systems aimed to clean the Danube raw water from dirt and prevent the floating particle intrusion into the circulating water system and the raw water system.

The building is seismically designed according to the Romanian Standard P100 - 81.

2.2.2.1.3. Common Facilities on Cernavoda NPP Site

Cernavoda NPP Unit 3 and Unit 4 are serviced by some structures and buildings common to the other nuclear units as well, such as:

- Solid Radwaste Intermediate Storage (extension);
- Spent Fuel Interim Storage Facility (extension);

- Start-up Thermal Plant;
- Water Chemical Treatment Plant (extension);
- Fire Fighting Building;
- Hydrogen Generating Station;
- Water Intake and Discharge Channels;
- Administrative Building.

2.2.2.1.3.1. Solid Radwaste Intermediate Storage (DIDR)

The storage is located in the physical protection area of Cernavoda NPP - Unit 1, on NE direction, on the left branch of Valea Cismelei and it is common to several units. It consists of the main storage hall for general wastes, the spent filtering cartridges storage and the quadricell.

The general waste storage hall is made of reinforced concrete panels at its lower part and light panels, at the upper part, the roof being made of steel trusses and ROMPAN type panels.

The spent filtering cartridges storage is a building made of reinforced concrete with concrete slab roof, provided with 126 storage enclosures for the filtering cartridges. The quadricell is a reinforced concrete structure with concrete slab roof, provided with 8 storage enclosures. The filtering cartridges storage and the quadricell are seismically qualified to DBE and the general radwaste storage hale is seismically designed according to Romanian standard P100 -92.

Initially, the storage of the solid radwaste coming from the two nuclear units was provided. The extension of the storage complying with the U3 and U4 solid radwaste production will be analyzed.

2.2.2.1.3.2. Interim Spent Fuel Dry Storage (DICA)

Interim Spent Fuel Storage is located on Cernavoda NPP site, close to Unit 5, at above 700 m far from Unit 1.

DICA is designed for temporary storage for a period of time, of minimum 50 years, to store the spent fuel resulted from Cernavoda NPP Unit 1 and Unit 2 operation.

This storage is a dry one, made in MACSTOR concrete modules (AECL project), after a 6 years cooling period in Spent Fuel Bay.

For the spent fuel storage resulted from Units 3 and 4, DICA is necessary to be extended. This will be made in order not to affect the technological solution already accepted and approved by the respective Authorities.

2.2.2.1.3.3. Start -Up Thermal Plant

The basic function of the auxiliary steam generation by the Start-Up Thermal Plant is to provide an additional steam supply source for the nuclear unit start-up or shutdown condition. During normal operation the steam required to the NSP and auxiliaries is provided by self-supply. When one of the units is shut down, the steam required for that unit is supplied by the adjacent unit or by the Start-Up Thermal Plant.

The Start Up Thermal Plant (STP) is outfitted with 3 × 30 t/h steam boilers with the generated steam parameters: 15 bar pressure, 250 °C temperature; one 4 t/h steam boiler with the steam parameters: 15 bar pressure, 200 °C. The 30 t/h boilers are connected to a common steam header which is provided with a ϕ 273 × 8 mm pipe connection to NPP Main Building. The steam generated by the 4t/h boiler is supplied to the NPP Main Building via a ϕ 108 × 4 mm pipe.

The Start - Up Thermal Plant becomes operational when the steam supplied from the plant is unavailable and it can supply the steam within 30 minutes from the start-up signal of the fast start-up boiler (4 t/h) and after 3.5 hours, the maximum steam flow of 90 t/h from the 3 × 30 t/h boiler.

2.2.2.1.3.4. Water Intake and Discharge Ducts

The cold water source for the NPP service cooling water systems (circulation water and service water) is the Danube River. The water is taken from the Danube at the Danube - Black Sea Canal (DBSC) intake, passed through the DBSC - Race 1 and the derivation canal to the intake duct and distribution basin where from it is directed to the screen house and pump station for the NPP units.

Sizing of the circulation cooling water system for each unit was based on the following data:

- number of pumps: 4;
- circulating cooling water flow: 46 m^3/s;
- cooling water nominal temperature at the intake: 15 °C;
- water heating in the condenser: $\Delta t_{avg} = 7.05$ °C;
- maximum water heating in the condenser: $\Delta t_{max} = 10.3$ °C.

Sizing of the Raw Service Water System for each unit was based on following data:

- number of pumps: 4;
- service water flow: 7.8 m^3/sec;
- number of in-service pumps: 3 (normal operation);
- flow rate of one pump: 2.61 m^3/sec;
- minimum temperature for sampled raw service water: 2 °C.

The hydraulics part for the discharge of the warm water resulting from the condenser cooling and from the service water circuit chillers, consists of hydro-technical buildings which make possible the warm water discharge either to DBSC - Race 2 or to the Danube.

The cooling water discharge from Cernavoda NPP to DBSC - Race 2 is performed via a circuit including the following:

- circulating water warm legs discharge pipes and ducts;
- service water warm legs discharge pipes and ducts;
- siphoning pools and special chambers;
- warm water discharge channel to DBSC -Race 2;
- warm-cold water mixture discharge channel for the "recirculation" mode.

The warm water discharge channels were sized for a flow rate of 54 m^3/sec on each unit. These channels give the possibility to discharge the warm water via the siphoning pool and the valve station, either in DBSC - Race 2, or into the Danube.

2.2.2.2. NPP Equipment and Systems

2.2.2.2.1. Cernavoda NPP-U3 and U4 Reactor Characteristics

The reactor consists of a cylindrical horizontal vessel (calandria, see Figure 2.2.2.2-1) provided with 380 horizontal fuel channels arranged in a square lattice and reactivity control units. The reactor vessel is filled in with heavy water as moderator and reflector of neutrons resulted from the nuclear fission reaction.

Except for the pressure tubes in the fuel channel assembly, all the reactor assembly components, including the reactivity mechanisms, operate at low pressure and temperature conditions. The fuel channels consist of pressure tubes concentrically arranged with calandria tubes that are rolled-join into the reactor vessel inside tube plates. Between the pressure tubes and calandria tubes an adequate separation by the use of some spacers rings (Figure 2.2.2.2-2) is maintained.

The inner space between one pressure tube and the corresponding calandria tube is called annulus gas space and it is filled in with dry carbon dioxide aimed to provide thermal isolation and it allows the pressure tube cracks detection. The nuclear fuel is inserted into pressure tubes by the fuelling machine.

Calandria is designed to withstand the pressure resulted from a simultaneous brake of the pressure tube and calandria tube. To limit such a pressure effect, four pressure relief ducts located in the upper part of calandria and provided with rupture disks, were provided.

Calandria vessel is provided with end shields (biological protections) which lower the level of the radiation to allow access of personnel to the pressure tube area (fuelling machine maintenance rooms) after the reactor shutdown. The end shielding is part of the reactor vessel and they also are supporting the calandria tubes passing through them.

Calandria vessel is located in a steel plated concrete enclosure filled with light water (calandria vault). The light water provides an additional shielding and a proper cooling of calandria vessel.

The reactor containment (Figure 2.2.2.2-3) is the most important plant protective system in case of unexpected failure of equipment (reactor, steam generator, etc) mitigating the release of radioactive materials and contamination of the environment.

Calandria assembly is seismically qualified to DBE.

Reactor core reactivity is controlled through liquid and solid neutron absorbers. During normal operation the reactivity control is performed by the Reactor Regulating System that includes:

- Mechanical Control Absorbers Units;
- Adjuster Assembly Units;
- Liquid Zone Control System;
- Moderator Liquid Poison Injection System which allows the insertion of boron and gadolinium as neutron absorbers;
- Moderator D_2O Purification System which allows the extraction of the absorbers from the moderator;
- Neutron flux monitoring devices (flux detectors and ion chambers).

Platinum and vanadium flux detectors are located in the reactor core and provide the neutron flux measurement. The detectors are supplemented by the ion chambers installed on the outer sheath of calandria vessel.

The neutron flux measurements performed by the platinum and vanadium flux detectors are used to adjust the local and bulk power distribution. The local values are adjusted through the light water level change from liquid zone control compartments. The variation of the light water level into these liquid zone control assemblies modifies the local neutron absorption in 14 areas of the reactor core, providing thus the control of the neutron local flux level.

In case that the liquid zone control compartments cannot provide the proper control of the neutron flux level and of the reactivity change rate, the reactor is provided with 4 absorber rods, vertically operated in the reactor core and aiming to control the neutron flux level and reactivity change rate. Normally the absorber rods are kept out of the reactor core.

The bulk long-term or slow variation of the reactor core reactivity is controlled through addition of some chemical neutron absorbing substances (gadolinium or boron solutions called poisons) in the moderator. The reactivity control is obtained by the variation of these "poisons" concentration in moderator.

For example, to compensate for the excess of core reactivity, when the reactor is initially start up and the reactor core is fresh fuel loaded, the concentration of the poison in the moderator is varied.

In order to assure an optimum shape and flattening of the neutron flux, 21 adjusting rods are provided. These rods are normally inserted in the core.

CANDU - 6 type NPP reactor project is provided with two shutdown systems, SDS # 1 and SDS # 2. Any of these systems is capable to shut down the reactor power inserting an amount of negative reactivity in the core, in the form of neutron absorbent materials, in order to stop the fission reaction chain. The provision of two independent, different and highly reliable shut down systems, make extremely improbable the occurrence of a Loss Of Shut Down accident. Both shutdown systems will trip when a postulated accident occurs, as a result of neutron and process signals or upon the operator's request maintaining the reactor sub-critical for an indefinite period of time. The two shutdown systems are physically and operationally independent to each other and from the reactor regulating system.

Figure 2.2.2.2-1. Calandria

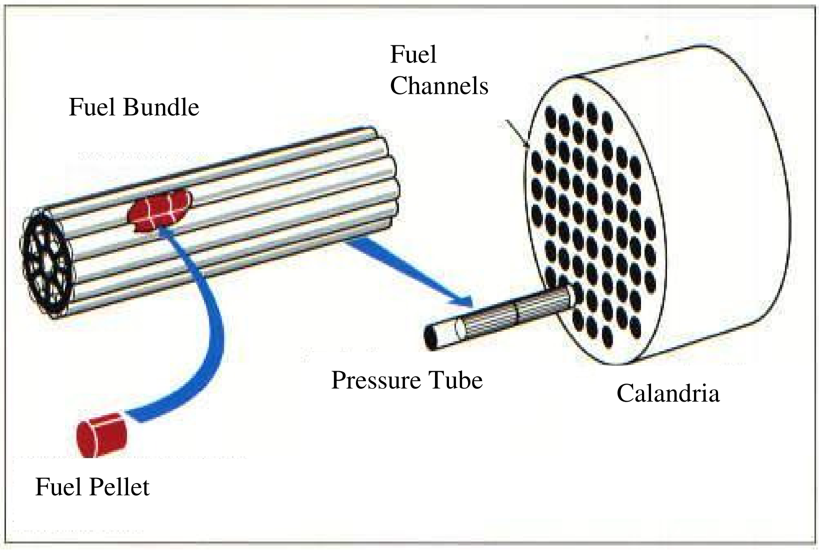


Figure 2.2.2.2-2. Pressure Tube

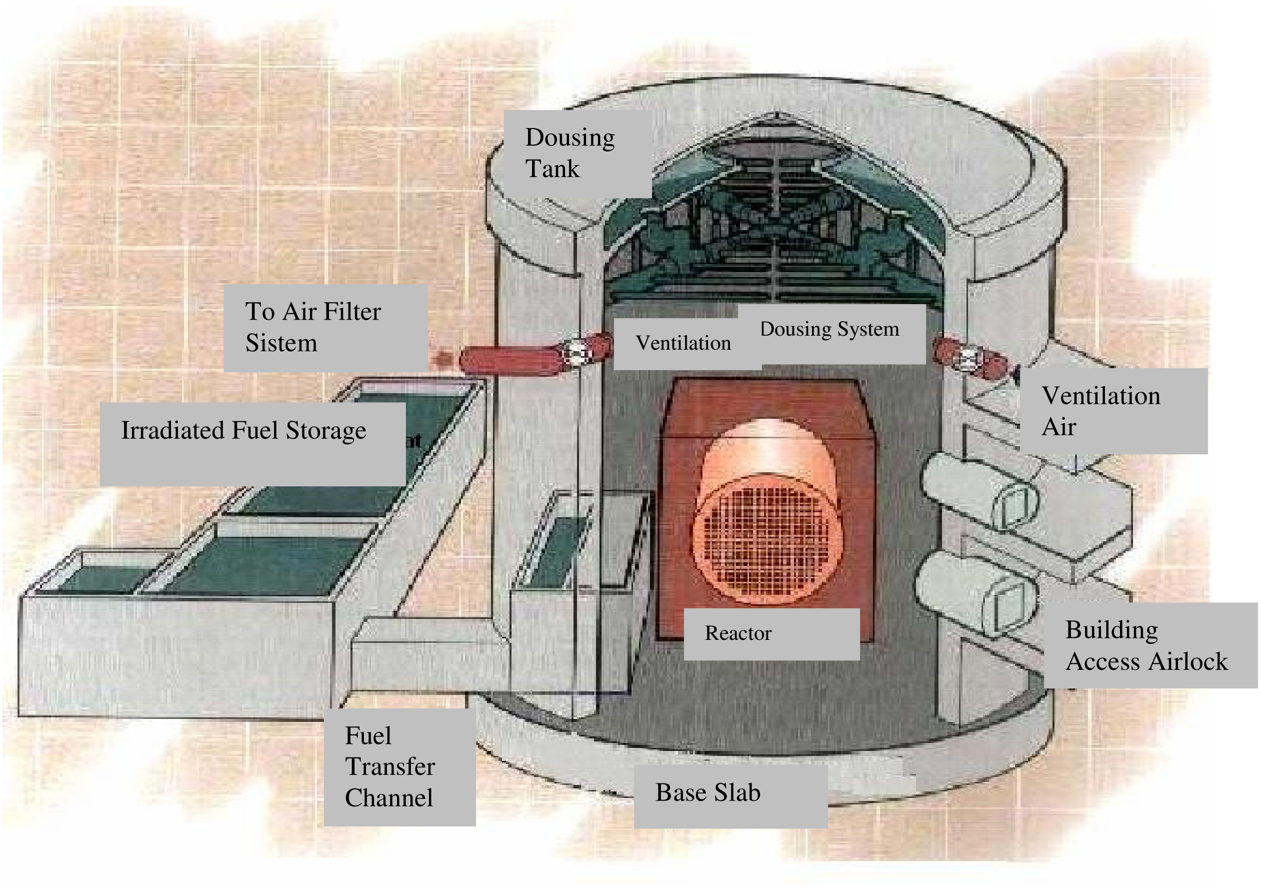


Figure 2.2.2.2-3. Containment System

2.2.2.2.2. Reactor Process Systems

For each the nuclear units, U 3 and U 4, the main reactor process systems are Primary Heat Transport System (PHTS) and Main Moderator System (MMS).

2.2.2.2.2.1. Primary Heat Transport System (PHT)

PHT system is designed to provide the pressurized heavy water circulation through the reactor fuel channels to take over the heat generated by the nuclear fuel during the nuclear reaction. Heat transported by the coolant is transferred to light water in the steam generators. By vaporization, saturated steam required for the turbine-generator set operation is produced.

The pressure boundary of PHTS is the third protective barrier against the potential radioactive releases to the environment, after the fuel matrix and the fuel element sheath. PHTS is designed in order to maintain its integrity both under normal operation and anticipated transients occurrences.

PHTS is seismically qualified to DBE in order to provide the pressure boundary integrity. During and after an earthquake, the system pump rotors shall rotate freely in order to maintain the coolant circulation. PHTS can be isolated after an earthquake in order to keep the D_2O inventory required for thermal-siphoning process. A D_2O make-up circuit is also provided in order to compensate for the small D_2O leakage which might occur after a seismic event.

The main design objective with respect to PHTS nuclear safety is the adequate cooling of the fuel for any operation regime throughout the plant lifetime with a minimum maintenance. As such, heat is transferred to the condenser or to the atmosphere via the steam generators, or is carried out by the service water system via the shut down cooling system. When the PHTS pressure boundary is intact, the system is capable to remove the decay heat in order to prevent the fuel damage.

Should the PHTS pressure boundary integrity is lost, the system is designed such that, along with the protective systems initiation (e.g., emergency core cooling system), it may limit the fuel failure. PHTS is a safety related system.

PHTS is mainly consisting of: 4 circulating pumps, 4 inlet headers, 4 outlet headers, 380 fuel channels, headers to channels connecting feeders and 4 steam generators (U tubes primary circuit) (see Figure 2.2.2.2-4).

PHTS is divided in two separate loops. If a loss of coolant accident occurs, the intact loop is automatically isolated against the affected loop and against any auxiliary system, reducing thus both the loss of coolant inventory and the nuclear fuel failure rate.

The primary coolant inventory and pressure control system is aimed to control the PHT pressure to the corresponding value required by operating condition and to add/extract the coolant when a loss/excess of PHTS coolant inventory occurs. With the reactor at power, the pressure is controlled through pressurizer and the feed and bleed circuits are adjusting the PHT inventory. During shut down state, the pressurizer can be isolated from PHT and pressure will be controlled solely by the feed and bleed system.

PHT D_2O purification system is controlling the coolant chemistry in primary circuit and is preventing the buildup of radiation fields around the equipment, by minimizing the presence of activated corrosion products and of fission products in reactor coolant.

Minimizing the PHTS coolant leakage and collecting the D_2O in liquid (**by D_2O collection system**) or vapor (**by D_2O vapor recovery system**) states are of major importance.

The shut down cooling system, SDCS, assures the nuclear fuel cooling during plant normal shutdown process or for certain accident sequences.

2.2.2.2.2.2. Main Moderator System

The fast neutrons produced by nuclear fission are thermalized in D_2O existing in calandria vessel. The moderator heavy water is circulated by the main moderator system (MMS) pumps being cooled down by its heat exchangers. The system operates at reduced temperature and pressure values. The heat exchangers remove the heat produced by neutrons slowing down in the moderator, as well as the heat transferred by radiation to the moderator from the fuel channels. The coverage gas for D_2O is He, being controlled in a closed circuit. The moderator purifying circuit maintains the control of moderator water chemistry within the specified limits (Figure 2.2.2.2-5).

The MMS is capable to remove the residual heat from the fuel immediately after reactor shutdown generated by certain postulated initiating events, whose characteristics are due to the total loss of electric power supply, LOCLIV, or to the loss of coolant, LOCA, including also a sufficient additional cooling for moderator ("crash-cooling"), as well as an adequate suction corresponding to normal pumps operation are required.

Calandria vessel containing both the moderator and its re-circulating system are seismically qualified to DBE. In addition, the part of this system that penetrates the containment wall is also seismically qualified to DBE.

The MMS is environmentally qualified so that it can meet the nuclear safety functions (residual heat removal), under harsh environmental severe conditions, due to a LOCA. The environmental qualification of the system is not required for the condition which occur following a main steam line break (MSLB) as there is no requirement of residual heat removal by means of the moderator, in case of such an accident.

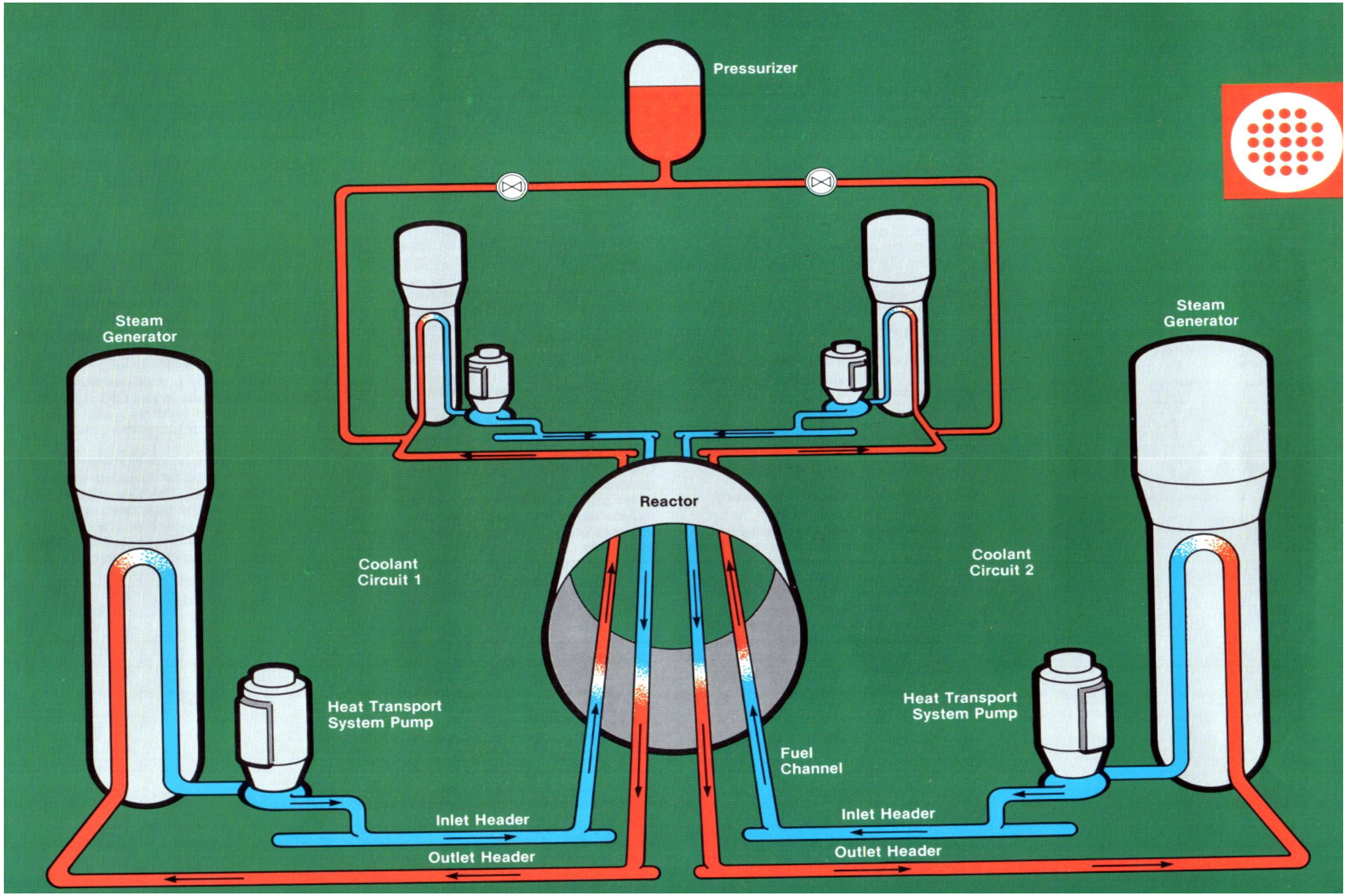


Figure 2.2.2.2-4. Primary Heat Transport System

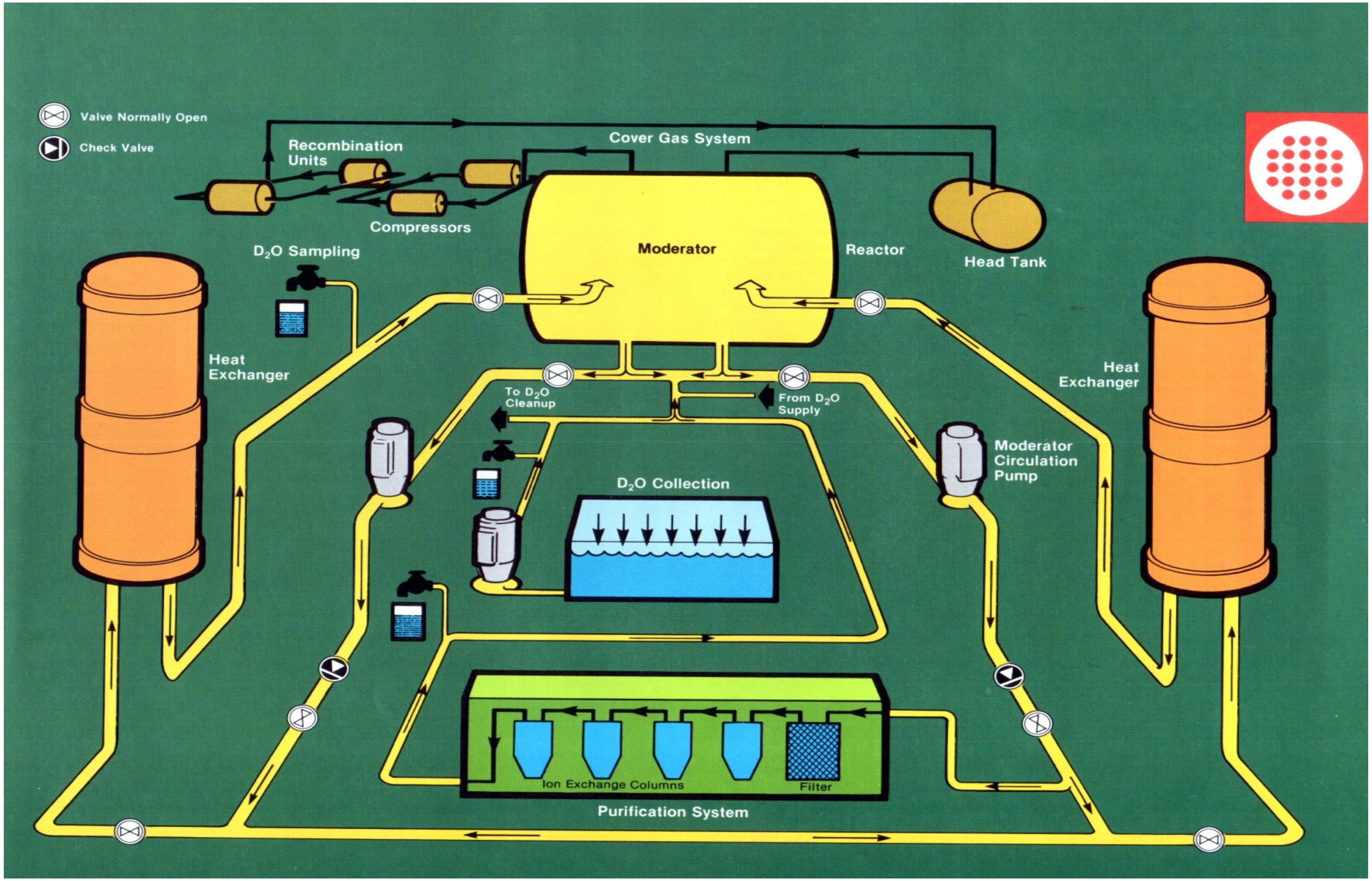


Figure 2.2.2.2-5. Main Moderator System and Auxiliaries

2.2.2.2.2.3. Auxiliaries Systems

There are auxiliaries systems associated to the heat transport system and to the moderator system, meeting both process and nuclear safety functions. The most important auxiliaries systems are the followings:

- Shutdown Cooling System;
- D_2O Coolant and Moderator Purification Systems;
- Pressure and Inventory Control System;
- D_2O Coolant and Moderator Deuteration and De-deuteration systems;
- D_2O Coolant and Moderator Collection Systems;
- D_2O Coolant and Moderator Sampling Systems;
- D_2O Coolant Storage, Transfer and Recovery System;
- End Shield Cooling System;
- Annulus Gas System;
- Resins Transfer System;
- D_2O Management System.

2.2.2.2.3. Fuel and Fuel Handling

2.2.2.2.3.1. Fuel

The fuel used in the nuclear reactors of U3 and U4 Cernavoda NPP is natural uranium processed as ceramics pellets of UO_2 which, assembled in Zircalloy-4 claddings, constitutes fuel elements. 37 fuel elements assembled together are making up a fuel bundle (Figure 2.2.2.2-6). Each of the 380 channels contains 12 fuel bundles resulting thus a total of 4560 bundles in the reactor core.

Nuclear fuel matrix, together with the corresponding cladding, constitutes the first two protective barriers against radioactive materials releases to the environment.

The fuel is designed to withstand the transients and the anticipated events during operation. It is considered that the cladding remains undamaged, whether the following criteria are met:

- there are no melting points in the fuel;
- there is no excessive sheath deformation (less than 5 % of uniform deformation for temperatures of the cladding lower than 1000 °C);
- there are no significant cracks in the oxide layer existing on the sheath surface;
- the sheath embrittlement due to the oxygen is not occurring.

2.2.2.2.3.2. Fuel Handling

For fresh fuel handling and storage, for reactor loading and unloading, as well as for spent fuel storage special equipment will be used.

The reactor is reloaded during operation with fresh fuel by means of two fuelling machines, F/M (Figure 2.2.2.2-7), one at each end of the reactor. The fuel machines operate at the opposite ends of the same fuel channel, one of them introducing fresh fuel and the other one extracting the spent fuel from that channel. During normal plant operation, the fuelling system remove the decay heat from the fuel located in F/M head over the all time period in which F/M is attached to the reactor, including when the fuel is transferred through the fuel transfer port to the spent fuel bay.

The spent fuel is unloaded from F/M through the fuel discharge ports, in its reception bay, from where it is transferred under water to the spent fuel bay, SFB, that is located in the Service Building.

The spent fuel bay is provided with a sufficient storage capacity in order to cool down the accumulated fuel during at least 6 years, and a backup capacity sufficient until the transfer to the other storage facilities. The spent fuel bay is provided with hoist devices and submerged transport of the spent fuel and a cooling and cleaning water system in order to remove the spent fuel decay heat from the water bay and to maintain the water chemistry and radioactivity to allowable levels.

Figure 2.2.2.2-6. Fuel Bundle

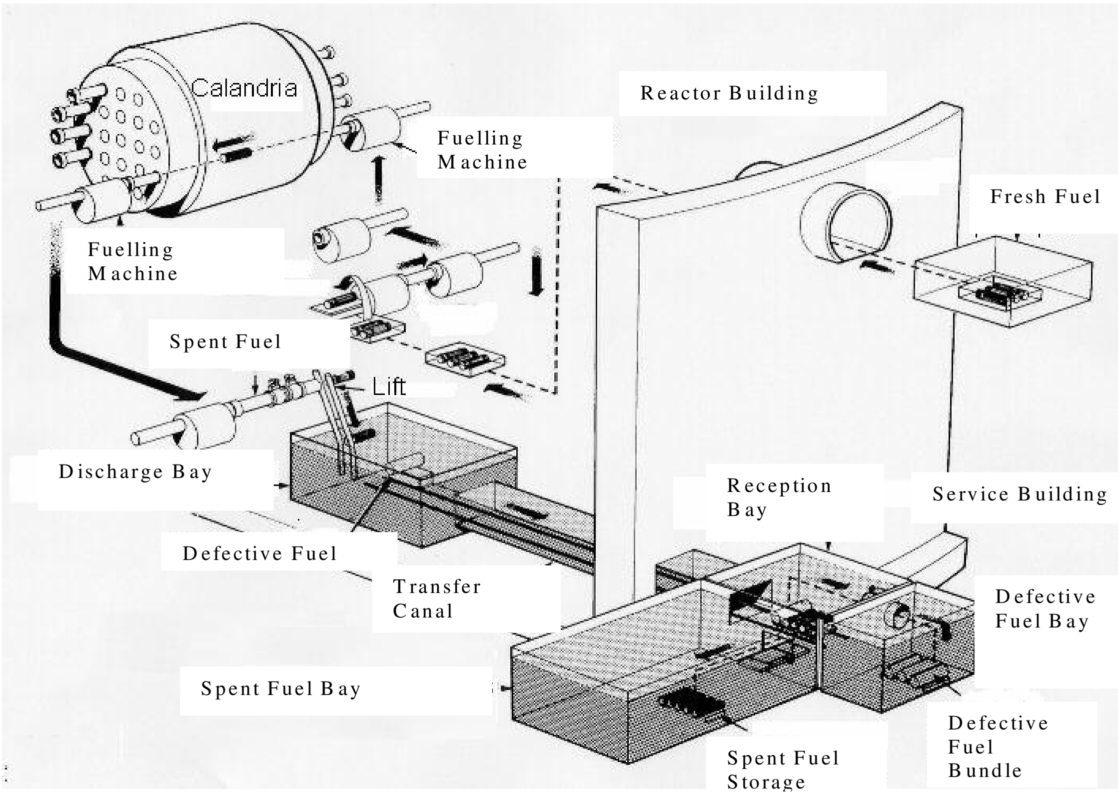


Figure 2.2.2.2-7. Fuel Handling System

2.2.2.2.4. Electric Power Production Systems

The electric power production system is made up of the turbine-generator set. The turbine-generator set contains two basic components, namely the turbine and the generator.

2.2.2.2.4.1. Turbine

The turbine type provided for Cernavoda NPP - U3 and U4 project is of action/reaction with condensation type, using saturated steam, being warranted to produce an active power at the shaft corresponding to a rated gross nominal output of 720 MWe, at a speed of 1500 rot/min, and the condenser cooling water temperature of 15 °C. Constructively, the turbine comprises one double flow high pressure cylinder and three double flow low pressure cylinder.

The turbine is provided with five fixed steam extraction connections from different stages of expansion, in order to heat up the steam generators feed water by means of regenerative re-heaters chain.

The turbine condenser is made up of three independent shells, each of them being connected to a turbine low pressure cylinder. The condensate from the turbine condenser is pumped by 3 main condensate pumps (3 × 60 %) through the regenerative circuit, which is constituted of three low-pressure heating stages, transferring the condensate to the de-aerator. By means of three main feed water pumps (3 × 60 %), the condensate is overtaken from the de-aerator and heated in two high pressure re-heaters, in parallel configuration, and then being transported through 4 pipes to a regulating valve station of the water rate for the supply of steam generators.

Both the main condensate system and steam generators feed water system, are provided with auxiliary pumps, namely: auxiliary feed water pump and auxiliary condensate pump.

2.2.2.2.4.2. Generator

The turbine mechanical power is converted into electric power by means of the electrical generator directly coupled to the turbine.

The generator is of synchronous type, with stator in star connection. Its apparent power is of 800 MVA, at 1500 rot/min and frequency of 50 Hz at a voltage of 24 KV and a power factor cos ϕ = 0.9.

The generator is supplied with a static system of excitation, EX2000, and with auxiliaries cooling systems – with water, for stator windings and with hydrogen for rotor. The shaft jacking system realized with oil.

2.2.2.2.5. Electrical Power Systems

The electrical systems supply power consumers different to process and safety equipment, to control and instrumentation systems, ventilation and lighting, in both Unit 3 and Unit 4 Nuclear Steam Plant (NSP) and Balance Of Plant (BOP).

The power sources of electrical systems are the following:

- offsite sources, which assure the required electric power during the startup, the shutdown and the normal operation of the unit; the offsite electric system is represented by national grid;
- onsite back-up sources, which are able to supply power to safety related systems and to safety support systems to achieve the basic safety functions, in the case of unavailability of the offsite sources; they are represented by Stand-by Diesel Generators (SDG) and accumulator batteries. In addition, an Emergency Power Supply system (EPS) is provided, with Emergency Diesel Generators, to cope with a Design Basis Earthquake or another event which render the other power supplies or the control room unavailable.

The physical and functional interface between Unit 3 and Unit 4 and the national grid is materialized at the interconnection switchyard level, the 400 kV switchyard and the 110 kV switchyard.

The Unit 3/Unit 4 internal services power supply from the grid can be achieved either from 400 kV switchyard or 110 kV switchyard, through two Unit Service Transformers or two System Service Transformers respectively.

Under normal operation, the unit services are supplied from Unit Service Transformers, with back-up from System Service Transformers.

The provision for a medium voltage automatic transfer of power sources ensures the continuous supply of unit internal services.

The electrical distribution system supplying power to unit internal services of the either nuclear unit is divided into four power classes, depending on their degree of availability:

- *Class IV power system* is an AC distribution system normally supplied from unit turbogenerator and/or the off-site grid. Class IV system supply power to all loads which can tolerate long term interruptions of power supply without endangering the equipment and personnel.
- *Class III power system* is an AC system consisting of two redundant and separated sub-systems, normally supplied with power from Unit Service Transformers. A 100 % capacity stand-by power supply is provided for each redundant division (Stand-by Diesel Generators). The loads fed from this system can tolerate short interruptions of power supply, required for starting-up and sequentially loading the Diesel generators.
- *Class II power system* is an AC distribution system fed from the battery/inverter or Class III power supply bus bars. The loads fed from Class II system can not tolerate any interruption of power supply.
- *Class I power system* is a DC system, normally supplied from the Class III power system via rectifiers. The back-up supply from batteries, continuously charged in floating mode by rectifiers, assures a nominally uninterrupted power supply to the assigned loads.

The voltage levels corresponding to power supply classes are the following:

- Class I: 220 VDC; 48 VDC; 400 VDC;
- Class II: 380/220 VAC; 220/127 VAC; 220 VAC; 170 VAC three-phase;
- Class III: 6000 V; 380/220 V;
- Class IV: 10000V; 6000V; 380/220V.

The 50 Hz frequency is the only frequency for Class II, III and IV.

The power supply of the essential equipment in safety related systems is designed with the required redundancy to assure the safety functions are met in any conditions.

The electrical power is supplied to loads on the basis of ODD/EVEN concept, that is two functionally independent and physically separated distribution systems have been provided. The redundant loads (e.g. 2 × 100 % or 3 × 100 %) are assigned as balanced as possible to ODD and EVEN sub-systems. In addition, for control and instrumentation systems, triplicated functionally independent and physically separated power supply channels have been provided.

The Emergency Power Supply System (EPS), is provided as a back-up power supply required for reactor safe shutdown and heat removal when all other power supplies and/or MCR are rendered unavailable. The EPS system consists of two redundant, seismically qualified, functionally independent and physically separated sets of equipment. Two Diesel generator sets are provided as power sources, each being able to supply all emergency loads. The system is initiated by operator from controls located in Secondary Control Area (SCA).

2.2.2.2.6. Instrumentation and Control System

Instrumentation and Control system, I&C, for Unit 3 and 4 is provided to perform monitoring, control and display functions for both the parameters and plant systems, on their full range, in normal operation and anticipated operational occurrences, including in accident conditions, maintaining the values of these parameters under allowable limits.

Central part of I&C consist in two subsystems, namely: Distributed Control System (DCS) and Plant Display System (PDS), both compatible with the X-Y arhitecture (structure) of the existing I&C System for U2. The Distributed Control System (DCS) is controlled and monitored by a HIACS 7000 type hardware panel while the Plant Monitoring System will employ an Advanced CANDU Control and Information System (ACCIS).

The continuous operation of the I&C main equipment is assured by redundant supply sources that provide protection against the loss of power supply sources from station service load.

Should the main control room becomes unavailable, the unit control is performed from the secondary control area so that the main plant safety function will be accomplished.

The I&C functions (measurement, control, regulation, annunciation) will be performed, for the NSP systems, using the process sensing and transducers signal amplifiers, signals processing equipment and logic and acting relays.

A distributed control system, DCS, was provided for the plant BOP systems, that includes all analogical and numerical control functions used to control the process systems, including both the safety related systems and support systems. DCS is an integrated system performing data acquisition and control functions, based on the programmable digital controllers, PDC, linked through data lines. The process system instrumentation and control devices will be connected to DCS input - output local stations.

2.2.2.2.7. Safety Systems

2.2.2.2.7.1. Special Safety Systems

The special safety systems are those systems provided in order to assure the plant key safety functions accomplishment when a postulated initiating event occurs. As such, the group of special safety systems should shut the reactor down (SDS#1, SDS#2), remove the decay heat (ECCS) and limit the radioactive releases (containment system) which might occur, should a safety related process system failure occur. The fulfillment of these functions can be monitored and controlled both from MCR and SCA.

As shown previously, the special safety systems comprise shut down system #1, SDS#1, shut down system #2, SDS#2, emergency core cooling system, ECCS, and containment system.

2.2.2.2.7.1.1. Shut Down System # 1

SDS#1 uses a number of 28 shutoff units, introducing in the core the associated solid absorber rods. Each shutoff unit consists of a drive mechanism, suspension cable, a helical spring insertion device, the shutoff rod and the guide tube. Each shutoff rod is of pipe type containing a neutron absorber material. The shutoff rods are suspended through the cables attached, at the bottom to the inner central tube and at the top, to the cable sheave. When the rods are extracted from the core, the cable is winded on the sheave of the driving mechanism, being situated on the platform of the reactivity mechanism deck above the reactor. In the position WITHDRAWN, the rods will be located on the top of the compressed helical spring thimble, above the guide tube. The rods are kept on position by means of the electromagnetic friction clutch that is coupled to the sheave shaft, such that the sheave rotation and the unwinding of the cable are avoided.

A trip signal will de-energize the electromagnetic clutch such that the shaft sheave is released and the cable is free to unwind under the action of the expanding helical spring. Consequently, the shutoff rods are gravitationally inserted into the core being assisted by the expanding springs. In the position INSERTED, the rods are symmetrical in the vertical plan, versus the core axis. As the shutoff rods are inserted into the core, the thermal fission neutrons are absorbed and the neutron flux is drastically reduced, and consequently, the fission power decreases with the same rate. The negative reactivity inserted into the core depends on the neutron absorption rate that is controlled by the insertion speed and the amount of solid absorber entering the core. Otherwise, for a given geometry of the shutoff rod, the negative reactivity and the neutron flux decrease are proportional to the number of shutoff rods and their insertion speed.

SDS # 1 trip is initiated whenever the specified parameter values exceed the associated setpoints. The actuating relays are de-energized when the trip signal occurs on 2-out-of-3 channels, through the 2-out-of-3 matrix logic of each channel, and the contacts of the actuating relays are opened, interrupting thus the power supply of the shutoff rods clutches. Consequently, the shutoff rods are inserted into the core and the reactor shuts down.

The system initiation prevents nuclear fuel failure during the anticipated operational transients, maintaining thus the reactor under sub-critical status for an indefinite period of time. The system is designed to provide the safe reactor shut down for the following postulated initiating events:

- Loss of regulation, LOR;
- Loss of coolant accident, LOCA;
- Loss of coolant circulation (loss of flow), LOCLIV;
- Loss of heat sinks secondary sides, LOHS;
- Loss of moderator cooling.

2.2.2.2.7.1.2. Shut Down System # 2

SDS#2 represents the second independent way to stop the chain fission reaction. This system injects a strong neutron absorber solution into the moderator volume, when a trip signal occurs on at least 2-out-of-3 protective channels, irrespective of their combination. The trip setpoints of these two shut down systems are different being thus selected as to prevent the SDS#2 firing before SDS#1, unless this system is unavailable (fails to trip).

SDS#2 consists of a helium supply tank, 6 quick acting valves, QAV, in a matrix configuration of 2 valves in series on each of the 3 parallel pipes, 3 venting valves of the adjacent gap between two QAV on each pipe, a helium injection header, 6 injection tanks containing poison solution and the corresponding isolation valves, 6 injection units installed in calandria vessel and the corresponding pipes. There are also additional equipment for the helium tank pressurizing, overpressure protection, pressure equilibration with the moderator cover gas system, sampling, emptying, filling in, control of injection tank poison solution concentration, equipment ventilation and drainage, as well as for the surveillance of equipment status and process variables. Injection tanks contain a concentrated gadolinium nitrate solution, a strong neutron absorber, known as ‘poison solution’. The helium injection header communicates with the cover gas from calandria through an equilibrium pipe that may be isolated by a high pressure signal actuating valve. Consequently, the moderator level in the calandria matches the level of the poison solution from the injection tanks.

A trip signal will open QAV allowing the high pressure He from the supply tank to expand quasi-adiabatically such that the injection tanks will be pressurized. Consequently, the solution will be rapidly injected in the moderator volume through the injection units, until the high-density polyethylene ball in the tank will be sit tight on its bottom flange, preventing He ingress to calandria vessel.

As the solution is injected into the moderator, the gadolinium will absorb the fission neutrons determining the neutron flux and hence the fission power decrease. The negative reactivity depends on the neutron absorption rate being controlled, in turn, both by the speed and amount of gadolinium entering the core. Consequently, the insertion rate of the negative reactivity (the shut down rate) relies on the number of available injection tanks, on the gadolinium quantity in each tank (level, concentration) and the injection rate of poison solution into moderator.

SDS#2 trip is initiated whenever the values of the specified parameters exceed the selected setpoints. A channel trip will de-energize its solenoid valve associated to one pair of QAV and its venting valve as components of the active protective channel. The 2-out-of-3 trip logic is set through the matrix configuration of these valves; their action will result in high pressure helium expansion which will inject the gadolinium nitrate solution into moderator and the reactor will shut down.

The postulated initiating events requiring the rapid shut down of the reactor include the accidents of LOCA type (L-LOCA, S-LOCA), loss of Class IV power supply, loss of reactivity control, main steam line break, boiler feedwater line break and main moderator system failure. The others DBA allow the operator a long enough period of time in order to reduce gradually the reactor power or to initiate its orderly shut down manoeuvres.

2.2.2.2.7.1.3. Emergency Core Cooling System

The system is designed and realized to provide the rapid injection of cooling water in primary heat transport system, when an event that generates loss of its inventory occurs (LOCA type events, or events producing coolant excessive shrinkage), in order to establish and maintain the long term core cooling.

ECCS consists of three sub-systems, HP (high pressure stage), MP (medium pressure stage) and LP - ECCS (low pressure stage), which are injecting the water contained in the specified cooling sources, through the injection headers into the primary circuit headers. The operating performances of ECCS are improved by automatic initiation of steam generators crash cooling.

The ECCS sequential operation comprises H_2O high-pressure injection contained in two accumulators, towards the common header, followed by the medium pressure injection that pumps the volume of water from dousing tank preserved to ECCS. After theses cooling water sources are exhausted, the quantity of warm water accumulated in R/B basement and sumps is pumped through the ECCS heat exchangers and returned into the primary circuit by injection headers. As a defense-in-depth measure, an alternative long term cooling source is provided by the initiation of the emergency water supply system, EWS. EWS injects water into PHTS through the injection headers, should the LP-ECC becomes unavailable on long term.

ECCS automatically initiates whenever the values of the specified parameters exceed the selected setpoints. ECCS action includes:

a - opening of the injection paths between the common header and PHT system;

b - pressurization of two accumulators by the expansion of the gas (N_2) contained in the gas tank;

c - ECCS pumps start up;

d - opening of the flow pathways between the dousing tank and the recovery pumps;

e - opening of the safety relief valves in the main steam circuit (MSSV) for steam generators crash cooling;

f - closing of the PHT loops isolation valves.

Additionally, the support systems that are active to maintain the ECCS in armed state are automatically isolated, if they are operational when ECCS trips.

HP-ECC is effective by using a pressurized gas tank and two water accumulators. The gas tank is maintained at high pressure and, during normal operating conditions, it is isolated from the accumulators. Water accumulators and the pipes up to the injection valves are kept pressurized at low pressure. The gas tank isolation valves opening time is delayed in regard to the D_2O circuit isolation valve and the H_2O circuit injection valve opening times, in order to reduce the effects induced by water hammer in the injection paths.

After ECCS tripping, the water pressure and level in the accumulators are getting lower in time. The water inventory from these two water tanks can provide injection for minutes (about 2.5 minutes in case of L-LOCA) or for hours (S - LOCA). The accumulators isolation valves are automatically closed upon a water level low signal in any one of the two tanks in order to avoid the gas ingress into PHTS.

MP-ECCS initiation starts immediately after the water inventory in HP-ECCS accumulators is over and their isolating valves are closed. MP-ECCS includes the dedicated injection water inventory from the dousing tank, the descending connection pipe between the common injection line (MP + LP) and the dousing tank, and the common injection circuit which includes two recovery pumps arranged in a parallel configuration, one or two heat exchangers (arranged also in parallel configuration) in series with the pumps, valves and corresponding pipes. One of the two recovery pumps is automatically started during HP-ECCS injection, locally recirculating a low water flow via the heat exchanger by-pass valve until the pressure in the common distribution header of the injection water gets down to a value lower than the pump discharge pressure.

The water from the dousing tank is maintained below the temperature maximum value that proved to be the optimum for an efficient fuel cooling and a maximum reduction of vaporization in the containment via local air coolers, LAC. The dousing tank is filled in with demineralized water that is shared between the containment dousing system and MP-ECCS, for the supply of the recovery pumps. ECCS medium pressure stage operation develops until the water inventory for ECCS is over (about

500 m^3 during at least 12,5 minutes) and the level in the tank reaches a minimum value. Water supply from the dousing tank is isolated from the common injection line via two in-parallel valves, each of them providing the flow rate control at each of the recovery pump suction.

Each discharge pipe of the two pumps is provided with a check valve to prevent back flow through the stand by pump. The discharge pipes are connected to the heat exchangers (isolated during MP-ECCS operation), at the level of a common header, wherefrom one heat exchanger by-pass pipe is directing the cooling water to the LP injection node. The injection node consists of two injection paths arranged in parallel configuration, each line including two isolation valves and a check valve in series. The two injection pathways get united in a common header wherefrom two common flow lines (HP-, MP-, LP-) provide the distribution of the injection flow to the PHTS loops.

LP-ECC action provides the core nuclear fuel long time cooling (3 months) when a postulated initiating event type LOCA occurs. Tripping of the LP-ECCS happens immediately after the water inventory from the dousing tank is over and the minimum specified level is reached. Since the injection of the first two stages of ECC is long enough function of the break characteristic, for at least 15 minutes, LP-ECC may be initiated either manually or automatically at low dousing water level in the dousing tank. LP-ECCS circuit includes the inventory of the water collected in R/B sumps and R/B basement (due to the discharge of a significant PHT-D_2O inventory, of the HP-ECCS injection water inventory and dousing water inventory) and two connection pathways between R/B sumps and the common injection circuit (MP+LP) at the level of the recovery pump suction pipes.

Each connection pathway includes a screen system to retain the impurities, a LP-ECCS isolation valve and the associated piping. One of the two recovery pumps is already operational when the LP-ECC is activated.

Water collected from the R/B sumps and basement is cooled in the heat exchangers below the maximum value of the temperature that was demonstrated to be optimum for core nuclear fuel cooling and a maximum reduction of vaporization in R/B. Cooled water is directed to the LP injection node and from there into the affected PHT loop.

The injected water flow is gradually cooling the nuclear fuel taking over the decay heat and than it is discharged via the PHT break (heat sink) mixing with the volume of water collected in R/B basement and the cycle is resumed.

Water supply from the R/B sumps is isolated from the common injection line via two valves disposed in two ducts, each of them providing the flow rate control at each of the recovery pump suction. Each discharge pipe of the two pumps is provided with a check valve to prevent back flow through the stand by pump. The discharge pipes are connected to the heat exchangers (one in operation and other in stand by), at the level of a common header, wherefrom one heat exchanger by-pass pipe is directing the cooling water to the LP injection node. The injection node consists of two injection paths arranged in parallel configuration, each line including two isolation valves and a check valve in series. The two injection pathways get united in a common header wherefrom two common flow lines (HP-, MP-, LP-) provide the distribution of the injection flow to the PHTS loops.

Each instrumentation loop for process monitoring that is important for ECCS operation, is redundantly designed and constructed, duplicated or triplicated, so that one component failure or power supply unavailability will not determine the unavailability or unexpected initiation of ECCS.

The events that activate ECCS include small LOCA (S-LOCA), transitory LOCA, large LOCA (L-LOCA), failures of PHTS auxiliaries, main steam and boilers feedwater supply systems failures, and SDCS failures.

2.2.2.2.7.1.4. Containment System

The containment system represents one of the four special safety systems of the plant and the 4th physical barrier against radioactive releases to the environment. The system is provided in the plant project to confine the radioactive materials during plant normal and abnormal operating regimes and to maintain the radioactive releases to the environment within the allowable limits, whether an accident accompanied by such release occurs. The system includes the protection containment, the dousing system, the local air cooler system and the isolation system.

The requirements imposed to the containment system are as follows:

i - to control the pressure in R/B in order to:

- maintain the structural integrity in case of an accident generating the highest pressure inside the containment envelope (main steam line break inside containment);
- maintain the atmosphere pressure in R/B at values lower than the design pressure in case of accidents that generate radioactive releases inside the containment (LOCA).

ii - to provide a fast leak-tight barrier around the reactor, steam generators and primary process systems (PHTS and its auxiliaries) in case of an accident that generates radioactive releases.

The above set of requirements is satisfied by:

- limiting both the magnitude and duration of the overpressure peak after an accident;
- designing the containment at a minimal leakage rate;
- isolating the containment after receiving of a high pressure/radioactivity alarm signal.

The Containment System includes the following passive and active sub-systems:

a - *structural and isolating sub-systems* including:

- 1 - R/B and the penetration systems, including the airlocks, spent fuel reception bay and the containment doors;
- 2 - containment isolation system

b - *pressure suppression sub-systems* including:

- 1 - dousing system;
- 2 - R/B local air cooler system.

I - Containment Structure

The containment structure is designed to confine the radioactive material that might be released from the reactor core after a postulated initiating event L-LOCA. Additionally, the structure provides protection against the radiation generated by the fission products that may be present in the Containment atmosphere in accident conditions. The containment includes the R/B structure, its extensions and penetrations. R/B is a pre-stressed concrete structure with epoxy liner that consists of three major parts namely, slab base, cylindrical perimetral wall and dome. Under the dome, there is a circular structure which, along with the adjacent perimetral wall, is making up a pool (tank) in which the required demineralized water inventory for containment dousing is stored. The epoxy liner is aimed to provide the control of leakage and facilitate the post-accident decontamination operations of the internal walls.

The containment design and test pressures have been set at the values of 1.24 bar(m) and 1.43 bar(m), respectively. The design pressure was so selected to provide coverage of any pressure peak associated to the postulated initiating events considered in design. Versus other containment designs, the above value is relatively low but acceptable because of the larger volume of the containment and of the use of a fast pressure suppression system.

From the point of view of essential safety function the containment must fulfill (i.e., the radioactive material retention), the containment leak tightness represents one of the important characteristics of the system. For that purpose, the maximum leakage rate is defined as representing the containment atmosphere air flow rate which is discharged outside environment through leakage when the inside pressure reaches the design value. The design leakage rate is 0.1 %/day of the containment free-air volume; the value employed in the safety analyses is 0.5 %/day.

R/B airlocks are part of the containment pressure boundary aimed to provide a means of access for the operating personnel and to transfer the components that require repairing/maintenance in the shop, during all the operation modes. Each airlock is designed with a cylindrical metallic structure, horizontally arranged and provided with an access door at each end. The doors are interlocked to prevent their simultaneous opening. During the periods the containment system is inoperable, the interlocking mechanism may be override allowing for both doors to stay open for quite a long period, if the access to R/B is frequent. Each airlock door was designed and tested to certify its performance level regarding the capacity to withstand a pressure higher than the pressure associated to a L-LOCA. Under these circumstances, one door of each airlock may provide the containment system operability. Each door is provided with a double-sealing system with inflatable elements and a system to locally test the leakage rate, in order to provide the pressure boundary integrity.

In order to provide the containment leak tightness, the inside of the airlocks must be maintained under the following conditions:

- the airlock inside and outside doors must be closed to maintain the containment pressure; during the reactor normal operation at least one of the doors is always closed;
- the inside and outside pressure of the airlock door must be matched before opening;
- the door seals must be operable.

II - Dousing System

The dousing system is designed to suppress the pressure increase which develops inside the containment as a result of an accident represented by the break of the largest steam pipe (such an event generates the highest overpressure peak in the containment), and such being capable to suppress any peak pressure resulting from a postulated initiating event type L-LOCA.

The system includes six independent dousing units and a demineralized water storage tank (dousing tank) located in R/B dome area. Each dousing unit includes a distribution header that can cover a 60° radius surface of the R/B and a downcomer header provided with 2 valves in series independently driven. Diversity that actually divides the dousing units into two sub-systems, is accomplished by dividing the 6 independent units into 2 x 3 groups. The valves of one group are pneumatically actuated and no electric power is required; the valves of the other group are actuated by electric-pneumatically devices. Each down comer is serviced by an instrument air tank that is provided in order to assure the two series valves actuation, when the normal instrument air supply is lost. The dousing tank contains an adequate cooling water inventory corresponding to dousing system operation (about 2000 m^3) and to MP-ECC injection (about 500 m^3). The MP-ECC water intake is constituted from an extension of descendent header inside the dousing tank up to 1.54 m elevation; should the water level reaches that value, the dousing inventory is depleted.

III – Isolation System

The containment isolation system is designed to isolate the R/B upon receiving the high pressure or activity signal due to a LOCA or MSLB type accident. In this manner the containment system accomplish its allotted function to retain the radioactive materials inside R/B providing a tight barrier against the release of these materials to environment.

The isolation process is effective by closing the isolation valves/dampers on the process pipes and ducts that are passing through the perimetral wall. The R/B airlocks (equipment and personnel) and a number as much as 32 isolation valves (dampers) associated to different process systems that penetrate the containment, are parts of the isolation system. The system's isolation valves (dampers) are arranged in series pairs. Any pair contains two identical valves (dampers) and each valve (damper) is able, alone, to isolate the containment. The I&C loops for any valve/damper are independent each other in order to reduce the common mode failure hazard.

The closing time required for isolation valves (dampers) gives an indication on the containment isolation speed, after high pressure or activity signal receiving, in order to limit the potential radioactivity releases to environment below the maximum allowable dose limits. By the isolation valves (dampers) operation, a physical separation between the containment atmosphere and the surrounding environment is achieved, should an initiating event that results in radioactive material release or pressure increase inside containment occurs. In order to accomplish the isolation function, at least 1-out-of-2 isolating valve (damper) provided for a process penetration should be fully capable to close.

Following containment isolation signal initiation 2 seconds will last till the complete closure of the valves (dampers). The isolation devices are fail safe designed, such that on loss of auxiliary services the valves (dampers) will close. The radioactivity is measured on the ventilation and D_2O vapor recovery outlet ducts. The radioactivity setpoint specified for containment automatic isolation (the level of radioactivity concentration at which the isolation signal is triggered) was selected as to prevent a cumulative release higher than 11 Ci of I^{131} or $2.1 \cdot 10^4$ Ci noble gases for initiating events that do not necessarily imply a containment pressure increase at or beyond 3.45 kPa. The functional effectiveness associated to the minimum allowable performance requirements is assured whether:

- 2-out-of-3 R/B high pressure signal channels are available;
- 2-out-of-3 R/B high activity monitoring signal channels are available;
- at least 1-out-of-2 valve (damper) for each penetration is operable.

IV - Local Air Cooler System

The containment cooling system is provided as a heat sink for the R/B atmosphere, both during normal operating and accident conditions. During normal operating condition the system assures the heat removal from the R/B accessible and inaccessible areas; during postulated initiating event (LOCA) the system is capable to supplement the capacity of heat removal from the containment via the dousing system, including on long time after the dousing water inventory is depleted.

The System includes 36 LAC's for cooling the areas in R/B and 4 fans (2 for each reactor end), to cool the calandria vault and the end shields. The purpose of the system is to maintain the temperature in the controlled areas below the design specified limit values. The quantity of heat from the controlled areas is transferred to the heat sinks, namely the recirculated cooling water system and the chilled water system. A number of 16 LAC's from 36 are designated for F/M rooms (8) and the steam generator rooms (8), their control being both manually and automatically. The automatic control means the start-up of all the 16 LAC's upon the initiation of the containment isolation on high-pressure signal when a LOCA accident may occur. After the containment isolation procedure is over (24 h after the initiation of the event), LAC's are automatically disconnected.

From the remaining 20 LAC’s, 2 LAC’s are provided to cool the moderator system equipment rooms and the remainder LAC’s will cool other rooms in accessible and inaccessible areas that are not provided with reliable cooling sources. Generally, the threshold value of the temperature in the controlled areas must not exceed 40 °C. The 4 fans must provide the cooling of the structural elements at the two ends of calandria vessel so that the reinforced concrete structural elements must not reach the "dry-out" value (abut 65 °C).

The containment system, as a whole, is a special safety system designed to confine an important quantity of radioactive materials which could be generated after a maximum credible design basis accident (DBA), an event characterized both by radioactive material release and the development of some thermodynamic processes which result in an increase of temperature and pressure within the free volume of the containment. This safety function is fulfilled by complying with the derived criteria associated to the containment isolation, dousing inside the containment and cooling of the free volume atmosphere.

Each instrumentation loop required to monitor the process and which it is important for the containment system operation, is redundantly designed and constructed, duplicated or triplicated, so that the failure of one component or loss of power supply will not generate the unavailability or undue tripping of one containment sub-system.

The events that initiate the actuation of the containment system are LOCA type accidents, main steam line break and the nuclear fuel failure in F/M rooms.

2.2.2.2.7.2. Safety Related Support Systems

The safety related support systems represent the implementation of a unique concept of the CANDU 600 type NPP, a concept which derives from the safety design philosophy that imposes, in turn, the separation of the safety related systems into two groups (see Section 2.2.1). Each group must be capable to accomplish four essential nuclear safety functions, namely: the reactor safe shut down, the decay heat removal, the radioactive material confinement and the post-accident plant monitoring, in order to provide the continuous protection of the public and the environment. The two groups are physically separated and completely independent from initiating logic point of view. Moreover, an entire group of systems, respectively

Group 2, is seismically qualified such that its operational state be maintain during a design basis earthquake (DBE).

The safety related support systems are supplementing the special safety systems group in order to fulfill the essential safety functions after or during any postulated initiating or common cause events such as: earthquakes, flooding, fires, aircraft crashes, missiles attack.

This group of systems comprises the following:

- emergency water supply system (EWS);
- emergency power supply system (EPS);
- secondary control area (SCA);
- emergency heating, venting and air conditioning system (EHVAC);
- stand-by Diesel generators (SDG);
- post-LOCA instrument air supply system (PLIAS);
- fire fighting water system;
- boiler make-up water system (BMW);
- turbine over-speed protection system;
- PHTS overpressure protection system (LRV);
- secondary circuit overpressure protection system (MSSV).

2.2.2.2.7.3. Safety Related Process Systems

From nuclear safety point of view, the NPP systems are classified into safety-related process systems and essential process systems.

The safety related structures, systems and components (SSC) represent all the SSC or procedures and the operator’s actions, including their associated support systems and corrective action initiation or detection or removal of an accident sequence that might generate a severe event.

The essential process systems represent those systems including their support systems, which are provided to produce energy and do not fulfill any nuclear safety function. Consequently, by reciprocity, the failure of one safety related system may directly or indirectly affect the plant safety by the gradual failure of the nuclear fuel or can produced significant releases of radioactive products to the environment. Yet, at the same time, the failure of one essential process system cannot affect the unit nuclear safety.

In turn, the safety related process systems are divided in two categories, namely: preventive safety related process systems and protective safety related process systems. The preventive systems are represented by the systems which, by their operation, can provide the proper cooling of the nuclear fuel and the radioactive product releases to the environment is prevented during normal operating conditions. Failure of such systems could result in radioactive releases which may exceed the allowable dose limits for the normal operation regime, unless subsequent mitigation measures are not considered, or their failure may affect the accomplishment of a safety function allotted to another safety related system. In case of an accident occurrence, a preventive process system does not change its status or operational characteristics in order to mitigate or remove the accident consequences. The below specified process systems belong to this category:

- Hydrogen Addition System;
- Main Condensate System;
- Heat Transport Pumps Gland Seal Cooling System;
- Primary Heat Transport System (PHTS);
- Heat Transport D_2O Purification System;
- Reactor Regulating System (RRS):
  - RRS Setback function;
  - RRS Stepback function;
- Secondary Circuit;
- Recirculated Cooling Water System (RCW);
- Moderator Cover Gas System;
- Heavy Water Management System;

- End Shield Cooling System (ESCS);
- Plant Control System;
- Turbine -Generator;
- Compressed Air System;
- Heating, Venting and Air Conditioning System (HVAC);
- Fuel Handling System (FHS);
- Spent Fuel Bay Cooling System (SFBC);
- Radioactive Waste Management;
- Normal Power Supply System;
- Moderator D_2O Collection System;
- Main Control Room (MCR);
- Structures;
- Support Structures;
- Operating Procedure System.

The protective process systems are those systems that must respond by the modification of their state (regime), by increase their capacity or increase their velocity (dynamic characteristics) in order to mitigate the consequences of an event. The below specified process systems belong to this category:

- Boiler Auxiliary Feedwater System (BAFW);
- Shutdown Cooling System (SDCS);
- Heat Transport D_2O Storage, Transfer and Recovery System;
- Heat Transport Pressure and Inventory System (PI & C);
- Main Moderator System (MMS);
- Moderator Liquid Poison System;
- PHTS Isolation Valves;
- Spent Fuel Bay Cleaning and Purification System;

- Gaseous Effluent Monitoring System (GEM);
- Normal Power Supply System;
- Electro-hydraulic Control System (EHC);
- Annulus Gas System;
- Gaseous Fission Products Monitoring System (GFM);
- Delayed Neutron Monitoring System (DNM).

2.2.2.2.8. Process Systems and Common Services

The most important essential process systems are the followings:

- Circulating Cooling Water Supply System (CCW);
- Demineralized Water Supply System;
- Drinking Water System;\
- Gas Control System;
- Heating, Venting and Air Conditioning System (HVAC);
- D_2O Upgrading System;
- Active and Inactive Drainage Systems.

The common service systems represent the systems which serve both the NSP loads and BOP loads. The most important such systems are:

- Raw Service Water System;
- Chilled Water System (CWS);
- Clean Water System ;
- PHT and MMS D_2O Collection System;
- Plant Off Site Power Sources;
- Communication and Plastic Suits Systems;
- Testing and Calibration Instrumentation System;
- Environmental Monitoring Equipment;

- Meteorological Equipment;
- Seismic Detector;
- Maintenance Equipment and Instrumentation;
- Chemistry Control Laboratories;
- Washing Equipment;
- Dosimetric Control Laboratory;
- Fire Protection System;
- D_2O in H_2O Detection System;
- Instrument Air Supply System;
- Breathing Air Supply System.

2.3. Decommissioning Activities

2.3.1. The Decommissioning Concept

2.3.1.1. Definition

The term “decommissioning” used in the nuclear industry, means the technical and administrative actions taken at the end of a useful life of a nuclear objective in order to liberate, partial or total, from the regulatory authority (Ref. 2-36).

The actions involve, through other activities, the decontamination, dismantling and removal of the radioactive materials, components, structures and wastes.

The actions are carried out to achieve a progressive and systematic reduction in the radiological hazard; they are taken on the basis of a preplanning and assessment to ensure nuclear safety during decommissioning operations.

In some situations, in case that the nuclear objective for the decommissioning process rests a part of a functional nuclear facility, this objective will remain under the Reglementation Authority.

The time period for the decommissioning activities for a nuclear power plant depends on the radioactive inventory, the chosen decommissioning option and the used decommissioning techniques, and may range from a few years to decades.

The basic document, which regulation the decommissioning of a nuclear objective, elaborated by the Romanian regulatory body in nuclear field (CNCAN), is NSN-15 “The decommissioning norms of the nuclear objective and facilities” (Ref. 2-18). The requirements of these norms are applicated, in a special way, on the decommissioning of the research nuclear reactors, subcritical assembly, treatment facility of the radioactive wastes, spent fuel intermediate storage facility, radioactive waste intermediate storage, but these can be applicated to a nuclear power plant also.

Both the internal norms and the international standards specific to the decommissioning (Ref. 2-18, 2-19, 2-20), recommend preparing a Decommissioning Plan in three stages:

- initial plan for decommissioning, prepared during the nuclear objective design and construction phase;
- ongoing plan for decommissioning, prepared during the operation phase of the nuclear objective;
- final plan for decommissioning, prepared at the beginning of the nuclear units decommissioning.

The Decommissioning Plan prepared will be submitted for approval by the regulatory body, in order to obtain the license for accomplishing the decommissioning activities, according to NSN-15.

The most important technical, administrative and financial items in a decommissioning plan are considered below:

- facility description and operational history;
- details of the radioactive and toxic materials;
- assessment of decommissioning alternatives;
- details of the decommissioning activities;
- project management team;
- nuclear safety;
- security and safeguards of the nuclear materials;
- safety, performance and environmental assessments;
- quality assurance requirements;
- wastes management;
- financial estimation and available funding;
- records.

The decommissioning strategy of a nuclear objective is defined on three stages. It begins with the safe shut down and ends with realization of a green field site. According with Ref. 2-21, the three basic stages are:

- Stage 1 or “Storage with surveillance”;

- Stage 2 or “Restricted site release”;
- Stage 3 or “Unrestricted site use”.

These stages are not mandatory for the decommissioning of a nuclear objective, but it must take into account that the final goal of any decommissioning activity is the release for unrestricted use of site. This means that the remained radiological risks at the end of decommissioning process must be acceptable for the safety of workers, public and environment.

Each stage described above is defined as a physical state of the plant in conjunction with one certain mode of surveillance (Ref. 2-22). In this way, a complete nuclear objective can go through all three stages during the decommissioning activity, but a small nuclear facility can be directly decommissioned until the stage of unrestricted site use.

2.3.1.2. The Generic Options of Decommissioning

According to international practice in this field (Ref. 2-23, 2-24, 2-25), three decommissioning strategies are used now:

- **The “SAFESTORE“ option**

In first phase, this option assumes a minimum of decommissioning activities and the conversion of the plant into one safety containment, meant to prevent the uncontrolled scattering of radioactive contamination. Here, the spent nuclear fuel is removed from the nuclear unit or stored on the site. Same way, the operating waste and radioactive liquids are collected, processed and removed from the nuclear unit or temporally stored on the site. 20 - 50 years after these activities the decontamination and decommissioning of the facilities and liberation from authority requirements can be produced.

The advantages of this option:

- The existence of a “waiting period”, when radiation fields will be reduced due to the radioactive decay, assuring, in this way, the smaller doses for the decommissioning personal;

- Less radioactive waste;
- Less cost because short time of decommissioning funding spending.

All that contributes to reduce the impact of decommissioning activities on the personal involved, population and the environment.

Disadvantages:

- The existence of some maintenance and operation activities on long term (20-50 years) on- site;
- Unavailability of the trained personal (ex-operators) at the moment of final decommissioning activities;
- The possible raises of costs at the moment of final decommissioning activities.

In the IAEA terminology this option corresponds to “Deferred dismantling” option.

- **The “DECON” option**

This option assumes the decontamination and decommissioning of the equipment and structures that are contaminated and/or activated in short time after the final shutdown of the nuclear plant. The spent fuel and radioactive waste are collected and transferred to repository. These actions ensure site availability for other activities, short time after shutdown of the operating activities.

The advantages of this option:

- The availability of the plant personal, with a good know-how of the plant, for planning and executing a part of decommissioning activities;
- Non-existence of supplementary costs for long-term monitoring of the plant;
- The site availability for other utilization, in the shortest time compared with other generic strategies.

All that contributes to reduce the impact of decommissioning activities on the personal involved, population and the environment.

In the IAEA terminology, this option corresponds to “Immediate dismantling“ option.

- **The “ENTOMB“ option**

This option assumes the contaminated and/or activated parts of the nuclear unit to be sealed in an appropriate material (for ex: concrete). The structures are monitored, until the radioactive content is reduced by decay to the level when the site is liberated from authority requirements.

The advantage of this option is that the environmental impact of the activities to prepare the “safety enclosure” is almost the same as the impact of a non-nuclear objective decommissioning.

The disadvantage is the need to rigorously evaluate the long-term environmental impact of safety enclosure at the moment of this option selection. From this way, the option is, in principal, used for nuclear research reactors or small nuclear objectives decommissioning, but isn’t considered a possible alternative for Cernavoda NPP Unit 3 and Unit 4.

2.3.1.3. Events and Activities with Significant Impact on Nuclear Safety

As nuclear safety means the assembly of technical and administrative measures to ensure the protection of personnel, population and environment, some events and activities with significant impact on nuclear safety have to be considered to define a specific decommissioning option.

These elements, including the preliminary evaluation of the possible impact, are:

- **Operating period** of a nuclear unit has a major financial impact due to the rising of the accumulation period of decommissioning founds.
- **The removal of spent fuel and operating wastes from a nuclear unit** have a major importance from nuclear safety point of view and implicitly for the impact of these activities on personal, population and the environment. Actually, all the decommissioning options take into account the removing, in safe conditions, still in the initial phase of decommissioning, of the fuel bundles and the wastes from the

operational phase, in order to obtain a significant reducing of the risks associated with the plant.

- **The waiting period** is a necessary temporal scale mark for the implementation of “SAFESTORE” option. A long waiting period can have a financial impact, in the way that a rising of the waiting period leads to lower decommissioning rates. Also, a waiting period can have a reduced impact on the personal, population and environment, caused by lower radiation fields determined by the radioactive decay.
- **Terminal state of the site** at the completion of decommissioning activities can vary from the “green field” state to the variant of the partial remove of the radioactivity to subsequent use in other radiological activities; an intermediate alternative is the total remove of the site radioactivity, but keeping the unit buildings for non-radiological uses.
- **Availability of the facilities for the spent fuel and wastes repository** has a significant influence on the decommissioning strategy: without these repositories one can consider a longer waiting period within the “SAFESTORE “ strategy, until these repositories become available or the construction of some facilities for the wastes interim storage are provided. In similar way, the spent fuel and wastes repositories can have an environmental impact.

2.3.1.4. Criteria for the Selection of Decommissioning Strategy

To establish the specific option some criteria have to be considered. In accordance with Ref. 2-23 recommendations, criteria to be considered for Cernavoda NPP decommissioning are:

- **The existence of more units on the same site** requires the operation of common systems, until the final shutting down of all the nuclear units. The correlated execution of decommissioning for all the units on the site, brings supplementary profits resulted from integrated planning and execution of disassembling activities;

- **The uncertainty level** – as far as the decommissioning activities are furthest in time, the uncertainties of costs increases, until the extreme case of insufficiency of financial resources;

- **Using in the decommissioning activities of a operating personal with good know-how of the facility,** prevent the possible problems in case of hiring and training of a new personal or loosing of so called “corporate memory”;

- **Unavailability of the repositories for radioactive wastes** at the moment of final shutdown requires the option for a “SAFESTONE” strategy or resource allocation for the construction of intermediate storage facilities. In the case that needed costs for these facilities are included in decommissioning founds estimation, is benefit to use a waiting strategy, from financial point of view and the possibility to use the technological developments in this field;

- **The necessity of site reusing** can leads to adoption of a strategy of accelerate decommissioning or can determinate the terminal state of the site;

- **Political and social restraints** can result from the existing national politics on decommissioning activities in sense of preference for a special option (accelerate decommissioning –DECON) or the terminal state of the site. Similarly, at the evaluation of this criterion, the social impact of the electrical energy cost and the aspects related to the manpower occupancy rate in the area have to be considered.

It is evident that for establishing the Cernavoda NPP U3 and U4 decommissioning option, the U1 and U2 decommissioning experience will be taken into account.

2.3.2. Equipment, Installations, Tools and Buildings which are Going to Be Decommissioned

2.3.2.1. The Concise Description of NPP Cernavoda U3 and U4 and the Significant Features in the Decommissioning Process

Because the design and development project of U3 and U4 is, in a large measure, similar, after wards, we will refer to one of them, only.

As it was presented in the Section 2.2, a nuclear unit is a complex objective that includes both a variety of buildings, installations and systems used for the achievement of the technological process as well as their auxiliaries.

As part of the decommissioning process of the nuclear unit, a series of buildings, structures, installations and equipment belonging to it will be decommissioned /dismantled/ demolished and removed from the unit.

Thus, the buildings, structures, installations and equipment part of the nuclear unit which are going to be subjected to the decommissioning process are the following:

a) Main buildings of the nuclear part

- The Reactor Building, constructed from prestressed concrete with internal structures made in reinforced concrete, shielding walls made in concrete and metallic structures (such as platforms, staircases, supports, supporting structures). Epoxy liners cover both the interior surface of the perimeter wall of the Reactor Building and its internal structures:
- The Nuclear Auxiliary Services Building (a monolith reinforced concrete substructure and a metallic superstructure being closed with metallic panels lined with thermal insulation);
- Emergency Power Supply System and Secondary Control Area (a reinforced concrete substructure and a metallic superstructure);
- High Pressure Emergency Core Cooling Building (a reinforced concrete vat substructure and a metallic superstructure, being closed with light panels of thermal insulation pleat plates);

- The D_2O Upgrading Tower and the Ventilation Stack (a metallic superstructure).

b) Main buildings and structures of the Balance of Plant

- Integrated Main Building (a monolith reinforced concrete substructure and a metallic superstructure);
- Cooling Water Pump house (monolith reinforced concrete substructure and a metallic superstructure, plated with concrete plates);
- The Workshop;
- The Non-Destructive Examination Building.

c) Equipment and systems of the nuclear part

- The calandria vessel and its auxiliaries, located inside a steel plate lined concrete vault and filled with light water;
- The reactor process systems, the safety systems and the auxiliary systems (stainless steel and carbon steel);
- The electrical and the control systems (ebonite, electrical cables, metal);
- The ventilation systems (metallic tubing, filters).

d) Objects corresponding for Units 3 and 4 investment

In the enclosure:

- High Voltage Station (110 kV) for Common Service Power Supply;
- Transformers for Common Service High Voltage Power;
- Emergency Water Supply System Building;
- Transformers for Common Service Power Supply;
- Service Water Network;
- Drinking Water Distribution Pipe Network;
- Fire Water Distribution Pipe Network.

Out of enclosure:

- Hot Water Discharge Canal;
- Grid Connection (400 kV and 110 kV);
- Telex, phones, weak current connections;
- Canteen (in Cernavoda Town);
- 400 kV Transformer Block;
- 110 kV Cernavoda NPP Electric Power Station.

e) Buildings and structures common for the nuclear units of NPP Cernavoda

Cernavoda NPP U3 and U4 are served by o series of common objects to other units, which will be partially decommissioned according to situation:

In the enclosure:

- Common Services Channels and Cable Ducts;
- Distribution Pool;
- Pipe Support;
- Sewage Water Pump Stations, stage I and stage II;
- Sewage Water Pipe System including the Pump Station;
- Intermediate Storage Facility of Radioactive Wastes.

Out of enclosure:

- Water Intake;
- Cooling Water Admission Canal;
- Hot Water Discharge Ducts and Canal to the Danube;
- Hot Water – Cold Water mixture canal;
- Special structures and hydrotechnical works for site protection;
- Secondary Access Road;
- Guard House;

- Outer Equipment Warehouse (Seiru Area);
- Guard House for NPP Security Personnel (Saligny);
- Aerial Electric Lines and Cables for Power Supply (20 kV).

f) The construction materials used to the Cernavoda NPP U3 and U4 structures / buildings

The main construction materials used in the carry out of the structures/buildings of Cernavoda NPP U3 and U4 are presented in Table 2.3-1.

The partitioning of the service building was carried out by concrete blocks walls, by IAFS and BASF and metallic panels, sustained by a structure of zinc plated sheet.

The doors garnitures (including the locks) are made of silicone based elastomers.

The execution joints between the pouring plots of the baseslab and the subbase are tightening with PVC bands and Thiokol liner.

The types of used epoxy liner are:

- the epoxy liner E2 easy to decontaminate, which is used in all areas where the radionuclides contamination can arising and periodical decontamination is necessary or in other situations which require a high decontamination factor;
- the tight and easy to decontaminate epoxy liner E1 used on the perimetral walls, on the containment baseslab and in other areas where a good tightening and a easy decontamination of the surfaces can be required;
- the tight epoxy liner reinforced with fiberglass E3, which must assure a specific tightening of the concrete surfaces, some of those are immersed in water and are exposed to an intense radioactive field. These liners are used for the reactor building dome, the concrete tanks for storage of the radioactive liquid wastes and eventually for other areas.

Table 2.3-1. List of principal construction materials used to the Cernavoda NPP U3 and U4 structures/buildings

Material	Grade Romanian
Normal weight concrete (2400 Kg/m^3)	
Plain concrete as mudmat in the Turbine Building	C6/7.5
Plain concrete as mudmat and completion	C8/10
Reinforced concrete in subbase	C16/20
Prestressed concrete in containment base	C32/40
Prestressed concrete in perimetral wall	C32/40
Prestressed concrete in ring beam	C32/40
Prestressed concrete in domes	C32/40
Reinforced concrete in internal structure	C16/20
Reinforced concrete in internal structure	C20/25
Reinforced concrete in internal structure	C25/30
Reinforced concrete in infrastructure of service building	C20/25
Reinforced concrete in infrastructures of other nuclear buildings	C20/25
Reinforced concrete in Turbine Building structure	C16/20
Heavy weight concrete with baritine aggregate&scrap iron slugs (3500 kg/m^3)	
Shielding concrete in internal structures	C20/25
Reinforcing steel (mild steel)	
Constructive reinforcement	OB 37
Designed reinforcement in nuclear buildings and turbine building	PC60N
Wire for prestressing cables	SBPINΦ7
Steel structures	
Laminated shapes in steel structures	OL52.2.K
Laminated shapes in anchor heads	41MoC11
Laminated shapes in bearing plates	OL52.4.kf
Electrodes for welding	E51X
High strength bolts (the bolt proper)	41MoC11
High strength bolts (the nut)	33 MoC11
High strength bolts (the shim)	41MoC11
Freyssinet flat jacks	imported

The contamination or activation condition should be established for each of the systems, buildings and equipment that are submitted to the decommissioning process. A part of the plant connected directly to the reactor or systems that transported radioactive fluids may be activated or contaminated. The rest of the buildings and systems may be less contaminated as function of operation history of the plant.

According to Ref. 2-20, the radiological condition of a nuclear facility represents the main condition in the process of choosing the decommissioning options and techniques.

In order to carry out the safe decommissioning process of the nuclear unit, it is important to take into account the following facts, which are significant from point of view of the impact on the personnel, public and environment:

- the radiological condition of the plant (the contamination or activation level presented of structures or systems);
- the layout and the geometry of the radiation sources;
- the physical status of the plant structures and systems which are going to be used as auxiliary or protective systems during the decommissioning activities (their deterioration degree, their functionality);
- the physical status of the plant control systems which are going to be used during the decommissioning operations;
- the status of the electrical systems (considering the deterioration of the cable's insulation, the possible accidental break of routes, the improper connection);
- the conveying and hoisting units status (their integrity, the existing control possibilities, the safety shutdown of this units, the protection of the transported loads);
- the physical status of the structures which are going to be used as confinement structures and as radiation shielding during the decommissioning process (the integrity and the efficiency of those shields, the confine structure's integrity, the possibility of leakage monitoring, etc);

- the existence of records concerning the operating history of the plant;
- the existence of the general plans and lists containing the equipment, components and structures (which to contain the materials used in their construction, their overall dimensions, the component's mass and, if possible, an estimation of their radiological status, etc.).

2.3.2.2. Main Operations Required in the Cernavoda NPP U3 and U4 Decommissioning Process

The removal of spent fuel from the core reactor (the least loading) at the end of its operational lifetime should be performed as part of decommissioning operations or as one of the preparing activities in decommissioning process, according to (Ref. 2-26).

Taking into account, first of all, the decommissioning of the more contaminated parts and then the less contaminated parts, the usual decommissioning operations, in random order, are described below:

- the sampling of solid, liquid and gaseous materials (by analysis will be established the activation/contamination level and also will be provided the nuclear zoning);
- the discharging of the radioactive fluids from the circuits;
- the cutting/disassembling/dismantling of the pipe sections from circuits and the radioactive fluids storage tanks;
- the cutting/dismantling and removing of the calandria vessel;
- the dismantling of the radiation shielding surrounding calandria vessel and of the other equipment provided with shielding walls;
- the cutting/ disassembling/dismantling of the large components of systems and circuits (steam generators, pumps, heat exchangers, ion exchangers, pressurizer, deaerator, delay tanks);
- the dismantling/disassembling and removing of the control panels;
- the removing of the plugged filters in the ventilation systems;
- the dismantling/cutting of the ventilation ducts;

- the dismantling of the electrical panels and cables for the power supply systems to be disassembled and also the racks and supports associated;
- the cutting/dismantling of the supply pipes with compressed fluids (gas, compressed air);
- the dismantling and removing of the electronic devices (dosimetric devices, control devices, etc.);
- the decontamination of walls, floor, equipment;
- the uncovering of the floor and the scarifying of concrete in order to remove the contaminated or activated layers;
- the demolition of all unit buildings and the removing of the resulting material, including the basements, in case of the site final liberation for unrestricted use;
- the segregation of the waste, mainly in two groups: radioactive and non radioactive;
- the selection of the hazardous chemical waste such as asbestos, hydrides and other toxic, flammable, explosive waste;
- the separate packing of the non-radioactive wastes, in order to transport them for treatment or, by case, for final disposal;
- the sorting of the all types of radioactive solid wastes in: metallic wastes, textiles, concrete, wood, etc.;
- the separate packaging (in containers/drums) of these types of wastes;
- the temporary storage of wastes containers inside the nuclear unit;
- the handling of the cutting/dismantled radioactive components from the nuclear unit to the place where they will be crumbled for packaging;
- the handling of the waste drums by the means of conveying and hoisting units (bridge crane, monorails, chutes);
- the loading of the solid waste containers into the vehicle which will transport them outside the nuclear unit for treating and final disposal;
- the transport of all types of waste outside the nuclear unit;

- the cleaning/decontamination of the surfaces, protection equipment and tools used in the process, intense contaminated components, storage systems and tanks used for transporting the radioactive liquids during the plant operating period;
- the decontamination of the construction surfaces (concrete surfaces);
- the fixing of the internal or external contamination of the components which will be removed by the use of base coats and paint coats (where the decontamination is not possible);
- collecting of all the effluents resulted in the decontamination process;
- collecting of the radioactive or possible radioactive effluents which can be retained into the sumps, after discharging of fluids in the process systems;
- collecting of the combustible wastes (paper, plastic materials, paper filters, cotton etc.);
- the handling of the combustible wastes;
- removing of the furniture and laboratory glass-ware;
- cutting of the radioactive wastes for packaging into the drums.

For the site reconstruction, after the building demolition, the following operations will be achieved:

- the collection of soil samples from the place where the nuclear unit was located and from its neighbourhood;
- the analysing of the soil samples (from the point of view of radioactivity or the toxic substances presence);
- the removal - if the existence of soil contamination is signalled- of one layer of soil; the thickness of this layer will be chosen accordingly to the depth up to where the contamination was detected; the removed soil will be considered as radioactive waste and adequate treated.
- the reconstruction of the site's geometry by the means of zone levelling and the developing of a fertile soil layer and of a vegetal layer (composed of grass, shrub) in order to fix the soil.

2.3.2.3. Hazardous Chemical Substances Stored in the Nuclear Units

According to Ref. 2-26, before the beginning of the decommissioning activities, an inventory of all-hazardous chemical materials and substances present in the nuclear unit should be conducted.

The toxic materials must be emphasis in order to avoid their negative impact on the health of people and on the environment. Similarly, hazardous substances, such as oils from the reactor, must be carefully handling, due to their present significant risk of fire or explosion.

A list of the hazardous chemical substances stored in the nuclear unit is presented below:

- Ammonia 25 %; it presents toxic and chemical danger; it is a corrosive and a strongly fumigating liquid;
- Cyclohexilamina (purity 79,2 %); is volatile, alkaline reactive, corrosive, toxic, inflammable liquid; during operational time, it is used for pH control in the secondary circuit;
- Hydrazine 15 % (hydrazine hydrate 35 %); is a reductant for oxygen and a corrosion inhibitor used in secondary circuit;
- Hydrazine 35 % (hydrazine hydrate 35 %); is a reductant for oxygen and a corrosion inhibitor used in secondary circuit;
- The morpholine (purity 99 %); is used during the operating period for pH control, in secondary circuit;
- Soda nitrite (corrosion inhibitor -detergent);
- Soda nitrate (flomate 537-corrosion inhibitor);
- Biomat 5716 (biocid), used in raw service water system (RSW);
- Glycol; is used for diesel generators in the operating period;
- Dowacal 10 or Dowtherm SR1 (monoethylene glycol solution with corrosion inhibitor); it is used in operating period for Glycol Water System Hydrochinona (photographically purity);
- Potassium hydroxide;

- Soda hypoclorite;
- Citric acid;
- Hydrochloric acid 36 %;
- KT2 T00379 (coagulator);
- Soda hydroxide - 48 % purity;
- Iron chloride – 40 % purity;
- Sand for the filtration bed (SiO_2 - 95 %), 9 - 15 mm;
- Sand for the stoppage stratum of the mechanic filter (95 % purity), 4 - 9 mm;
- Sand for the stoppage stratum of the mechanic filter (95% purity), 2 - 8 mm;
- Water Chemical Treatment Plant (STA) resins;
- Nuclear systems resins;
- Activated carbon; is used during the operating period for the D_2O Purification System;
- Lithium hydroxide; it is used during the operating period for pH control in the Primary Heat Transport System and in the Chemical laboratory;
- EDTA;
- hexahydrated gadolinium nitrate (min. 99.9 %);
- boric anhydride;
- Amberjet 12000H; it is used polyethylene alcohol non-toxic detergent); is used for avoiding the oxidizing agents.for the filter cleaning operation;
- Ion-exchange resins as a powder form; it is used for the treatment of the liquid radioactive waste;
- Renex 36 (threedecil polyethylene alcohol non-toxic detergent); is used for avoiding the oxidizing agents.

A part of these chemical substances is used for the chemical conditioning of the systems and also in technological process, as well as for the use in laboratories (in small quantities).

The decommissioning methods, the costs of the decommissioning process and also the personnel's safety can be strongly affected by the existence of these materials/substances.

According to Romanian legislation (Ref. 2-27, Ref. 2-28), all types of polychloride biphenyl (PCB) and similarly components, as well as the asbestos (powder and asbestos fibres), are considered hazardous substances due to their toxicity and so they require a special administrative and monitoring measures.

According to Ref. 2-29 the chloride aromatic hydrocarbons, by type PCB, are not used in Cernavoda NPP U3 and U4, due to their high level of toxicity. Also, according to Ref. 2-30, for both the structures/constructions and the thermal isolation operations in Cernavoda NPP U3 and U4 is not provided the use of asbestos.

Consequently, these hazardous materials/substances (PCB, asbestos) will not find as wastes at the end of decommissioning process of the nuclear units.

2.3.3. Techniques of Cernavoda NPP U3 and U4 Decommissioning

The techniques and equipment that will be used in decontamination/ dismantling/demolishing of Cernavoda NPP U3 and U4 are, in general, similar to those used in the non nuclear industry, but they present certain modifications and utilization specific of nuclear field. One of these techniques and equipment were specially developed to be used in the decommissioning process of the nuclear objectives (e.g. spalling of the contaminated concrete surfaces).

2.3.3.1. Decontamination Operations

The term decontamination as used in the nuclear field means the removal of radioactive contaminants in order to reduce the residual level in/on the radioactive materials within the plant and site.

Depending on result of contamination characterization, doses optimization and minimization criteria of the resulted wastes, the decontamination will be applied on the internal surface of components and systems, the external surface of equipment, floor, walls, fluids and contaminated solids and instrumentation implied in the decommissioning activities of nuclear unit.

For the components of equipment that can not be decontaminated, the fixing of the surface contamination is recommended, on condition that all openings to be sealed in order to reduce the contamination spread possibilities.

The objectives of decontamination include:

- reduction of exposures of the personnel and public during decommissioning activities;
- reduction of the residual radiation sources on the site so as to minimize any potential hazard to the personnel and public;
- recycle and reuse of equipment and materials which was decontaminated;
- reduction of the volume of radioactive wastes.

To achieve a good decontamination factor, the following factors must be taken into account:

- the type of plant to be decommissioning;
- the operating history of the plant;
- type of material: steel, Zircaloy, concrete, etc;
- type of surface: rough, porous, coated, etc.;
- type of contaminant: oxide, crude, sludge, loose, etc.;
- composition of the contaminant: activation products, fission products, actinides, etc.;
- external or internal surface to be cleaned;
- the decontamination factor required;
- destination of the components being decontaminated: disposal, reuse, etc.;
- time required for decontamination;
- the proven efficiency of the decontamination method;
- the type of component: pipe, tank, etc.

Other elements which are important in selecting the decontamination method but which do not affect the decontamination factor are:

- availability, cost and complexity of the decontamination equipment;
- the need to condition the secondary waste generated, in order to disposal it;
- occupational and public doses resulting from decontamination;
- other safety, environmental and social issues;
- availability of trained operators;
- extent of the plant decontamination plan to achieve decommissioning;
- extent of the facility to be decontaminated: isolated systems, enclosed and ventilated spaces, etc.

The decision of decontamination and the choice of decontamination method will depend on the balance of these factors on the decommissioning and the impact on the population and environment.

The decontamination processes recommended by the international guides (Ref. 2-31) are:

a) Chemical decontamination (the removal of the contaminants by dissolution with a liquid chemical reagent);

b) Electrochemical decontamination (the removal of the contaminants by electropolishing and electropickling);

c) Mechanical decontamination (the physical removal of the contaminants):

- Vacuum cleaning;
- Washing, swabbing or scrubbing with or without solvents;
- Water-steam jetting;
- Abrasive jetting (carborundum);
- Spalling (mechanical in force process);
- Flame spalling;

- Scarifying (similar to spalling but offering deeper penetration per pas);
- Normal radioactive decay, up to the allowed levels for the beginning of the decontamination;
- Ultrasonic cleaning;
- Melting.

2.3.3.2. The Disassembly Techniques

The main components of the nuclear reactor to be dismantled/demolition are the reactor pressure vessel, vessel internals and concrete structures such as the shielding walls.

In order to remove activated or contaminated areas of the nuclear unit structures will be necessary demolition or scarification of some concrete surface.

The activities and techniques used in these operations include:

a) Segmenting activated vessels and internals:

- Arc saw;
- Plasma arc torch;
- Oxyacetylene cutting;
- Thermal reaction lance

b) Segmenting piping, tanks and other components:

- Explosive cutting;
- Hacksaws and guillotine saws;
- Abrasive cutters;
- Circular cutters.

c) Concrete demolition and surface decontamination:

- Controlled blasting;
- Wrecking ball or slab;

- Backhoe mounted rams;
- Flame cutting;
- Rock splitter;
- Demolition compounds;
- Wall and floor cutting;
- Core stitch drilling;
- Pavement breakers;
- Drill and spall;
- Scarifies.

d) Remotely controlled equipment

2.3.4. Protection Measures for the Personnel, Public and Environment

The main objectives of decommissioning activities are (Ref. 2-36):

- Insuring the safety of all the operations, in order to protect the personnel, population and environment;
- The safe management of all the equipment and tools needed in the handling and disassembling activities;
- The safe management of the removed radioactive materials (from the cutting, treating, conditioning, intermediate storage, transport and final disposal operations).

2.3.4.1. Measures for the Personnel Protection

2.3.4.1.1. The Nuclear Zoning of the Facility

The nuclear zoning is specific for each decommissioning stage; the zoning activity will evolve, as the areas will be decommissioned. The zoning activity will be accomplished based on the dates concerning the estimated dose rates and contamination levels. The existing zoning is maintained in the initial phase of decommissioning.

2.3.4.1.2. The Access Control

The access control in the nuclear zones is assured by establishing of fixed access routes between these zones, which will be periodically decontaminated and checked from point of view the contamination. Also, points of access provided with systems for radiation monitoring and access control will be provided between these zones.

2.3.4.1.3. The Protection Measures against External and Internal Irradiation

In addition to the zoning and access control, in all the working areas and for all the operations in which the personnel can be exposed to irradiation or contamination, the following protection means will be provided (in order to assure the application of the ALARA principle that all the exposures should be kept as low as reasonably achievable):

- establishment of special working areas (depressurized, sealed enclosures for cutting the reactors components, provided with ventilation systems equipped with filters);
- use of fixed and mobile protection screens;
- use of adequate individual protection equipment (consisting of overall with hood, protective gloves, protective eyeglasses, face mask and, if needed, integral suit with protecting cap);
- decontamination of the working equipment used for the cutting/dismantling operations;
- specific measures for minimizing the personnel's exposure during each of the operations of radioactive waste dismantling/disassembling/cutting, handling and decontamination;
- use of radiological alarm systems (tested and calibrated each time is necessary) in every working area, in order to signal any rise of the radioactivity in the area above the admitted limits.

2.3.4.1.4. The Activities Planning

In order to reduce the occupancy period in the zones with a high radioactivity level, the decommissioning activities must be optimum planned.

2.3.4.1.5. The Radioactivity Monitoring

The decommissioning project will include:

a) Individual dose monitoring systems; two components will be considered in the evaluation of the individual doses: the external doses and the internal contamination.

b) Radioactivity monitoring systems provided in the access points, working areas, in the handling and temporary storage areas of radioactive wastes, both for direct monitoring and resultant effluents radioactivity monitoring (residual waters, radioactive air).

2.3.4.1.6. The Decontamination of Personnel

The equipment and facilities of decontamination used in case of occupational exposed personnel contamination consist of the following:

- the lock, which contains:
  - i) the shower, fitted out with equipment for the washing activities of injured and uninjured contaminated personnel; the contaminated water will be evacuated to recycling or storing facilities, depending on the radioactivity level;
  - ii) locker rooms;
  - iii) medical emergency case;
- surgical suits;
- plastic covers;
- surgical capsules;
- different kind of gloves (plastic and rubber);
- sterile bandages;
- protection suits or cleaned covers for the patients and socks;
- overshoes;
- big towels;
- electric razor and shaving foam for the aerosols;
- scalpel for removing the bandages;
- big plastic bags for collecting of the contaminated clothes;
- oxygen tubs for the decontamination team;
- notification plates (e.g. “Attention! Contamination hazard”, ”Do not enter”);
- individual dosimeters (ionization chambers with direct reading) and thermo luminescence dosimeters;
- adhesive band for medical user (min. 3 mm width);
- marked containers for collecting biological samples;

- blankets;
- adhesive labels and plates for marking contaminated materials and tissues;
- specimen flasks (filled with formalin, if the freezer facilities are not available);
- notebooks, maps, writing tools;
- portable beta-gamma radiometers and equipment for dose rates detecting;
- scintillation portable alpha detectors;
- absorbent paper roles;
- plastic foils;
- decontamination specific consumables: usually detergents, with titanium dioxide (abrasive), $KMnO_4$ (potassium permanganate), acid sodium sulphite for removing the contaminated stains, bleaching agents (sodium hypochlorite NaOCl 5 %); all these items must be stored in a special marked box;
- containers for storing the waste and baskets for storing the clothes and other contaminated objects;
- fixed or portable beta-gamma shielding equipment used for treating the personnel who suffered a beta-gamma contamination.

Overshoes and gloves will be provided for the personnel who will effectuate the decontamination of the persons involved in an accident; in case of alpha emitter significant contamination they will be also provided with boxes respirator.

2.3.4.1.7. Training

The involved personal in the decontamination activities will be properly licensed and trained for development of nuclear activities, including the knowledge of all applicable procedures. Also, the site personal must have the safety culture.

2.3.4.2. Measures for the Public and Environment Protection

During normal decommissioning operations of the nuclear unit, the population radiation exposure pathways are the following:

- the gamma-radiation direct exposure of population, when radioactive waste (including massive components) is transported from the site to the treatment or disposal areas.

In order to avoid this situation, the protection measures that must be considered are the following:

  - use of the special means of transportation, compatible with both the shape and size of the component that must be transported;
  - use of proper radiation shielding;
  - use of the proper straining systems;
  - use of the licensed personnel for the handling and transport operations.

If the transport routes pass through areas accessible to the public, the usual protection measures as specified in the radioactive wastes transport regulations must be considered.

- the inhalation of the air contaminated with radioactive dust; this risk may appear during the processes of final demolishing of buildings or during the uncovering of the site. In such cases, the use of demolishing and uncovering techniques producing less dust are recommended as protection measures
- the inhalation of the air contaminated with possible radioactive gaseous effluents
- the internal exposure, due to contaminated water or aliments ingestion, in case of evacuation in the environment of the potential radioactive effluents (depending on the evacuated quantities);

The elements concurring to assure an adequate protection of the population and the environment, are mentioned below:

2.3.4.2.1. Sources Control

This element must assure the accomplishment of the necessary measures and activities for the radioactive materials confinement.

In order to prevent and to reduce the spread of radioactive material in the environment, the following measures will be considered:

- providing of constructions isolating the areas containing radioactive systems or structures, susceptible to leaks (depressurized, monitored and proper ventilated working enclosures);
- assuring the proper functionality of local drainage and ventilation systems (including dust vacuum systems);
- assuring the availability of decontamination systems;
- providing of some handling and dismantling/disassembling mechanisms to avoid secondary activation or contamination (e.g. it is recommended to use the clove hitch instead of the hand saw for pipes cutting operations);
- working equipment for the decommissioning activities will be choose in a way that:
    - will prevent the radioactive material accumulation in difficult to decontaminate cavities or places;
    - their decontamination process will be an easy one;
    - will limit the spread of radioactive material arising from decommissioning activity;
    - can be remotely controlled or from the back of the protection shielding screens.

The decommissioning activities will be performed in a way to minimize the airborne radioactive material production; means of controlling the airborne radioactivity will be provided.

Waste resulted from decommissioning activities will be isolated and sealed either in standard drums or with plastics foil, in order to avoid the spread of the radioactivity during their handling and transporting activities.

2.3.4.2.2. The Effluents Control

This element must include the all-necessary measures and activities for the control of the environmental radioactive releases, in order to comply with the limits imposed by the competent authority.

The following means will be used for the effluents control:

- measures of limiting the effluents radioactivity level to values not exceeding the maximum admitted values specified in the regulations (by using the special working enclosure, special local and general ventilation systems and special systems for collecting and evacuating the resulted liquid effluents);
- the control systems for the environmental effluents releases, in order to assure that the unrestrictive release limits are not exceeded.

During the nuclear unit decommissioning activities, the contaminants can be released in the environment on different ways: air, surface waters, underground waters and the transport of the wastes or other evacuated materials/equipment, animals.

The air quality protection measures will consist on the gaseous effluents filtration by the equipment provided with various types of filters (including HEPA filter) and, also, radioactivity level monitoring and O_3, NO_2, CO, SO_2, solid particles content.

The surface water protection will be achieved by the radioactive and non-radioactive contaminants spread prevention (sources control).

The underground waters protection will be assured by the drainage developments, the control, prevention and organization of the interventions in case of floods, by the evacuation of meteoric waters and avoiding the ground contamination.

The radioactive and non-radioactive materials transportation on the public roads will be performed in a way that will not generate potential risks for the public and the environment. The packaging and transportation means, the attendant personnel and the transport documents will be previously licensed; the documents must include information about radionuclides content, radiation dose rate on the package exterior, the external contamination of the surface.

2.3.4.2.3. Environmental Monitoring

The environmental monitoring is an essential element of the nuclear unit decommissioning process. This element is necessary in demonstrating the control efficiency regarding the radioactive materials releases in the environment and for providing data regarding the environmental impact of the decommissioning activities.

As part of the documentation package needed for obtaining the nuclear unit decommissioning license, a personnel, public and environment Radiological Monitoring Program, including the afferent procedures, will exist for all the duration of decommissioning activities.

Usually, continuing the monitoring programs developed in the operating period of the nuclear plant proves to be sufficient. If necessary, the program will be modified according to the conditions existed on the duration of decommissioning.

The monitoring operations executed before the beginning of the decommissioning activities will provide information about any possible environmental effect derived from those activities.

The following principal elements will be contained in the environmental monitoring program:

- establishment of the monitoring points;
- frequencies of the monitoring;
- detection limits.

The monitoring program will be based on the laboratory analyzes of the environmental samples.

2.3.4.2.3.1. Location of the Sampling Points

The type of samples to be analyzed and the frequency of sampling and analyzing will be established by the afferent procedures of the Radiological Monitoring Program for the personal, public and environment during the decommissioning activities.

Similarly, the sampling locations will be established in the Radiological Monitoring Program by the appropriate procedures.

Into the protection area (in the nuclear unit perimeter) will be established, through the appropriate procedures of the program, the location of the water, soil, vegetation, air and atmospheric deposits samples.

The samples of grow vegetation and milk will be bought from the producers of the localities in the vicinity of Cernavoda NPP.

2.3.4.2.3.2. The Sampling, Conditioning and Analyzing of the Samples

Samples from the following location will be sampled, conditioned and analyzed:

- water from the evacuation channels, surrounding hydrographic system, sewage and inspection pits;
- uncultivated soil and sediment;
- spontaneous and cultivated vegetation, milk;
- dry or total atmospheric deposits;
- atmospheric aerosols.

The sampling locations, the frequency of the sampling, the samples conditioning and analyzing will be established according to the working procedures of the personnel, public and environment Radiological Monitoring Program during the decommissioning activities.

2.3.4.2.3.3. Assurance and Control of Analyses Quality

Periodically verification of the device response as well as its calibration will be effectuated, according to working procedures in the environment Radiological Monitoring Plan. Duplicate samples are also periodically analyzed.

2.3.4.2.3.4. The Reevaluation of the Monitoring Program

In case of some conditions modification, the personnel, public and environment Radiological Monitoring Plan during the decommissioning activities, elaborated in order to obtain the nuclear unit decommissioning licensing, will be reevaluated. The accumulated results, the operating system changes of the nuclear unit, the introduction or withdrawing of some sources during the operation and the new scientific results will be considered in these reevaluations.

2.3.4.3. Emergency Plan

The package of documents needed to obtain the nuclear unit decommissioning licensing will contain the Emergency Plan in case of nuclear events with impact on the people, environment and personnel; this Plan will be periodically revised. Also, application exercises of the Emergency Plan, with all the involved factors, will be performed. The following Emergency Plan measures will be included in the decommissioning project:

- emergency specific procedures;
- training requirements of the personnel involved in the decommissioning activities;
- emergency communication and annunciation systems;
- special routes and areas for protection/evacuation of the personnel;
- possibility of the plant condition surveillance during the emergency procedures application period, in order to have the possibility to evaluate the consequences of an accident and to be able to announce the end of the emergency state.

2.3.4.4. The Physical Protection and Security of the Nuclear Plant

For the proper decommissioning activities, the project will contain the following measures meant to assure the physical protection and security of the plant:

- existence of the physical protection systems of the plant (the access control of the personnel in/outside the unit, control of materials radioactive or with potential radiological consequences);
- prevention and detection measures of potential accidents: fires, explosions, incidents with radiological consequences.

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