ROMANIA

National Commission for Nuclear Activities Control

National Report on the Implementation of the Stress Tests

December 2011

FOREWORD

This report provides information on the measures taken by Romania for the implementation of the stress tests required by the European Commission following the Fukushima accident.

The information provided in this report reflects the status of Cernavoda NPP activities relevant for the completion of the stress tests as conducted by the 31st of October 2011 (date of the submission of the final report by the licensee).

The report has been prepared by the National Commission for Nuclear Activities Control, based on the stress test report for Cernavoda NPP submitted by the National Company Nuclearelectrica and on the regulatory reviews conducted by the end of 2011.

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EXECUTIVE SUMMARY

Following the Fukushima Daiichi accident occurred in March 2011, the Romanian authorities and the nuclear industry have started to perform reassessments of nuclear safety and emergency preparedness arrangements and to implement improvements, in line with the international efforts in this direction.

The safety reassessments conducted in response to the Fukushima accident included the "stress tests" review required by the European Council, in compliance with the specifications and criteria issued by the the European Commission, based on the work done by the European Nuclear Safety Regulators' Group (ENSREG) and the Western European Nuclear Regulators' Association (WENRA).

This report has been elaborated by the National Commission for Nuclear Activities Control (CNCAN) taking account of the standard format and content recommended by ENSREG and provides information on the outcome of the implementation of the "stress tests" in Romania, for Cernavoda Nuclear Power Plant.

The regulatory reviews performed in relation to the implementation of the “stress tests” have focused on verification of the completeness and validity of the reports submitted by the licensee. It was found that the stress test specifications and methodology have been complied with by the licensee. No concerns have been raised from the regulatory reviews performed to date. The conclusion of the review conducted by CNCAN is that the risk to the public from beyond design basis accidents at Cernavoda NPPs is low and is kept under control.

A significant effort has been made by the licensee to respond to the lessons learned from the Fukushima accident in a timely manner. Several potential design and operational improvements have been identified and their implementation is in progress to further enhance the existing safety margins and reduce the risk from severe accidents. The licensee is fully committed to implement these improvements in a reasonable timeframe and all the financial resources needed for this purpose have been already secured.

CNCAN will monitor the licensee's progress in the implementation of the planned improvements and will continue to perform safety reviews and inspections with a focus on the implementation of the lessons learned from the Fukushima accident, to ensure that all applicable opportunities for improvement are identified and addressed.

CHAPTER 0 - LEGAL AND REGULATORY FRAMEWORK IN ROMANIA

0.1. Background

The nuclear policy of Romania encompasses the development and use of nuclear energy and other nuclear fuel cycle activities, as well as oversight of the development and enforcement of nuclear legislation and regulations to ensure that all nuclear activities are strictly regulated and controlled to the highest standards to ensure public health and safety.

Romania has only one nuclear power plant, Cernavoda NPP, with two units in operation. Cernavoda NPP Units 1 and 2 cover approximately 18% of Romania's total energy production. The Government has plans to further increase nuclear generating capacity through the commissioning of Units 3 and 4 of the Cernavoda NPP.

Long term commitment to nuclear power development, considered one of the drivers of the Energy Strategy of Romania, builds on the well developed national nuclear infrastructure, proven and safe technology and excellent performance of Cernavoda NPP.

Following the Fukushima Daiichi accident occurred in March 2011, the Romanian authorities and the nuclear industry have been actively involved in providing information to the public and the media on the development of the accident, on its significance at national and international level, as well as on the measures taken to improve nuclear safety and emergency preparedness in Romania.

0.2. Legal and Regulatory Framework

The Law no. 111/1996 on the safe deployment, regulation, licensing and control of nuclear activities, republished on the 27th of June 2006, provides the legislative framework governing the safety of nuclear installations. In this report, it will be further referred to as "the Law".

The Law empowers the National Commission for Nuclear Activities Control (CNCAN), which is the national nuclear regulatory authority, to issue mandatory regulations on nuclear safety, to issue licences for nuclear installations and activities, to perform assessments and inspections to verify compliance with the nuclear safety requirements and to take any necessary enforcement actions.

CNCAN is thus responsible for regulation, licensing and control with regard to nuclear safety, radiological safety, non-proliferation of nuclear weapons, physical protection of nuclear installations and materials, transport of radioactive materials and safe management of radioactive waste and spent fuel.

CNCAN reports to the Prime Minister, through the General Secretary of the Government. CNCAN is completely separated and independent from all the organisations concerned with the promotion or utilisation of nuclear energy. The responsibilities assigned to CNCAN by the Law are concerning solely the regulation,

licensing and control of nuclear activities. CNCAN exercises its functions independently from the ministries and other authorities of the central public administration subordinated to the Government.

The Law clearly stipulates that the prime responsibility for the safety of a nuclear power plant rests with the licence holder. As required by the Law, a licence is needed for each of the stages of the life time of a nuclear installation. For a nuclear power plant, the licensing stages include design, siting, construction, commissioning, trial operation, operation, repair and/or maintenance (as major refurbishment), modification (as major upgrades), preservation and decommissioning.

All the regulations issued by CNCAN are mandatory and enforceable. The regulations are developed in observance of relevant international standards and good practices. Various sources of information relevant for updating the system of regulations and guides are used, including international cooperation as well as feedback from the operators and from CNCAN inspectors based on their experience from the enforcement of the regulations.

The priorities for the development and revision of national nuclear safety regulations have taken account of the harmonisation process in the WENRA countries. During the harmonisation study, the national regulations have been benchmarked against the reference levels established by the Reactor Harmonisation Working Group based on the Safety Requirements and Safety Guides of the IAEA Safety Standards Series. Due to CNCAN's participation in the harmonisation process within the WENRA (Western European Nuclear Regulators' Association) countries, the use of IAEA Safety Standards has become more systematic.

Examples of regulations issued in the period 2005-2010 include the regulations on Fire Protection for NPPs, PSR (Periodic Safety Review) and PSA (Probabilistic Safety Assessment), which make extensive use of the requirements and guidance provided in the IAEA documents.

The revision of the nuclear safety regulation establishing general design criteria for nuclear power plants has been finalised in 2010. The revision of this regulation was aimed at endorsing more of the NS-R-1 requirements, in addition to those which served as basis for the reference levels of WENRA. The regulatory requirements on siting have also been revised.

The new regulations, "Nuclear Safety Requirements on Siting of Nuclear Power Plants" and "Nuclear Safety Requirements on Design and Construction of Nuclear Power Plants", have entered into force at the end of 2010.The most important elements introduced by these new regulations are summarised below:

- the establishment of new numerical nuclear safety targets / quantitative nuclear safety objectives;
- requirements on the consideration of severe accidents in the establishment of design bases and in the choice of site for nuclear power plants and on the analysis of severe accidents for demonstrating compliance with the quantitative nuclear safety objectives;

- requirements on accident analysis, including on the way in which deterministic and probabilistic safety analyses should be used together in the design of nuclear power plants;
- detailed requirements on the format and contents of the safety analysis reports which need to be elaborated by the applicants for site and construction licences;
- formulation of nuclear safety requirements for generic plant systems in a technology-neutral, function oriented manner, without prescribing technical design solutions,
- establishment of requirements on the safety classification of nuclear power plant systems, structures and components based on their safety importance, i.e. their contribution to ensuring the essential nuclear safety function.

This approach represents a change from the prescriptive national requirements used for the regulation of siting and design of nuclear power reactors. Although the new requirements are more demanding, taking into account the technological development and the current safety standards worldwide, they leave flexibility to the designers, provided that the choice of technical solutions is thoroughly justified.

A regulation containing requirements on the commissioning and operation of NPPs is also planned to be finalised in 2012, taking account and making use of the latest IAEA requirements.

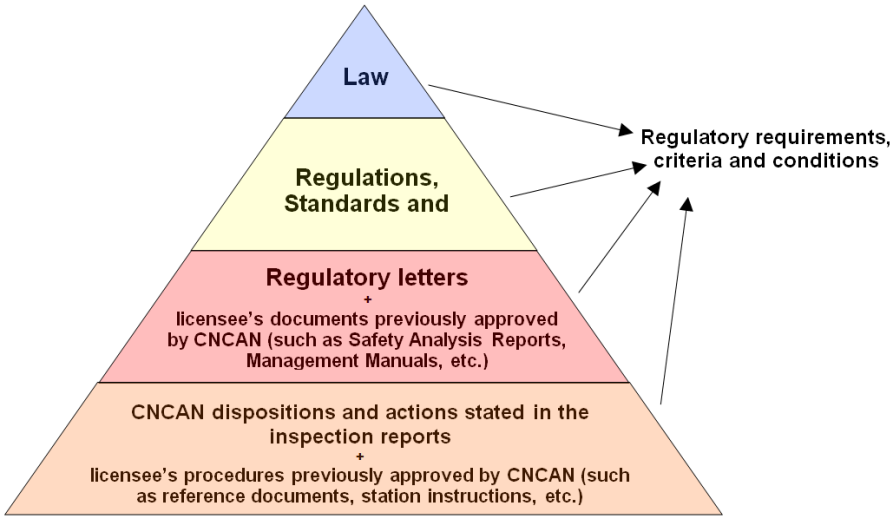

Fig. 0.1 - Requirements used by CNCAN in the licensing process

The detailed regulatory requirements, as well as the assessment and inspection criteria used by CNCAN in the licensing process are derived from a number of sources, such as:

- Romanian regulations;
- Limits and Conditions specified in the different licences;
- IAEA Safety Standards and Guides and WENRA Reference Levels;
- ICRP recommendations;
- Regulatory documents developed by CNSC and US NRC;
- Applicable Standards and Codes (CSA, ANSI, ASME, IEEE, etc.);

- Safety related documentation produced by the licensee and approved or accepted by CNCAN (e.g. Safety Analysis Reports, Safety Design Guides, Design Manuals, reference documents, station instructions, operating manuals, technical basis documents, etc.).

Apart from the formally issued (published) regulations, the requirements established by CNCAN in the licensing process are imposed through regulatory letters. Requirements and dispositions are stated also in the inspection reports.

In accordance with the provisions of the Law, CNCAN is empowered to request from the licensees, or from the applicants for a licence, all the documentation needed for the regulatory decision making process on safety related matters. The documentation that needs to be submitted to CNCAN for review and approval is usually specified in the regulations. Additional support documentation is requested on a case by case basis and specified in regulatory letters, minutes of the meetings between CNCAN staff and licensee's representatives, etc. According to the Law, the licensees and applicants have the obligation of facilitating CNCAN inspections and access to documentation and to provide all the information required by CNCAN.

The safety related documentation made available to CNCAN includes a large variety of documents, such as safety analysis reports, (quality) management manuals, different kinds of safety assessments and technical evaluations, information reports and procedures (reference documents, station instructions, operating procedures, work plans, etc.).

The assessment and inspection activities performed by CNCAN staff are documented in assessment reports, inspection reports and written minutes of the meetings with licensee's representatives. These documents are distributed to the licensee, in addition to the regulatory letters that summarise the main regulatory requirements and dispositions based on findings arising from the review process.

In accordance with the provisions of the Law, CNCAN has in place a system to enforce compliance through graded measures. Therefore, the possible actions that CNCAN can take in the event of non-compliance are:

- dispositions for licensee action (these are stated in each inspection report);
- action notices/directives through regulatory letters;
- licence amendments;
- restricted reactor operation;
- revocation or suspension of the license;
- prosecutions.

The Romanian legal and regulatory framework provides all the necessary elements for ensuring a strong regulatory control of all the nuclear facilities and activities. In line with the international efforts to improve nuclear safety, the lessons learned from the Fukushima accident will be taken into account by CNCAN in further strengthening the regulatory framework and its review and inspection processes.

0.3. Use of WENRA Reference Levels

The WENRA Reactor Safety Reference Levels (RLs) have been incorporated into the Romanian regulatory framework through the following regulations:

- Requirements on Fire Protection in Nuclear Power Plants (2006) – incorporating RLs in Issue S;
- Requirements on Periodic Safety Review for nuclear power plants (2006) – incorporating RLs in Issue P;
- Requirements on Probabilistic Safety Assessment for nuclear power plants (2006) – incorporating RLs in Issue O;
- Nuclear Safety Requirements on the Design of Nuclear Power Plants (2010) – incorporating RLs in Issues E, F, G & N.

The revision of the set of 13 regulations on quality management systems, covering activities related to all the phases of the lifetime of nuclear installations, started in 2007, takes account of the latest IAEA Requirements and Guides on Management Systems (GS-R-3, GS-G-3.1 and GS-G-3.5). The external consultation process for the new regulations has been finalised and they are due to be published in 2012. These new regulations cover all reference levels in Issue C.

A regulation on commissioning and operation of NPPs will incorporate the remaining RLs. The intention is to have it published before the end of 2012. This regulation will include provisions for the management of severe accidents.

The compliance with the requirements in the reference levels is currently re-assessed by the licensee as part of the periodic safety review that is ongoing. This assessment has been required by CNCAN. The assessment of the licensee's implementation of the RLs has been performed by CNCAN in support of the benchmarking conducted within the RHWG in 2005 and also as part of the assessment of the implementation of the regulations issued.

The outcome of the benchmarking was that most of the RLs were actually implemented. The evidence that has been used for benchmarking is heavily relying on documentation that has been approved by CNCAN, having the updated safety analysis report as the major source of information for verification of the implementation. A number of plant's procedures, especially operating procedures and their technical basis' documents, inspection and maintenance procedures, as well as procedures relevant for the control of modifications, have also been checked for more detailed information relevant to specific reference levels. In addition, the industrial standards and codes used for the plant design and various operational programmes (e.g. periodic inspection programme, fire protection programme, etc.) have been consulted. As part of the verification process, CNCAN staff has also conducted inspections and interviews with different technical managers from the plant. For specific issues related to design, the design manuals for various systems and the accident analyses, as well as the probabilistic safety assessments have been consulted for ensuring the accuracy of the information presented during benchmarking.

The RLs that were not implemented at the time of the benchmarking (2005) are related to the severe accident management programme (issue LM), development of PSA Level 2 (issue O) and performance of a PSR (issue P). Since then, the licensee

has implemented severe accident management guidelines and is preparing to develop the PSA Level 2 study. The first PSR for Unit 1 of Cernavoda NPP is ongoing and it is expected that the resulting reports will be submitted to CNCAN in 2012.

0.4. Actions taken post Fukushima

Following the Fukushima accident, CNCAN has requested the licensee to do a reassessment of the protection against beyond design basis events, including extreme external events and the emergency preparedness and response arrangements.

The licensee also initiated measures in response to WANO SOER 2011-02, including:

- a thorough plant walkdown for verifying protection against seismic, fire and flooding events;
- acquisition and testing of mobile diesel generators;
- development of new operating procedures for response to Station Blackout and to total and extended Loss of Spent Fuel Pool Cooling events.

CNCAN requested the licensee to perform a reassessment in compliance with the stress test specifications developed by WENRA and endorsed by ENSREG and the European Commission. For this purpose, the licensee established a dedicated team of plant specialists, supplemented with experts from the plant designers (AECL and ANSALDO Nucleare) in order to perform the safety reassessment in compliance with the ENSREG specifications. In support of the reassessment required by the stress tests, new analyses have been performed where required. The licensee has submitted their final stress test report by the 31st of October, in compliance with the schedule requested by the European Commission.

The licensee's stress test report provides extensive information covering all the aspects outlined by the stress test specifications. Due to security reasons and property rights, the licensee's detailed report cannot be made publicly available. Instead, a summary containing all the information relevant for the public was included in this report, which has been elaborated following the ENSREG specifications for national reports on the stress tests.

CNCAN has performed a number of regulatory reviews and inspections in order to verify the completeness and the accuracy of the information provided in the licensee's report and to ascertain whether the stress test specifications and methodology have been complied with by the licensee.

Based on the reviews and inspections performed up to date, CNCAN has confidence that the licensee is able to support all the claims made in the report and that any issues and opportunities for improvement arising from the stress test will be adequately addressed. Further information on the planned improvements is provided in Chapter 7 of this report.

CHAPTER 1 – GENERAL DATA ABOUT THE SITE AND NUCLEAR POWER PLANT

Romania has one nuclear power plant, Cernavoda NPP, with two units in operation, pressurised heavy water reactors of CANDU 6 design (CANadian Deuterium Uranium), each with a design gross output of 706.5 MWe. Unit 1 and Unit 2 started commercial operation on the 2nd of December 1996 and on the 1st of November 2007, respectively. Cernavoda NPP Units 1 and 2 cover up to 19% of Romania's total energy production.

Cernavoda NPP is owned and operated by the National Company Nuclearelectrica (Societatea Nationala Nuclearelectrica, further referred to as SNN). SNN is the license holder for Cernavoda NPP.

The Romanian Government has plans to further increase nuclear generating capacity through completion of the project of Units 3 and 4 of the Cernavoda NPP. SNN has started the procedure for analysing the opportunity for resuming construction of these two units and the feasibility studies and investment organisation has been delegated to EnergoNuclear, a joint venture between SNN and other investors.

As the detailed design for Units 3 and 4 is not yet finalised, CNCAN agreed that these would not be covered under the scope of the current stress tests, but required that any potential design improvements resulting from the stress tests for the operating units will have to be implemented also in Units 3 and 4.

1.1. Brief description of the site characteristics

Cernavoda Nuclear Power Plant (NPP) is located in Constanta county, latitude 44.3˚N and longitude 28.01˚E in the Dobrogea Region (Figure 1.1-1). The nuclear site lies about 2 km southeast of the Cernavoda town boundary, at 4 km southeast of Danube River and at about 1.5 km northeast from the first lock on the Danube-Black Sea Channel (DBSC).
The Cernavoda NPP gets its cooling water from the DBSC. The operational requirement for two units is about 90 m^3/s of cooling water. The cooling water is returned to the Danube River. The DBSC (64.2 km long) is a waterway beginning near Cernavoda and ending at Agigea – Constanta, at the Black Sea. It was opened to traffic in 1984. The canal has two locks: Cernavoda (km 60.3), at the Danube end and Agigea (km 1.9), at the Black Sea end.

The Cernavoda site grade level has been established higher than the highest credible flood water level that can theoretically originate from either the Black Sea via DBSC or Danube River.

The site license for Cernavoda NPP (intended for five units) has been granted in 1979. The safety documentation for demonstrating the fulfillment of regulatory requirements and criteria comprised of the Initial Safety Analysis Report (ISAR) and the supporting technical studies and evaluations.

The factors taken into account in the evaluation of the site from the nuclear safety point of view included both those related to the characteristics of nuclear reactor design and those related to the specific site characteristics. The natural and man-made hazards analysed for the site include, for example: extreme temperatures, snow fall, high winds, flooding, earthquake, low Danube level, explosions, release of toxic and explosive gases, fires, missiles, aircraft crashes.

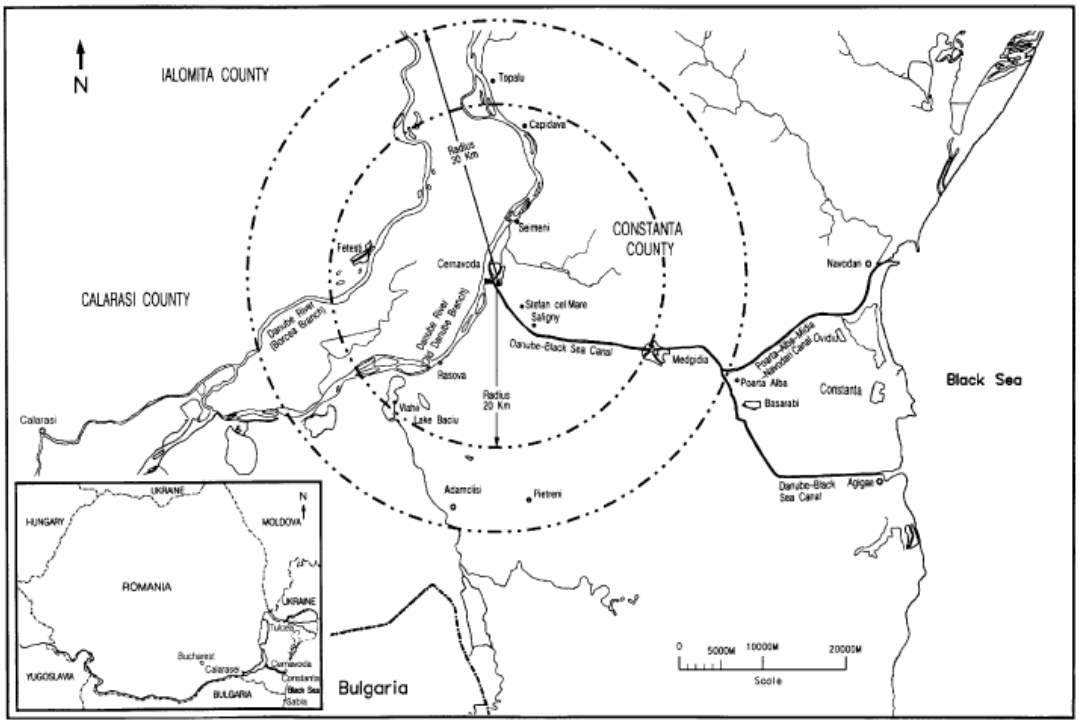

Figure 1.1 Cernavoda NPP Site

In accordance with the regulatory requirements, comprehensive safety assessments have been performed to demonstrate that the reactor design ensures a very low probability for accidents resulting in significant radioactive releases and that the site choice and the technical measures taken to mitigate the consequences of the accidents, should these occur, ensure adequate protection of the public and environment. The latest studies of the site-related factors relevant to safety are reflected in the current Final Safety Analysis Reports (FSAR).

1.1.1. Main characteristics of the units

All the units are pressurised heavy water reactors (PHWR), of CANDU 6 type, designed by AECL (Atomic Energy of Canada Ltd.).

Each unit is provided with a dedicated Spent Fuel Bay (SFB) for the spent fuel temporary storage. The SFB is designed to accommodate the fuel discharged during 8 years. After 6-7 years of operation, the spent fuel bundles are transferred to the on-site, naturally air cooled dry storage facility (IDSFS) for the spent fuel long term storage. The IDSFS is designated to provide safe, reliable and retrievable storage for spent fuel produced by the Cernavoda NPP Unit 1 and Unit 2 for a period of time of at least 50 years.

Table 1.1. General data on Cernavoda NPP Units

Reactor	Type	Gross Capacity MW(e)	First Criticality	Operating Status
Cernavoda-1	CANDU-6	706.5	16th of April 1996	in operation
Cernavoda-2	CANDU-6	706.5	6th of May 2007	in operation
Cernavoda-3	CANDU-6	720	-	under preservation, plans for resuming construction
Cernavoda-4	CANDU-6	720	-	under preservation, plans for resuming construction
Cernavoda-5	CANDU-6	-	-	under preservation

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

1.1.2.1 Description on main systems

Cernavoda NPP is the only nuclear power plant in Europe based on the CANDU (CANada Deuterium-Uranium) technology. Therefore a brief description of the main systems of the CANDU-6 plant is provided in the following.

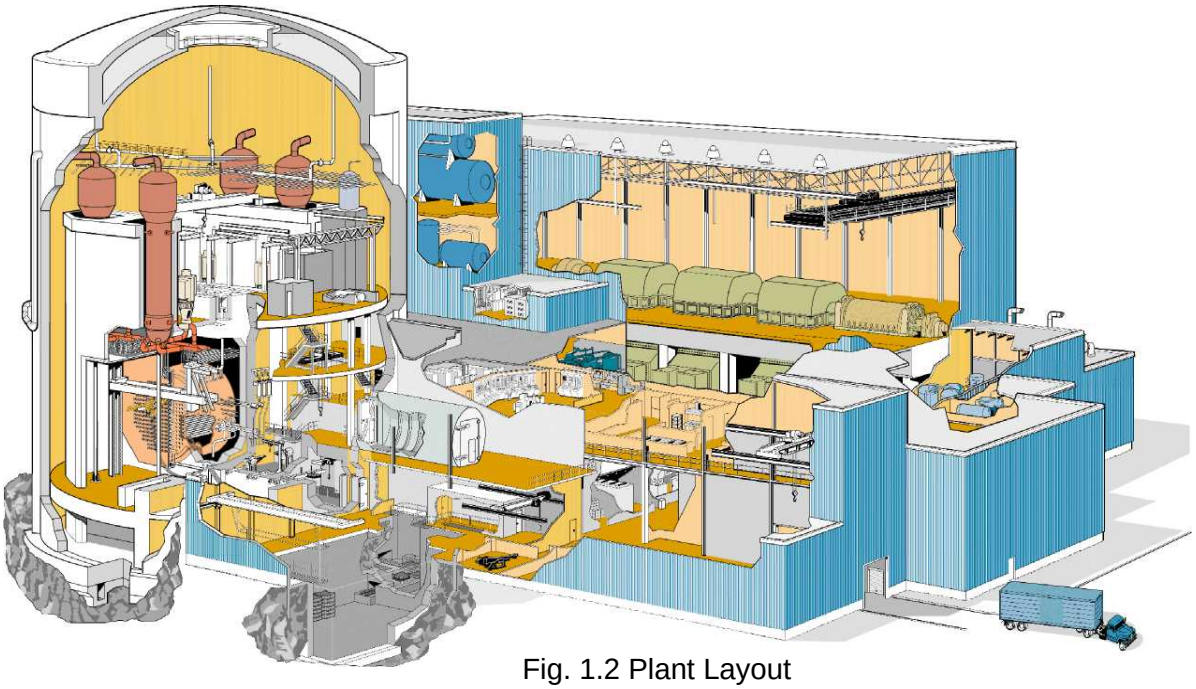

Fig. 1.2 Plant Layout

Reactor

The CANDU-6 reactor is fuelled with natural uranium fuel that is distributed among 380 fuel channels. Each six-meter-long fuel channel contains 12 fuel bundles. The reactor comprises a stainless steel horizontal cylinder, the calandria, closed at each end by end shields, which support the horizontal fuel channels that span the calandria, and provide personnel shielding. The calandria is housed in and supported by a light water-filled, steel lined concrete structure (the reactor vault) which provides thermal shielding. The calandria contains heavy water (D_2O) moderator at low temperature and pressure, reactivity control mechanisms, and 380 fuel channels.

The fuel channels are housed in a horizontal cylindrical tank (called a calandria vessel) that contains cool heavy water (D2O) moderator near atmospheric pressure. Fuelling machines connect to each fuel channel as necessary on both ends of the reactor to provide on-power refueling; this eliminates the need for refuelling outages. The on-power refueling system can also be used to remove a defective fuel bundle in the event that a fuel defect develops. CANDU reactors have systems to identify and locate defective fuel.

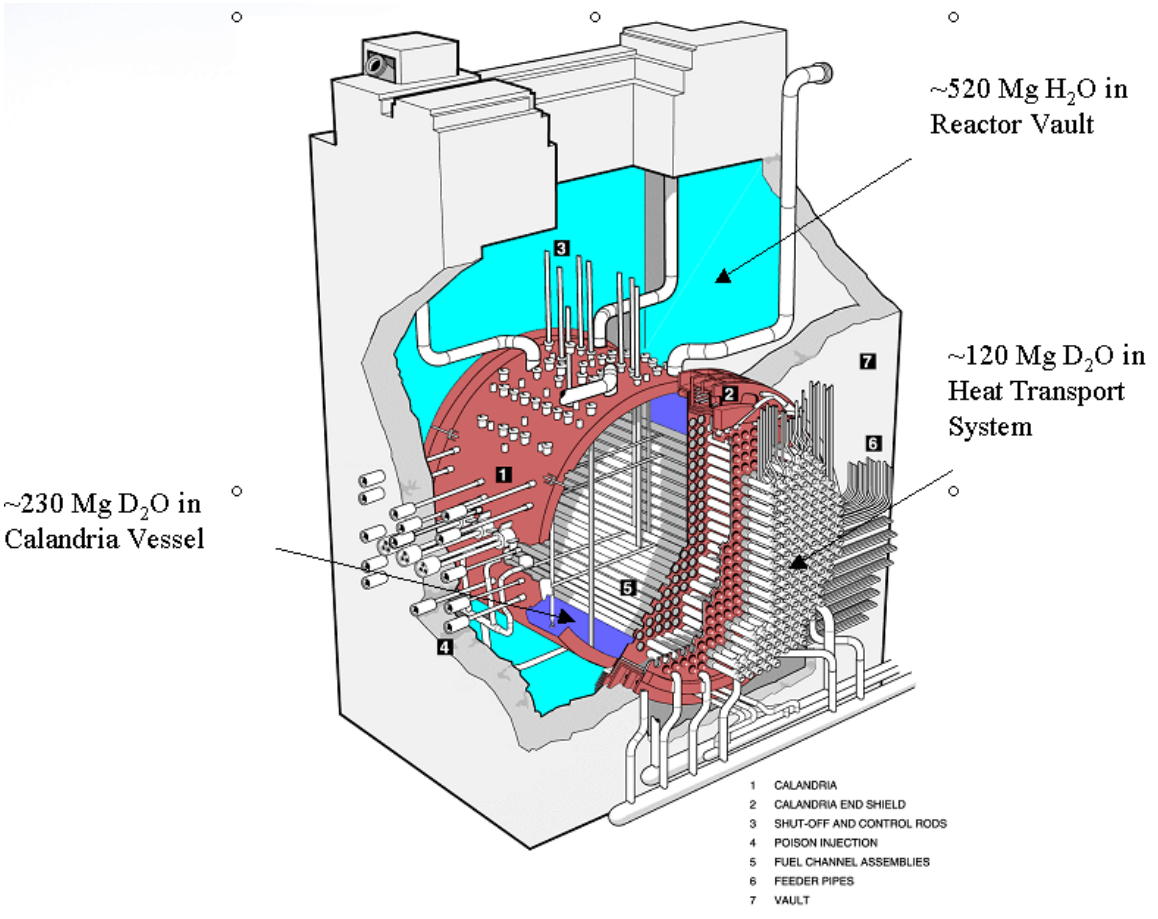

Fig. 1.3 Reactor Core Structure and Calandria Vault

Fuel Handling System

The fuel handling system refuels the reactor with new fuel bundles without interruption of normal reactor operation; it is designed to operate at all reactor power levels. The system also provides for the secure handling and temporary storage of new and irradiated fuel.

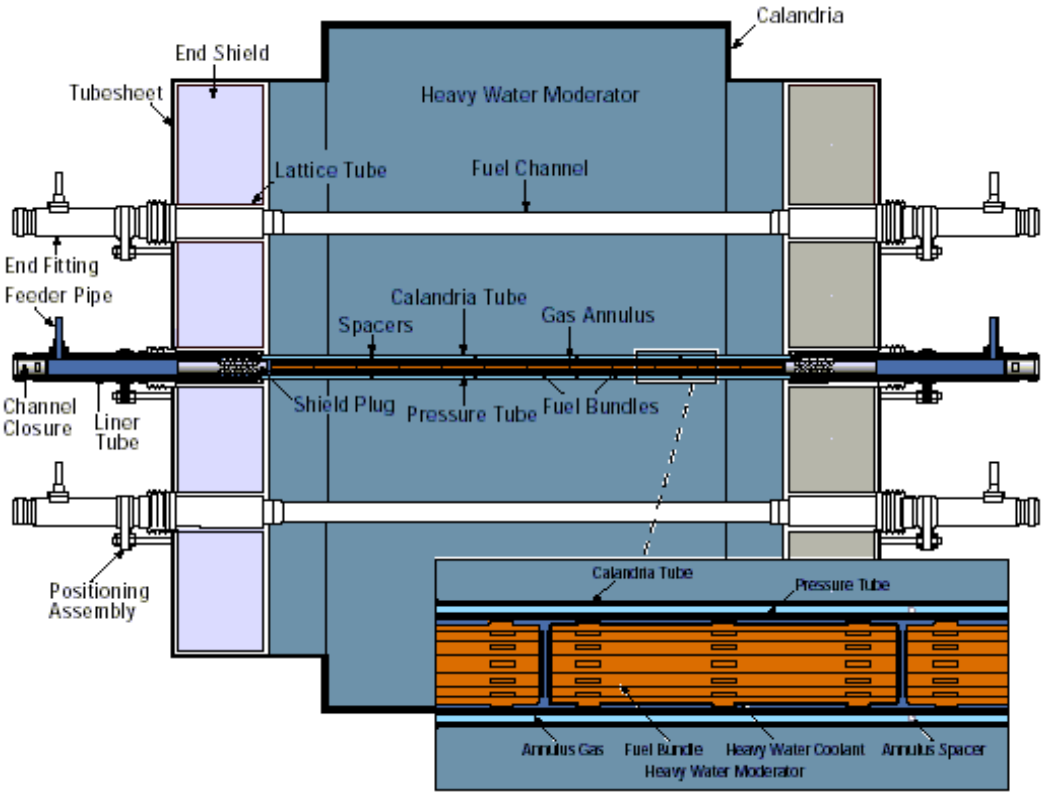

Fig. 1.4 Fuel Channels

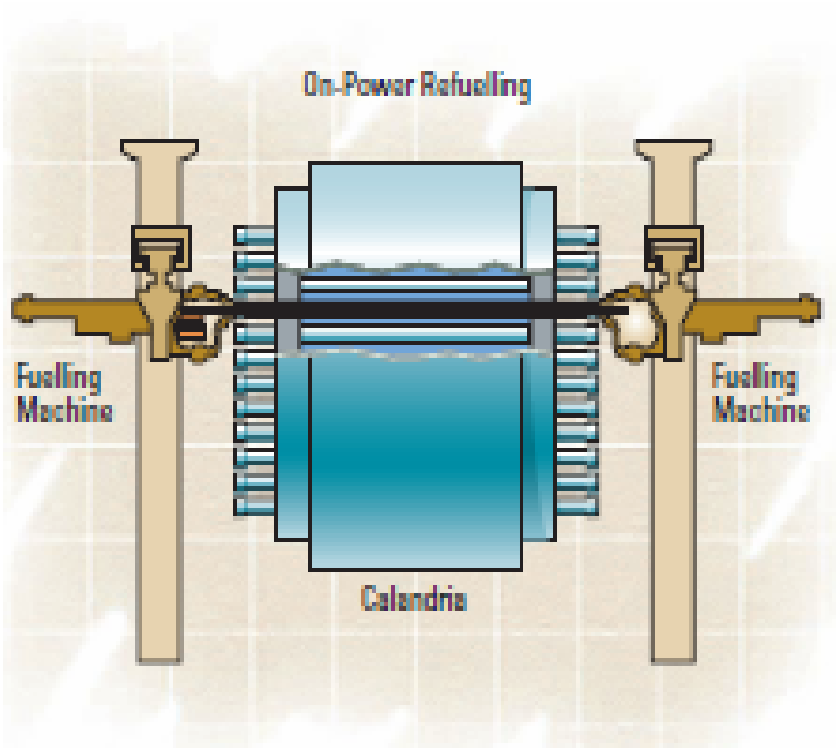

Fig. 1.5 Fuelling Machines

Heat Transport System

The heat transport system circulates pressurized heavy water coolant (D2O) through the reactor fuel channels to remove heat produced by fission in the uranium fuel. The heat is carried by the reactor coolant to the steam generators, where it is transferred to light water to produce steam. The coolant leaving the steam generators is returned to the inlet of the fuel channels.

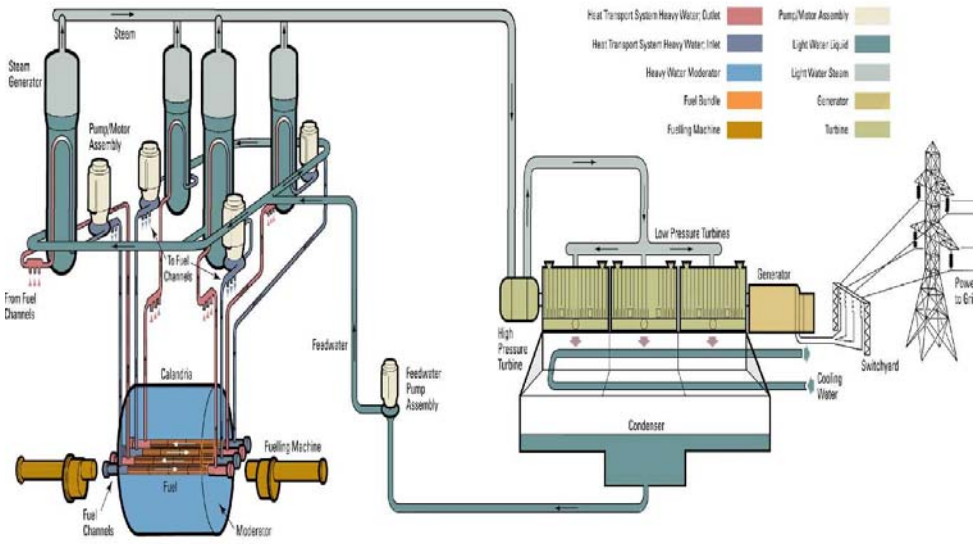

Fig. 1.6 Primary Heat Transport System and Balance of Plant

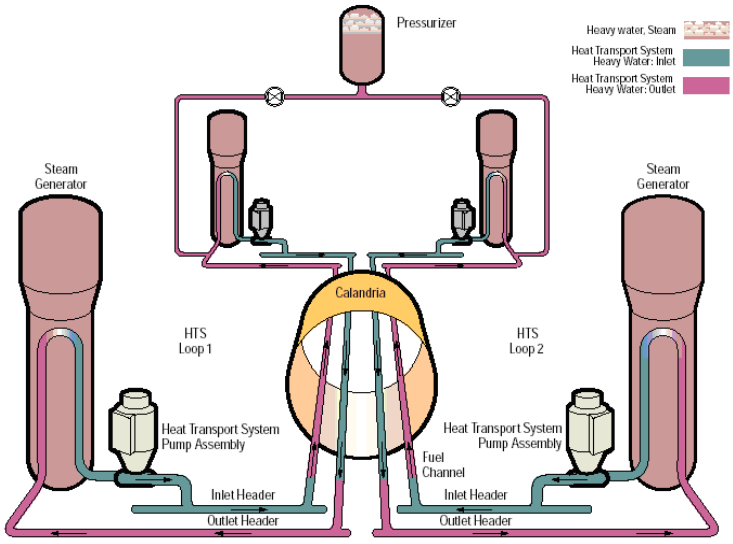

Fig. 1.7 Schematic of the Primary Heat Transport System

Moderator System

Neutrons produced by nuclear fission are moderated (slowed) by the D2O in the calandria. The moderator D2O is circulated through systems that cool and purify it, and control the concentrations of soluble neutron absorbers used for adjusting the reactivity.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss of coolant accident in the heat transport system coincident with a failure of emergency core cooling.

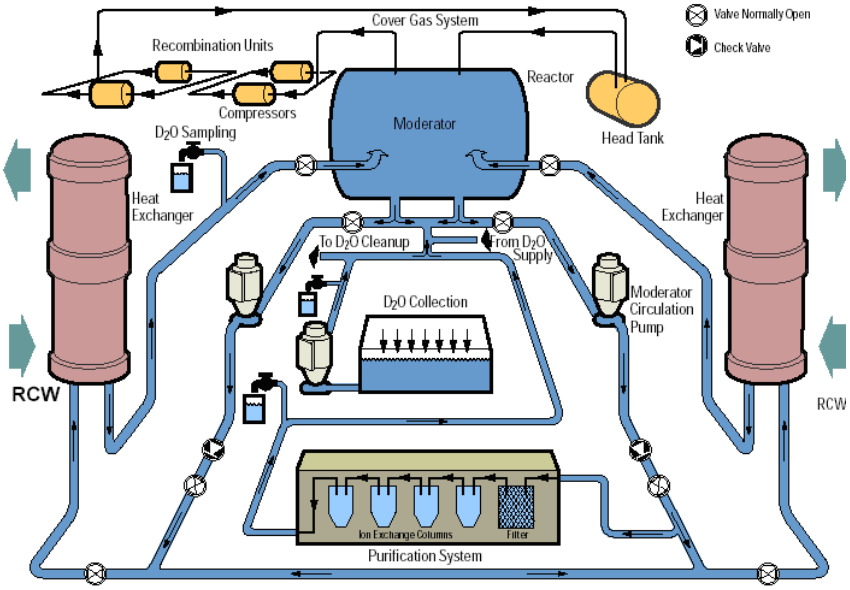

Fig. 1.8 Schematic of the Moderator System

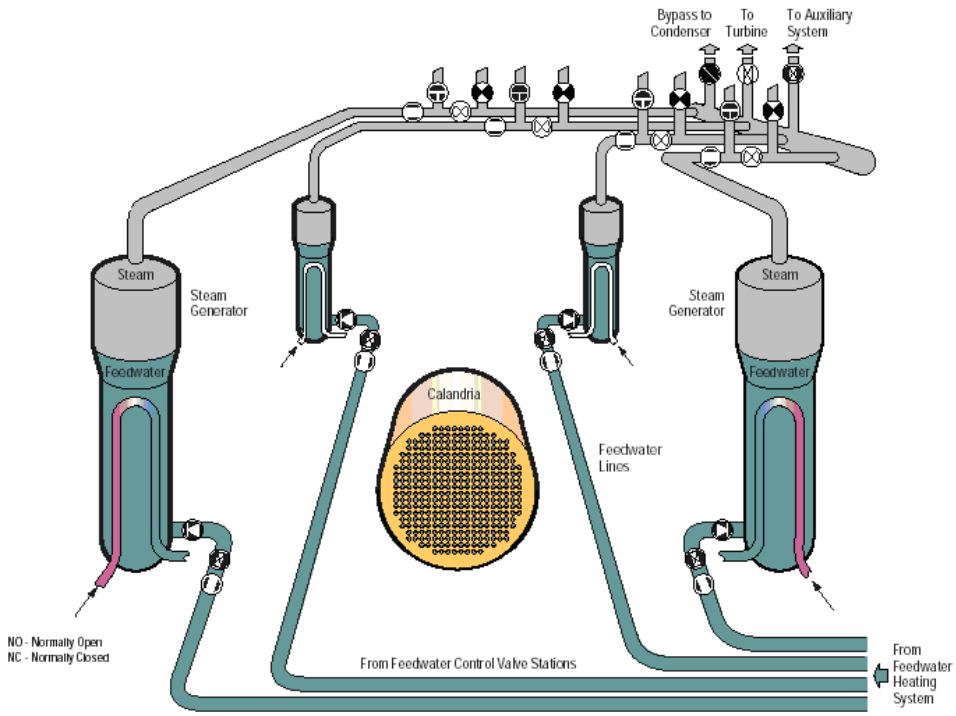

Fig. 1.9 Schematic of Secondary Heat Transport System

Feedwater and Steam Generator System

The steam generators transfer heat from the heavy water reactor coolant to light water (H_2O) to form steam, which drives the turbine generator. The low pressure steam exhausted by the low pressure turbine is condensed in the condensers by a flow of condenser cooling water. The feedwater system processes condensed steam from the condensers and returns it to the steam generators via pumps and a series of heaters.

Reactor Regulating System

This system controls reactor power within specific limits and makes sure that station load demands are met via two independent (master / slave) digital control computers (DCC). It also monitors and controls power distribution within the reactor core, to optimize fuel bundle and fuel channel power within their design specifications.

Safety Systems

Four seismically qualified special safety systems (Shutdown System No. One (SDS1), Shutdown System No. Two (SDS2), the Emergency Core Cooling (ECC) System, and the containment system) are provided to minimize and mitigate the impact of any postulated failure in the principal nuclear steam plant systems. Safety support systems provide services as required (electric power, cooling water, and compressed air) to the special safety systems.

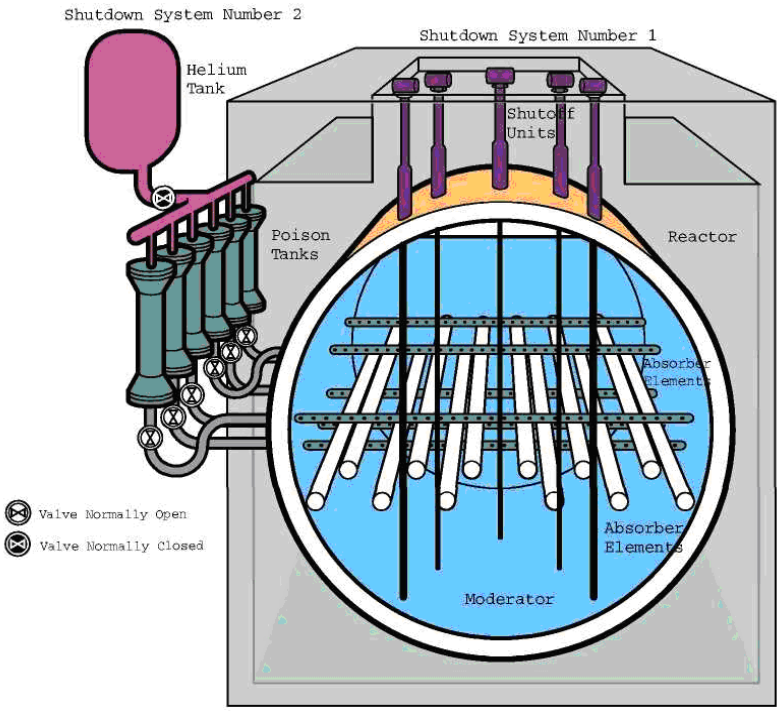

Fig. 1.10 Reactor Shutdown Systems

Reactor Shutdown Systems

There are two 'full capability' reactor shutdown systems, each able of shutting down the reactor during any postulated accident condition. The two shutdown systems are functionally and physically independent of each other; and from the reactor regulating system. Functional independence is provided by utilizing different shutdown principles: solid shutoff rods for System number 1, direct liquid poison injection into the moderator for System number 2. Physical independence of the shutdown systems is achieved by positioning the shutoff units vertically through the top of the reactor and the poison injection tubes horizontally through the sides of the reactor.

Emergency Core Cooling System

The Emergency Core Cooling (ECC) system is a special safety system, designed to provide light water to the PHT for fuel cooling and inventory makeup in the event the PHT system is breached causing a Loss of Coolant accident. The ECC performs no normal operating functions, it operates only to mitigate LOCA events.

The ECC is composed of three stages: High Pressure (HPECC), Medium Pressure (MPECC), and Low Pressure (LPECC) or Recovery Stage. The HPECC Stage uses gas pressure from the gas tank to inject water automatically into the reactor core from the two ECC water tanks located outside the reactor building (see Figure 1.11). The MPECC Stage automatically supplies water from the dousing tank, part of which is reserved for ECC, to the reactor core using the ECC recovery pumps. When this water supply is depleted the LPECC or Recovery Stage is initiated to recover water that has collected in the reactor building basement, and to pump it back into the reactor core via the ECC recovery pumps and ECC heat exchanger.

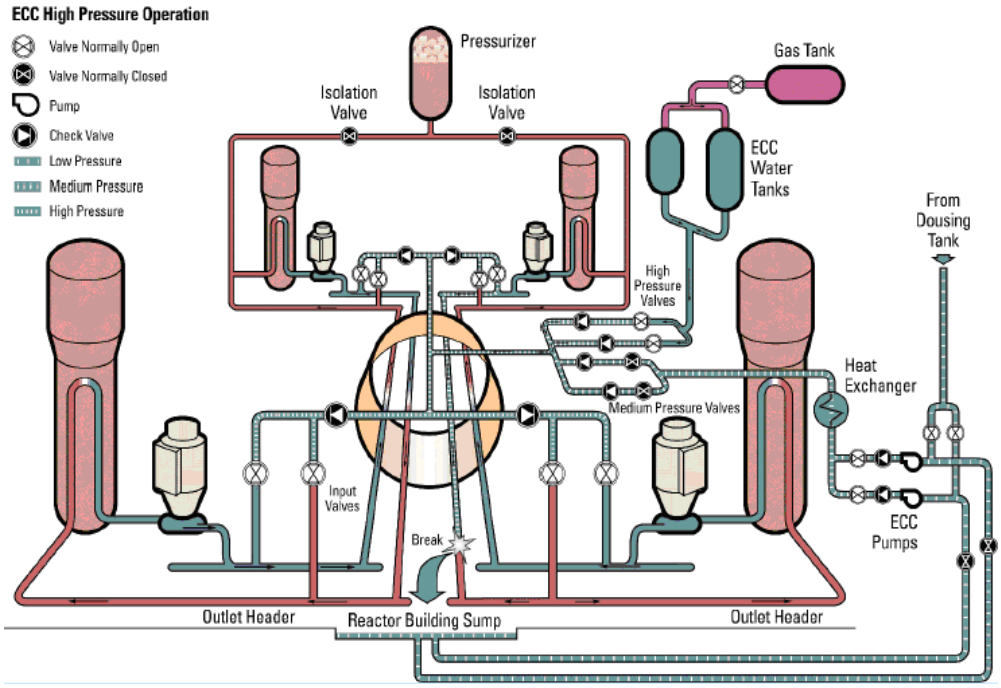

Fig. 1.11 Emergency Core Cooling System

Containment System

The containment comprises of the structures and systems that operate to provide a sealed envelope around the reactor systems if an accidental radioactivity release occurs from these systems. The structures and systems that form containment are:

• a lined, post-tensioned concrete containment structure

• an automatic dousing system

• air coolers

• a filtered air discharge system

• access airlocks

• an automatic containment isolation system.

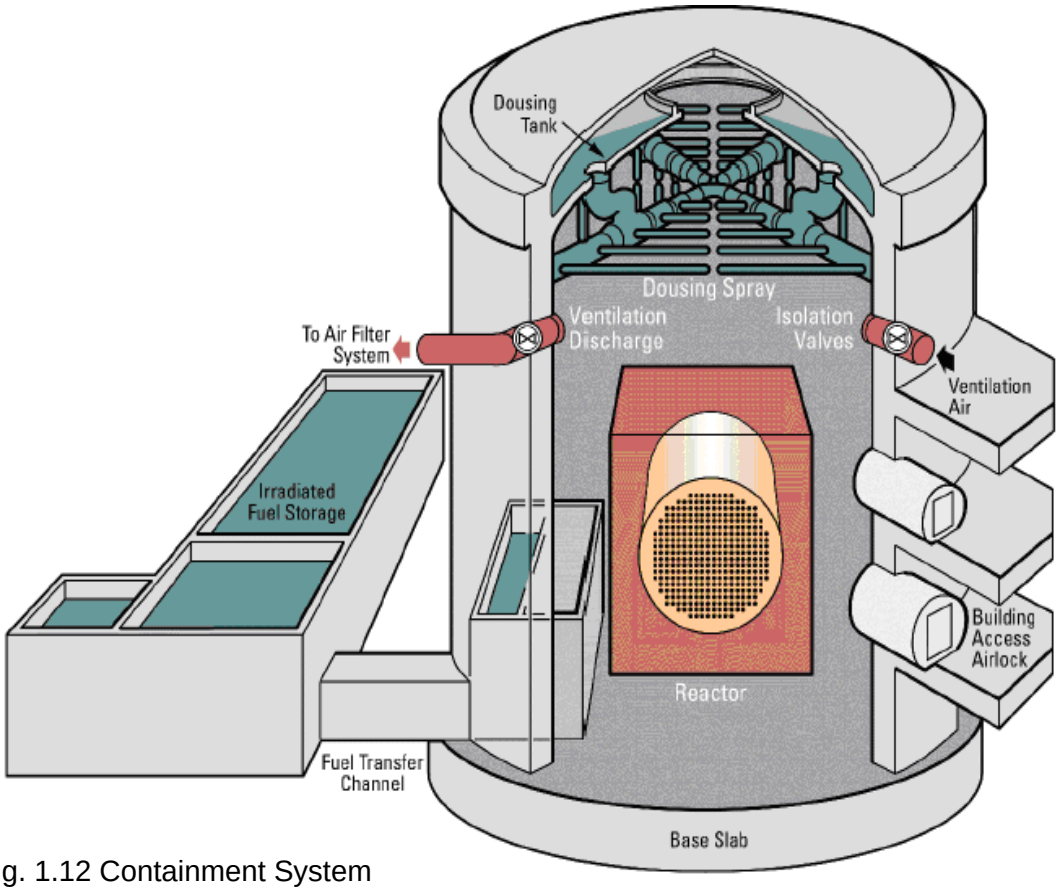

Fig. 1.12 Containment System

Electrical Power Systems for Cernavoda NPP

The power supply sources for the Cernavoda NPP Units are as follows:

- Redundant offsite sources, which provide electrical power required during startup and shutdown of the unit and can also supply power during normal operating conditions;
- The turbine generator (onsite), which provides electrical power required during normal operation;
- On site standby sources which provide the electrical power required in case of loss of the normal power supply: Class III Standby Diesel Generator (SDG), batteries, Emergency Diesel–generator (EPS, Emergency Power Supply).

The onsite power Distribution system is divided into redundant load groups (EVEN and ODD) so that the loss of any one group does not prevent the minimum safety function from being performed. Furthermore the onsite station service power supplies are classified as four classes that range from uninterruptible power to that which can be interrupted with limited and acceptable consequences, provided as follows:

- **Class I**: Uninterruptible direct current (dc) supplies for essential auxiliaries, control, protection and safety equipment. Batteries provide uninterruptible power for 8 hours.
- **Class II**: Uninterruptible alternating current (ac) supplies for essential auxiliaries, control, protection and safety equipment. Uninterruptible power is provided by batteries, through inverters or by Class III during unavailability of the inverters.
- **Class III**: Power supplies to the safety-related systems. Normal supply of class III distribution system is from Class IV via the service transformers, and it is backed-up by 100% redundant standby diesel generators with 100% redundancy. Any interruption of power is of a short duration (maximum 180 sec), which is necessary for start-up and loading of the standby diesel generators. Also, class III is the charging source to the class I batteries and back-up supply to Class II loads.
- **Class IV**: Normal alternating current supplies to auxiliaries and equipment, which can tolerate long duration interruptions without affecting nuclear safety, personnel or equipment safety. A complete loss or a loss of either odd or even division of Class IV power will initiate a reactor shutdown. Partial and total loss of Class IV power, including loss of offsite power are design basis events which do not pose any safety threat to the plant.

EPS: A completely independent, seismically qualified, emergency power supply (EPS) system designed to 100% redundancy and separation requirements is also provided to cope with common mode events, ensuring the safety functions are maintained. This system is intended for back-up supply supporting essential safety functions when all the others electrical supplies are unavailable or when the main control room is uninhabitable. A design change has been implemented which allows an electrical conection facility to the EPS buses for a mobile Diesel Generator. This change has been made as part of the action plan resulted from the analyses performed in response to the WANO SOER 01-2011.

Mobile diesel generators: Following the Fukushima accident, Cernavoda NPP procured two mobile diesel generators (one for each unit), to provide power if the EPS is not available. The capacity of each mobile diesel generator is almost equivalent to that provided by the design non-mobile EPS diesel generators. The mobile diesel generators have autonomy of 6 hours at full load without external support. The available fuel supplies on site will ensure more than 3 days of operation without external support.

Shutdown Cooling System

The purpose of the Shutdown Cooling System (SDCS) is to provide cooling to the fuel after a reactor shutdown. It is also designed to provide core cooling with the PHTS drained to the Reactor headers level for maintenance and internal inspections on primary side of the boilers and on the PHTS main circulating pumps.

The system consists of two separate shutdown cooling loops one at each end of the reactor. Each SDCS loop contains a pump, a heat exchanger, valves and piping connected between the Reactor Inlet Headers (RIH's) and Reactor Outlet Headers (ROH's) of each heat transport system loop.

Raw Service Water

The RSW system is a safety support system for the safety related systems. It ensures that an adequate heat sink for decay heat removal is available during normal plant operation, Loss of Class IV electrical power supply (LOCLIV), as well as during Loss of Coolant Accident (LOCA).

Connections have been installed to allow the addition of water to the primary side of the RSW/RCW heat exchangers using fire fighting pumpers and flexible conduits in the event of an emergency condition such as a very low level in the Danube River.

Recirculated Cooling Water

The Recirculated Cooling Water System (RCW) is a safety support system for the safety related systems and ensures that an adequate heat sink for decay heat removal is available during normal plant operation, Loss of Class IV electrical power supply (LOCLIV), as well as during Loss of Coolant Accident (LOCA).

The RCW system consists of a closed loop circulating demineralized water through the nuclear and Balance of Plant users in order to supply them with the required amount of cooling water under Class IV and Class III power conditions.

The heat received by RCW is rejected further to the Raw Service Water System (RSW).

Emergency Water Supply / Boiler Make-up Water

The Emergency Water Supply System (EWS) system ensures that an adequate heat sink for decay heat removal is available following a loss of the normal heat removal systems. Facilities are provided for a separate water supply to the steam generators

(SG), emergency core cooling (ECCS) heat exchangers and primary heat transport system (PHTS).

Examples of the accident modes that could lead to the loss of normal heat removal systems include loss of boiler feedwater (FWS), loss of service water (RSW), loss of Class IV and Class III power, as well as common mode failures such as earthquakes or fires.

EWS/BMW to Boilers

The steam generators can serve as a back-up long term heat sink for decay heat removal as long as the source of feedwater and circulation in the primary heat transport system can be maintained. Three Class IV powered main feedwater pumps and one Class III powered auxiliary feedwater pump provide the normal feedwater supply to the Steam Generators. This normal feedwater supply is backed up by two sources of water via the Emergency Water Supply system.

One is the gravity fed Boiler Make-up Water (BMW), The BMW supplies the Steam Generators with water from the Dousing tank via the ECCS downcomer, through BMW pneumatic valves that are automatically opened when Steam Generators pressure falls below certain value.

The other supply is from the EWS reservoir via the EWS pumps manually started by the operator and through the BMW pneumatic valves, to the Steam Generators.
The EWS is a low pressure system and in order to achieve flow path from BMW or EWS pumps, the secondary side of the Steam Generators must be depressurized. There are two possibilities to depressurize the secondary side: auto depressurisation and manual depressurisation. The auto depressurization is initiated automatically on an abnormal steam generator low level in two or more Steam Generators for at least twenty consecutive minutes, conditioned by low feedwater header pressure.

Connection facilities have been added to allow the addition of water to the EWS via fire pumpers, in the event the EWS pumps become unavailable.

EWS to PHTS

Full PHTS inventory may be necessary for natural circulation to be effective and thus allow decay heat removal via steam generators. The EWS flow is initially provided from the Dousing tank via the ECCS piping through motorised valves powered from EPS.

EWS to ECCS Heat Exchanger(s)

The EWS provides cooling water for ECCS heat exchangers to support the ECCS system availability during long term operation and following a seismic event.

Normal system segregation is achieved by check valves on the connection lines to RSW system in Unit 1 and to RCW system in Unit 2 design. Motorized valves direct flow from Emergency Water Supply pumps to the raw water side of ECCS heat exchangers and to drainage via the normal cooling water discharge.

Main Steam System

The Steam Generators are connected to the Main Steam header by four lines. Each Main Steam line is provided with one Atmospheric Steam Discharge Valve and four Main Steam Safety Valves. Each of the three lines from the Main Steam header to Condenser A / B / C is provided with one Condenser Steam Discharge Valve.

The safety functions of the Main Steam system are:

- Atmospheric Steam Discharge Valves (ASDVs):
  - Provide pressure control and normal cooldown of Steam Generators;
  - Provide a heat sink (under certain upset and emergency conditions).

- Main Steam Safety Valves (MSSVs):
  - Provide suitable overpressure protection of the Steam Generators;
  - Provide the ability to crash-cool the Steam Generators and the Heat Transport System.

- Condenser Steam Discharge Valves (CSDVs):
  - Provide pressure control and normal cooldown of the Steam Generators;
  - Discharge the main steam to the condenser during transient anticipated operational modes (turbine trip or load rejection), to avoid the opening of MSSVs.

Spent Fuel Bay

Spent (irradiated) fuel is removed from the reactor channels and is transferred to the Spent Fuel Bays where it is stored under water. The water provides shielding from radiation emitted by the fuel and also provides means to remove decay heat, which is given off by the fuel.

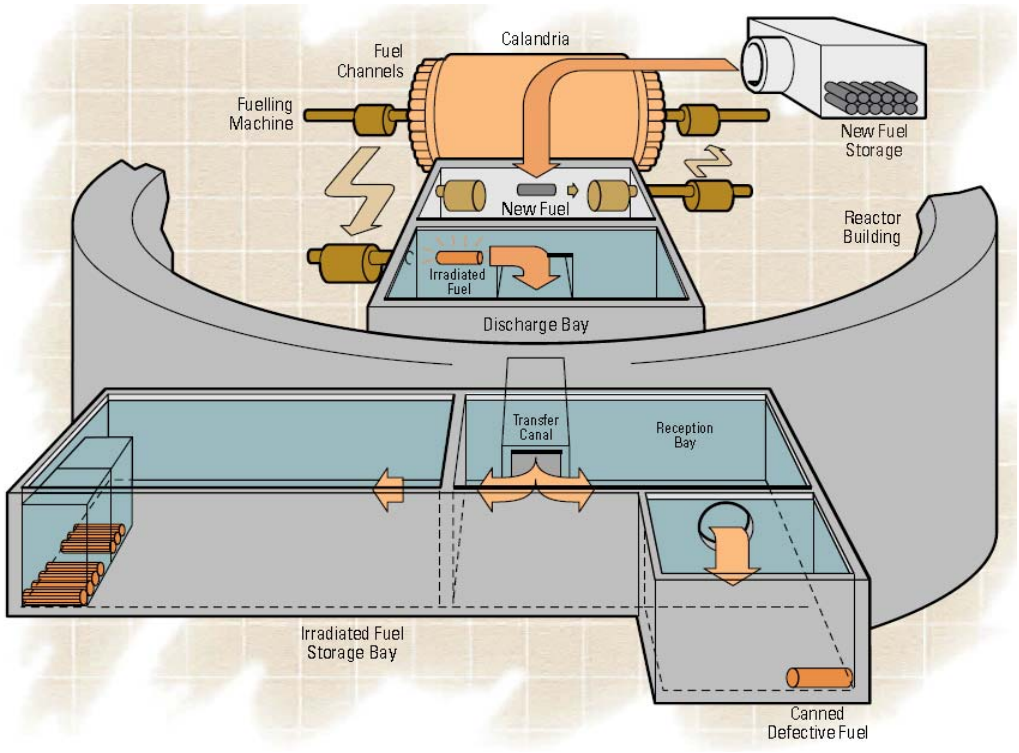

Fig. 1.13 Spent Fuel Route

The water in the Spent Fuel Bays is maintained in a clean and pure condition so that the spent fuel can be handled by station personnel with long handled tools working through down below 8 m of water.

The Spent Fuel Bay Cooling and Purification System are provided to remove the decay heat of the stored fuel and to remove dirt and radioactive particles (fission and corrosion products) from the Storage and the Auxiliary Bays. Although the system is designed to operate a common cooling and purification circuit for all Bays, it is normally operated as two isolated circuits to prevent contaminating the Storage Bay water in case of a failed fuel transfer from the Reactor. Skimmers are provided to clean the water surfaces of the Discharge, Reception, Failed Fuel and Storage Bays. In addition, the operation of a portable Underwater Vacuum Cleaning subsystem is provided.

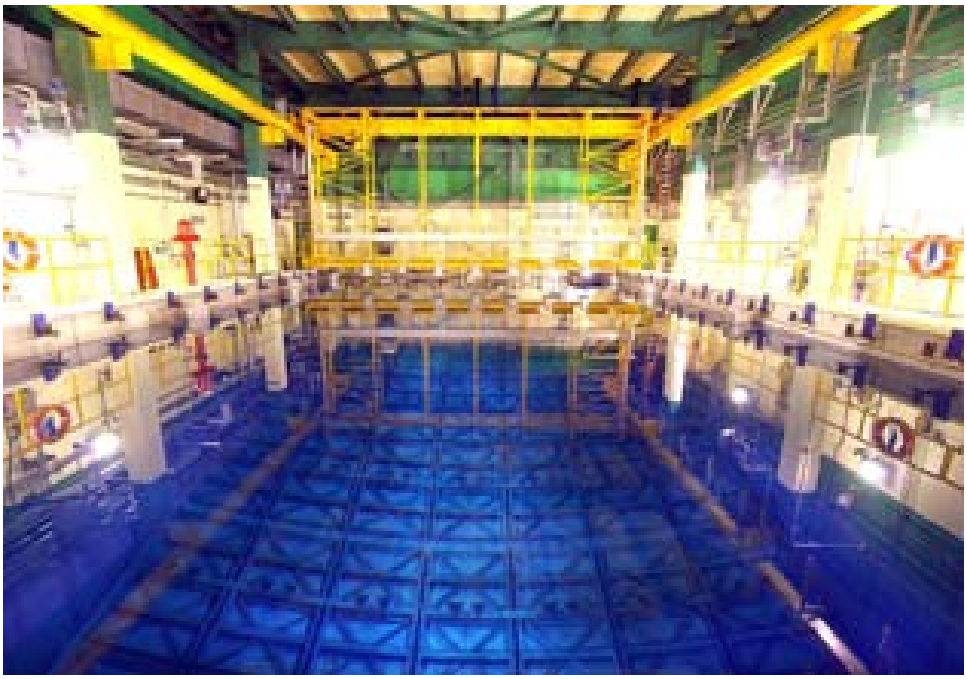

Fig. 1.14 Spent Fuel Bay

Intermediate Dry Spent Fuel Storage Facility

The facility will consist of 27 seismically qualified MACSTOR 200 modules. At present, 4 modules are built and in operation and 3 modules are in construction phase. Each MACSTOR-200 module is a parallelepiped structure made of reinforced concrete, which embeds 20 metallic Storage Cylinders positioned vertically. Once filled, the cylinder is covered with a reinforced concrete shield plug and a welded metallic cover plate, both of which are seal-welded to the upper flange of the storage cylinder.

Fig. 1.15 Intermediate Dry Spent Fuel Storage

1.1.2.2 Safety Philosophy and Defence in Depth

The safety philosophy of CANDU reactors, based upon the principle of defence-in-depth, employs redundancy (using at least two components or systems for a given function), diversity (using two physically or functionally different means for a given function), separation (using barriers and/or distance to separate components or systems for a given function), and protection (seismically and environmentally qualifying all safety systems, equipment, and structures).

An important aspect of implementing defence-in-depth in the NPP design is the provision of a series of physical barriers to confine radioactive material at specified locations. In CANDU design these barriers are the fuel matrix, the fuel sheath (clad), the Heat Transport System (HTS), and the Containment. An additional administrative barrier is the exclusion area boundary.

For design purposes, the safety related systems and structures have been defined as those which, by virtue of failure to perform the safety functions in accordance with the design intent, could cause the regulatory dose limits for the plant to be exceeded, in the absence of mitigating system action.

The safety related systems and structures of a CANDU NPP can be broadly categorised as follows:

- Preventative: Systems and structures that perform safety functions during the normal operation of the plant, to ensure that radioactive materials remain within their normal boundaries. These are systems and structures whose failure could cause a release exceeding the regulatory dose limits during normal plant operation, in the absence of further mitigating actions, or whose failure as a

consequence of an event could impair the safety functions of other safety related systems.

- Protective: systems and structures that perform safety functions to mitigate events caused by failure of the normally operating systems or by naturally occurring phenomena.

Some systems may perform both protective and preventative safety functions, and therefore may have more than one safety category designation.

The protective systems defined above are further identified as:

- Special Safety Systems, which include Shutdown System No. 1, Shutdown System No. 2, Emergency Core Cooling, and Containment.
- Safety Support Systems, which provide essential services needed for proper operation of the Special Safety Systems (e.g., electrical power, cooling water). These systems may have normal process functions as well.

The Special Safety Systems are always in standby during the normal operation of the plant and ready to mitigate the consequences of any serious process failure. They are totally independent from the process systems.

The Special Safety Systems and standby safety related systems have been physically separated by their assignation into two groups (Group 1 and Group 2) in order to provide adequate protection against common cause failures from events such as:
i) Turbine disintegration and resultant missiles;
ii) Fires that can lead to uninhabitable control centre, wide spread system damage, etc.;
iii) Aircraft crash;
iv) Failure of a common process e.g. Electrical Power Systems, Service Water System, etc.;
v) Common adverse environment e.g. extremes of temperature, pressure, humidity, radiation, toxic gases, etc.

In addition, within each group, there is separation between each the Special Safety Systems and between the channels of a system. The separation is achieved by the physical arrangement of equipment and of protective channels.

The essential safety functions that can be performed by either Group 1 or Group 2 are:

- reactor shutdown;
- fuel cooling;
- confinement of radioactivity;
- providing the operators with the alarms and indications required to assess the state of the unit and to take the necessary actions to mitigate the consequences of an accident.

Each group includes one SDS and either the ECCS or the Containment, because the analyses of the most severe cases, as presented in the Safety Report, assume one SDS system is unavailable and that either the ECCS or Containment is unavailable. As it is not possible to suffer more than those unavailabilities, it follows that the safety of the

facilities is ensured at all times. Component redundancy is built-in for the Special Safety Systems to ensure that the single failure criterion is satisfied. Special Safety Systems satisfy an unavailability target of 10^{-3} years/year, which effectively requires redundancy of all critical components.

The availability of these systems is verified during operation by regular safety system component tests. Specific requirements are applied to the triplicated instrument cables and the duplicated power and control cables for safety-related systems. The odd and even concept of on-site power distribution is applied to equipment, the raceway system and junction boxes, in order to maintain physical separation between the odd and even systems to achieve maximum reliability under normal and abnormal conditions

To satisfy reliability requirements to meet safety objectives, the Group 1 Electrical Power System is equipped with standby Diesel generators supplied with support services from Group 1 systems. The power distribution system is designed to prevent propagation of electrical faults to the Group 2 Emergency Power Supply System and vice-versa. The portions of the distribution system needed to supply electrical power from the Group 2 Emergency Power Supply System to components required for the earthquake events are seismically qualified.

For the purpose of safety assessment all major systems in CANDU reactors are categorised as “process systems” and “special safety systems”. All special safety systems are independent from all process systems and from each other.

The CANDU safety philosophy is based on the concept of single/dual failures. “Single failure” is a failure of any process system which is required for the normal operation of the plant and “dual failure” represents a combination of the single failure events and a simultaneous failure or impairment of one of the special safety systems. Coincident failure analysis is a systematic assessment of postulated dual failures.

Each postulated process failure is systematically coupled with a failure of one of the special safety systems. Loss of the shutdown systems is excluded from required dual failure sequences because the design includes two independent shutdown systems which are each capable of shutting down the reactor.

A distinguishing feature of dual failure assessment is that the analysis of CANDU 6 reactors must show that:

- coolable core geometry is retained, even if the ECCS were to be impaired;
- radioactive releases are adequately prevented, even if the containment system were to be impaired.

The deterministic analyses, including the description of initiating events, event sequences, acceptance criteria, methodology, results and interpretation are provided in Chapter 15 of the FSARs. Each of process systems failures (initiating events) considered were analysed for the case in which the ECCS and the containment subsystems are available, and also in combination with various failures/impairments to either ECCS or containment subsystems. Feedwater events and main steam line breaks were also analysed in combination with loss of Class IV power. Large LOCA and small LOCA events are analysed also in combination with loss of off-site power

and with impairments to either ECCS or containment system functions.

CANDU-6 is a proven design and sufficient information is publicly available on the general design features and on the CANDU safety philosophy and approach to prevention, mitigation and management of accidents. Therefore, this section only gives some examples of CANDU design features relevant for each of the levels of the defence in depth.

Prevention

- The reactor coolant pressure boundary is designed in accordance with ASME Section III - Class 1 requirements, as supplemented by Canadian Standards in the areas not covered by the ASME Code. The pressure tubes of the PHTS have "leak-before-break" characteristics. The plant is provided with extensive and sensitive leak detection systems. The presence of tritium in the PHTS makes the leak detection very efficient even for very small leaks.
- The on-line tritium in water detection system is used for revealing leaks to heat exchangers and to the S/G tubes.
- PHTS leaks open to Reactor Building atmosphere are revealed by the increasing of D_2O vapours recovery or by balance of heavy water into PHTS.
- The probability of occurrence of a sudden large-size break in a pressure tube is extremely low, in view of the following considerations:

i) as per design, the tube-wall thickness was selected such that leakage will precede tube rupture ("leak-before-break" concept);
ii) a leak of a pressure tube can be detected quickly (by means of the surveillance system analysing the gas contained in the annular space between pressure tubes and calandria tubes) thus allowing ample time for corrective action;
iii) the pressure tubes and their end-fittings can be inspected by means of ultrasonic techniques, thus providing an up-to-date overview of the state of the pressure tubes;
iv) although the pressure tubes are designed to serve for the entire life time of the plant, they can be replaced with relative ease, thus permitting early elimination of tubes showing any signs of faults.

- On-power refuelling implies that the power distribution reaches an equilibrium in less than a year from initial start-up, and remains virtually unchanged for the reactor's operating life. This greatly simplifies the analysis of core behaviour as a result of postulated accidents.
- CANDU fuel is highly reliable, being composed of natural uranium oxide. On-power refuelling allows for defective fuel to be detected, localised and removed from the core, reducing the contamination of the reactor coolant piping and simplifying maintenance.
- There is no criticality hazard in the handling or storage of the UO_2 fresh/spent fuel because it is not enriched and cannot be arranged in a critical array, except for in heavy water.

Control

- CANDU NPPs are provided with extensive instrumentation and control systems, capable of monitoring those variables and systems that can affect the fission process, the integrity of the reactor core, the PHTS pressure boundary and the containment. Most control functions for the reactor and the Balance of Plant, including automatic start-up, are performed by two identical, independent digital computers, each capable of complete station control. The two computers run simultaneously, one acting as instantaneous back-up to the other. Protection functions are, however, not performed by the digital process control computers but by Programmable Digital Controllers (PDCs), there being strict separation between control and protection systems.
- The Reactor Regulation System (RRS) is part of the fully computerised control system. This computerized control system is also responsible for boiler pressure and level control, unit power regulation, primary heat-transport pressure and inventory, and turbine run-up.
- The design philosophy for the RRS is to limit the maximum rate of reactivity additions to a value low enough to achieve safe control in all conditions. The neutronic flux spatial control system is designed to maintain stable control of the power distribution for any of the normal movements of other control devices such as adjuster rods or liquid zone controllers. The reactivity change due to refuelling is also adequately controlled by liquid zone controllers.
- The low excess reactivity of the CANDU core leads to relatively low reactivity worth of the control devices, limiting the potential severity of postulated loss-of-regulation accidents.
- Apart from the four systems employed by RRS, using control rods, adjuster rods, light water compartments and poison addition into the moderator region, two independent and diverse fast-shutdown systems are provided.
- Furthermore, the relatively open core lattice of the CANDU reactor permits complete separation between control and protection functions also for the neutron poison devices (i.e. the control rods used by RRS are the 4 mechanical control absorbers - MCA, while the SDS #1 uses 28 shutoff rods; poison addition to the moderator is done by RRS through the moderator liquid poison system, while the SDS #2 inserts poison from its own liquid injection shutdown units).
- To insure that localised overrating of the fuel does not occur an array of self-powered flux detectors is provided for application in the regional overpower protective (ROP) system. A separate array of detectors is provided for each of the two shutdown systems.
- The self-protection functions of the RRS (Stepback and Setback) are essential to ensure that station operation is within the boundaries assumed in the analyses. In the majority of event scenarios, the above mentioned self-protection functions can avoid reaching the trip set points of the Shutdown Systems (SDS#1 & SDS#2). The availability of the Reactor Regulating System (RRS) is absolutely required for maintaining the reactor in the critical state. Consequently, on a loss of RRS, the reactor is tripped immediately, with no attempt at re-start.

- Heavy-water neutron kinetics is slower by several orders of magnitude than light-water kinetics, this making the control easier because of the inherent kinetic behaviour of the delayed neutrons.
- Provision of passive heat sink after common mode events like loss of electrical power is ensured by thermosyphoning through the steam generators.
- The plant is provided with two separate control rooms in different locations, each with capability of shutting down and cooling the reactor to cold conditions, and providing continuous monitoring-of-the-plant information to the operating staff; this capability is still maintained in each control room even if total failure of all equipment in the other control room is assumed.

Protection

- The Safety Systems are fully automated, although they can be actuated also manually if required. Each system is independent of the others, employing its own sensors, logic, and actuators. Each system uses triplicated logic in two out of three logic configuration, (three sensor circuits, with two-out-of-three voting), with the ability to be tested on-line. Also, the fail-safe design principle has been implemented in the design of the Safety Systems.
- SDS#1 uses solid shutoff rods (stainless steel sheathed cadmium absorbers), dropping by gravity into the core, and is capable of shutting down the reactor for the entire spectrum of postulated initiating events. SDS#2 uses high-pressure liquid poison (gadolinium nitrate) injected into the (low-pressure) moderator, and is also capable of shutting down the reactor for the entire spectrum of postulated initiating events.
- Each SDS, acting alone, is capable of shutting down the reactor within less than 2 seconds and maintaining it subcritical under cold conditions, for all accident scenarios. In safety analysis, the two most effective of 28 shutoff units for SDS#1 are assumed unavailable. Likewise, one of six liquid poison injection nozzles for SDS#2 is assumed unavailable. Prompt criticality is not reached in accident conditions, as shown by analysis.
- The positive reactivity that would be introduced by loss of coolant accidents is well within the capability of mechanical and hydraulic shutdown systems.
- An important intrinsic safety feature of the CANDU reactor is that all neutron control devices are installed in the low-pressure moderator region, where, in case of a postulated LOCA due to a break in the headers or feeders, they are not subjected to potentially severe hydraulic forces. The moderator also provides a low-pressure environment for the control rods, eliminating the "rod-ejection" scenarios. In addition, the location of neutronics measurement devices in the moderator avoids subjecting this equipment to a hot, pressurised environment.
- Under any operating state, the CANDU 6 has a number of heat sinks. At full power, the main heat sink is provided by the four steam generators. The other heat sinks become more important when in a shutdown state or during abnormal events. This can be either through the Shutdown Cooling System (SDCS), the Emergency Water Supply System (EWS), or the Boiler Make-up water system (BMW).

- The steam generators with the Feed Water System remove reactor heat during normal plant operation. The Auxiliary Feedwater System and/or the Shutdown Cooling System removes the decay heat during plant shutdown. These systems belong to Group 1, they are designed to remove normal and decay heat and are powered by the normal (Class III, II and I) electrical power systems.

- The Shutdown Cooling System (SDCS) is designed for the full nominal operating pressure and temperature of the PHTS, so it can, if needed, be connected to the PHTS immediately following reactor shutdown, precluding the need for depressurisation after a loss of heat sink.

- Following a common mode event that may disable the above means of decay heat removal, a second independent means of decay heat removal is provided by the Emergency Water Supply (EWS) System which is powered independently by the Emergency Power Supply (EPS) System. Accordingly, the EWS and EPS Systems belong to Group 2.

- The EWS system has a function/feature known as the Boiler Makeup Water (BMW). This subsystem automatically feeds water under gravity to the secondary side of the boilers when they become depressurised following a loss of boiler feedwater. The source of BMW system is the water stored in the dousing tank.

- It should be noted that the Group 1 and Group 2 means of removing decay heat have the PHTS and the steam generators in common. Open path to atmosphere is ensured by Group 1 (ASDV) and Group 2 (MSSV) relief devices.

- The ECCS can maintain or re-establish core cooling by supplying coolant to all reactor headers. It consists of three phases: high-pressure water injection (used during the early stages of an event), medium pressure water supply from the containment building's dousing tank (used during the intermediate stages), and low-pressure water supply based on recovery from the building's sump. The ECCS is designed for LLOCA - 100% break of the largest pipe (reactor header). The discharge area is equal to twice the cross-sectional area of the pipe assumed to fail. Sensitivity analysis for the comparison of a 100% longitudinal break and a double ended guillotine break has shown very similar results, so longitudinal breaks have been modelled for all break sizes up to 100%.

- Considerations with regard to the ECCS:

i) the simple configuration of the individual fuel channels facilitates coolant delivery to all core locations;

ii) the correct performance of the ECCS does not constitute the final defence against core meltdown in case of LOCA; the accident analyses, supported by experiments, indicate that a LOCA combined with ECCS failure, though resulting in limited fuel damage (including partial melting of the cladding) and some deformation of the pressure and calandria tubes, does not result in fuel melting; the decay heat can be removed by conduction through the walls of the pressure and the calandria tubes into the moderator, and rejection by the moderator cooling system, which can remove than 4% of the total thermal power, enough to accept decay heat indefinitely.

- The Containment System forms a continuous, pressure-confining envelope around the reactor core and primary heat-transport system. In the CANDU 6 design it consists of a pre-stressed, post-tensioned concrete structure, an automatically-initiated dousing system, building coolers, automatic isolation system and a filtered

air discharge system. The containment system prevents releases of radioactivity to the public in the event of failure of the nuclear components of the heat transport system. The design basis event considered is any LOCA event concurrent with dousing failure. This event presents the highest potential in terms of peak pressure. However, the events related to steam systems breaks are also considered in terms of maintaining structural integrity of containment. The containment structure and all other parts of the containment boundary, are pressure and leakage tested before first criticality and leakage tested periodically thereafter. Another inherent safety characteristic of CANDU 6 plants is the low ratio of reactor thermal power to containment volume.

Mitigation

- The large-volume, low-pressure, low-temperature moderator surrounding the fuel channels acts as a heat sink in LLOCA + LOECC scenarios (which for CANDU are included in the design basis), rendering negligible the risk of fuel meltdown. The pressure tubes will sag and/or strain into contact with the calandria tube where further deformation will be arrested by the cooling of the moderator system.
- In a loss of heat sink or loss of flow event (such as a total station blackout), the reactor coolant will heat up and pressurise which can cause the pressure boundary to fail. In a CANDU reactor experiencing the same initiating event the fuel heat-up in the fuel channels will cause one of the many pressure tubes to rupture, depressurising the system by blowdown into the moderator well before boiler tube might fail and before a high pressure melt ejection can occur. The pressure tubes act like fuses in this instance. Failure of one channel is sufficient to limit widespread channel failures because it results in rapid heat transport system depressurisation and induced blow down cooling. Furthermore, heat transport system depressurisation occurs well before potential formation of molten core conditions, thereby assuring that high pressure melt ejection does not exist as a containment challenge in CANDU reactors.
- A large volume of light water surrounds the calandria vessel in the calandria vault. Thus, the design ensures a passive heat sink capability which, in many event sequences, would provide significant time delays in the progression of the accident. The calandria vault provides the third line of defence (after the ECC and the moderator) in cooling the reactor core during a severe accident. The large volume of water in the calandria vault has adequate thermal capacity to passively prevent calandria vessel failure. Water in the calandria vault can provide continued external cooling of the core debris relocated at the bottom of the calandria. During this process, the significant volume of water inside calandria vault cools the outer calandria vessel wall, maintaining the external cooling of the vessel. As long as calandria vessel is mostly submerged in water and the calandria vault water inventory can be maintained, it is expected that corium will be retained in the calandria vessel and accident progression arrested in-vessel. The externally cooled calandria vessel acts as a “core catcher” containing the core debris. Core disassembly and relocation take place only at low heat transport system (PHT) pressures and that melting of core materials is avoided until after the debris has relocated to the bottom of the calandria vessel.
- Overall, high volumes of water in the Heat Transport System, in the calandria vessel and in the calandria vault, notwithstanding the water volume from the dousing

tank, all ensure a CANDU-specific extensive heat sink capability that confers a slow progression of severe accidents

- Since the geometry of the CANDU core is near optimal from a reactivity standpoint, any rearrangement under severe accident conditions ensures shutdown. Therefore, re-criticality under is not a concern for a CANDU reactor.
- The bottom of the large calandria vessel provides a spreading and heat removal area for core debris following a severe core damage accident.

1.2. Significant differences between units

Unit 1 of Cernavoda NPP was commissioned in the period 1993 - 1996. The design installed and commissioned in Romania has incorporated most of the significant safety related design changes already made by other organisations operating CANDU-6 up to late 80's. Supplementary, during commissioning and in operation, a few other hundreds design changes were incorporated that originated from:

- operating experience from other CANDU-6 stations, especially Point Lepreau, Gentilly 2 and Wolsung;
- the probabilistic safety evaluations performed to verify the adequacy of the design.

Unit 2 was commissioned in the period 2005 - 2007. In the period for which the construction of Unit 2 was stopped, there have been many developments in the nuclear industry worldwide. For example, CANDU plants similar to Cernavoda 1 and 2 have been built and placed in service in South Korea (3 units at Wolsung) and in China (2 units at Qinshan). In addition, during this period, additional experience has been gained from the operation of CANDU plants worldwide. All the improvements resulting from the commissioning and operating experience were considered in the process of identification of the feasible design changes for Unit 2.

The reference design for Cernavoda Unit 2 was the design of Cernavoda Unit 1 as of July 1997. In addition, the Unit 2 design incorporated safety significant design modifications implemented at other CANDU 6 plants. These modifications can be categorised as follows:

- Design changes in response to the revision of codes, standards or regulatory requirement documents (in general, these changes can be categorised as safety improvements).
- Changes due to development of CANDU technology (in general, these changes result in improved performance or reliability of operation).
- Design changes to replace equipment where similar equipment used in Unit 1 was approaching obsolescence, and for which modernisation would result in improved availability of spare parts and maintenance.
- Other design improvements for enhancing systems or station performance.

Examples of safety improvements for Unit 2 are given below:

- Provision for manual initiation of Shutdown System number one (SDS#1) from the Secondary Control Area;
- Emergency Core Cooling System initiation on low sustained pressure in the heat transport system and automatic initiation of low pressure ECC;

- Provision for an alternate power supply for the LACs (Local Air Coolers) for mixing the Reactor Building atmosphere during a severe accident;
- Provision of hydrogen igniters to prevent hydrogen accumulation in the Reactor Building in case of severe accidents.

Cernavoda NPP has a feed-back program to assess and implement the design modifications and improvements from Unit 2 to Unit 1, in order to maintain an equivalent level of nuclear safety with Unit 2.

The potential for safety improvement from the implementation of Unit 2 design changes also in Unit 1 has been evaluated using a probabilistic safety assessment and it was found that the impact would not be significant (see Section 1.3).

An analysis of these modifications has been made for Unit 1, based on which the most beneficial design modifications from safety perspective have already been implemented. For the rest of the changes, the assessment of the reasonable practicability of their implementation will be completed in the framework of the first Periodic Safety Review (PSR) for Unit 1.

In the "stress test" report, design differences between the Cernavoda NPP Units are highlighted whenever they have an impact on the development of the scenarios analysed.

1.3. Use of PSA as part of the safety assessment

A level 1 probabilistic analysis has been performed for both Cernavoda NPP Units. The Level 1 PSA results for all operating stages, including external (seismic) and internal events, show a core damage frequency (CDF) of 3.3E-5 events/year for Unit 1 and 3E-5 events/year for Unit 2. These results are three times less than the internationally accepted target of 1E-4 event/years (IAEA 75-INSAG-3) for operating plants.

In order to support operational decisions with input from probabilistic assessment, Risk Monitor applications are developed based on the plant specific PSA models, providing on-line / off-line users with friendly interface. The Cernavoda NPP Risk Monitor is based on the Equipment Out Of Service (EOOS) application developed by EPRI, commonly used in nuclear power plants. The use of EOOS for risk-informed decision making was reviewed and approved by CNCAN.

For both Cernavoda Unit 1 and 2 the risk monitoring results show that the medium Annual Cumulative Recorded CDF is lower than the Average PSA Level 1 CDF.

The licensee has started actions to perform a Level 2 PSA for both Cernavoda Units 1 and 2. Meanwhile, the fault tree analyses for containment systems demonstrate that the unavailability of 1E-3 years/year, imposed by the design standards, is met.

The annual cumulative CDF together with the containment systems performance are monitored and reported quarterly to the Romanian nuclear regulatory authority. The results confirm that the probabilistic safety goals related to core damage and radioactive release frequency are met

CHAPTER 2 - EARTHQUAKES

2.1. Design basis

2.1.1. Earthquake against which the plants are designed

The seismologic data for the Cernavoda NPP site area have been determined by the National Research and Development Institute for Earth Physics (INCDFP) authorised by CNCAN for seismological studies of NPP sites. The maximum possible earthquakes have been defined in terms of both magnitude and intensity, their location in relation to the site, and characterization of the faults.

The determination of the site seismicity was made employing the seismic-tectonic method and the probabilistic method. The research was based on historical data on more than 100 important earthquakes that occurred between 984 and 1980. The data for pre-1940 earthquakes is based primarily on earthquake catalogues, while post-1940 also data from instruments was available. The seismic analyses considered the NPP site area on a radius of up to 300 km.

After the selection of the site for Cernavoda NPP and the establishment of the geologic and seismologic design data, the on-site investigations and observations were periodically reviewed and the seismologic design data were reconfirmed taking into consideration new information from recent seismic events (i.e. events occurred after 1980). The most recent update of the seismological catalogue was done in September 2011 by INCDFP and GeoHazard Ltd. Company to cover the earthquakes recorded in the period 2004 to 2011. All the analyses and all methods for analysis have taken account of the IAEA recommendations in the relevant safety guides.

The analysis of the geology, tectonics and seismologic data of the area indicates that the Cernavoda NPP site may be influenced by 3 tectonic provinces: Vrancea region, Balkanic region (Sabla-Dulovo) and Dobrogea region. The seismicity around the site is low, with no active faults.

The most important seismic source for the Cernavoda NPP site is represented by the Vrancea seismo-tectonic province. The maximum observed earthquake had a magnitude M = 7.5 (on Richter scale) and the value of the maximum possible magnitude estimated is M = 7.8.

Considering the seismic movement attenuation with epicenter distance for each seismo-tectonic province, the seismic intensity in Cernavoda area was determined for the maximum observed and for the maximum possible earthquake given by each seismo-tectonic province. The results of the analysis of these data show that, conservatively for Cernavoda area, the maximum observed earthquake may have the intensity I = VII degrees MSK-64, and the maximum potential earthquake I = VIII degrees MSK-64, with a peak ground acceleration (PGA) of 0.2 g.

In 2004 a probabilistic seismic hazard analysis (PSHA) was carried out for the Cernavoda NPP. The PSHA, carried out to support a seismic Probabilistic Safety Assessment (PSA), was performed by a Romanian team under the technical supervision of Paul C. Rizzo Associates, Inc. (RIZZO). Staff from the University of

Bucharest developed the seismo-tectonic and seismic source models, staff from the Technical University of Civil Engineering, Bucharest (UTCB) developed the ground motion models, and Stevenson and Associates-Bucharest (SAB) performed the probabilistic hazard computations.

The main conclusions from this study are:

- The Vrancea sub-crustal seismic source dominates the seismic hazard at the Cernavoda NPP.
- For the maximum historical recorded event, with a magnitude M = 7.5 the corresponding PGA at the NPP rock surface is 0.11g.
- For the maximum estimated event with a magnitude M = 7.80, the PGA at the NPP rock surface is 0.18g.

Compared with the design basis earthquake (DBE) PGA value of 0.2g, the above mentioned results confirm the adequacy of the design basis.

As a confirmation of design basis for earthquake, the mean seismic hazard curve calculated as part of PSHA shows the value of PGA for a probability of 1E-3 as being 0.2g, the same value as the one used as part of the initial design data input for Cernavoda NPP for a return period of 1000 years.

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

CANDU reactors are designed for safety with a philosophy to deal with design basis accidents (DBA) and DBE events with significant margins. Both diverse and redundant systems are implemented to ensure safe reactor shutdown and fuel integrity.

The safety-related systems, structures and components (SSCs) are divided into two groups as follows:

- Group 1: PHT, shutdown system one (SDS1), main and auxiliary moderator, steam and feedwater system, emergency core cooling (ECC) system, shutdown cooling (SDC), Local Air Coolers (LACs), Class I, II, III and IV power, and the main control room (MCR).
- Group 2: shutdown system two (SDS2), Containment Structure, Containment Isolation System, Dousing, Air locks, Hydrogen Control, EWS, EPS and the secondary control area (SCA).

The safe shutdown state for a CANDU reactor is achieved when all the following conditions are satisfied:

- Reactivity of the reactor is kept to a sufficient margin below criticality and maintained at that level for an indefinite period of time;
- Heat is removed from the fuel to prevent thermal design limits from being exceeded;
- The release of radioactive material is kept within the allowable limits by maintaining the physical barriers;
- The condition of the plant is monitored and the actions required to maintain

the above safety functions are performed.

DBE is a common cause event, which affects both Group 1 and Group 2. The design approach is to seismically qualify those SSCs (all of Group 2 and some of Group 1 are considered appropriate) necessary to carry out the essential safety functions following a DBE. The key Structures, Systems and Components (SSC) dedicated to post-DBE operations are: PHT, ECCS and MSSVs (in Steam System) in the Group 1 systems, and SDS2, Dousing system, EWS, EPS, SCA, Containment system (including containment isolation system and Air Locks), and the Instrumentation and Control in the Group 2.

As per CANDU Safety Design Guides, the separation of Group 1 and Group 2 SSC is implemented in the plant so that failure of any Group 1 Systems Structures and Components (SSCs) would not cause failure to the Group 2 SSCs. This has been confirmed through the seismic walk-downs in the reactor building, service building and turbine building performed to ensure that any structures or equipment that are not seismically qualified will not interfere with the seismically qualified SSCs.

2.1.2.1 Achievement of safe shutdown state following DBE

The success path identifying the key structures, systems and components to remain available after a seismic event has been defined for both units of the Cernavoda NPP and includes only the DBE qualified systems.

The design basis for the CANDU plant is to shut down the reactor if an earthquake causes a process parameter to exceed the limit requiring a shutdown, e.g., on low flow trip as a result of loss of Class IV power to the pumps in the PHT system. The CANDU plant has two shutdown systems: SDS1 and SDS2, seismically qualified.

The feedwater system removes reactor heat during normal plant operation. The auxiliary feedwater system and/or the Shutdown Cooling System remove the decay heat during planned plant outage. These systems are powered by the Class IV, III, II and I electrical power systems and belong to Group 1.

For the case when a DBE event could disable the Group 1 means of decay heat removal, there is a Group 2 independent means of decay heat removal, provided by the EWS system. The EWS is powered independently by the EPS system.

Both the Group 1 and Group 2 means of decay heat removal have the PHT and the steam generators (SGs) in common. The PHT and the SGs are seismically qualified, therefore no LOCA could occur as a result of the DBE. The cooling of the reactor after the DBE is by thermo-syphoning in the heat transfer loops with the SGs as the heat sink, when active heat sinks are not available.

The reactor and the PHT system are housed inside the containment building. Isolation of the containment following a DBE will provide for the control of a potential radioactive release.

A secondary control area (SCA) is incorporated into the plant design for the event that the DBE renders the MCR uninhabitable. Emergency operating procedures are

available in the SCA for operations from this location.

The SCA enables the monitoring, testing and manual operation of Group 2 safety-related systems; sufficient signals are re-transmitted from the SCA to the MCR so that the conditions of the Group 2 systems could be available in the MCR. These re-transmitted signals are buffered to ensure that failure of the cables or of the MCR would not result in failure of the Group 2 systems.

The following outlines the reasoning behind seismic qualification of systems that are important to the safety of the plant in the event of DBE:

(1) PHT is seismically qualified so that a LOCA will not be caused by a DBE;

(2) SDS1 and SDS2 are seismically qualified so that the shutdown capability is available by two independent means;

(3) EPS and EWS systems are seismically qualified so that power and water, respectively, are available to ensure decay heat removal following a DBE;

(4) SCA and the indications taken to the SCA are seismically qualified so that adequate monitoring capability is available following a DBE;

(5) The Containment System and all its structures and subsystems are seismically qualified to ensure that the containment is isolated and secured for radioactive release control;

(6) The ECC system is qualified for DBE for PHT inventory make-up and core cooling;

(7) The Spent Fuel Bay is seismically qualified to DBE together with the Service Building structure.

(8) The Intermediate Dry Spent Fuel Storage Facility is also seismically qualified to DBE.

The justification for the success in achieving safe shutdown state for DBE is summarized as follows:

(1) The reactor will be shut down by process trip parameters if any parameter exceeds its trip setpoint. Additionally, manual shutdown capability is available to the operator in both MCR and the SCA.

(2) PHT thermosyphoning is a proven effective mechanism for transporting the decay heat from the reactor core to the steam generators. This has been demonstrated by analysis and tests of natural circulation and verified through commissioning tests. Thermosyphoning is ensured by a full PHT with inventory make-up available from the HPECC water accumulators and/or EWS operation.

(3) Steam Generators as heat sink:

- MSSVs open to reject the energy into the environment for decay heat removal;

- Steam generators secondary side water make-up is provided promptly by gravitational flow of dousing water inventory in the event of a Main Steam Line Break (MSLB) and by EWS pump operation for the other seismic induced events.

(4) The integrity of the fuel is demonstrated by analysis for the DBE or other seismic events. Group 2 SSCs are capable of performing the essential functions required to

safely shut down the reactor and cool the fuel following the DBE. In addition, the ECCS has been seismically to provide make-up to the PHT following a DBE.

(5) The Containment System will perform as designed for containment isolation and contain any radioactivity, for protection of the public from radiation exposure. For design basis events, the LOCA + loss of emergency core cooling (LOECC) event presents the highest potential for fission product release. The safety analysis has shown that, for this limiting event, the public dose limits are not exceeded. By comparison, DBE events are much less severe transients with fuel cooling assured using the steam generators as the heat sink. Any radioactive release during and after the DBE would result in radiation exposures well below the public dose limits.

(6) Cernavoda Units 1 and 2 have control panels in the SCA with enough display and control instrumentation to allow the plant to be shut down, monitored and maintained in a safe shutdown state (including the assurance of adequate fuel cooling and the containment isolation).

2.1.2.2. Ensuring spent fuel integrity

The DBE qualified design of the Spent Fuel Bay (SFB) concrete structure will ensure no leakage from the bay during and after the DBE. The seismically qualified equipment, such as tables, overhead cranes, etc. in the bay area will not fail, and thus, will not cause damage to the spent fuel in the pool. The spent fuel trays, seismically qualified with secure anchoring for stability, ensure the spent fuel will not topple to cause damage.

As regards the provisions for adequate cooling of the spent fuel after DBE, since the normal pump cooling for the SFB is assumed to be unavailable due to loss of power, an emergency operating procedure is in place for establishing an alternate source of water make-up to the SFB.

A supply line is installed on site, outside the SFB area, for emergency water makeup to the bays by connection with the fire main hydrants of the fire protection system. If station fire water is not available, fire trucks loaded with water or mobile pumps taking water from various locations, such as the fore-bay, fire water tanks, de-mineralized water tanks or deep underground water wells, can provide the makeup water to the bays via hose connections.

After a DBE, the decay heat from the spent fuel will be transferred to the SFB pool water by natural convection, as long as the fuel is submerged. The evaporation of water from the bays to the atmosphere will slowly carry away some of the decay heat from the spent fuel. Emergency water make-up will be supplied to maintain normal water level in the bay.

A heat balance for the SFB shows that it will take about 3 days before the bay water starts to boil. On the assumption than no action is taken to replete the SFB inventory, there would be 15 days until first row of fuel bundles become uncovered.

Given this length of time, prompt operator action should be successful in restarting SFB pump operation to prevent boiling of the bay water. Therefore, the scenario of spent fuel dryout and damage is improbable.

As regards the Intermediate Dry Spent Fuel Storage Facility, this is a seismically robust structure made of reinforced concrete designed for a DBE of 0.3g. The seismic magin assessment performed as part of the "stress test" demonstrated that the structure can witstand an earthquake having a PGA > 0.4g without posing any threat with regard to spent fuel cooling capability or structural integrity.

2.1.2.3. Protection against indirect effects of the earthquake

The structures of the vital areas hosting the seismically qualified safety systems that are required to ensure a safe shutdown path are also seismically qualified for the DBE (0.2g).

In conformance with the "stress test" specifications, the indirect effects of a seismic event such as loss of external power supply and station blackout have been considered. The operation of the SSCs that ensure the safe shutdown path following DBE does not rely on external power supply.

As part of the response to the WANO SOER 2011-02 Recommendation following the Fukushima accident, walk-downs and inspections of important equipment needed to mitigate fire and flood events were performed at Cernavoda NPP to identify any equipment whose function could be lost during seismic events.

Based on the Fire Hazard Assessment performed during the early design stages of both Cernavoda NPP Units, measures have been taken to minimize the fire ignition sources and combustible material for the vital areas. These analyses have been recently reassessed in view of their use as an input for a complementary systematic analysis - Fire Probabilistic Safety Assessment performed for each unit during the operation stage. The studies confirm that the fire protection design measures are effective.

The inspections conducted in response the WANO SOER 2011-02 confirmed the design robustness and good material condition regarding the fire barrier preservation pertaining to the vital areas. The strategy and mitigation actions for fire suppression in the vital areas have been confirmed. The use of portable fire extinguishers located in the vicinity of seismically qualified equipment is efficient and can be used even by operating crew if a fire occurs at anytime following a safe shutdown. Combustible materials were found to be very limited, as per the design intent and the fire barriers were found appropriate, so no significant fires are expected to occur. Although the firefighting plans have been validated, some opportunities for improvement have been identified during the walk-downs. Both remedial actions on some degraded fire barriers and improvements on the fire fighting plans and manual firefighting equipment location have been implemented.

For seismic induced flooding or concurrent flooding and seismic events the essential safety functions of reactor shutdown, containment isolation, reactor cooling and monitoring of critical safety parameters are ensured from the Secondary Control Area (SCA) that is seismically qualified and physically separated from internal flooding sources. The potential for seismic induced flooding propagation and impact is localized and has no impact on the vital areas hosting the equipment qualified to

perform the essential safety functions after an earthquake. This statement is based on deterministic hazard analyses conducted in support of the Level 1 PSA for internal flooding.

For situations where the access to the site could be hindered due to extreme meteorological conditions, natural disasters (earthquakes, flooding, etc.) or other traffic restrictions, Cernavoda NPP has established a protocol with other authorities involved in emergency response, to ensure the provision of the necessary support in an emergency (transportation of Cernavoda personnel, fuel supplies, etc.). This aspect is addressed in Section 6.1.3 of the report.

2.1.3. Compliance of the plant with its current licensing basis

There are no known deviations from the seismic design / qualification licensing basis. The plant was designed, constructed and licensed to operate taking into account the requirements of nuclear codes and standards, as well as best practice procedures applicable to the seismic qualification of safety related SSCs.

Cernavoda NPP has programmes in place to preserve the original qualification and to provide evidence that all the qualified SSC meet the seismic qualification requirements for the entire station life. Arrangements to ensure compliance with the licensing basis involve generating, documenting and maintaining evidence that SSCs (including the seismic qualified SSCs) can perform their safety function during their installed service life. This is a continuous process, from the plant design to the end of service life, and plant ageing, modifications, repairs and refurbishment, equipment failures and replacements, and abnormal conditions are taken into account.

It should be mentioned that the mobile equipment and supplies that are planned to be used after an earthquake are stored in a seismically qualified location and are addressed as critical equipment by the plant surveillance and testing programs.

The following arrangements ensure that all the plant activities that could impact safety, design or licensing basis (including those that could have an impact on the seismic qualification of SSCs) are reviewed and assessed, the necessary actions are taken, and all documents affected are updated:

- Assessment of the impact of plant modifications (permanent or temporary) on the seismic qualification requirements of SSCs. This activity is managed through the Configuration Control Program and the procedures governing the modification process.
- Ensuring that the SSCs performance has been preserved by ongoing application of measures such as scheduled maintenance, testing and calibration and has been clearly documented (Maintenance Procedures, Seismic House Keeping Program).
- Seismic Monitoring and Inspections to confirm the actual condition of seismically qualified SSCs and their operability (Seismic Monitoring Program and Monitoring of Plant Buildings Behaviour for site and structures).
- Surveillance of seismic qualified SSCs and the analysis of the results to ensure that the SSCs are not affected by ageing process (Surveillance Program, PLiM).

- The assessment of SSCs failures and their impact on seismic qualifications, including the necessary corrective actions/ improvements to preserve seismic qualification (OPEX Program).
- New regulatory requirements that could have an impact on seismic qualification of SSCs are managed through the station process called "Register of Licensing Documentation and Tracking of CNCAN Action Items". This process, documented into an internal department procedure, ensures that all license conditions and other CNCAN specific requirements (including those that could have an impact on seismic qualification) are identified, analyzed, translated into specific plant procedures and tracked for their status until implementation.

Besides the activities carried out to ensure compliance with the current licensing basis, CNE Cernavoda is currently conducting its first Periodic Safety Review (PSR). The interim results show that activities and procedures addressing equipment maintenance as per the design basis conditions are in agreement with international good practices.

2.2. Evaluation of safety margins - assessment of the seismic margin

The first step in the assessment conducted in the framework of the "stress test" was a systematic review of the original analyses done as part of the Cernavoda NPP siting. The level of earthquake against which the plant is designed and the methodology used to evaluate the DBE have been found adequate as they are in compliance with the international standards. The validity of data in time and the conclusion on the adequacy of the design basis has been confirmed by the most recent studies done as part of Probabilistic Hazard Analyses in 2005, updated in August 2011 and independently confirmed by the Romanian National Institute for Earth Physiscs.

Furthermore, the provisions to protect the NPP against the DBE provided by the design have been assessed in terms of both key SSCs that are part of the safe shutdown path and operating provisions to prevent reactor core or SFB damage after an eartquake. No gaps have been identified in respect with the design basis. The licensee's process to ensure compliance is considered adequate by CNCAN.

The seismic margin analyses have been based on the well established methodology and on the reports elaborated as part of the Seismic PSHA performed in the period 2004 - 2005 for both Cernavoda NPP Units. The fragility analyses performed as part of these studies, that have been succesfully verified by an IAEA IPSART mission in 2005, have been consequently confirmed and complemented by rigourous plant walk-downs performed in May – July 2011 by a team built-up from plant designer (AECL) engineers and operating organisation engineers. The aim of the walkdowns was to confirm preservation of the original design in terms of seismic interaction. The evaluation of seismic margin of Cernavoda NPP was done based on the methodology illustrated in Fig. 2.1.

A review level earthquake (RLE) was established at a reasonably high level seismic ground motion, based on site seismicity and plant specific design features. The selected RLE has a return period of less than 10000 years, with a Ground Motion Response Spectrum (GMRS) with a PGA of 0.33g.

Based on a review of the DBE qualified systems required for performing the safety functions, a complete Safe Shutdown Equipment List (SSEL) for a seismic-induced station blackout has been compiled.

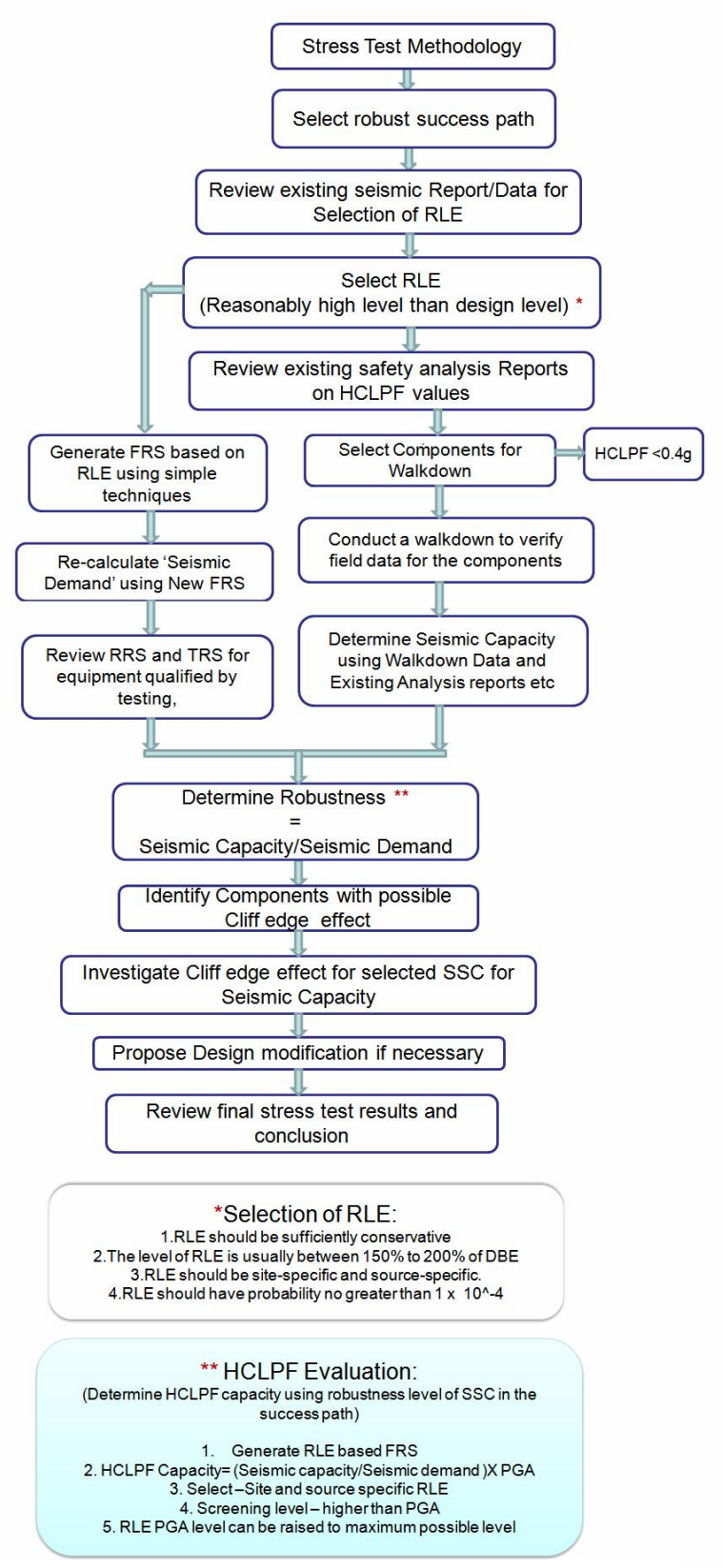

Fig. 2.1. Seismic Margin Evaluation Methodology

The SSCs in the path were evaluated against the RLE, making use of the HCLPF (High Confidence Low Probability of Failure) values from existing studies conducted in support of the PSHA (Probabilistic Seismic Hazard Analysis) in conjunction with walkdown observations. A screening level of 0.4 PGA, expressed in HCLPF capacity, was selected, to include additional margin above 0.33 g. The goal of the screening was to limit the number of items that needed to be assessed to calculate seismic fragilities.

The screening was based on walkdown observations and design basis review (as-built and as-operating conditions) supplemented by generic calculations that quantify the margin in the design basis. The robustness of the SSCs in the success path, as well as their protection against adverse effects of seismic interactions, including seismic initiated internal flood, have been verified through walk-downs. The data gathered from the walkdowns for each SSC in the Success Path has been reviewed and used in the determination of the seismic capacity. Only SSCs judged to be capable of sustaining less than 0.4 g have been subject to further examination.

Most of the information necessary for seismic assessment of the selected SSC was extracted from the documents and calculations prepared as part of licensing process for Cernavoda Unit 1 and Cernavoda Unit 2. Wherever such information was not available for Cernavoda Unit 1 or Cernavoda Unit 2, the corresponding information of the same or similar SSCs from other CANDU 6 projects was used in conjunction with engineering judgment.

In order to determine the seismic demand, the Floor Response Spectra (FRS) were chosen corresponding to various locations where the SSEL equipment is located. However, the FRS are based on 0.33 ZPGA and for the stress test, the values were pro-rated to assess the seismic capacity of SSC for the targeted value of 0.4 g.

The original design FRS defined for a DBE (0.2g) has been compared with the FRS pro-rated from 0.33g to 0.4g. The comparison shows that the seismic responses based on 0.4 g PGA, generated using the FRS for 0.33g, are almost equivalent to the FRS based on DBE of 0.2g GRS in terms of magnitudes and shapes. Therefore, although the SSCs in Cernavoda NPP units were seismically qualified using the original FRS based on DBE, their seismic qualification can be considered applicable also for a 0.4g PGA. This means that the seismic demand at 0.4g PGA remains very close to the seismic demand at design level.

As regards the investigation of seismic cliff-edge effects, in seismic structural dynamics, a cliff-edge effect is referred to as an abrupt catastrophic collapse of a structure without warning. Three such potential effects have been identified and analyses have been performed for the stability of the long vertical column support of the steam generator, the skirt support with openings of the vertical HPECC water tank, and the hanger rods of the dousing platform. It was determined that no cliff-edge effect would occur for PGA up to 0.4g (additional margins exist beyond this value).

The seismic margin assessment shows that in comparison with the original design basis earthquake of 0.2g, which has a frequency of 1E-3 events/year, all SSCs which are part of the safe shutdown path after an earthquake would continue to perform their safety function up to 0.4g, which has a frequency of 5E-5 events/year. This margin is

considered adequate as it meets the safety goals applied internationally for new NPPs.

2.2.1. - 2. Range of earthquake leading to severe fuel damage or loss of containment integrity

The plant seismic capacity was assessed for 0.4g. Up to this value, all the fundamental safety functions can be maintained. Since this result shows that a significant margin exists and given the estimation that the maximum credible earthquake would not yield a PGA equal to or greater than 0.4g, assessments for higher PGA values are not deemed necessary or relevant.

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

The potential of Cernavoda NPP units flooding induced by an earthquake exceeding the DBE has been analysed by considering all the failure mechanisms consisting of failure of dams and other hydrological or civil structures collapsing and the tsunamigenic potential of a Black Sea originating earthquake. The results of these analyses show that the effect of these failure mechanisms has physically no potential for seismically induced flooding of the Cernavoda site.

The potential for seismic induced internal plant flooding was also analysed and it was concluded that this does not pose a threat to the equipment qualified to perform the essential safety functions after an earthquake.

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

The seismic walk-downs and subsequent seismic robustness analyses done as part of the seismic margin assessment have not revealed a need for any safety significant design change. However, several recommendations resulted from these inspections, which have been considered by the licensee as part of the regular plant seismic housekeeping program.

CHAPTER 3 - FLOODING

This chapter provides the status of Cernavoda NPP design and operation capability to cope with external flooding events.

3.1. Design basis

3.1.1. Flooding against which the plant is designed

3.1.1.1. Initial assessment performed at siting stage

Cernavoda NPP site is located adjacent to the Danube River that is providing required cooling water flow. The site is 60 km away from the Black Sea coast.

The site is bordered on the Northeast by Cismelei Valley and on the Southeast by a bypass channel of Danube-Black Sea Channel (DBSC). Cooling water for the plant is taken from DBSC through a Bypass Channel, Intake Channel and Distribution Basin.

At the time of the selection of Cernavoda site, it was assumed that two future dams would be built on the Danube River, one upstream of Cernavoda at Calarasi and one downstream at Macin. The supporting studies carried out at that time analysed the different regimes to determine the maximum (flood) water level of the dam accumulation lake, and the extreme case of the upstream dam breaking while the downstream dam holds. The elevation of +16.00 mBSL for Cernavoda NPP site was selected with due consideration of this extreme postulated failure mode. It was later decided not to build the two dams envisaged at the time Cernavoda NPP site was selected.

Based on the original study, since the dams were not built, the maximum water level for the return period of 1 in 10000 years, chosen as the design basis flood (DBF) for Cernavoda NPP, is of +14.13 mBSL. However, the Cernavoda site platform elevation of +16.00 mBSL was not changed.

The top elevation +13.50 mBSL of Cernavoda water lock is below DBF level (+14.13 mBSL) and well below the Cernavoda NPP platform elevation (+16.00 mBSL). This means that there is an unobstructed path for water to flow towards the Black Sea and also to the surrounding Cernavoda territory lowlands, which are at +10.00 mBSL elevation. This was not taken into account in establishing the DBF of +14.13 mBSL, therefore the DBF value is conservative.

The evaluation of the maximum water level also considered the severe failure of the Portile de Fier hydro-electrical plant located 600 km upstream of Cernavoda, resulting in a high-flood wave. The results of the evaluation showed that the impact on the Cernavoda site would be negligible and within the normal fluctuation of the Danube River level.

Various elevations (expressed in metres Baltic Sea Level – mBSL) comprising the design basis related to the plant and main watercourse near Cernavoda NPP are marked in Fig. 3.1.

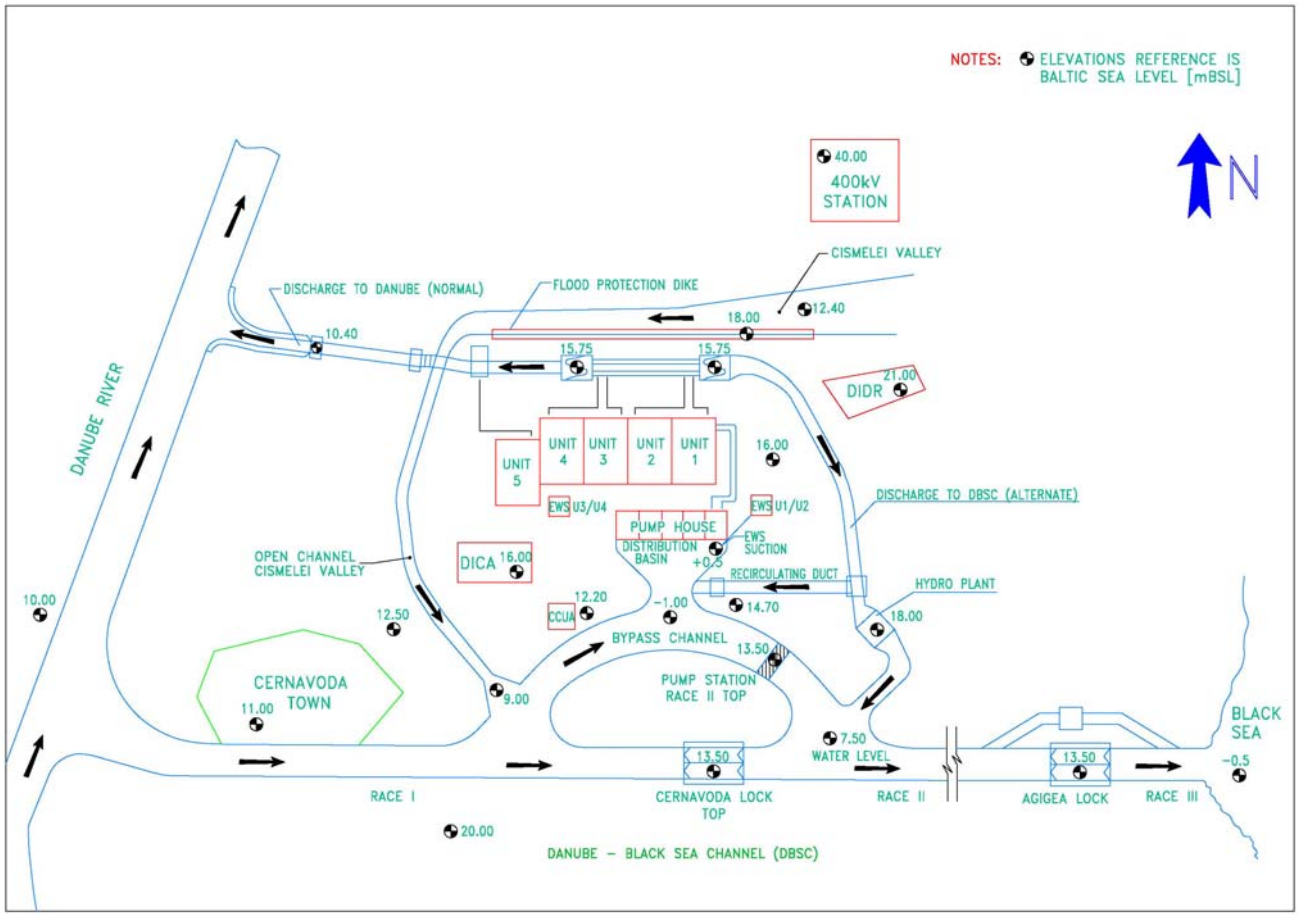

Fig. 3.1 Cernavoda NPP site elevations

3.1.1.2. Reassessment of the design basis flood

The assessment conducted in the framework of the "stress test" started with a systematic review of the original analyses done for Cernavoda NPP at the siting stage. The sources of flooding have been identified as Danube River, Heavy Rains and combination of these.

Potential external flooding sources considered by design for Cernavoda NPP site are the following:

- Extreme Danube River water flow / level;
- Extreme local rainfalls.

Danube Maximum Flood levels

The original estimations of the Danube River maximum annual level at Cernavoda, used in the establishment of the DBF, are presented in the table below:

Table 3.1 Cernavoda maximum Danube River water level

Probability of occurrence (%)	**1**	**0.1**	**0.01**
Annual Max. level [mBSL]	12.10	13.03	13.75

The DBF was calculated by adding a correction factor to the level corresponding to 0.01% probability of occurrence. This resulted in the DBF value of +14.13 mBSL.

In addition to the original design studies, two studies were performed by national institutes GeoEcoMar (2005) and INHGA (2008) in order to review and derive extreme water flow and level with 1%, 0.1% and 0.01% probability of occurrence making use of statistical data recorded during 1940 – 2003 time interval and empirical functions. Another study, more recent, has been performed in 2011 by INCDDD national institute. The results from these studies related to water level are similar to each other, although there are some differences explained by using different methodologies and set of statistical input data. The results are presented in the table 3.2:

Table 3.2 Danube River calculated extreme water levels at Cernavoda

Probability of occurrence (%)	**1**	**0.1**	**0.01**
GeoEcoMar study [mBSL]	11.64	12.37	12.99
INHGA study [mBSL]	11.83	12.73	13.53
INCDDD study [mBSL]	11.93	12.64	13.31

The results obtained present even lower values for the maximum annual water level showing that the originally calculated values are conservative and confirming the adequacy of flood design requirements for Cernavoda NPP site, established based on previous performed studies.

The original DBF calculation has been therefore reconfirmed by more recent data and studies, the latest one having been performed in September 2011, using the modern tool of Digital Topographic Model (DTM) to create the external flooding hazard map for Cernavoda NPP site and adjacent area.

The DTM used as input LIDAR scan data of the Cernavoda NPP site and surrounding territory. Subsequently, a specific hydraulic model was developed based on the methodology used in the European project "Danube Flood Risk - Stakeholder Oriented Flood Risk Assessment for the Danube Floodplains" within the South East Europe Transnational Cooperation Program (SEE Program).

The results of this study confirmed the validity in time of the original design topographic measurements and concluded that, in comparison with the design basis flood (DBF) corresponding to +14.13 mBSL, taking into account also the ground floor elevation for the plant buildings, a margin of +2.11 m exists. The adequacy of this margin is supported by the calculations showing that the Danube flow required to challenge this margin is not physically achievable.

The methodology used to derive estimated frequency (return period) of flow values, that could be greater than the recorded values in the set of statistical analysed data, considers a regression function applied to the set of recorded flow data and logarithmic distribution for the return period. Uncertainty evaluation with 95% confidence for the maximum flow and corresponding water level was considered in the results.

The results of estimated maximum flow and level of the Danube River at Cernavoda, based on 1971-2010 set of statistical data, are presented in Table 3.3 below. Data before 1971 was not used in this study because the construction of dikes along the Danube River for agricultural developments was completed in 1975. These new dikes remodeled Danube borders and bottom shape with direct influence on flow and level.

Table 3.3 Estimated return period of maximum Danube River flow and level at Cernavoda

Return period (years)	Flow (m^3/s)	Elevation [mBSL]	Notes
100	7385	11.93	
1 000	8690	12.64	
10 000	10113	13.31	
324 000	12201	14.13	Design Basis Flood
1 000 000	12878	14.37	
16 000 000 000	18688	16.00	Cernavoda platform
140 000 000 000	19990	16.30	

The flooding hazard maps were developed for each estimated return period in years and corresponding flooding parameters flow and level. The hazard maps have shown that the critical plant structures are above the water level. The dry spent fuel storage modules are also not affected by flood.

According to the flooding hazard maps, the updated estimations of the Danube River maximum water flow and level at Cernavoda are in accordance with the previous estimated maximum values, confirming adequacy of the site design in relation with external flooding events.

Huge lowlands areas are available at both upstream and downstream Cernavoda site location to discharge extreme floods, preventing Cernavoda site from flooding. These results combined with the results presented in Table 3.2 demonstrate that the Danube water flow higher than 20,000 m^3/s at Cernavoda, required for flooding the site, is not physically achievable.

Additional, one direct and unobstructed flowpath from the Danube River to the Black Sea is available for water levels greater than +13.50 mBSL, that is the top elevation of the Cernavoda and Agigea water locks. This pathway was not considered in development of the Cernavoda site flooding hazard map.

In conclusion, the Cernavoda NPP design basis flood DBF of +14.13 mBSL allows for significant safety margin against external flooding, as the site platform elevation is +16.00 mBSL.

Flooding due to rainfall on the Cernavoda site platform

The effect of rainfall inside the Cernavoda NPP site perimeter was calculated according to the national regulations for the frequencies, intensities and duration for

the maximum rainfall.

The extreme meteorological characteristics were based on statistical analysis performed as per the methodology specified in national standards. The values obtained from the records of neighbouring Meteorological Stations Fetesti (1943–1985) and Medgidia (1946–1986) were statistically processed and extrapolated to determine the extreme meteorological data. Considering the location of Cernavoda NPP site within the area of both meteorological stations, the greatest value estimated at one of the 2 meteorological stations was used for Cernavoda NPP. All the extreme meteorological characteristics were estimated using the same methodology.

The site main drainage header includes drains from 5 units and was sized based on national standard requirements, using the IDF regional rainfall maps for the following conditions:

- Design maximum rainfall: 270 l/s * ha;
- Design rain duration: 10 minutes.

These data correspond to an hourly maximum rainfall of 97.2 l/m^2.

A recent study (Evaluation of maximum rainfalls in Cernavoda area, Meteorological National Agency ANM, September 2011) using statistical analysis based on Gumbel distribution function, estimated the maximum rainfalls with various probabilities of occurrence on a certain time-interval. This study updates and confirms the adequacy of the original design rainfall intensity (IDF) curves.

A second study (External flooding hazard map, evaluation of extreme minimum and maximum water flows and levels of the Danube River in relation with Cernavoda NPP site elevation, National Institute INCDDD Tulcea, September 2011) was performed to evaluate the maximum water height resulting from rainfall exceeding the capacity of the drainage system on site platform. Several multiplication factors with respect to the maximum rainfall of 97.2 l/m^2 were considered.

Calculations were performed using the Digital Topographic Model (DTM) of Cernavoda NPP platform. In order to derive the average water height above the site platform at +16.00mBSL, the area occupied by plant buildings was subtracted from the total area.

It was found that, even in case of rainfall one order of magnitude greater than the design rainfall, the maximum average water height of 20 cm is accumulated on site mainly on pathways.

Local topographic measurements have been recently performed in order to validate the floor elevations of the buildings and confirm that adequate slopes are provided in the vicinity of buildings entrance. The results of these measurements show that buildings floor elevations range from 24 to 70 cm above ground elevation, with doorsteps heights of 8 to 16 cm and there are no reverse slopes. Therefore extreme rainfalls will not result in flooding the buildings on the site.

To compare the design against actual measurements of rainfall, it has to be mentioned that the hourly maximum quantity of rainfall recorded at Cernavoda was 47.3 l/m^2 (July 2010). This shows that the site drainage system capacity (97.2 l/m^2) is more than double the greatest hourly recorded rainfall at Cernavoda during the period 1986 - 2010. For reference, the maximum rainfall recorded in Cernavoda in 24 hours was 120.5 l/m^2 (July 2010).

Flooding due to rainfall on catchment area

One natural valley (Cismelei Valley) captures rainfall over an area of 22.2 km^2 around the Northeast plant site and discharges the water into the bypass channel. The discharge capacity of the valley is sized for coincident DBF = +14.13 mBSL and intense rainfall resulting in maximum flow Q= 458 m^3/s.

The characteristic water flow rates that are collected by Cismelei Valley from the plant and adjacent area surrounding the site in torrential regime are presented in Table 3.4:

Table 3.4 Maximum flow resulted from rainfall

Probability of occurrence (%)	1	0.1	0.01
Flow [m^3/s]	127	240	458

Therefore, the design basis related to extreme rainfall is considered Q = 458 m^3/s flow. The pluvial water flow rate of 458 m^3/s (collected from the catchment area) is drained into Race I of the Danube-Black Sea Channel, at the outlet of Cismelei Valley.

In these conditions, the water level upstream of the valley could reach an elevation of +17.50 mBSL and consequently a dike was raised for plant protection with upper elevation +18.00 mBSL.

Evaluation of water height on site platform resulted from Cismelei Valley dike failure was performed by the national institute INCDDD using the Digital Topographic Model (DTM) of Cernavoda site and adjacent area.

The analysis conservatively considered simultaneously occurrence of the following events:

- Danube River at Cernavoda DBF level +14.13 mBSL;
- Extreme rainfall on Cernavoda catchments area is resulting in maximum water flow rate of 458 m^3/s on Cismelei Valley;
- Breakage of Cismelei Valley dike.

The following data were used in support of evaluation:

- 1D2D Sobek Rural software;
- Digital Topographic Model (DTM), including buildings;
- Water level on Cismelei Valley is maintained at maximum dike elevation +18.00 mBSL level for the full duration of scenario development, that is considered for 1 hour;

- Dike failure within 30 minutes from the highest +18.00 mBSL elevation to minimum +16.00 mBSL elevation (site platform);
- Various break size, ranging from 10 m to 50 m wide;
- Different number of breaks and locations, from singular break to simultaneous four breaks;
- Conservatively, the site drainage system capacity was not considered.

For the worst case scenario, flowing water could increase up to 1m around the 110 KV Station and up to 0.5 m behind the Turbine Buildings were the main output and unit service transformers are located.

It was conservatively assumed that this scenario would lead to Total Loss of Class IV power (initiating event) occurrence, bounded by the potential unavailability of Class III Standby Diesel Generators. Plant response is provided by the existing abnormal operating procedures at both Unit 1 and Unit 2. The buildings housing SSCs required to perform essential safety functions (EWS/BMW and SCA/EPS) are not subject to flooding.

The sensitivity analyses performed show that water height accumulated on site platform is not dependent on the breaks number or size that influence only the flooding rate. It was assumed that dike failure at any location will significantly decrease water pressure on the rest of the dike and occurrence of additional breaks is not credible.

Tsunami induced flooding

The Cernavoda NPP site is located at 60 km from the Black Sea coast, at an elevation of +16.50 m above the Black Sea level (compared to the Baltic Sea, Black Sea level is at -0.5 mBSL).

Based on the IAEA Safety Guide SSG-18 (Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations, para. 5.44), no specific further investigations and studies need be performed to analyse the tsunami hazard for the plant site, provided that the site is located in an area that shows no evidence of past occurrences of tsunamis, and is located:
- At more than 10 km from the sea or ocean shoreline, or more than 1 km from a lake or fjord shoreline, as appropriate; or
- At more than 50 m elevation from the mean water level.

Even though the IAEA guidelines allow this event to be screened out for the Cernavoda NPP site, the potential for generation of tsunami in the Black Sea was evaluated and the results showed that extreme waves that could be generated are lower than 10m high. The evaluation showed that a wave would not propagate past the Agigea lock, which is +13.5 mBSL high.

3.1.1.2. Conclusion on the adequacy of protection against external flooding

The Design Basis Flood level for Cernavoda NPP is DBF = +14.13 mBSL (meters referenced to Baltic Sea Level) and the minimum buildings floor elevation is +16.24 mBSL, as confirmed by the local topographic measurement results. Therefore, significant margin of 2.11 m exists between the DBF and the lowest buildings floor elevation.

Sensitivity cases performed simulate the dynamic flooding on site based on increasing flow / level at platform elevation or above. The actual results support the expectations related to flood design basis since the Danube water flow required to reach platform level +16.00 mBSL is not realistic.

Flooding because of extreme rainfall in the catchment area surrounding the site was also considered. The combination of DBF level and extreme rainfall was found to result in level of 17.50 mBSL upstream of the valley; therefore a dike with maximum height of 18.00 mBSL was raised for plant protection. A margin of 0.5 m exists between the calculated maximum level and the top of the dike.

Even in the worst case scenario considered for dike failure, coincident with Danube River at Cernavoda DBF level +14.13 mBSL and extreme rainfall on catchment area resulting in maximum water 458 m^3/s flow rate, that might impair Class IV power and Class III power safety related systems, there is no a major threat on plant operation as long as plant response is provided by the existing abnormal operating procedures at both Unit 1 and Unit 2. For this extreme scenario, the buildings housing SSCs required to perform essential safety functions (EWS/BMW and SCA/EPS) are not subject to flooding.

The design maximum rainfall rate calculated using Romanian National standard is 97.2 $l/hr/m^2$. By comparison, the maximum recorded hourly rainfall at Cernavoda site is 47.3 $l/hr/m^2$ in July 2010. This shows that there is sufficient margin in the design basis of the drainage system above the maximum recorded rainfall rate.

In case of rainfall rate exceeding the capacity of the drainage system, the runoff is by overland flow to the drainage channel or the DBSC. A calculation for these scenarios demonstrates that the rainfall water is accumulated on the site access pathways and maximum average height is less than 20 cm for rainfall rate about 10 times the design maximum rainfall rate. The minimum buildings floor elevation is 24 cm above the site ground elevation; therefore this will not result in flooding the buildings on the site.

The existing margins are considered as being adequate and no additional measures are required to protect the plant against external flooding.

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

The key Structures, Systems and Components (SSC) required for achieving safe shutdown state are presented in Table 3.5. Based on the assessments described in Section 3.1.1, it is concluded that the safety functions will be maintained in case of external flood.

Table 3.5 Safety related systems separation into two groups

REQUIRED CAPABILITY	Normal Shutdown	GROUP 1	GROUP 2
(a) Shutdown Capability	- RRS - Class IV Power - Instrument Air (Normal Distribution)	- SDS#1 - Class III, II, I Power - Instrument Air (Normal Distribution)	- SDS#2
(b) Decay Heat removal Capability	- Steam to Condenser (CSDV) - Class IV Power - Main Feedwater System and Service Water System (RSW & RCW) - SDCS - Instrument Air (Normal Distribution)	- Steam Reject to Atmosphere (ASDV, MSSV) - Class III, II, I Power - Auxiliary Feedwater System and Service Water System (RSW & RCW) - ECCS - SDCS - Instrument Air (Normal Distribution)	- Steam Reject to Atmosphere (MSSV) - Emergency Power Supply System - BMW / Emergency Water Supply System - Instrument Air (Local Air Reservoirs)
(c) Capability to Limit Release	Containment	Containment (LACs)	Containment (Isolation valves, Airlocks, Dousing)
(d) Monitoring Capability	- Main Control Room - Class II, I Power	- Main Control Room - Class II, I Power	- Secondary Control Area - Emergency Power Supply System

The elevation of the site was designed to accommodate flooding level that could be reached in the extreme case of the upstream dam failure; however, the plans to build the two dams were not implemented. Therefore the elevation of +16.00 mBSL for the Cernavoda NPP platform provides additional margin to the calculated maximum flooding level.

Since the DBF of +14.13 mBSL is more than 2m below the measured buildings minimum elevation of +16.24 mBSL, no specific provisions were required for flood protection, in addition to the regular site drainage system and the dike with upper elevation +18.00 mBSL rose on Cismelei Valley.

The roads and the general ground elevation of the plant are designed to accommodate drainage of the maximum rainfalls and from concrete platform washing towards the discharge openings of the collecting channels. These channels as well as the remainder of the on-site drainage systems are sized for the maximum design flows.

Electrical power cables that are located in trenches outside buildings, potentially exposed to flooding from rainfalls, are made of one segment and resistant for water submergence since they are insulated.

The main operating provisions to prevent flood impact to the plant include:

Hydro-technical structures monitoring - The monitoring of the behavior in time of the Cismelei Valley channel is performed by visual inspection of the channel course and of the two banks. The inspection (monthly performed) aims to evaluate the status of the channel flowing section, the status of the bank slopes and of the dam, as well as to maintain the cleanliness status of the channel surface without excessive growth vegetation. Periodic maintenance activities and repair of the observed deficiencies are performed based on a support services contract. Topographical measurements are performed twice per year, using the reference (fixed) points defined in the design stage, in order to identify potential horizontal and vertical movements of the structures and buildings. As per up to date monitoring results there are no developing movements, confirming stability of the plant structures and buildings. Internal inspection of the site drainage system is periodically performed as part of Plant Preventive Maintenance Program.

Flood alerting system - As a preventive measure to protect against potential external flooding events there is in place a national forecast and warning system. This system is providing daily forecasts and warnings, if the case, making use of water level thresholds that are developed for specific locations along important rivers in Romania, including for Cernavoda. The warning thresholds are established by the state authorities in order to protect the Cernavoda local community and are not related to Cernavoda NPP operation, but they provide early warnings compared to DBF. The recorded Danube water level at Cernavoda exceeded the danger code limit of +10.86 mBSL four times in 40 years, with the maximum recorded value of +11.72 mBSL. The danger code level is more than 3 m below the DBF level of +14.13 mBSL for Cernavoda NPP.

Operating procedures - The emergency operating procedures (APOP) for Cernavoda NPP are symptom and event based procedures and cover the entry conditions that could be initiated by internal or external flooding events (e.g. Loss of Service Water, Loss of Feedwater, Loss of Class IV power, etc.), including unit response to transient events or seismic events that require operation from the Secondary Control Area. Therefore, development of specific emergency operating procedure for external flooding events was not necessary.

Plant Meteorological System - The plant meteorological system is an Automatic

Weather Station (AWS) that includes a set of sensors (wind speed and direction, air temperature, precipitation), a data acquisition unit, a radio data transmission system and two computers used as displays - one in the Main Control Room and another one in the Secondary Control Area. Meteorological sensors set consisting of temperature, wind speed and wind direction sensors are installed at three elevations (i.e. 10, 30, 80 m) in a Meteorological Tower of 90 m high. One sensor for precipitation is installed on the roof of the Meteorological Tower Equipment Room. The radio transmitter and its power supply are located in the Meteorological Tower Equipment Room, at about 1.4 km westwards from Unit 1 on a hill, at +43.8 mBSL. In addition to AWS there is one Cernavoda meteorological station reporting local data to the Meteorological National Agency (ANM). Also, Cernavoda NPP benefits from daily forecast updates from meteorological (ANM) and hydrologic (INHGA) national institutes.

The measures in place for ensuring access of personnel and equipment to the site in case of extreme external events that could affect the infrastructure rely on existing protocols between Cernavoda NPP and the local authorities involved in emergency response and are addressed in Section 6.1.3 of the report.

3.1.3. Plant compliance with its current licensing basis

There are no deviations from the licensing basis revealed by the existing inspection programs related to design and operating provisions for external flooding.

The Cernavoda NPP site has been selected and plant has been designed, constructed and licensed to operate taking into account the requirements of nuclear codes and standards, as well as best practice procedures applicable to cope with all the possible flooding hazards.

Arrangements to ensure the compliance with the licensing basis that involves generating, documenting and maintaining evidence that SSCs (including those that are required to cope with a flooding hazard) can accomplish their safety function during their installed service life have been implemented in the plant. This is an ongoing process, from the plant design to the end of service life, and plant ageing, modifications, repairs and refurbishment, equipment failures and replacements, and abnormal conditions are taken into account.

These arrangements ensure that all the plant activities that could impact safety, design or licensing basis, including the impact from the flooding hazard point of view, are reviewed and assessed, the necessary actions are taken, and all documents affected are updated:

- As per the Configuration Control Program and its package procedures, the assessment of the plant modifications (permanent or temporary) take the flooding hazard into account in order to ensure the capability of the SSCs to maintain their safety function.

- Ensuring that the SSCs performance has been preserved by ongoing application of measures such as scheduled maintenance, testing and calibration and has been clearly documented (Maintenance Program, Testing Program);

- Inspections to confirm the actual condition of SSCs (Inspection Program, Monitoring Program for site and structures). Specifically for structures and buildings, a monitoring program is performed twice per year with support from specialized contractors and includes performance of visual inspections, topographic measurements and bathymetric measurements.

- Surveillance of the SSCs (including those that are required to cope with a flooding hazard) and the analysis of the results to ensure that the SSCs are not affected by ageing process (Surveillance Program, PLiM);

- The assessment of the SSCs failures and their impact on the capability to maintain their function to cope with a flooding hazard, including the necessary corrective actions/ improvements to preserve their safety function (OPEX Program);

- New regulatory requirements, including those that could be related to a flooding hazard are managed through the station process called "Register of Licensing Documentation and Tracking of CNCAN Action Items". This process, documented into internal department procedure ensures that all license conditions and other CNCAN specific requirements are identified, analyzed, translated into specific plant procedures and tracked for their status until implementation.

As a response to the WANO SOER 2011-2 walkdowns and inspections were performed at both Unit 1 and Unit 2 by teams that gather staff from technical, operations, safety and licensing departments using general arrangement drawings and other plant information resources, including Flood Equipment Lists (safety related equipment location / flood areas) developed in support to internal flood and HELB events PSA.

The locations inspected are the Secondary Control Area and EPS Building, HP/ECCS Building, EWS House, Screen House, Pump House (P/H), Integrated Building (T/B), K-L Gap, Service Building (S/B), Chiller Building, Class III Standby Diesel Generators Building and the on-site drainage system used for mitigation of potential external flooding induced by local intense precipitations. There were no deviations from the design basis found. This conclusion is confirmed by an independent review performed in support of the present stress test report.

Mobile equipment and supplies are stored in a location that is not threatened by flooding.

The Internal flooding events are not in the scope of Stress Test specifications and therefore not addressed in this chapter. However, it should be mentioned that the adequacy of the Cernavoda NPP design bases with respect to internal flooding was demonstrated by a systematic deterministic flood hazard analysis done as part of the Internal Flooding Probabilistic Safety Assessment.

The results from PSA Level 1 Internal studies for full power, intermediate power levels and shutdown states shows that internal flooding events contribute 3% to the total Core Damage Frequency (CDF) for Unit 1, and also for Unit 2, being the lowest initiating events group contribution. The results of these studies confirm the design robustness and Cernavoda NPP capability to mitigate internal flooding events as required by the station design.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

The Cernavoda plant can withstand credible external floods without compromising any of the safety functions. This is achieved by:

1. Designing the site platform at a high elevation (+16.00 mBSL) and confirmed design provisions for the protection level of +16.24 mBSL for the lowest buildings ground floor elevation, compared with the highest Danube River water level DBF = +14.13 mBSL ensure adequate margin of 2.11m (DBF level of +14.13 mBSL has a return period of less than once in 100 000 years);
2. Protecting the site from Cismelei Valley flooding induced by extreme rainfall in the catchment area surrounding the plant, in coincidence with DBF = +14.13 mBSL, using a dike elevated at +18.00 mBSL. The dike top elevation ensures a margin of 0.5 m compared to the calculated maximum level of +17.50 mBSL;
3. Designing the buildings ground elevation (+16.30 mBSL) higher than the site platform (+16.00 mBSL) with ground sloping away from the buildings. As confirmed by the local topographic measurement results, the buildings minimum elevation is +16.24 mBSL;
4. Designing the site pluvial drainage system for maximum rainfall rate of 97.2 $l/m^2/h$ ensures adequate margin since for heavy rainfall rate 10 times greater than the maximum rainfall rate (972 $l/m^2/h$); it is thus ensured that the water level increase on the site platform is of about 20 cm, less than 24 cm, representing the minimum height about buildings ground floor, as confirmed by the local topographic measurement results;
5. Designing multiple independent heat sinks and power supplies to provide defense-in-depth against system failures (details provided in Chapter 5).

On the basis of the flooding assessments reported in Section 3.1, the plant buildings will not flood since the water level can not reach the ground elevation of any building due to any flooding scenario.

It can be concluded that there is significant margin to flooding of the plant buildings by external events. If these margins are exceeded, fuel cooling is assured because there are passive means to keep the fuel cool that do not rely on equipment at low elevations nor electrical power. This is gravity feed from the dousing tank to the boilers via the BMW system, with PHT flow driven by thermosyphoning. Chapter 5 of this report describes this heat sink and shows that there will be at least 23 hours of passive cooling available. Actually the time available will be longer if the plant was shut down and cooled while electrical power (Class III or EPS) was still available, as would be the case for a slowly-evolving event with early warning of the risk of flooding.

As for weak points and cliff-edge effects, there have been assessed by determining where water ingress could occur in external flooding events regardless of the source. Cliff edge effects were considered as failures of components or systems that reduce the available methods for core cooling. For water to be able to enter buildings at

ground level, the flood level has to be higher than 16.24 mBSL. There is no credible scenario for this to occur, based on the assessments performed for this stress test.

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

Based on the analysis results obtained by making use of the latest deterministic tools and complemented by probabilistic approach, it is concluded that Cernavoda NPP design intent in relation with flooding hazards provides sufficient margins, therefore no further measures are envisaged in this area.

CHAPTER 4 - EXTREME WEATHER CONDITIONS

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

As part of the "stress test" assessment, a screening and bounding analysis for Cernavoda NPP response under severe weather conditions has been performed.

In order to derive a comprehensive list of severe weather events to be considered for Cernavoda Units 1 and 2, a review of CNSC regulatory documents and guides, IAEA guides, ANSI/ANS standards, and US NRC documents and IPEEE experience has been performed.

The preliminary screening criteria, established in accordance with internationally recognised standards, consist of the following (any one of these criteria was considered to be sufficient to screen out the event):

Criterion 1: The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external event.

Criterion 2: The event has a significantly lower mean frequency of occurrence than another event taking into account the uncertainties in the estimates of both frequencies. The event in question could not result in worse consequences than the consequences from the other event.

Criterion 3: The event cannot occur close enough to the plant to affect it. This criterion must be applied taking into account the range of magnitudes of the event for the recurrence frequencies of interest.

Criterion 4: The event is included in the definition of another event.

Criterion 5: The event is slow in developing and it can be demonstrated that there is sufficient time to eliminate the source of the threat or to provide an adequate response.

Table 4.1 - Initial Screening of Severe Weather Events for Cernavoda NPP

Event	Screening Criteria	Remarks
Avalanche	3	The nature of the topography surrounding the plant is such that avalanche is not possible. The plant is situated on a plain zone, and there are no hills or mountains close to the site that have the potential to generate an avalanche.
Coastal erosion	3	The Cernavoda site is not located within a coastal region and thus coastal erosion is not applicable for this site.
Drought	5	The main source of water for the plant is the Danube river and in the event of a drought there would be adequate warning and time so that remedial action could be taken. The consequence of drought for a long period would result in reduced river levels and this is addressed in "Low river level"
External fire	-	Addressed

Table 4.1 - Initial Screening of Severe Weather Events for Cernavoda NPP		
Event	**Screening Criteria**	**Remarks**
External flooding	-	Covered by external flooding analysis in Chapter 3.
Extreme Winds and Tornadoes	-	Addressed
Fog	4	It is a quite frequent phenomenon in the boundary areas of the Danube River. At Cernavoda, the annual average number of foggy days is about 47 and the maxim number is 87 days. Throughout a year, fog is mostly frequent in winter seasons. Fog affects the likelihood of man-made hazards, particularly transportation accidents, and is outside the scope of this report.
Frost	1	The effect of frost is bounded by the effects caused by snow and ice (loading) and is therefore screened out on the basis of criterion 1.
Hail	1	At the Cernavoda NPP site, hail is rare. The annual maximum number of hail days is 2 and the average number is 0.9 days. Hail may result in a loss of Class IV power, the effects of which are covered in the accident sequences covered by Chapter 5, and is screened out on the basis of criterion 1.
High summer temperature	1, 5	This event is bounded by loss of all heat sinks, including alternate ultimate heat sinks, a scenario covered in Chapter 5.
High river level	-	Covered by external flooding analysis in Chapter 3.
Hurricane	3	Romania is far remote from any oceans where hurricanes occur. It is judged that any such hurricane having travelled that far over land to reach the station would have lost much of its strength and is not a credible threat to the Cernavoda NPP.
Ice Cover	5	This is a slow developing event with adequate measures in place to prevent loss of safety functions.
Intense local precipitation	-	Covered by external flooding analysis, Chapter 3.
Landslide	3	The Cernavoda NPP site topography is a flat platform and the geotechnical conditions are such that a landslide is not credible.
Lightning	1	Considered in plant design, walk-downs have been performed to confirm protection against lightning. Effects of lightning have been shown to be extensive for electrical equipment and can develop into transformer explosions, fires, spurious signals and loss of Class IV power. Loss of on-site or off-site power covered in the accident sequences discussed in Chapter 4.
Low river level	5	This is a slow developing event that has occurred in the past and resulted in safe shutdown of the plant. Adequate assessments and operating procedures in place to deal with this event.
Low winter temperature	1, 5	Accounted for in building codes. Prolonged low temperatures considered to be a slow developing event such that any remedial actions necessary to maintain ultimate heat sink could be undertaken in a timely manner. River icing due to low temperatures is addressed in ice cover.
Sandstorm	1	In the Cernavoda NPP site area, the frequency of the phenomenon is very low. Sandstorms resulting in blockage of

Table 4.1 - Initial Screening of Severe Weather Events for Cernavoda NPP		
Event	**Screening Criteria**	**Remarks**
		air intakes accounted for in the design. Loss of instrument air accident sequence is also covered in the existing safety analysis.
Seiche	3	A seiche is a standing wave and may be caused by changes in atmospheric pressure, wind, earthquakes and possibly landslides. Landslides are already screened out as a credible hazard and do not require consideration here. Seismically induced waves, also called tsunamis, originate in large bodies of water such as seas and oceans and has been screened out from Cernavoda site. The effect of wind and changes in atmospheric pressure are covered by consideration of a storm surge that was also found not to be a credible event for Cernavoda.
Snow	1, 5	This is a slow developing event with adequate measures in place to prevent loss of safety functions.
Soil shrink/swell	3	Soil shrink and swell due precipitation or periods of drought for example was considered however all buildings foundations are on rock (limestone) and therefore this event is not applicable for the Cernavoda site.
Storm surge	3	A storm surge is the onshore pileup of ocean or lake water caused by a combination of low pressure and wind of which 75% of the contribution comes from the latter. Canadian experience with a CANDU 6 plant on a river site similar in nature to Cernavoda NPP has shown that the area of the water surface is not large enough to have a storm surge to represent a credible hazard for the plant. Moreover, large storm surges often accompanied hurricanes and tropical storms which were screened out for Cernavoda NPP.
Tsunami	3	Covered under external flooding analysis in Chapter 3.
Waves	4	The effects of waves are already covered under coastal erosion, storm surge and seiche. All of these events were screened out and therefore the wave hazard is also screened out.

As outlined in Table 4.1, the conditions that would result in events bound by the scenarios analysed in Chapter 5 of the report (i.e. SBO, LOUHS and a combination of these) have been therefore screened out and are not addressed in the following.

However, the events screened out based on criterion 5 (slow developing event) are discussed below.

Low winter temperature - Ice Cover

The ice cover event has been screened out based on the preliminary screening criterion 5 (slow developing event).

Ice accumulation can interfere with the function of Class IV and Class III power supply. In the event of a loss of Class IV power, the effects are covered in the accident sequences discussed in Chapter 5 of the “stress test” report.

Operational provisions are in place to prevent ice accumulation (from freezing rain for example) on the ventilation louvers for the rooms housing the Class III Standby Diesel Generators and Emergency Power Supply Diesel Generators.

Icing of the distribution bay has the potential to interfere with condenser cooling water, raw service water and emergency water supply system intakes in combination with low river levels. This situation has been addressed by an emergency operating procedure. Scenarios involving loss of heat sinks are covered in Chapter 5 of the "stress test" report.

Snow

Two types of effects are considered:
1) The short-term effects of snow
2) The longer term effects of snow accumulation.

At Cernavoda the annual maximum number of snowstorm days is 9, and the average number is 2.2 days. This event has a low frequency, and occurs in winter and early in spring. In the period of 1999-2011, there were registered, according to plant control room logs, at least 2 days with snowstorm (in the winters of 2007 and 2008). Snowstorm may result in a loss of Class IV power, the effects of which are addressed in Chapter 5 of the report.

For the long term effects, the primary safety concern with respect to snow accumulation is the collapse of buildings and structures. Delineation between non-safety related structures and safety related structures from a safety perspective is important here because non-safety related structures may collapse due to excessive snow load without impact on nuclear safety. The safety related structures, however, must remain intact above maximum expected snow loading conditions to ensure safety systems are available.

It is recommended in IAEA Guide NS-G-3.4 that *"an indicator of snow pack hazard, the expected extreme value and its confidence interval for the lifetime of the plant should be determined"*. Annual maximum expected snow load (and thickness) was derived from statistical data from the nearby meteorological stations. From the derived hazard curves if a return period of 50 years (characteristic of the lifetime of the plant) is considered, the snow load hazard would likely be as high as 150 kg_f/m^2 based on the Medgidia station data and 100 kg_f/m^2 based on the Fetesti station data. Two notable extreme snowfall events at the Cernavoda site include an extreme accumulation event in 1954, where 136 cm accumulated corresponding to a load of 450 kg_f/m^2, and in April 1981, an extreme duration event occurred where snow fell for 230 hours consecutively.

All safety related structures at Cernavoda NPP are designed taken as reference snow loads of 100 kg_f/m^2 with a recurrence period of 50 years as per Romanian legislation and vendor's recommendations. However, a factor of 1.4 was applied to the calculations to compensate for the small amount of historical data related to snow. In addition, based on the geometry of the building (height, slope, etc.), a further factor is applied. Therefore, the actual snow loading ranges from 156 kg_f/m^2 to 680 kg_f/m^2.

Snow accumulation is a slow developing event and the plant has field inspection routines and procedure to deal with extreme weather events. Therefore, it is expected that adequate actions will be taken to avoid any damage to the safety related structures such as Service Building, EPS, SCA and EWS buildings. The Reactor Building is a robust structure. The dome shape does not allow large snow accumulation, so it is not expected to be affected by extreme snow loading.

Considering the above, this event is screened out based on the slow development of the event, allowing sufficient time for response actions.

Drought - Low River Level Assessment

Cernavoda Unit 1 had experienced low Danube river level that required extended plant shutdown in August 2003 (Unit 2 was still under construction at that time). The low river level event in 2003 was due to prolonged period of drought all over Europe, the Danube level and flow reached the lowest values in the last 160 years. Recently, the Danube level was abnormally low in the period of September – December 2011 September, but the level did not reach the critically low level that require a plant shutdown.

Following the event in 2003, a number of safety assessments on low Danube level have been performed by the licensee and the vendor. Some design changes have been performed, two deep underground wells have been dug Cernavoda site to provide water also for worst case drought scenarios and a specific emergency operating procedure (Abnormal Plant Operating Procedure - APOP) has been established to provide instructions to the operating staff in safely dealing with critically low river level situations.

Based on the long time available for response, actions can be implemented in accordance with the specific APOP and Severe Accident Management Guidance response for maintaining the cooling safety function in the worst case drought scenario and. Chapter 5 of the “stress test” report addresses the bounding event of the loss of alternate ultimate heat sink.

After the application of screening criteria, the following events have been retained for further analysis:

- Forest fires
- External flooding
- Extreme winds and tornadoes

Forest fires

External fires could result from lightning strikes igniting local ground cover and the situation would be aggravated during hot weather and or drought conditions accompanying the lightning strike.

In accordance with plant specific procedure for responses to high temperature conditions warning provided by ANM (National Meteorological Administration), the on-site firefighters are instructed to check during their routine for the possibility of

dried vegetation fire occurrence. A fire truck and fire brigade are available on-site at all times.

Also, the site security personnel are posted at different points all around the site perimeter and would be able to quickly notify the fire brigade in such an event. All the paths outside the plant are made accessible (as per plant procedure) for fire trucks intervention.

External flooding

External flooding has been addressed in Chapter 3 of the present report.

Extreme winds and tornadoes

Extreme winds

The maximum wind speed recorded by the Cernavoda meteorological station during 1986 – 1999 was 65 km/h (18 m/s) on December 10th, 1991 at 1 pm. The absolute maximum wind speed recorded by the 3 meteorological stations at Cernavoda, Fetesti and Medgidia are 126 km/h (35 m/s), 122 km/h (34 m/s) and 101 km/h (28 m/s), respectively.

Statistical data was used to derive a correlation between the maximum wind speed and the velocity of wind gusts. For the 1000 year return period, the direct wind speed on the basis of the data at the Fetesti and Medgidia stations is of 184 km/h (51 m/s) and of 148 km/h (41 m/s), respectively, while the maximum gust speed with the same return period is 220 km/h (61 m/s) and 173 km/h (48 m/s), respectively.

Cernavoda NPP structures are constructed in accordance with the Romanian standards to withstand, as a minimum, a loading figure of 140 kgf/m^2 (1.37 kPa), that translates to a wind speed of 166 km/h (46 m/s). Therefore, the Cernavoda NPP structures at a minimum would withstand all maximum wind speeds historically recorded in the site region.

The statistics extrapolated for the 1000 year return period indicate that wind and/or wind gusts may exceed the minimum 166 km/h design value. However, based on seismic design requirements, the structures would be more robust than the requirements for wind loading alone and would have sufficient margins to withstand higher wind speeds.

Furthermore, Canadian experience has shown that plants very similar to Cernavoda have been exposed to above design basis winds with no consequences. In one particular case, a plant for which the design basis wind speed was 108 km/h (significantly less than what Cernavoda is designed for), was exposed to wind gusts of 148 km/h without suffering any damage.

Since the plant is not explicitly designed to withstand higher than maximum wind speeds predicted for the 1000 year return period, potential damage to the plant caused by high winds has been considered (including secondary effects of missiles generated

by high wind). The bounding cases for event scenarios resulting from beyond design basis winds are covered by the events analysed in Chapter 5 of the report.

Given that the plant is not explicitly designed for wind speeds of greater than 166 km/h, the appropriate proactive corrective actions (e.g., shutdown and cooldown) to be taken at various high wind conditions should be specified in a procedure. A specific procedure is already in place for extreme weather conditions exceeding pre-established setpoints (action levels), which covers also the actions to be taken in case that the wind speed exceeds 180 km/h (50 m/s).

Tornadoes

Tornadoes are commonly classified on the Fujita scale devised in 1971 by the Japanese American meteorologist Tetsuya (Ted) Fujita. A description of the first five levels in the scale is as follows:

F0 - light (winds of 64 - 116 km/hr; some damage to chimneys, TV antennas, roof shingles, trees, signs, and windows), accounts for about 28% of all tornadoes.

F1 - moderate (winds of 117 -180 km/hr; automobiles overturned, carports destroyed, and trees uprooted), accounts for about 39% of all tornadoes.

F2 - considerable (winds of 181 -252 km/hr; roofs blown off homes, sheds and outbuildings demolished, and mobile homes overturned), accounts for about 24% of all tornadoes.

F3 - severe (winds of 253 -330 km/hr; exterior walls and roofs blown off homes, metal buildings collapsed or severely damaged, and forests and farmland flattened), accounts for about 6% of all tornadoes.

F4 - devastating (winds of 331 - 417 km/hr; few walls, if any, left standing in well-built homes; large steel and concrete objects thrown great distances), accounts for about 2% of all tornadoes.

F5 - incredible (winds of 418 -509 km/hr; strong frame houses lifted off foundations and carried considerable distances; automobile sized objects fly through the air in excess of 100 metres; trees debarked; steel reinforced concrete structures badly damaged), accounts for about 0.1% of all tornadoes.

Several confirmed tornado events have occurred since 2002 in Romania. On August 12, 2002 a tornado was confirmed (and recorded for the first time in Romania) in Facaieni in southern Romania. On May 7, 2005 a squall line with an embedded bow echo formed over southern Romania producing 3 tornadoes. One tornado occurred in Buftea and another in Ciobanu village and both were classified as category F0 on the Fujita scale. The third tornado occurred at Movilita village and was classified as an F1.

There is a lack of information to establish a credible estimate of an annual frequency of tornadoes per unit area in order to derive a Cernavoda specific frequency for a tornado direct hit on the plant. It should be noted that damage from a tornado is limited to the direct hit area.

Assuming conservatively that a tornado could damage the Emergency Power Supply building and the Emergency Water Supply building and that the equipment therein were unavailable, which corresponds to a loss of alternate ultimate heat sink, this

scenario was examined in Chapter 5 of the "stress test" report. In this case, fire water trucks are ultimately used for SG make-up water via the EWS lines and they draw from the fire water tanks that store an inventory of 2 x 1500 m^3.

The fire water tanks can be supplied with water by a mobile diesel driven pump with a capacity of 500 m^3/h, and can also be used to supply water directly to the fire trucks. Should this source of water be unavailable such as the case would be if the tornado also rendered the fire water tanks unavailable, the domestic water system can be used and is provided with water from two underground wells within the Cernavoda site. The water comes from the underground phreatic water (700 m deep). The water in the domestic water system is provided by two pumps that can provide 90 m^3/hr. Normally the pumps are powered from Class IV power, but the two mobile diesel generators that have been procured for electrical power supply would be stored in a robust structure on site that would withstand the effects of a tornado (wind pressure and tornado generated missiles). On the basis that steam generator make-up water is ensured with these measures, thermo-syphoning would continue to remove decay heat and core damage would be precluded.

For a lesser damage scenario, such as availability of the EWS building, the onsite mobile diesel (stored in a robust structure) may be used to connect directly to the EWS distribution panel to enable makeup water to the steam generators for decay heat removal. Due to the large spatial separation between the EWS building and the onsite mobile diesel generator storage site, it is unlikely that both the EPS building and the mobile diesels will be hit by the same tornado (if one was to directly hit the Cernavoda NPP). It is therefore judged that core damage scenarios can be prevented also in case of tornadoes.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

Adequate safety margins exist in relation to extreme weather conditions, taking account margins provided in the design of the safety related SSCs as well as the time available for preventative measures in slow developing scenarios.

For cases in which the extreme weather conditions addressed in Section 4.1.1. could affect the availability of the off-site power supply and / or the transfer of heat to the ultimate heat sink, an assessment is provided in Chapter 5 of the report. Based on the review of severe weather conditions and their impact on the plant, it was concluded that these would not generate worst accident scenarios as compared with those analysed in Chapter 5 of the present report.

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

Even though the possibility to have on site winds corresponding to the 1000 year return period is very remote, the specific procedure which is in place for extreme weather conditions (and covers also the actions to be taken in case of high wids), will be revised to include more proactive actions.

CHAPTER 5 - LOSS OF ELECTRICAL POWER AND LOSS OF ULTIMATE HEAT SINK

In compliance with the stress test specifications, the licensee has analysed the following scenarios:

- loss of offsite power;
- station blackout (SBO);
- loss of primary ultimate heat sink (UHS);
- loss of both primary and alternate ultimate heat sinks;
- loss of primary ultimate heat sink with station blackout.

These events could be initiated by a seismic or flooding event or by other external hazards. The "stress test" report submitted by the licensee has provided an analysis of the above mentioned scenarios, in accordance with the ENSREG specifications.

For each scenario, the licensee has identified the plant design capabilities to fulfill the safety functions (shutdown reactor, cooldown the reactor core, contain and monitor the plant parameters). In the same time the potential cliff edge effects associated were identified and evaluated. The design and supplementary measures available on site were presented.

5.1. Loss of electrical power

Loss of electrical power has been considered progressively, starting with a loss of off-site power, which is a design basis event for Cernavoda NPP, both as a independent event and also combined with other design basis accidents, such as Loss of Coolant Accidents. In order to cope with Loss Of Off-site Power (LOOP), the plant is provided with a range of on-site power generation and support facilities.

The following level of severity regarding the loss of electrical power supply is the sequence of loss of all off-site and on-site AC power generation capacity, generally known as Station Blackout (SBO). In the framework of the stress tests, an evaluation of the safety margins for SBO has been required and also a review of the improvements which could be implemented, if necessary, in order to increase safety margins. The impact of loss of electrical power on the spent fuel bays is covered in Section 6.

Overview of the electrical power supply and distribution systems

From the main functional processes point of view, Cernavoda Nuclear Power Plant is organised in three different sections:

- Unit 1 – main function: nuclear electrical power production;
- Unit 2 – main function: nuclear electrical power production;
- Unit 0 – main function: services supply for Unit 1, Unit 2 and other plant facilities.

To enable understanding of the events analysed, a brief description of the Cernavoda NPP electrical power supply system has been provided in Chapter 1 of this report and

is further detailed in this section. A schematic diagram of the essential services power supply is presented in Fig. 5.1.

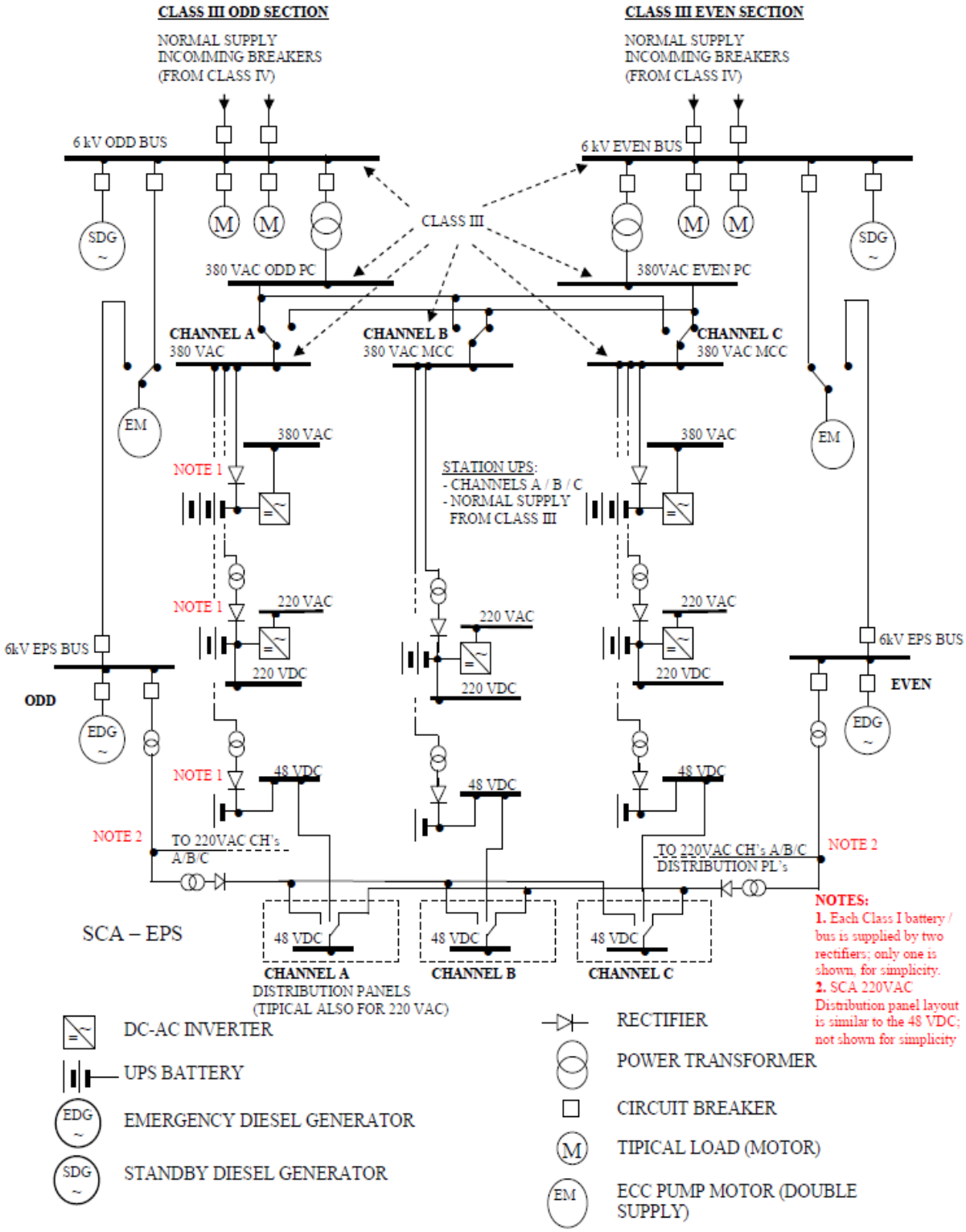

Figure 5.1 - Simplified one-line diagram of essential services power supply

The Unit 1 and Unit 2 internal power supply/distribution are organised on 4 classes of power, as follows:

- Class IV: Normal electrical supply to equipment and auxiliaries, which can tolerate long term interruptions without affecting nuclear safety, personnel or equipment safety. The class IV power supply is divided in two independent and separate divisions (odd and even). A complete loss of Class IV, or a loss of either ODD or EVEN division of Class IV power will initiate a reactor shutdown.

  Class IV consists of:
  - Redundant off-site sources, which provide electrical power required during startup and shutdown of the unit and can also supply power during normal operating conditions;
  - The turbine generator (onsite), which provides electrical power required during normal operation;

On-site stand-by sources which provide the electrical power required in case of loss of the normal power supply:

- Class III: Power supplies to the safety-related systems. Normal supply of Class III distribution system is from service transformers and is backed-up by the stand-by diesel generators with 100% redundancy. Stand-by electrical diesel can provide power to essential loads (around 7 MW for each Unit) to ensure an orderly shutdown. Also, Class III is the charging source to the Class I batteries and back-up supply to Class II loads;
- Class II: Uninterruptible power supply (UPS), alternating current 380 VAC and 220 VAC, supplies for essential auxiliaries, control, protection and safety equipment. Uninterruptible power is provided by batteries, through inverters or by Class III during unavailability of the inverters;
- Class I: Direct current (DC) uninterruptible power supply (UPS) 380 VDC, 220 VDC, 48 VDC. Uninterruptible direct current (DC) supplies for essential auxiliaries, control, protection and safety equipment. Batteries provide uninterruptible power for 8 hours.

The onsite Power Distribution system and each of the internal power supply distribution systems are divided into redundant load groups (EVEN and ODD) so that the loss of any one group does not prevent the safety functions from being performed. Electrical AC and DC power is distributed throughout the plant at multiple voltage levels with provisions incorporated for diversity, redundancy and segregation.

Apart from Unit 1 and Unit 2, Unit 0 of Cernavoda NPP provides the following support functions:

- System service supply: 110 kV switchyard operation for Unit 1/Unit 2 back-up and Unit 0 internal services;
- Water treatment plant (clarified/demineralised water production);
- Fire water supply;
- Auxiliary steam plant (light oil fuel);
- Domestic water supply;
- Non-essential buildings and structures power distribution.

Unit 0 has its own DC power (UPS).

In addition to the electrical sources mentioned above, for each of the Cernavoda NPP Units, an Emergency Power Supply (EPS) system is provided by design, representing a completely independent, seismically qualified, emergency power supply system designed to 100% redundancy and separation requirements, provided to cope with common mode events, ensuring that the safety functions are maintained. This system is intended for back-up supply supporting essential safety functions when all the others electrical supplies are unavailable or when the Main Control Room (MCR) is inhabitable. For such conditions, a seismically qualified alternative control room is available - the Secondary Control Area.

EPS provides back-up for Class III, II and I seismically qualified equipment (cca. 1 MW each Unit). The EPS distribution has the same separated ODD / EVEN layout as the other station power supplies. The ODD and EVEN buses can be tied together via tiebreakers for flexibility. There are also several local EPS services panels powered from either Class III ODD/EVEN or EPS ODD/EVEN.

Only one EPS DG is required to be operational at any time, the other one acting as a back-up. The plant design ensures at least 30 minutes until the manual start of the EPS Diesel generators is credited. Under SBO conditions, EPS continuous operation is ensured for at least 5 days for each unit, from the point of view of the fuel oil reserve. The EPS DGs are 100% redundant, Design Basis Earthquake (BDE) seismically qualified to maintain integrity and functionality. Also, the entire building, support systems including diesel fuel storage and fuel transfer, structures and equipment of the EPS are DBE seismically qualified.

Mobile Diesel generators: following the Fukushima Daiichi NPP accident, Cernavoda NPP procured two mobile diesel generators (for both units), to provide power if the EPS is not available. The capacity of each mobile diesel generator is almost equivalent to that provided by the design non-mobile EPS diesel generators. The mobile diesel generators have autonomy of 6 hours at full load without external support. The available fuels oil on site will ensure more than 5 days of operation without external support, considering only the fuel oil stored in the seismically qualified EPS storage tanks. A maintenance and mandatory testing program of the mobile diesels is being implemented at site in order to ensure that the systems are available on demand.

5.1.1. Loss of off-site power

Loss of off-site power (LOOP) is a design basis event which does not pose any threat for the plant fuel cooling, in the reactor or in the spent fuel bays. However, the LOOP event leads immediately to automatic plant actions that will ensure the electrical power supply to plant consumers important for safety. These actions are different, depending on the cause that determined the LOOP:

Islanded operation is an abnormal operating mode defined as the separation of a part of the station 400 kV transmission network from the rest of the grid. The separation could be the result of remote transmission lines breaker trips. The main breaker of the plant generator remains connected to the internal services and plant output transformers.

The resulting island could operate almost normally (however, with reduced power supply reliability) if the islanded power generating unit(s) can rapidly adjust the power output to match the loads remained in the island. Otherwise, the unbalance between the power produced and power consumed could lead to the local grid collapse: possible power generating units remaining isolated (supply only their internal electrical services) or even units tripping (the risk of LOOP is increased during this regime).

The *loss of line* event represents the sudden unit disconnection from the 400 kV output switchyard caused by a protective trip of the 400 kV grid interconnection breakers (two breakers for each unit at Cernavoda NPP), with the unit remaining at high power - isolated on their own internal services supply. If the loss of line event is caused by a progressive or sudden transmission grid failure (rather than a local / single unit output system malfunction), and if the regional 110 kV distribution grid - that provides the back-up for internal services supply - also fails, the units will experience a LOOP event. The difference between a single unit loss of line and the whole station being affected by a LOOP is the operation of equipment in Unit 0, which provides services (demineralized water, etc.) to both Unit 1 and Unit 2.

When grid connection is lost for both units, the Units 1 and Unit 2 will operate independently of each other. Either unit shutting down in this operating mode has no effect on the other unit high power operation (and thus on the ability to supply its own services, without the use of stand-by power).

Following a LOOP event, the resulting *isolated* mode of unit operation is characterized as a high power operating mode of the reactor (35% up to 60 % full power - FP), with the turbine generator supplying only the internal services of Unit 1 or Unit 2 respectively (approximately 7% equivalent of FP), due to LOOP output connections. The difference from 7% to 35% up to 60% FP represents steam produced in the boilers that is diverted directly to the condenser via the turbine by-pass Condenser Steam Discharge Valves (CSDVs).

Any unit that disconnects its main generator from the internal services supply buses will not be able to remain at high power (the reactor will trip). This event is called Loss of Class IV electrical power and is part of the design basis. The plant design ensures safe reactor shutdown and cooldown under Loss of Class IV conditions.

Unit 0 will lose power supply and as a consequence, the pre-treated and demineralised water production in Unit 0 will cease. Still provided by Unit 0 is the supply of firewater, which relies on a Diesel driven pump that starts automatically in case of power supply failure. The demineralised water consumption will be reduced to a minimum by reducing the auxiliary steam load and the boilers blow-down.

In the case of a total loss of Class IV power, with the main generator disconnected from the internal services, all fundamental safety functions are fulfilled. The turbine and the reactor are automatically shutdown. The stand-by Class III diesel generators will automatically start up to energize the Class III buses within about 10 seconds. The Auxiliary Feedwater (AFW) pump would be loaded within 110 seconds to supply feedwater to Steam Generators (SG). As the power to SGs drops rapidly following the

reactor shutdown the reserve of water in boilers is enough to ensure the heat sink for the primary coolant.

The electrical power and instrument air will be also available for pressure and inventory control on the primary and secondary circuits. Following the trip and the rundown of the main Primary Heat Transport (PHT) pumps, the cooling of the heavy water in the primary circuit is ensured by thermosyphoning. The water to SGs will be supplied by the AFW pump (on Class III) with inventory provided from the secondary side tanks. The time available exceeds 4 days.

Containment functions are provided as per normal operation state and are not affected by LOOP.

Regarding reactor cooling, in the long-term the operator may start the Shutdown Cooling System (SDCS), which is also supplied by the Class III power, to preserve the demineralized water inventory. The same action is required in case the auxiliary feedwater pump is unavailable (a specific Abnormal Plant Operating Procedure will be followed). The heat transferred through the SDCS heat exchangers will be removed to recirculated and service cooling water (RCW/RSW) systems (Class III power) and finally to the Danube River.

If SDCS pumps and auxiliary feedwater pump (AFW) are not available, SGs can be used as heat sinks by depressurization and addition of water from the dousing tank (by Boiler Make-up Water system – passive make-up) and Emergency Water System (electrically supplied by EPS).

The necessary fuel oil for standby Diesel generators is ensured on the site for at least 5 days for each unit. The stand-by Diesel fuel oil is stored on-site in 4 fuel tanks in Unit 1 and 4 similar fuel tanks in Unit 2. These are installed in individual semi-buried concrete structures. The stand-by diesel generators and associated diesel fuel tanks, even though are robust structures, are not seismically qualified as per the original design. In case the loss of Class IV power is determined by a DBE, Stand-by Diesel generators are considered unavailable. This case is covered by Section 5.1.2., when power supply is ensured by EPS Diesel Generators.

The EPS diesel generators fuel tanks are seismically qualified (DBE) and capable to ensure the necessary fuel for more than 72 hours for EPS Diesel generators (at least 6 days). Specific arrangements are in place to have the necessary fuel oil on site after the depletion of the fuel reserves (addressed in Section 6.1.3.)

Stand-by Diesel generators, electrical connections and fuel reserves cannot be affected by a flooding event, as discussed in Chapter 3 of the report.

The robustness of the Cernavoda NPP is ensured by four additional levels of defence in depth for electrical power supply, apart from the off-site power:

- the Class III electrical power supplied by the first set of diesel generators with 100% redundancy built-in;
- the Class I / II electrical power supplied from batteries; according to the current evaluation, the batteries can provide power supply for 8 hours;

- the emergency electrical power supply provided by the second set of diesel generators (seismically qualified) known as emergency power supply (EPS) (designed with 100% redundancy and separation requirements);
- the mobile diesel generators.

Except for the Class I / II batteries, the other electrical power sources ensure at least 5 days of continuous power supply without any external support. The Class I / II batteries ensure the cooling function (basically actuating valves) for a limited time interval but sufficient to allow the operator to secure another heat sink.

Based on this, the licensee concluded that additional design changes or operational changes are not required in this case.

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

Loss of all Class IV (External Grid and Class IV Distribution Buses) and Class III (Standby Diesel Generators) power sources represents a design basis scenario.

In case of Loss of Class IV power and loss of Class III power, the operator has to take the necessary actions to ensure the electrical power supply to important systems and components in order to maintain the nuclear safety functions and to place the plant in a safe shutdown state.

Loss of Class IV determines the event sequences presented in Section 5.1.1: turbine trip, reactor trip, heavy water feed pump trip, main feedwater pumps trip, pressure in the primary circuit increase with a consequently Liquid Relief Valves (LRVs) opening and closing for overpressure protection, the Steam Generators (SGs) steam pressure increase with a consequently opening and closing of the Main Steam Safety Valves (MSSVs) for SG overpressure protection.

In case that, following a Loss of Class IV power, the automatic or manual start-up of the stand-by diesel generators does not succeed (any one of the two independent trains), and consequently the Class III buses remain de-energised, the operator will initiate SGs depressurization in order to bring into service low pressure water supply systems to SGs: Boiler Make-up Water (BMW) or Emergency Water Supply (EWS) system. The operator will open MSSVs and when the SGs pressure decrease below 345 kPa, the water from the dousing tank will start to be fed by gravity into the SG after BMW isolating valves are manually open from MCR, SCA or directly from the field.

Since Class I and II electrical power (batteries) are available, in case the operator does not depressurize the boilers then boiler auto-depressurization is automatically initiated when specific conditions of very low SG water level and low feedwater header pressure are reached. In this case, the MSSVs open automatically.

In both cases, of manual or auto depressurization of the SGs and BMW isolating valves opening, the water inventory will be available from the dousing tank. The minimum available demineralized water inventory from the dousing tank is about 2000 m^3. The opening of BMW isolating valves can be also adjusted by the operator from the field in order to maximize the period in which the dousing tank water inventory is available to SGs. This period is at least 23 hours, and can be extended by adjusting manually from the field the BMW valves opening. The onset of significant void in PHT due to PHT reaching saturation and start boil off is expected after 27 hours from the start of the event.

As long as the flow path will provide water to the SGs, the thermosyphoning process will ensure decay power removal and the fuel damage is not expected.

Until Class I/II depletion (in approximately eight hours), the operating team will remain in the MCR due to better monitoring, control and communication facilities. In the mean time, operator tries to start the EPS diesels.

In case of the Loss of Class IV and Class III electrical power supply, all important safety functions can be fulfilled using Class I, Class II and Emergency Power Supply system:

- Reactor shutdown occurs in the first seconds after Loss of Class IV power, by automatic action of either one of the two Shutdown Systems; the reactor shutdown systems do not require power supply for shutdown and for maintaining the reactor in a a sub-critical state indefinitely.
- The nuclear fuel cooling is ensured using the heat sinks as described above;
- Containment isolation: in case of no automatic actuation, manual box-up will be performed from MCR (using Class I batteries) or SCA (by Class I batteries if available or EPS Diesel Generators). Even if Class I batteries are not available, containment valves will fail in a close position (due to loss of electrical power or loss of instrument air) to assure containment box-up. Airlocks are available as long as the seals pressure is maintained by back-up air tanks (which are adequate for 24 hours). After depletion of the back-up air tanks, the plant design provides another supplementary supply from nitrogen bottles. A set of full N_2 bottles is in place at the Airlocks.

Containment pressure suppression can be ensured by the Dousing System. The LACs are lost since the RCW and RSW are lost following loss of Class III. For Unit 2, some of the LACs are also powered from the EPS to provide air circulation inside containment. However, containment pressurisation will not occur during these events.

Regarding the robustness of the plant, it can be mentioned that SCA, EPS and EWS are seismically qualified to DBE and are not affected by flooding. The monitoring and the control of the plant are ensured even after the batteries depletion period from SCA, using EPS. The water source for EWS system is a separate intake structure from Danube river water, DBE seismically qualified.

In case that loss of Class IV and Loss of Class III (Standby DGs) are determined by a DBE, the Class I/Class II batteries are not currently credited. Actions are in progress to increase the seismic robustness of batteries for DBE. In case the batteries are not available, EPS will ensure the power supply in less than 30 minutes, this being a situation included in the plant design.

The EPS Diesel generators can ensure the electrical power supply for fulfilment of the fundamental safety functions, even in case of an earthquake. The EPS system can also ensure the electrical power supply to instrumentation and control for safety systems (Group II and HPECC) and for monitoring of the critical safety parameters.

The case with EPS Diesel generators unavailable is considered in the scenario involving loss of all installed AC sources due to DBE, addressed in Section 5.1.3.

Cooling of the spent fuel:

The Spent Fuel Bay cooling and purification system provides cooling and purification of the water within the spent fuel bay. The Spent Fuel Bay system consists of 4 bays filled with water in which the spent fuel is discharged from the core using fuelling machine facilities. Cooling and purification for these bays is ensured by 3 Class III power pumps, 3 heat exchangers, 2 purification lines with filters and ion exchange columns, associated piping and instrumentation.

The SFB is a pool which is stainless steel plated in Unit 2 and epoxy liner covered in Unit 1. All Spent Fuel Bay s are filled with demineralized water that is circulated with the provision of pumps in order to cool the spent fuel stored on fuel racks inside the pool. The heat transferred from the spent fuel is removed by plate type heat exchangers using RCW - recirculated cooling water (in Unit 2) or RSW - raw service water (in Unit 1). The water in the Spent Fuel Bay also provides shielding from radiation.

The Spent Fuel Bays and associated piping are DBE seismically qualified in order to preserve their integrity after an earthquake.

Loss of off site power and loss of the ordinary back-up source on the spent fuel bay will lead to loss of cooling with service water (RCW/RSW) in SFB and this situation is addressed in Section 6.4.2.

5.1.3. Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources

Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up AC power sources is referred to as Station Blackout. This represents a beyond design basis case scenario, which involves, for Cernavoda NPP, the loss of External Grid, Stand-by Diesel Generators and Emergency Diesel Generators. This scenario is analysed considering that the batteries are available. In this scenario, the nuclear safety functions will be ensured as follows:

Shutting down the reactor and maintain in a safe shutdown condition: The reactor is shutdown immediately after the loss of Class IV power. All the indications and logic

of SDS1 and SDS2 are supplied from Class I and II electrical power. Both shutdown systems are designed "fail safe", and thus they are effective in performing their function in case of Class I and / or Class II power failure.

Containment integrity: Having the batteries available, the containment can be also isolated by the operator. Furthermore, the containment isolation valves will fail close (fail safe on loss of power or loss of instrument air).

Providing a heat sink for reactor fuel cooling;
Regarding the reactor cooling safety function, this scenario is similar with that one presented in Section 5.1.2, for the first 30 minutes after the loss of Class IV power.

The main focus here is to maintain the thermosyphoning by ensuring:

- Boiler pressure at lowest practical value by opening and then blocking open 4 MSSVs;
- Boiler make-up: BMW (medium term) followed by EWS (long term);
- PHT make-up: HPECC (medium term) followed by EWS supply on long term.

All these functions can be performed either with battery supply or mobile diesel generators supply.

As it was presented in Section 5.1.2, the SG will be depressurized, manually or by automatic action, in less than 30 minutes by opening MSSVs.

The SGs level control can be performed manually from the MCR if the SBO is not seismically initiated or from the SCA (DBE seismically qualified) by opening BMW isolating valves. These valves can be also manually operated from the field in order to control the SGs water level, by adjusting the valves opening. The entire inventory from the dousing tank (about 2000 m^3) will ensure more than four days for decay power heat removal from the PHT system, if the dousing tank water inventory is used in a controlled manner.

SG depressurisation will cause a rapid depressurization of the PHT system, the complete draining of the pressurizer due to shrink and will trigger automatic loop isolation. The primary inventory make-up can be performed using ECC high pressure injection.

During this time, the operators will attempt to restore EPS in order to start EWS pumps to ensure the long-term heat sink. If EPS cannot be recovered, the plant procedures are directing the operator to use the mobile Diesels generators. Once the mobile Diesel generators are operational, the EWS pumps can be used to provide water supply to SGs. Mobile Diesel generators have autonomy of 6 hours and the fuel oil from EPS fuel tanks can ensure their operation for at least 5 days. If EWS system can not provide water to SGs, the fire water trucks will be used to provide water directly to the SGs through the EWS pipes.

As long as water is provided to the SGs, the thermosyphoning process will ensure decay power removal and the fuel damage is not expected.

Critical safety parameter monitoring. During the SBO, the critical safety parameters can be monitored from MCR or SCA, having instrumentation and control supplied by

batteries, in case the SBO is not determined by DBE. If the MCR is not available or after the depletion of the batteries, the plant monitoring can be performed from SCA, using mobile Diesel generators (connected within 2.5-3 h from the event initiation).

Regarding Spent Fuel Bays cooling, SBO will lead to loss of cooling with service water (RCW/RSW) in SFB and this event is covered in Section 6.4.2.

Seismic induced loss of all electrical power supply (all AC and DC sources) was also assessed by the licensee, based on specific safety analyses performed for the plant. In this case, the unavailability of the batteries is also considered, besides the SBO.

Since there is a direct effect on the primary and secondary side behaviour, two different cases were considered:

a) The steam lines break at the weak-link location;
b) The steam lines do not break at the weak-link location.

According to the seismic qualification for the steam lines (secondary circuit), in order to ensure that the SGs heat removal capacity is ensured, the end of the steam pipes (outside reactor) that are part of the first welding after the MSSVs are specially designed and mechanically processed. This place becomes the “weak-link” - the point expected to break following a DBE. In the same time, the PHT system is designed leak tight, so only very small leaks may be expected (via PHTS seals, packing and gaskets).

a) The case when the steam lines break at the weak-link location is the most likely event during an earthquake, since "weak links" were designed to break following a DBE. The SGs secondary side will be immediately depressurised, as a result of the weak links break. Since the instrument air and electrical power are lost, the BMW pneumatic isolating valves will fail open and will provide water to SGs secondary side when SGs pressure decreases below 345 kPa (g).

The BMW pneumatic isolating valves can be operated manually from the field in order to control the SGs level. The minimum available demineralised water inventory from the dousing tank (about 2000 m^3) will gravitationally flow to ensure the SGs water inventory. As it was already presented, this inventory will last at least 23 hours, even in the case the BMW valves fail open. The onset of significant void in PHT due to PHT reaching saturation and start of boil off is expected after 27 hours from the start of the event. The control of these valves can be done manually from the field or from SCA. If the entire inventory from the dousing tank is used in a controlled manner, this inventory will ensure at least 7 days for decay power heat removal from PHT system.

Even if the HP ECC is not available (because the batteries are assumed to be lost), as long as the boilers are available, thermosyphoning can be credited and will ensure the decay power removal. For low SGs secondary side temperatures, the two-phase thermosyphoning cooling process is effective up to 60% void in the PHT system.

The void in PHT system is generated due to PHTS coolant shrinkage from normal operating conditions to low temperature conditions (determined by the secondary side depressurisation) and by PHT system leaks following DBE (which are estimated via PHT seals, packing and gaskets). These are small pipes and the 60% void is expected only after 29 hours after the event.

The EPS diesel generators are expected to be available even for an earthquake that will produce a PGA of 0.4 g. In the improbable case that the seismic even exceeds this limit, the mobile diesel generators can be used to provide power supply. The time needed to establish the connections, including the time for the access of the mobile equipment from the storage location to the areas designated for use, is of 2.5 - 3 hours. According to actual operating procedures, the operator is required to start the mobile diesels generators in case of the seismic induced loss of all power sources (SBO). Consequently, the safety parameters monitoring can be performed from SCA, which is DBE seismically qualified and also has a seismic capacity of 0.4 g (See Chapter 2). After the electrical power from mobile diesel generator is available, electrical power to HP ECC and EWS pumps is restored and make-up to PHTS is performed, the single-phase thermosyphoning can be credited.

The autonomy of mobile Diesel generators and the on site reserves of fuel oil that can be used by these mobile Diesel generators after a seismic induced SBO (EPS fuel tanks), have been previously mentioned.

If the mobile diesel generator is not available, the operator is instructed to ensure the heat sink by connecting the fire trucks directly to the EWS pipes.

b) In the unlikely case when the steam lines do not break at the weak-link location, the SGs will remain pressurised. The main focus is to maintain the long term thermosyphoning by ensuring:

- Boiler pressure at lowest practical value by opening and then blocking open 4 MSSVs;
- Boiler make-up: BMW (medium term) followed by EWS (long-term) powered by mobile diesel generators or fire water trucks connected to EWS lines;
- PHT make-up: HPECC (medium term) followed by supplying water from dousing tank (long-term).

The above safety functions can be ensured using mobile Diesel generators supply since Class I/II batteries are not credited in this event.

If the SGs secondary side remains pressurised (assuming the steam lines remain all intact after the seismic event and the MSSVs are not opened) the PHT system will be also pressurised. The loss of D_2O feed pumps and pressurizer HTRs on loss of Class IV and loss of Class III will cause PHT pressure and pressuriser level to decrease slowly - due to the heat losses, and also a very small amount of leakage from PHT (e.g. from PHT pump seals) may occur. Continuous inventory depletion from the boilers will cause the boiler levels to decrease continuously. The initial water stored in the SGs secondary side ensures adequate heat sink for at least 2 hours, heat being removed by simmering opening of the MSSVs (acting as spring valves).

As long as SGs contain water, the PHT cooling will be ensured by single or two-phase

thermosyphoning or by the intermittent buoyancy induced flow (IBIF) when the SGs inventory is depleted. As mentioned previously, there are 2 hours until the SGs get dry following a seismic induced loss of all electrical powe supply (AC and DC). Starting from this moment, there are 2 more hours (for a total of about 4 hours since the event initiation) until at least one fuel channel gets dry. Because the mobile DGs can be installed within 2.5 - 3 hours, there is time to connect the mobile DGs until PHT is affected. The Mobile Diesels and necessary cables are stored in protected locations on site.

With the MSSVs manually opened, the water from the dousing tank will makeup the SGs secondary side inventory (through the BMW isolating valves, failed open) and ensure the reactor cooling by thermosyphoning and IBIF until mobile Diesel generators become available and allow HPECC injection for PHT inventory recovery. Once the SGs are depressurised, the inventory from the dousing tank could provide an additional time frame of 23 hours for core cooling.

Besides the reactor cooling safety function, the other safety functions (reactor trip, containment isolation) are fulfilled as described in the case of SBO with batteries available.

In case of seismic induced SBO, the monitoring of the critical safety parameters is possible from SCA only when the electrical power is supplied by mobile Diesel generators. The operator is instructed to try to start the seismically qualified EPS to reduce the period without power supply to the plant systems.

Cliff-edge effects

The cliff edge effect for the scenarios in this section is defined by the loss of inventory to all SGs as a heat sink.

If Class I/II power is available, eight main steam safety valves (MSSVs) will be open based on the auto-depressurisation logic long before the dry-out condition allowing timely make-up of water to the steam generators. The MSSVs must remain block open for long term cooling. After the SGs are depressurised, gravity-fed make-up from the dousing tank is available through the BMW pneumatic valves in the medium term. If the PVs are uncontrolled, the time available for the make-up inventory for decay heat removal from PHT is 23 hours. If the BMW PVs are controlled, more than 4 days are available. In the long term, mobile diesels can power the EWS pumps supplying water to the SGs until the EPS power is restored.

In case of seismic induced SBO, when the batteries are considered unavailable, for the scenario when steam lines do not break at weak-link, in the short term the steam generators have an inventory that will last for about 2 hours. However, there are about 4 hours from the initiation of the event until dry-out conditions are expected in one of the PHTchannels. The fuel channel failure can occur if the necessary electrical power is not supplied to those systems which can ensure the reactor cooling. As the mobile Diesel generators can be installed in 2.5-3h, this cliff edge can be avoided. Furthermore, even though this scenarios is highly unlikely (a seismic event which disables EPS Diesels having a seismic capacity of 0.4g but causes all the steam lines which are designed to break at the “weak link” for a 0.2g DBE to remain intact), in

order to prevent the cliff edge conditions from occuring, the licensee proposed some plant modifications, the implementation of which is already in progress, to:

1. Provide different modes of MSSVs opening (manual or pneumatic from the field, pneumatic from SCA using a dedicated battery), to ensure SGs depressurisation in case of SBO and batteries unavailable;
2. Increase the seismic robustness of the batteries;
3. Replace the two 880 kW, 0.4 kV mobile Diesel generators with others 2x1MW (to cover entirely the EPS loads), more versatile as they can supply also 6KV loads supplementary to 0.4 KV loads.

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

Loss of the off-site power to one or both units of Cernavoda NPP may lead either to the operation of the unit in the island mode or to the loss of Class IV power, depending on the specific conditions. An automatic start of Stand-by diesel generators is expected in less than 3 minutes (2x100% for Unit 2 and 4x50% for Unit 1). The manual start of these diesels is also possible, if the automatic start doesn't work. If these standby diesel generators are not available, the emergency diesel generators (2x100% for each unit, DBE qualified) will be started by the operator.

If all these supply alternatives fail, the plant has batteries for DC and AC power supply, which enable, together with accident management measures, the capacity to remove the residual heat. Also, two mobile diesels are available on site for both units in order to back up the emergency diesel generators and to supply EWS and other selected pumps/components.

In summary, it can be stated that, the current robustness and maintenance of the plant is compliant with its design basis against loss of electrical power. Also, taking into account the operating procedures and accident management measures, the plant units have a high level of defence against the loss of power and its consequences.

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

Taking into account the margins demonstrated for safeguarding the power supply as indicated and also considering the events imposed (earthquake, flooding and extreme weather conditions), the licensee determined that it is no need for measures to further increase the robustness of the plant in case of loss of electrical power supply, other than those mentioned in Section 5.1.3.

As it was mentioned, for the future, the licensee intends to improve some of the accident management measures and provide supplementary means by replacing the existing mobile Diesels with other two mobile Diesel generators (2x1MW) which are more versatile, by supplying also 6 KV loads in addition to 0.4 KV. Also, the station will provide different modes of MSSVs opening (manual or pneumatic from the field, pneumatic from SCA using a dedicated battery) and will increase the seismic robustness of the batteries.

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

The licensee evaluated the progressive effect of loss of heat sinks. For a better understanding of the effect of these losses on the safety functions, a short description of the normal, abnormal and emergency provisions for reactor cooling is presented below. A schematic diagram of the heat sinks is presented in Fig. 5.2.

For the CANDU reactor, the heat from the nuclear fuel is transferred to the heavy water in the primary heat transport system. The heat from the heavy water could be transferred to either SGs using PHT pumps (reactor at power or in shutdown conditions) or using natural circulation (thermosyphoning) or to RCW/RSW systems by the Shutdown Cooling (SDC) system using PHTS / SDC system pumps (for shutdown conditions).

The primary Ultimate Heat Sink (UHS) is based on decay power heat removal using forced cooldown circulation in PHT system. The preferred way to remove the decay power is using the shutdown cooling system (SDCS). SDCS transfers the heat to the Recirculating Cooling Water (RCW) system through its heat exchangers. The RCW system transfer the heat to the Raw Service Water (RSW) system through its heat exchangers and the heat is dissipated into the Danube River. Raw service water is taken from the suction bay. After reactor shutdown, the SDC system represents the main system of the primary UHS. The system is interconnected with the PHT system by opening the isolation motorized valves.

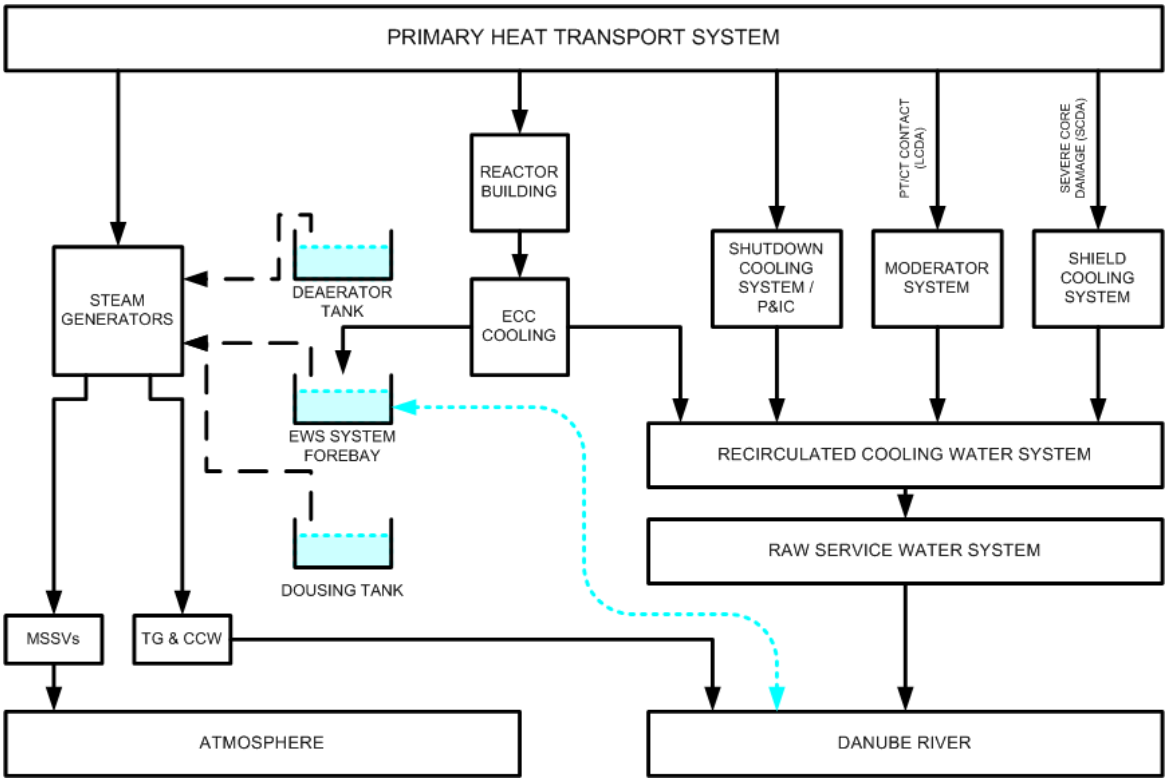

Figure 5.2 Schematic of Core Heat Sinks in CANDU 6

The alternate heat sink is used only if the primary UHS is not available (forced circulation through the core is lost). If the forced decay power removal is not available, the natural circulation in the primary circuit can be used while de energy is removed to the secondary side or to the atmosphere through the SGs.

During the use of the alternate heat sink chains, the difference in fluid density between the SG cold leg and hot leg ensures continuous flow circulation through the PHT circuit by natural circulation (thermosyphoning process). The SGs secondary side inventory is transformed to steam, which is released to atmosphere through the ASDVs or MSSVs.

When the alternate heat sink is used, two redundant and different paths can be used to provide cooling water to the SGs secondary side.

a) *Alternate heat sink: Demineralised water provided by "feedwater train"*

The "feedwater train" needs demineralised water provided by the Water Treatment Plant (WTP) by a process that requires off site electrical power available, the Condenser Cooling Water (CCW) system (Class IV electric power) or Back-up Cooling Water (BCW) system (Class III electric power) available. The demineralised water produced in the WTP is transferred to the Emergency Water Tanks and then to Condensate Storage Tank using the Class III demineralised-water transfer pumps.

The water from the Condensate Storage Tank flows gravitationally to Condenser hot wells or to the Deaerator tank once the isolation valves are open. Also, the demineralised water can be provided to the Deaerator from Condenser hot wells using the Auxiliary Condensate Extraction pump (Class III electric power) and from Deaerator to SGs using Main Feedwater pumps (Class IV power) or AFW pump (Class III power, EVEN bus).

As a result of the heat transferred to SGs secondary side from the primary side, the water will be transformed to steam and released to atmosphere using the ASDVs and MSSVs. In case the CCW system is available (Class IV available), the CSDVs and the turbine condenser can be used to condense the steam from SGs.

The level in the SGs is controlled by level control valves. The Auxiliary Feedwater (AFW) pump cooling is normally provided by RCW system. In case that the AFW pump cooling is lost, the back-up cooling flow can be provided by water inventory from the Condensate Storage Tank. The water is supplied gravitationally to AFW pump from Condensate Storage Tank or from Emergency Water Storage tanks using the demineralized water Class III pumps.

The AFW can provide to SGs a flow larger than the flow required for cooldown purposes after reactor trip, ensuring that a continuous heat sink is available as long as the AFW pump is running.

b) *Alternate ultimate heat sink: Water provided by BMW system / EWS system)*

Water supply from BMW (dousing tank inventory):
In case that MFW or AFW pumps cannot provide water to SGs secondary side, the alternate source of demineralised water that can be provided to SGs secondary side is represented by the water inventory available in the dousing tank. The water will flow gravitationally to the SGs secondary side once the BMW pneumatic isolating valves are open and the SGs are depressurised to atmospheric pressure. The BMW pneumatic isolating valves will be opened automatically by the autodepressurisation logic or can be operated manually from SCA or manually from field in order to control the SGs level. As long as the flow path will provide water to SGs, the thermosyphoning process will ensure adequate decay power removal. The large dousing tank inventory will ensure, for at least 23 hours, water supply to SGs secondary side, if the flow through BMW valves cannot be manually controlled, or for 7 days if this flow is controlled.

Continuous water flow to SGs by EWS

Once the demi-water inventory is depleted, the SGs secondary side will be directly provided from suction bay (Danube River) by 2 x 100% Emergency Water Supply pumps (for each unit) and injected in the SGs secondary side.

As long as cooling water will be provided to SGs, the fluid density difference between the hot leg and the cold leg in the SGs will promote the natural circulation (thermosyphoning) through the PHT system. Part of the water provided to SGs secondary side will be released to atmosphere (as steam) through open MSSVs. The electrical power supply for the MSSVs opening is from Class I or from EPS or from mobile diesel generators. Also MSSVs can be manually blocked open as per design.

Each EWS pump is powered by any EPS diesel generator (as per design) and by the mobile diesels. The intake for the EWS pumps from suction bay is separated from the intake for the RSW pumps. Also a separate suction pit is provided by design for EWS pumps.

As long as the EWS will provide water to SGs and enough inventory exists in the primary circuit (this can be also recovered by EWS), the thermosyphoning process will ensure adequate decay power removal.

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

Loss of the primary ultimate heat sink can be caused by:

- earthquake (case covered by Section 5.3, with details about systems' seismic capacity presented in Chapter 2);
- severe water level reduction in the suction basin due to very low level in the distribution bay / intake channel blockage or severe blockage of the filtering chains.
- other causes.

For the loss of primary Ultimate Heat Sink (UHS) caused by any other event, except for earthquake, Class IV off site electrical power is assumed available. In addition, the Class III, Class II and Class I are assumed available because they are supplied from Class IV power busses.

Loss of primary UHS consists basically in loss of the water in the RSW intake or loss of RCW/RSW systems (active or passive components). Electrical power is assumed available. The loss of primary UHS caused by SBO is presented in Section 5.3.

The licensee presented in the report the design provisions considered to prevent the loss of primary UHS. These are referring mainly to the robustness of the design of RCW/RSW and SDC systems, as:

- the independent and redundant trains of RCW/ RSW and SDC systems (shutdown conditions: 4 x 100% RCW and 4 x 100% RSW pumps; 2 x 100% SDC pumps - 6 hours after plant shutdown);
- the minimum suctions intake water level in the suction bay for the RSW pumps is adequate for extremely low river levels (the frequency of reaching 1.45mBSL Danube river level is less than 10^{-4}/year) and in addition, the RSW pump rotor of one RSW pump for each unit has been further lowered;
- existence of the two independent flow paths available to discharge the water used for cooling purposes for each unit;
- independent electrical power supply on Class III (different buses for different trains) of RCW/RSW and SDC systems;
- automatic start of pumps and selective loading of important consumers in order to ensure the fulfilment of the important safety function;
- other provisions to increase the reliability of RCW/RSW and SDC systems (alternatives for valves' opening and pumps' cooling, suction bay water level monitoring instrumentation, etc);

The evaluation presented the operational provisions, regarding mainly the operating procedures prepared for the plant to cope with different loss of flow conditions in order to prevent the loss of reactor cooling safety function but also the administrative measures taken in order to increase the confidence in this UHS. The main provisions refer to:

- Specific Abnormal Plant Operating Procedures prepared in order to guide plant operation following loss of Feedwater, loss of Raw Service Water events and for a generic loss of heat sink;
- Prognoses of Danube water level evolution (daily and for long term)
- Independent and reliable instrumentation provided for each unit for suction bay water level monitoring.
- Abnormal Plant Operating Procedure for the cases when the Danube water level decreases to very low levels. The operating units are directed to be shut down well in advance of the RSW becoming unavailable, in order to reduce the heat load that has to be removed from the core. A set of supplementary measures to enable the alternate water sources (domestic deep underground water wells, firewater) are included.

Regarding the provisions of the plant to avoid *the blockage of flow to the intake,* a special study ("Intake Cooling Water Blockage") was recently performed in response to specific WANO SOER reccomendations. One of the recommendations was: "Periodically (at least once every two operating cycles) review the capability of intake screens, screen wash systems, related system strainers and heat exchangers to perform their design functions during all credible scenarios identified from both the risk assessment carried out under recommendation 1-a and plant and industry operating experience associated with intake cooling water blockage". The licensee analysed all scenarios that can lead to the blockage of the intake (as required), the provisions available to deal with this recommendation and concluded that the present provisions are adequate to fulfil these requirements. A set of 2 filtering lines (trash rack and traveling water screens) are provided at the intake of the RSW pumps. The power supply to these equipments is provided from class III. The screens are started automatically on detection of a high differential level, also at specific time intervals in response to a master timer or by local operator control.

Another recommendation in the WANO SOER was: "Periodically (at least once every three years) inspect the material condition of the intake structure, equipment and associated systems and perform preventive maintenance on key equipment such as trash bars, fixed screens, traveling screens and screen wash systems, cathodic protection systems, heat trace systems, level instrumentation and chemical treatment equipment. Protective features, such as pump and screen trip and traveling screen start circuitry and instrumentation, should be periodically tested. Inspection intervals should be timed to ensure required maintenance is performed prior to those times when intake blockage risk is the highest". The licensee has also analysed this recommendation and concluded that the present provisions are adequate. In this order, there were verified all provisions related to the cleaning of the intake, periodically dragging of the channel, preventive cleaning of the shores, the physical barriers provided to block the entrance of foreign materials to the intake channel, prevention of oil spreading into the intake, etc.

The primary UHS is not seismically qualified to remain operational in case of a DBE, even if the SDC system and large portions of RCW/RSW systems are qualified to preserve their integrity in case of DBE.

In case one of the components of the primary UHS becomes unavailable, the heat removal from the core will be achieved through the alternate heat sink or alternate UHS, which relies on thermosyphoning process as described above.

The alternate UHS is the reliable path for the transfer of the heat produced by the reactor in case the primary UHS is not available, especially in case of an earthquake.

Related to the robustness of the provisions of alternate UHS in connection with earthquakes, it can be specified that:

- The dousing tank is a large water reservoir situated at high elevation in the R/B and can be used by the Boiler Makeup Water system to recover the water level in SGs when primary UHS or "feedwater train" are unavailable (as well as dousing spray and the medium pressure ECC). Dousing tank, pipes, valves and instrumentation and controls related to dousing system are all seismically

qualified to DBE.

- The BMW and EWS systems, including the valves, their controls and the instrumentation and control equipment, are seismically qualified to DBE. The intake of for the EWS pumps from suction bay is seismically qualified and separated from the intake for the RSW pumps: a separate suction pit is provided by design for EWS pumps.
- Each EWS pump (2 trains x100% for each unit) is electrically supplied by a EPS diesel generator, seismically qualified to DBE. EPS supplies also all valves necessary to bring the EWS water for make-up to the PHT loops and the SGs for decay heat removal. EWS system can be operated from SCA, also seismically qualified to DBE. EWS pumps and valves can be also supplied by the mobile Diesel generators.

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

During normal plant operation, overall there are small differences between the case the RSW system is lost versus the case when RCW system is lost. Due to delayed heat-up of the RCW system (large water inventory) after a loss of RSW, the operator has more time to respond compared to the case when RCW is lost. Conservatively, loss of RCW is described / presented for the loss of primary UHS.

The loss of primary UHS was taken into account in safety analyses, as part of the plant design (Loss of Feedwater, Loss of SDC cooling, Loss of RCW).
Following a loss of RCW, the main safety functions:

- reactor shutdown
- reactor cooling
- containment integrity
- plant monitoring

are ensured as it is described below.

Reactor shutdown safety function:
The functioning of SDS1 and SDS2 are completely independent of RCW or RSW. Therefore, these systems are not affected by an impairment of RCW / RSW systems. Following loss of RCW cooling flow, the temperature in the moderator system (the main load for RCW system) starts to increase; this will determine the automatic reactor power reduction by a setback or by a reactor trip. The reactor power reduction will be also performed by the operator based on temperatures increasing above the determined limits (in RCW system or moderator temperature).

Reactor cooling safety function:
Following a total loss of RCW and consequent reactor power reduction, the cooling by SDC system is not possible and the operator will take the necessary actions to use the alternate mode to cooldown the PHT system using “feedwater train”, by thermosyphoning process. The operator strategy, in parallel with any attempt to restore recirculated cooling water, is to cooldown the PHT system.

Early cooldown of the Heat Transport System is achieved by:

- controlled depressurization of the secondary side of the SGs to atmospheric

conditions (by ASDVs or MSSVs opening),
- maintaining SGs level,
- maintaining pressurizer level and PHT system pressure within acceptable ranges.

After the loss of RCW system, the operator will take specific actions as directed by a dedicated Abnormal Plant Operating Procedure (APOP). The main PHT pumps should be tripped if no D_2O feed pump is running, due to the loss PHT seals gland cooling. As the forced cooldown feature is lost, the PHT system circulation will rely on thermosyphoning process.

If the PIC system became unavailable or the subcooling margin has decreased below 10^oC, the operator will initiate HPECC injection for PHT system pressurization and subcooling margin increase.

The operator will open MSSVs and the steam produced in the SGs secondary side will be released to atmosphere. On the secondary side, boiler level and pressure will be controlled, either by control programs from DCC or by operator action. In order to maintain SGs level, the operator has to ensure feedwater to SGs.

Alternate heat sink: Due to loss of RCW, the cooling will be lost for both the MFW pump and AFW pump. The back-up cooling for MFW pump is provided by Fire Water system. If the cooling cannot be restored from Firewater system to none of the MFW pumps, the only available pump is the AFW pump. The water to SGs secondary side is supplied by AFW pump considering its normal suction from the Deaerator tank or from the condensate storage tank. The Deaerator tank is supplied with water from the condenser by the auxiliary condensate extraction pump or gravitationally from the Condensate storage tank if the Deaerator is depressurized. The Condensate storage tank is supplied with demineralized-water from the Emergency water tanks using the emergency water pumps. The Emergency water tanks are supplied with demineralized-water from WTP. The WTP can produce continuously demineralized-water as long as the off site power is available and the WTP is supplied with fresh water by any of the CCW or BCW systems (water taken from suction bay). Demineralised water is produced in Unit 0, which supplies with demi-water both Unit 1 and Unit 2 of Cernavoda NPP.

If the WTP becomes unavailable, for each unit at Cernavoda site, the minimum stored demineralized water inventory in different tanks and reservoirs in the "feedwater train" will ensure a continuously flow to SGs for at least 4 days. In case one of the emergency water tank was isolated for maintenance prior to event, the stored inventory ensures 3 days.

Alternate UHS: If for some reasons the AFW pump becomes unavailable, the operator is instructed to use the demineralized water from the dousing tank, through the BMW system. Following opening of the pneumatic isolation valves, the dousing water inventory can be used to provide water to SGs secondary side. The entire dousing tank inventory will ensure a continuous heat sink for at least 7 days. During this time, the operator will ensure that the open path to atmosphere is available (open and block open the MSSVs) and will control the SGs levels (successive close / open cycles for BMW pneumatic isolating valves).

If the loss of service water cooling occurs when the plant is in outage shutdown state, there is no need to trip the reactor since the Guaranteed Shutdown State (GSS) is already ensured according to outage operating procedures. In case that the loss of service water occurred at low power conditions, the SDC system cooling will be lost. The PHT core cooling will be provided by thermosyphoning process as long as the heat transfer to secondary side is ensured in the same manner as presented for the case from full power conditions.

If the “feedwater train” is not available to ensure the water to the SGs, the dousing tank inventory and BMW pneumatic isolating valves will be used to maintain the water level in SGs. EWS system is also a back-up. If the RCW is lost during the plant outage, the specific operator actions needed to be performed, in order to ensure the reactor cooling, are included in the Outage Heat Sink Operating Manual.

Containment safety functions and monitoring of the plant are not affected by the loss of RCW/RSW and the operations can be performed from MCR (and also from SCA, if needed).

Loss of primary UHS (RCW/RSW systems) will lead to loss of cooling to spent fuel bays. This event is covered in Section 6.4.2.

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

This event sequence corresponds to the complete failure of the Main Feedwater (MFW) pumps, the Auxiliary Feedwater (AFW) pump (or the unavailability of the "feedwater train"), the loss of SDC or RCW/RSW systems, as well as loss of the two trains of EWS system. This scenario is beyond the design of the Cernavoda NPP units. In this case, the heat removal via the SGs to the atmosphere can be ensured by thermosyphoning following the manual operator actions and accident management measures.

Reactor trip, plant response and operator actions following loss of RCW/RSW (primary UHS) were presented in section 5.2.2. Following loss of primary UHS, the operator is directed to use the alternate heat sink that consists of thermosyphoning process (use of AFW and demineralized-water inventories which will last for more than 3 days after reactor shutdown). A detailed procedure for loss of feed water from full power conditions is used by the operator in such conditions.

In case of loss of all heat sinks from full power conditions, with the plant operated from MCR (in case that MCR area is still available) or from SCA (in case the support systems are lost), specific generic procedures (symptom based generic APOPs) have to be used by the operator.

For the actual scenarios, RCW/RSW, Feedwater and EWS systems are assumed lost. Since the operator cannot restore any of the available heat sinks, the dousing tank inventory will be used to supply water to SGs secondary side. The decay power will be transferred to SGs secondary side. The difference between the SGs primary side cold leg and the hot leg densities will ensure continuous circulation in the PHT system. The steam produced in the SG secondary side will be released to atmosphere

through open paths (open MSSVs). In the long term, the operator has to mechanically block open the MSSVs to ensure an open path to atmosphere for the steam produced in the SGs. The primary side inventory and pressure control will be restored (following loss of PIC system) by using the HPECC water inventory (at least 170 m^3).

Due to loss of both primary and alternate UHS, the water supply to SGs secondary side can be ensured only by the water available in the dousing tank inventory (about 2000 m^3). The flow from the dousing tank will be provided to the SGs secondary side through one the BMW pneumatic isolation valves open. These valves can be controlled from MCR, SCA or manually from the field.

If the entire inventory from the dousing tank is used in a controlled manner, this inventory will ensure at least 7 days for decay power heat removal from PHT system.
In case that the SGs level cannot be controlled (by the opening and closing of BMW valves), the dousing tank inventory will ensure a continuously heat sink for at least 23 hours.

The PHT coolant reaches saturation and start to boil off after 27 hours from the start of the event. However, this is an unlikely situation since the BMW pneumatic isolating valves can be manually operated from the field.

As per design, following loss of heat sink, the EWS should be used to provide water to SGs (as the alternate UHS). However, for the actual scenarios, the EWS is assumed lost. The EWS can be lost due to loss of power supply, due to loss of EWS pumps / building or due to low level in the suction bay.

In case that the EWS is lost due to loss of power supply, the operator is instructed to use the mobile diesel generator for each unit to provide electrical power supply. The mobile diesel generator can be installed within 2.5 up to 3 hours for each of the operating units. The generator can be connected directly to the EPS buses or can be connected directly to the EWS pump local junction box. Following EWS pump electrical power supply restoration, it can be used as per design to provide water to SGs secondary side and to PHT system (if required). As long as sufficient cold water inventory exists in SGs secondary side, the thermosyphoning is working and the decay power is removed from PHT system. The mobile Diesel generators have autonomy of 5-6 hours. After this period the diesel fuel from EPS tanks will be used.

The fire water trucks will ultimately be used for SG make-up via EWS lines in case none of the other heat sink chains is available. A set of specially built connectors will be installed in the EWS lines in this case (only if the other sources of water are unavailable). The fire water truck can be supplied with water from the fire water tanks (2 x 1500 m^3) or water from the domestic water system.

In addition to the motorized pump provided by design, the fire tanks can be supplied with water from the suction bay using a mobile diesel driven pump. The procurement of the mobile Diesel pump is already contracted by the station.

The water make-up in the domestic water system is provided by two pumps. Running together, the pumps can provide 90 m^3/h. The pumps are powered from off site electrical power connection. Separate procurement related actions have been started to

get two mobile diesel generators for electrical power supply to these two pumps.

Loss of the alternate UHS after a loss of the primary UHS following a gradual water level reduction in the suction bay to less than 1.21 mBSL (the frequency of reaching this Danube river water level is far less than 10^{-4} /year) is a slow process which allows the licensee to apply the specific APOP. The procedure presents the operator actions for different water levels, including a case when the level decreases beyond the EWS minimum water level.

Considering the water level reduction in the suction bay, the procedure call for successive units shutdown, at least 3 day before the water level is expected to decrease to a level that inactivate the RSW system pumps (final component of the primary UHS). The units will be shutdown successively in an orderly manner. Once the entry conditions for this procedure are fulfilled, a set of manual connections using fire hoses are installed, ready to be used. The cooling water for Unit 1 RCW system is provided by the fire water system and by the domestic water system. The cooling water for Unit 2 RCW system will use fire water only. These connections are already installed in Unit 1 while for Unit 2 they are effectively activated only when the loss of RSW is imminent.

Loss of primary UHS (RCW/RSW systems) and alternate UHS will lead to loss of cooling to spent fuel bays. This event is covered in Section 6.4.2.

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

For the Cernavoda NPP diverse heat sinks are available to ensure heat removal via the SGs to the atmosphere. Adequate systems are installed for the SGs feeding, which do not require service water. A back-up to alternate UHS can be also assured by additional operator actions, i.e. by installation of flexible tube connections for, or by the accident management measures to ensure the residual heat removal in all plant operational states.

Section 5.2.1 presents the special design and operational provisions of the plant to prevent the loss of the primary UHS as well as the alternate UHS, considering all identified causes that can determine this unavailability.
The complete loss of the UHS can be coped with in Cernavoda NPP without leading to fuel failure.

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

Following Fukushima event, station ensured the availability (on site) of two mobile Diesel generators. Also, station initiated the procurement process for acquisition of a mobile diesel engine driven pump, for fire water tanks make-up or fire water truck supply. Also, 2 electrical mobile submersible pumps are already available on site.

Separate procurement related actions have been initiated to get two mobile diesel generators for electrical power supply to the two pumps that can provide water in the domestic water system from the deep wells which are available on site. The procurement has been already contracted by the station.

5.3. Loss of the primary ultimate heat sink, combined with station black out (see stress tests specifications).

According to the ENSREG specifications, loss of Ultimate Heat Sink followed by a Station Blackout event was requested to be evaluated. This sequence is an extremely low probability event. This review will provide information regarding the safety margins and the improvements that have to be considered, if necessary, in order to reduce or remove the identified cliff-edge effects.

The effect of a SBO event alone, (regardless of the intake water level), will result in a sudden loss of the equipments that are providing / using the primary UHS (main water intake pump station – CCW / RSW / BCW / Fire Water Make-up systems). According to the SBO definition, all the Class IV and Class III systems and the EPS system will be lost. As a consequence, the AFW pump will be lost and no demineralized-water supply from the "feedwater train" will be available to SGs secondary side. Since the SBO includes loss of EPS (and consequently loss of EWS pumps), the only source of water to SGs (as per plant design) will be water from the dousing tank.

Taking into account the above mentions, the event of loss of the primary ultimate heat sink, combined with station black out is covered by the station blackout (Section 5.1.3).

The reactor shutdown and containment isolation: these safety functions are performed from the begining of the event, by automatic or operator actions or by their fail safe characteristics, as it is described in Section 5.1.3.

The reactor cooling function:
After the loss of primary UHS and SBO, the PHT circuit inventory and pressure control will be restored (following loss of PIC system) by using the HPECC water inventory (at least 170 m^3). Since there is no inventory loss from PHT system, the pressure will be maintained by the HPECC gas tanks.

In order to ensure the HPECC injection (and the SGs secondary side depressurization), the operator is instructed to open and mechanically block open 4 MSSVs. The MSSVs can be open since Class I electrical power is available. Until the SGs manual depressurization is done, the pressure in the SGs secondary side will be maintained at high values.

The MSSVs will respond to pressure increase by staggered opening and pressure release. If the operator actions to open the MSSVs is delayed, SGs autodepressurization will occur in less than 30 minutes, when specific conditions are fulfilled by opening 8 MSSV's. SGs depressurization in turn will cause a rapid depressurization of the PHT system and manually or automatic HPECC injection, as long as Class I and Class II are available.

The available reserve of demineralized-water for this sequence in the short term will be the water available in the dousing tank. The water from the dousing tank can be provided to SGs through the BMW system following opening of the BMW pneumatic isolation valves. The operator can control these valves from MCR or from SCA or

manually from the field. The entire dousing tank inventory ensures at least 7 days for decay power heat removal. Even for the conservative case when the flow from the dousing tank cannot be controlled at all (pneumatic valves fail open), the water could ensure a heat sink for at least 27 hours, when it is expected the onset of significant void in PHT due to PHT reaching saturation and start boil off. However, this is an unlikely situation since BMW pneumatic isolating valves can be manually operated from the field.

Cernavoda NPP can also use the two Diesel generators of 880 kW (one for each of Unit 1 and Unit 2), already on site, that can supply the 380 VAC EPS buses and the EWS pumps. The minimum time to install these mobile DGs is within 2.5 up to 3 hours. The minimum time provided by the water available from the dousing tank will ensure enough time for the operators to install the mobile DG's. EWS can provide water to SGs secondary side. The thermosyphoning process will ensure decay power heat removal with steam release to atmosphere through open MSSVs.

The station response following loss of primary UHS event coincident with – or generated by – a SBO, is based on the EWS system, being powered from the mobile DG's. For the unavailability of the EWS system (loss of alternate UHS), the response has been presented in Section 5.2.3 and consist of fire water trucks connected directly to EWS pipes through special connections, after the depletion of water in dousing tank. These connections are installed only if there is no other heat sink available. The fire water trucks will use the water from the fire water tanks or from the domestic water system. The water pumped in the EWS lines will reach the SGs secondary side and will promote thermosyphoning process.

The monitoring of the critical safety parameters is ensured from MCR, as long as the batteries are available. Following EPS bus power re-energization (after mobile Diesel generators connection), plant monitoring is performed from SCA.

If the loss of Primary UHS and SBO occurs during the plant shutdown, depending of the PHT system configuration and status prior to the event, the operators required actions may be slightly different. However, the required operator actions to restore fuel cooling are provided in the Outage Heat sink operating manual and special Operating Instructions - used for contingency actions during performance of certain outage work plans. The event of Loss of the primary ultimate heat sink combined with station black-out due to a DBE is covered in Section 5.1.3.

The first cliff edge effect identified by the licensee for this scenario is defined as loss of batteries supply in 8 hours. The mobile diesel generators can be deployed within 2.5 to 3 hours to restore the EPS to supply power when the batteries are exhausted.
Loss of primary UHS (RCW/RSW systems) combined with SBO will lead to loss of cooling to spent fuel bays. This event is covered in Section 6.4.2.

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

According to the information provided by the licensee, on the Cernavoda NPP site there are all the necessary provisions which can ensure the cooling conditions of the

reactor core (e.g. no fuel failure due to the loss of water from the primary circuit) for more than 7 days in the case of loss of primary UHS combined with SBO, as it results from the Section 5.3.

Following the Fukushima accident, the Cernavoda NPP procured two 880 kW, 100% mobile diesel generators and tested them by powering the 380 VAC EPS buses and the EWS pumps.

The minimum time to install these mobile DGs is within 2.5 up to 3 hours. The minimum time provided by the water available from the dousing tank will ensure enough time for the operators to install the mobile DG's. EWS can provide water to SGs secondary side using mobile Diesel generators. The EPS fuel tanks will ensure the necessary fuel oil for Diesel generators for at least 5 days. The thermosyphoning process will ensure decay power heat removal with steam release to atmosphere through open MSSVs. Following EPS bus power re-energization, plant monitoring from SCA is restored.

5.3.2. External actions foreseen to prevent fuel degradation.

To ensure the availability of the equipment required for response, the decision was taken for all of the equipment to be stored on-site.

For the recovery phase, efforts from the "Transelectrica" National Power-Transport Company will combine with the licensee's efforts in order to restore off-site power supply.

Special agreements were also established with the local and national authorities involved in the emergency response in order to ensure that in case of a Station Blackout coincident with loss of primary UHS the plant has absolute priority to grid re-connection and supply of light and heavy equipment and the necessary diesel fuel. These arrangements are part of the emergency preparedness and response and they are presented in Section 6.1.3.

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

Following to Fukushima Dai-ichi NPP accident, for the purpose of the SBO event response, two 880 kW, 0.4 kV mobile diesel generators (one for each of Unit 1 and Unit 2) have been procured and tested by powering the 380 VAC EPS buses and the EWS pumps. The capacity of each mobile diesel generator is almost equivalent to that provided by the design non-mobile EPS diesel generators. The plant has already approved a contract in order to replace the above mentioned Diesel generators with other 2 Diesel generators 2x1MW (to cover entirely the EPS loads), which are more versatile as they can supply also 6KV loads supplementary to 0.4 KV loads.

Furthermore, the licensee initiated also the procurement process for acquisition of a mobile diesel engine driven pump. Also, 2 electrical mobile submersible pumps are already available on site. There are also plans to procure two smaller diesel generators for electrical power supply for the two pumps that can provide water in the domestic water system from the deep underground wells.

The plant proposed already some other plant modifications which have as objective to further increase the safety margins by giving the operator the possibility to manually operate important equipment for ensuring heat removal path (e.g. MSSVsS), in the extremely unlikely case of a SBO coincident with loss of batteries or instrument air (this case was discussed in Section 5.1.3) and to increase the seismic robustness of batteries.

In order to further decrease the time to connect the mobile Diesel generators, the plant has initiated a modification to install special connection panels to the loads which may be supplied from these Diesels.

CHAPTER 6 - SEVERE ACCIDENT MANAGEMENT

6.1. Organization and arrangements of the licensee to manage accidents

The licensee has specific procedures are in place to mitigate the effects of initiating events and direct the operator to bring the plant to a safe state that usually is defined as cold shut down state.

The response to anticipated operational occurrences and to accidents is controlled through a hierarchical system of station procedures as follows:

- Operating Manuals and Alarm Response Manuals – include procedures used by the plant operation staff during routine operation of the nuclear power plant and its auxiliaries and also information regarding abnormal operation and the alarm functions associated with the plant systems (set points, probable cause, operator response, etc.);
- Impairment Manual - includes actions to be taken by the operator in case that operation is close to or getting outside the specified limits of the safe operating envelope;
- Abnormal Plant Operating Procedures (also known as Emergency Operating Procedures (EOPs)) - which direct the operator during accident conditions (for design basis and design extension conditions) and are designed to restore the plant to a safe condition and ensure protection of the health and safety of the plant personnel and of the general public;
- Severe Accident Management Guidelines – which direct the operators and technical support groups during severe accident conditions and are designed to minimize the severe accident consequences and to bring the plant in a stable end state.
- Emergency Response Operating Manual - includes operator's actions in case of radiological, medical and chemical incidents, fire events, extreme weather conditions, spent fuel transfer/ transport incidents, spent fuel bays and spent fuel dry storage facility incidents, loss of Main Control Room; this manual provide the necessary criteria to classify the emergency and easy access to each of the sections containing the necessary measures to be taken for the different types of emergencies, with the overall process being governed by the on-site Emergency Plan.

Administrative procedures are in place to describe responsibilities for the operating crews when dealing with plant transients and accidents, aiming to obtain consistency in crew performance. These documents instruct the licensed operating personnel to recognise any abnormal event and mitigate its consequences.

Abnormal Plant Operating Procedures (APOPs), provided for response to design basis accidents and design extension conditions, include event-based type of procedures, as well as symptom based procedures. Two new APOPs, for responding to Station Blackout and Abnormal Spent Fuel Bays Cooling Conditions, have been issued as part of the response to lessons learned from the Fukushima Daiichi accident.

In addition, Cernavoda NPP has implemented a set of Severe Accident Management Guidelines (SAMGs), to cope with situations in which the response based on APOPs is ineffective and the accident conditions progress to severe core damage. The objectives of the SAMGs are:

- to terminate core damage progression;
- to maintain the capability of containment as long as possible;
- to minimize on-site and off-site releases.

The SAMGs for Cernavoda NPP have been developed based on the generic CANDU Owners Group (COG) SAMGs for a CANDU-6 type of plant. In developing the generic SAMGs, COG adopted the Westinghouse Owners Group (WOG) approach, with the necessary technical modifications suitable for implementation in CANDU plants, based on extensive CANDU specific severe accident analysis and research.

Preparation of plant-specific SAMGs was done by customisation of the generic COG documentation package for Cernavoda NPP, removing extraneous information not applicable to the station, incorporating station-specific details and information and making any other adjustments required to address unique aspects of the plant design and/or operation.

A total number of 48 documents were prepared (SAG's, SCG's, CA's, SACRG's, SAEG's, DCF, SCST and their associated background documents). Also, another 40 Enabling Instructions were prepared in order to support the line-ups for each strategy presented in the above mentioned documents.

The list of SAGs (Severe Accident Guidelines) and SCGs (Severe Challenge Guidelines) is provided in the Table 6.1.

Table 6.1 – SAMGs for Cernavoda NPP

SAMG	Priority	Scope of application
Severe Accident Guidelines (SAG)	SAG-1	Inject into Heat Transport System
	SAG-2	Control Moderator Conditions
	SAG-3	Control Calandria Vault Conditions
	SAG-4	Reduce Fission Product Release
	SAG-5	Control Containment Conditions
	SAG-6	Reduce Containment Hydrogen
	SAG-7	Inject into Containment
Severe Challenge Guideline (SCG)	SCG-1	Mitigate Fission Product Release
	SCG-2	Reduce Containment Pressure
	SCG-3	Control Containment Atmosphere Flammability
	SCG-4	Control Containment Vacuum

The interface between APOPs and SAMGs was established by introducing the severe accident entry conditions into the APOPs. The interface with the Emergency Plans was provided by making revisions to the existing EPP documentation, to reflect the new responsibilities and requirements arising from the implementation of the SAMGs. Also, all categories of plant personnel involved in the emergency response organisation were trained for SAMG use, and drills are currently being incorporated in the overall Emergency Response Training Program.

The SAMGs have been developed based on the existing systems and equipment capabilities. A limited and focused set of information requirements was defined to support SAMG diagnostics and evaluations. The primary source is from plant instrumentation, supplemented by additional measurements and data expected to be available through emergency response procedures and Computational Aids where appropriate.

6.1.1. Organisation of the licensee to manage the accident (on-site emergency response)

The stress test report submitted by the licensee provided extensive information on the organisation of the response to emergencies, covering all the aspects outlined in the stress test specifications.

An On-Site Emergency Plan is in place to adequately respond to any emergency, ranging from the lowest incident classification ("Alert" level) to the highest classification ("General Emergency") that requires the evacuation of all non-essential personnel on-site. Off-site emergency response is under the responsibility of the local, county and national authorities.

The human resources appointed for emergency response activities has been assessed and identified, based on the assumption that both Cernavoda NPP Units would be affected by an accident, in conformance with the specifications for the "stress test". 412 persons were specifically trained for the emergency response procedures:

- operation personnel;
- emergency management and technical support personnel;
- assembly area responsible persons;
- medical personnel;
- professional civil fire fighters.

The licensee has included in the stress test report a conservative evaluation of the on-site vital areas habitability and accessibility, based on selected severe accident scenarios.

The estimation of the doses to the operating staff was performed for a period of 7 days, assuming normal 12h shifts. The projected doses to the workers are below the values considered acceptable at international level for justifying emergency actions that reduce the risk for public exposure.

For the Main Control Room personnel, the alternative operating area is the Secondary Control Area (SCA), able to control and command all the safety systems. The SCA is seismically qualified and is continuously monitored by a qualified person trained to safe shut down the unit, keep the heat sinks available, alarm the on-site personnel, activate the emergency organization and notify public authorities.

A comprehensive emergency response program and provisions for responding to emergencies, including severe accidents, are in place, covering:

- Organizations and human resources;
- Emergency procedures, training and drills;
- Emergency facilities and equipment;
- Fuel supplies for diesel generators;
- Emergency monitoring and sampling;
- Dose calculations, personnel protection and evacuation;
- Communication provisions and equipment;
- Notifications to public authorities for off-site response.

All the provisions of the emergency program, including the associated documentation, have been approved by station management and the regulatory body and are subjected to periodic testing through emergency exercises.

The On-Site Emergency Control Centre is the headquarters for emergency response personnel to deal with the emergencies. It is appropriately equipped with the required instrumentation to assess plant status, and has filtered ventilation system, diesel power generator, food and water provisions to ensure availability for long term operation periods.

In accordance with "On-site Emergency Plan", document approved by CNCAN and reviewed by the Inspectorate for Emergency Situations "DOBROGEA" of Constanta County, any event which involve an actual or potential release of radioactive materials into environment that require urgent protective actions off site is classified by the affected unit Shift Supervisor (SS) as "General Emergency" (level 4 of the emergency classification system, other emergency levels being: level 1 – Alert, level 2 – Station Emergency, level 3 – On-Site Emergency).

Based on radiation hazards, General Emergency is declared in the following conditions:

- External dose rate ($\dot{H}_{ext}$) in normally occupied areas of the station (areas where in normal conditions the dose rates are smaller than 10 μSv/h): $\dot{H}_{ext}$ > 10 mSv/h, or
- External dose rate ($\dot{H}_{ext}$) at off-site / beyond the site boundary: $\dot{H}_{ext}$ > 1 mSv/h, or
- Total activity released to stack (confirmed release), averaged on 15 minutes, which lead in 1 hour the off-site doses: H > 1 mSv, or

- Total activity in the containment, based on the results from Post Accident Sampling and Monitoring System: Λ_{GN} > 9E14 Bq / Λ_{I} > E13 Bq.

In case of emergency classified as "General Emergency", the On-site Emergency Organization is activated and emergency response activities are initiated in the following sequence:

- The SS notifies the plant personnel about the initiated event through the public address system and site siren system requiring on-site personnel to assemble in assembly areas (done in approx. 5 minutes after the event classification), and all non-essential personnel evacuate the site;
- The SS activates the Intervention Support Center (ISC) from the main control room of the unit affected by the event: Intervention Coordinator arrives in the ISC, obtain information about the event from SS and set initial strategic objectives of response activities (done in approx. 10 minutes after the event classification);
- The Intervention Coordinator notifies the management and support staff in the emergency, which is composed of personnel working in the On-site Emergency Control Centre, staff working in the operational centers of the county and local public authorities and members of the on site / off-site monitoring teams (done in approx. 12-13 minutes after the event classification);
- The Intervention Coordinator, with the approval of the SS, makes the initial notification of public authorities, communicating critical information on plant status and submitting recommendations on protective measures for the population to the authorities involved in off-site intervention, if the accident is with immediate loss of containment integrity. In this case, in order to establish and transmit quick recommendations on protective measures, the emergency procedure "Determination of Population Protective Measures" defined pre-established recommendations on protective measures for all identified accidents, depending on the meteorological conditions (wind direction), affected sectors, localities placed in those sectors, distances from the plant, anticipated whole body effective doses and anticipated thyroid committed doses (done in max. 30 minutes after the event classification);
- The main control room personnel carry on operating activities in emergency situations, which aim to bring the plant into a safe state, ensure proper cooling of the fuel and reduce or stop radioactive emissions from the containment using APOPs or SAMGs, based on established conditions and criteria;
- The Intervention Coordinator initiates the following emergency response activities: checking of the accounting results and implementing protective measures for on-site personnel, and monitoring of the affected unit;
- Up to the activation of the On-site Emergency Control Centre, the SS fulfills the Emergency Manager duties regarding management and coordination of activities whose purpose is to protect the public, the environment, the plant and on-site personnel;
- Activation of the On-Site Emergency Control Centre: members of Command Unit (Emergency Manager, Emergency Technical Officer, Emergency Health Physicist and Emergency Administrative Officer) arrive in On-site Emergency

Control Centre, get information (status of plant, staff, notifications made, etc.) about the event from the SS and review the response activities strategic objectives (required time: 15 minutes after notification of the emergency management and support personnel during normal working hours / until 2 hours after notification of the emergency management and support personnel outside normal working hours);

- The Emergency Manager takes the lead in coordinating of the emergency response activities from the SS;
- The Emergency Health Physicist (EHP) establishes the on-site / off-site radiological monitoring strategy and coordinating of the monitoring teams (In-station Survey Team and two On-site / Off-site Monitoring Teams). The EHP, aided by the EHP Assistant processes the data received from the monitoring teams, Gaseous Effluent Monitors System (if the release is going on through the stack) and Off-site Gamma Monitoring System (this is an on-line gamma monitoring system which contains 15 gamma monitoring stations, two of them being installed at each stack of both units and 13 being installed in range of 3 km around the plant) and establishes protective measures for population and on-site personnel. In order to recommend the right protective measures where established:
    - Generic Intervention Levels, if the protective measures are recommended based on dose projections, and
    - Operational Intervention Levels, if the protective measures are recommended based on environmental radioactivity measurements;
- The Emergency Administrative Officer coordinates the implementation of protective measures for staff on site;
- The Emergency Health Physicist with the Emergency Notification Forms completed regularly updates information for Public Authorities regarding plant status and recommendations on protective measures for population;
- The Emergency Technical Officer coordinates the Technical Support Group to provide technical advice in timely manner to the Emergency Manager and to the Shift Supervisor. If SAMG entry condition is met, the Technical Support Group use the SAMG, based on Diagnostic Flow Chart (DFC) and Severe Challenge Status Tree (SCST) parameter values, to evaluate and recommend recovery actions/or strategies to reach a controllable, stable plant state;
- In case of a severe accident, the Main Control Room staff implements SAMG strategies recommended by the Technical Support Group, executing special SAMG Enabling Instructions and providing all necessary information about the status of plant equipment and conditions;
- The Technical Support Group monitors all potential hazards on long-term and confirms / records if implemented strategies continue to function. When all the monitored parameters are stabilized or a steady decline, the Technical Support Group then assesses the Severe Accident Management Exit Guides for "exit" conditions.

During emergencies, the following administrative activities are performed, initiated and coordinated by the Emergency Administrative Officer:

- notification of the emergency services and the families of the casualties;
- recording of information in the On-site Emergency Control Center log;
- sending by FAX the Public Authorities notification forms;
- notification for calling in the supplementary personnel during the events;
- notification for the traffic control;
- ensuring arrangements for additional supplies;
- ensuring heavy equipment for cleaning the debris;
- providing eating and sleeping arrangements for the On-site Emergency Organization personnel;
- ensuring transport, accommodation and eating arrangements for the off-site support personnel.

The public authorities will assist with any other needs, such as clearing roads, providing fuel, transportation of key emergency response personnel, food and other necessities, etc.

6.1.2. Possibility to use existing equipment

Mobile equipment and management of supplies

Emergency response personnel are provided with all necessary provisions to respond to the emergencies, from the initial response phase to post-accident recovery phase. Plant staff will make use of existing equipment, including innovative uses of plant systems and equipment.

Sources of supplementary water supply available for make-up to the reactor cooling systems and to the SFB include station fire water, fire trucks loaded with water or mobile pumps taking water from various locations, such as fore-bay, fire water tanks, de-mineralized water tanks or deep underground wells.

Where necessary, provisions have been made to bring on site mobile equipment such portable pumps, fire trucks, etc., to allow mitigation of the accident when existing equipment are not available. Two mobile diesel generators have been provided in a secure location on-site for hook-up to provide power for the unlikely situations where all the other electrical power supplies for the plant are lost. The mobile diesels can be made available on site and connected within 2,5 - 3 hours, if necessary.

The supplementary fuel procurement represents one of the most important administrative activity in order to ensure continues running of the Diesel Groups (Standby Diesel Generators – SDG, Emergency Power Supply – EPS, mobile diesel generators and mobile diesel engine driven pump) and the On-site Emergency Control Center Diesel Generator, if necessary.

The fuel procurement activity has to take in account the following issues:

- There are two fuel storage tanks dedicated for U1/U2 EPS. The minimum fuel inventory ensures 4 days of continuous running of EPS. The maximum fuel inventory ensures 7.5 of days continuous running of the EPS.
- There are four fuel storage tanks dedicated for U1 SDG. The minimum fuel inventory ensures 5 days of continuous running of the SDG. The maximum fuel inventory ensures 18 days of continuous running of the SDG.
- There are four fuel storage tanks dedicated for U2 SDG. The minimum fuel inventory ensures 7 days of continuous running of SDG. The maximum fuel inventory ensures 17 days of continuous running of the SDG.
- There is a common fuel storage facility belonging to the Transportation Service. The maximum fuel inventory ensures 100 days of continuous running of the On-site Emergency Control Center Diesel Generator.

It should be noted that all the diesel groups on the site use the same fuel and it is therefore possible to transfer fuel from one unit to other, if necessary.

A contract arrangement is in place to urgently deliver diesel fuel to Cernavoda NPP on short notice. Public authorities will help with transportation if this will be the case.

Management of radioactive releases

In general, the strategies for mitigating radioactive releases are in the following order:

- Isolate the leak path to the environment (close isolation valves, dampers, airlock seals, crimp);
- Reduce the driving force using a containment heat sink (local air coolers and dousing serve multiple purposes – they not only remove heat, but also condense steam to reduce pressure, and remove fission products by plate-out and wash-out);
- Reduce the driving force by venting through a filtered, monitored release path (this is the lowest priority strategy because there would be a temporary increase in release rate until the driving force for leakage is reduced); venting strategies are addressed in Section 6.3.3.

The criteria for declaring a General Emergency based on the radiation hazards and the related emergency response measures have been described in Section 6.1.1.

Communication and information systems

The dedicated equipment available for personnel and public warning consists of:

- On-site Public Address System;
- On - site Sirens (continuous current battery supplied);
- Sirens for communities (Cernavoda, Saligny and Stefan cel Mare) inside 3 km Precautionary Action Zone (PAZ) provided and maintained on regular basis by Cernavoda NPP (continuous current battery supplied).

All the emergency equipment are accounted for, tested and maintained based on a routine verification basis, ensuring their availability and reliability.

The dedicated communication and notification equipment available during emergency include regular phones and faxes available in the control rooms and emergency centre, but also satellite phones, E-LAN and special communications service phones and radio systems.

The diversity of these systems ensure that means for communication will remain functional even in the worst-case scenarios that would involve destruction of the infrastructure outside the site, and the contact with the response team members and public authorities involved in the intervention off-site will be maintained.

As a measure for increasing the reliability of the communication systems, the licensee has taken measures to acquire additional special communication service phones and satellite phones.

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

For situations where the access to the site could be hindered due to extreme meteorological conditions, natural disasters (earthquakes, flooding, etc.) or other traffic restrictions, Cernavoda NPP has agreed a protocol with the Constanta County Inspectorate for Emergency Situations, Police County Inspectorate, National Roads and Bridges Company, County Roads and Bridges Company and Territorial Structure for Special Problems of Constanta County to ensure the provision of the necessary support in an emergency (transportation of Cernavoda personnel, fuel supplies, etc.).

Cernavoda NPP has protocols in place also with medical centres and hospitals in the region, for the provision of medical services (first aid, initial treatment and decontamination, treatment of overexposed personnel).

As mentioned in Section 6.1.2, special communications services are ensured for Cernavoda NPP, which will remain functional regardless of the functioning of terrestrial or satellite networks.

Impairment of work performance due to high local dose rates, radioactive contamination and destruction of some facilities on site, as well as unavailability of power supply, have been taken into account in the emergency response measures. Also, in accordance with the specifications for the “stress test”, the resources allocated for emergency response have been determined and provided based on the assumption that both Cernavoda NPP Units would be affected by an accident.

The habitability of the main control rooms and secondary control areas has been assessed and it was concluded, based on conservative assumptions, that all the 5 shift crews can perform their work either from the MCR or from the SCA without exceeding an integrated dose of 100 mSv for the first 7 days of an accident, even for the case in which both units would be affected by an accident.

The Command Unit and the Technical Support Group carry on their activities during an emergency in the On-site Emergency Control Center, located at cca. 800 meters from the plant. This center can be operated for a very long period of time taking to account that it is equipped with filtered ventilation system, diesel power generator and food and water provisions.

For situations where the Main Control Room and The On-Site Emergency Control Centre would be unavailable, measures have been taken to ensure that emergency response will be directed from the Secondary Control Area, which is qualified to remain functional for all emergency scenarios, including those caused by extreme external events.

The set-up of an Alternative Off-site Emergency Control Centre is in progress. The Off-site Emergency Control Center will be located in an existing facility in Constanta City (approximately 60 km away from Cernavoda NPP). The Off-site Emergency Control Center will be able to fulfill the same functions as the On-site Emergency Control Center and will be available to support any type of emergency considered into the on-site emergency plan. Until then Off-site Emergency Control Center will be commissioned, the Secondary Control Area facilities (from Unit 1 or Unit 2) may be used by the Command Unit and Technical Support Group for emergency management.

Cernavoda NPP has also started actions in order to set-up a new seismically qualified location on-site for hosting the On-site Emergency Control Center and the Fire Fighters Facility. In the same location there will be sheltered the most important intervention equipment including: mobile diesel generators, mobile diesel engine driven pumps, firefighter's engines, radiological emergency cars, heavy equipment to unblock roads, etc.

The ability to correctly diagnose plant conditions and to monitor the implementation of mitigation strategies depends on the availability of reliable information about the state of systems, structures and components in the plant. A limited and focused set of information requirements is defined to support SAMG diagnostics and evaluations.

The primary source is from plant instrumentation, supplemented by additional measurements and data expected to be available through emergency response procedures and Computational Aids where appropriate. Generally, instrument response does not need to be as accurate as during normal operation.

The main parameters monitored to support the implementation of the SAMGs for Cernavoda NPP are:

- RIH Subcooling Margin (SCM);
- Moderator level;
- Calandria vault water level;
- Plant radiation measurements;
- Containment pressure;
- Containment hydrogen flammability;
- Containment water level.

It is recognised that the design requirements for most plant instrumentation are based on normal operation or accident conditions less severe than expected when a SAMG is entered. Therefore, the licensee has identified and has committed to implement actions to improve the reliability of the existing monitoring instrumentation by environmental qualification to severe accident conditions and by extending the measurement domain, as well as to provide additional instrumentation, such as hydrogen monitoring system inside the reactor building.

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

Based on the review performed in the framework of the "stress test", the organisation for accident management and emergency response has been found adequate. The resources allocated are sufficient also for the situation in which both units would be affected by an accident.

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

Improvement measures have been identified and will be implemented for increasing the reliability of the communication systems and of the on-site emergency control centre. The set-up of an Alternative Off-site Emergency Control Centre is in progress. Also, improvements to the plant instrumentation are planned to support the implementation of the SAMGs.

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

The CANDU-6 reactor has both preventative and mitigating features to ensure a robust design against severe accidents. It has inherent and engineered provisions to prevent core damage, terminate progression of core damage, retain the core within the calandria vessel, localize core debris within the calandria vault, maintain containment integrity, and minimize off site releases.

Progression of accidents in CANDU reactors from those involving little or no fuel damage to significant core damage and possibly core disassembly is strongly influenced by the unique aspects of the reactor design. In particular, the low pressure heavy water moderator in the calandria vessel surrounding the pressure tubes and the large volume of light water in the calandria vault which, in turn, surrounds the calandria vessel, provide a passive heat sink capability which will provide significant time delays in the progression of a severe accident sequence. Such delays are of benefit in that they provide decision and action time for accident mitigation and management measures to be taken.

The CANDU-6 design has inherent design robustness against core damage (i.e., no severe core damage occurs at high pressure, high pressure melt and direct containment heating are precluded, reactivity induced accidents are precluded by the two fast, highly reliable and diverse shutdown systems). The large water inventories surrounding the fuel and the entire core act as a heat sink to remove the decay heat

after reactor shutdown, even if all engineered heat removal systems fail, and allow for sufficient time for the implementation of severe accident management actions.

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

The accident management measures aimed at preventing fuel damage are covered by the emergency operating procedures (Abnormal Plant Operating Procedures – APOPs) which include event-based and symptom-based procedures for ensuring the essential safety functions and for bringing the plant to a safe and long-term stable state.

For events with an intact PHT (including predominantly steam line breaks or other events causing rapid cooldown and shrinkage of PHT inventory), the first measure required to give assurance of fuel heat removal is to keep the PHT full. There are three means of providing PHT make-up: Injection from high pressure ECC, from EWS, or by using the PIC system feed pumps (assuming they are available). Regarding the secondary side, when the initial inventory of the boilers is depleted, one of several other means of long-term heat removal must be established – there are several choices:

- MFW can be established and can operate indefinitely to remove decay heat. MFW is capable of operating at full SG pressure;
- AFW can be used as a heat sink until the available sources of water are depleted (a total time of 73 hrs); AFW is also capable of operating at full SG pressure;
- The SG can be depressurized to allow feedwater to flow by gravity from the dousing tank to the SG (i.e., 23 hours for uncontrolled flow, and 6,9 days for controlled flow);
- EWS can be established to provide make-up to the SG from the Danube river (indefinitely) after the SG are depressurized; or
- The SDC system can be established, supported by RCW as the long-term heat sink (sustainable as long as RCW and Class III power remain available).

For LOCA events, crash cooldown is initiated automatically upon detection of the LOCA. This depressurizes the PHT to facilitate passive high-pressure injection from the ECI system and diminishes the tendency of the secondary side to act as a heat source (particularly applicable in case of large breaks).

The ECC system, if available, begins recovery once the PHT is sufficiently depressurized. ECC takes suction from the sumps and injects coolant to the PHT, via the ECC heat exchangers for cooling. In the event of a small LOCA, a secondary side heat removal (via MFW, AFW, or EWS) may be required to provide the required long-term heat removal. Either feedwater or the SDC system may be used as the heat sink for the intact loop of the PHT.

A prerequisite for the majority of events to progress to conditions leading to severe core damage is a loss of heat transport system coolant coupled with failure of the ECC

system to inject cold water into the heat transport system (in accident analysis, the sequence is LOCA + LOECC). The loss of coolant may be due either to an initiating pipe break or a consequential induced failure of the heat transport system boundary resulting, for example, from events involving a loss of flow or loss of heat sink.

The LOCA + LOECC sequence is part of the design basis accident analyses for CANDU reactors. These analyses have demonstrated that progression of the event to core disassembly is effectively prevented by the passive rejection of heat from the fuel channels to the moderator fluid. For such scenarios, the heavy water moderator is credited as an effective heat sink. Heat will be removed by moderator heat exchangers supported by active circulation.

If forced circulation of the moderator system is not available, natural circulation inside the calandria (promoted by warming of the moderator at the centre of the core and cooling at the relatively cool outer surfaces of the Calandria Vessel) prevents further damage to the fuel channels (maintaining the fuel channels at temperatures below that at which channel failure will occur) and prevents severe core damage (fuel channel failures). The calandria vessel inventory will gradually steam into the R/B.

If the water level in the calandria drops below the level of the top row of fuel channels (after loss of all other active heat sinks have previously failed.), the event will progress to severe core damage (fuel channel failures). Similarly, for events that result in moderator draining, the fuel channels will become uncovered and will eventually fail. In the absence of active heat removal steaming to the R/B will result in an increase in R/B pressure that may be suppressed by the local air coolers (LACs) or by dousing.

6.2.2. Accident management measures after occurrence of fuel damage in the reactor pressure vessel/a number of pressure tubes

In case the moderator heat sink is lost, the moderator inventory will boil off gradually, during which time the heat removal will be sufficient to ensure the integrity of the calandria vessel. The fuel channels which become gradually uncovered will overheat, sag and fail and thus the core will eventually collapse to the bottom of the calandria. Event progression may be halted at this state by providing water make-up to the calandria vessel or by cooling the calandria vessel externally, via the calandria vault light water inventory.

In-vessel retention of core debris prevents MCCI (Molten Corium Concrete Interaction) and the associated build-up of hydrogen, and containment over-pressurization. CANDU 6 has been designed with redundant and diverse systems and design features to provide a high level of assurance that severe accidents will not progress beyond a sustainable in-vessel state.

A Shield Cooling System (SCS) is provided to remove reactor core decay heat from the end shields and calandria vault by circulating demineralised water through them and then transferring this heat to the RCW system by means of heat exchangers. If operational following a severe accident, the SCS and its supporting systems are the only required means of core decay heat removal and containment energy management (i.e., no other core heat sinks or containment heat sinks are required) to halt accident

progression and retain the corium within the calandria vessel. If active circulation and cooling are not available, the calandria vault inventory will boil off gradually, during which time heat removal will be sufficient to ensure continued integrity of the calandria vessel.

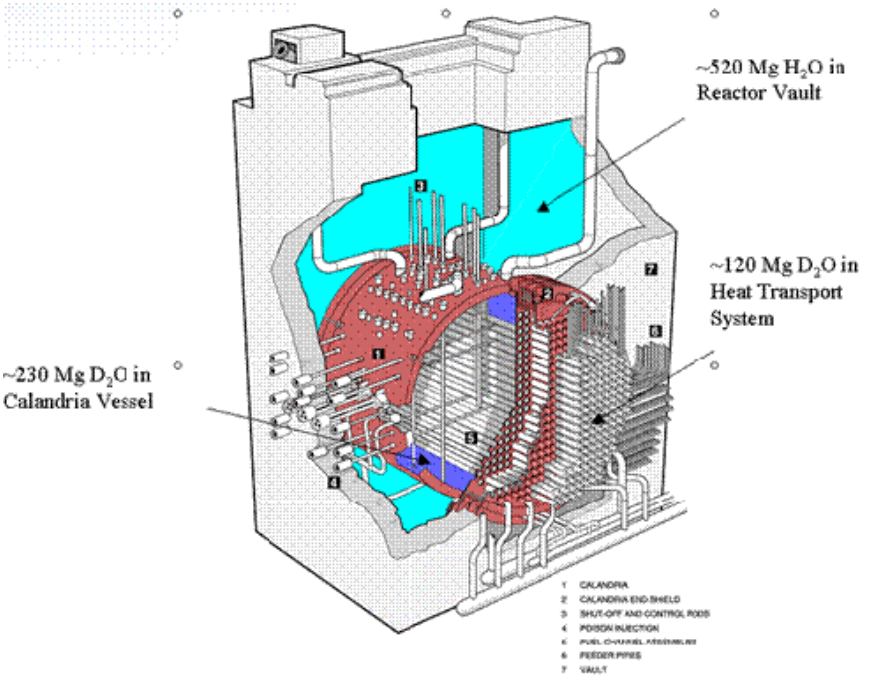

Fig. 6.1 – Calandria Vessel location inside Calandria Vault

The calandria vessel could deform but is not expected to break under any steam surges or other dynamic loading during the core disassembly process. In addition, the calandria vessel is expected to be able to maintain the load of a disassembled core as well as providing a sufficient heat transfer pathway to the calandria vault to reject the decay heat and maintain sufficient cooling of the calandria shell.

The significant volume of water inside calandria vault cools the outer calandria vessel wall, maintaining the external cooling of the vessel. Heat is removed from the exterior of the calandria vessel via the calandria vault inventory by steam relief from the calandria vault. It will take many hours for the calandria vault inventory to heat up and boil off. This long process would provide time for operators to take appropriate actions to mitigate the severe accident progression by providing make-up to calandria vault.

As long as calandria vessel is mostly submerged in water and the calandria vault water inventory can be maintained, it is expected that corium will be retained in the calandria vessel and accident progression arrested in-vessel. The externally cooled calandria vessel acts as a “core catcher” containing the core debris. It should be noted that core disassembly and relocation take place only at low heat transport system (PHT) pressures and that melting of core materials is avoided until after the debris has relocated to the bottom of the calandria vessel.

In order to retain the corium in the calandria vessel and to prevent relocation of the core debris to the calandria vault, the SAMGs provide for water make-up to the calandria vessel and to the calandria vault. Design modifications to support these SAMGs have been proposed and are planned for implementation (a modification for make-up to the calandria vessel has already been implemented in Unit 2).

6.2.3. Accident management measures after failure of the reactor pressure vessel/ calandria vessel

Assuming that no water make-up supplies are available, the calandria vessel would eventually fail due to creep and the debris would relocate into the calandria vault. In the unlikely event of ex-vessel failure there is provision to provide water to the calandria vault to keep the corium cool and remove decay heat. The reactor floor has a large area to facilitate the corium spread and cooling. The slow progression of the accident would allow sufficient time for the operators to restore failed mitigating systems or for accident management actions to be implemented, including water make-up to prevent the core melt through the 2.4m think concrete calandria vault.

The severe accident analyses for a CANDU 6 generic design have shown that, if the ex-vessel corium is not submerged in water and cooled, it would take at least two days after the accident initiation for the MCCI to occur, with complete concrete ablation at least four days after accident initiation. This provides sufficient time for operators to mitigate consequences in the containment and bring the accident into a controlled and stable state. These are conservative timeframes, since during the calandria vessel failure, there is calandria vault water surrounding lower part of the calandria vessel.

The ex-vessel phenomena have been considered and the design provisions have been reassessed for use in SAMGs aimed at protecting the containment function. The protection of the confinement function is addressed in the following section.

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

The containment building provides the fundamental barrier protecting the public in the unlikely event of a severe accident by limiting the radioactive releases to the environment. Protection of the confinement function requires limiting the interior temperature and pressure in the containment.

The severe accident analyses for a generic CANDU-6 design have been reviewed to identify scenarios that could pose a challenge to the containment integrity and design provisions and SAMGs are available for maintaining the confinement function in severe accidents. The main conclusions of this review are summarised as follows.

High pressure core melt ejection scenarios are not possible for a CANDU reactor, as explained in Section 6.3.1. The measures for hydrogen management, protection of containment against slow over-pressurization and prevention of containment basemat melt-through are addressed in Sections 6.3.2, 6.3.3 and 6.3.5, respectively. Potential re-criticality of core debris, addressed in Section 6.3.4, is not a concern for CANDU reactors.

Containment bypass through ruptured steam generator tubes is not possible during a severe accident in a CANDU reactor. In CANDU severe accident scenarios, the pressure tube is expected to rupture well before a steam generator tube might fail on PHT overpressure. The PHT relief valves in CANDU-6 will limit the pressures in the heat transport system to below those at which the steam generator tubes would rupture. Thus consequential steam generator tube ruptures leading to containment bypass are precluded.

Steam explosions (energetic fuel-coolant interaction) are considered unlikely during CANDU severe accident progression because of the core geometry that dictates the mode by which the hot debris materials are brought into contact with water.

6.3.1. Elimination of fuel damage / meltdown in high pressure scenarios

High pressure core melt ejection scenarios do not exist as a challenge for a CANDU reactor.

Primary heat transport system depressurisation (either directly through a break in the system or indirectly via automatic depressurisation of the secondary side by the opening of the main steam safety valves) occurs well before the potential formation of molten corium conditions.

Even if the engineered depressurisation mechanisms would fail, fuel overheating will cause a limited number of fuel channels to fail, depressurizing the PHT. Thus, the fuel channels of the CANDU-6 reactor act as ‘pressure relief fuses’ should an accident evolve and produce high PHT pressure and elevated PHT coolant temperature.

6.3.2. Management of hydrogen risks inside the containment

The hydrogen control strategy employed in Cernavoda Unit 1 relies on the reduction of hydrogen volumetric concentration by inerting the containment atmosphere so that in the longer term the containment integrity is not threatened. Cernavoda 2 is additionally equipped with igniters to deliberately ignite and burn the hydrogen as soon as it reaches flammable concentration, thus avoiding its detonation at higher concentrations.

The containment structures are designed to provide natural circulation mixing and the local air coolers’ fans, if available, promote forced air circulation for hydrogen mixing to avoid pockets of locally high concentrations.

Specific SAMGs have been developed for hydrogen control in severe accident situations. In addition, installation of passive autocatalytic recombiners will be implemented for both units, to further increase the safety margins and to ensure a hydrogen control feature independent of the availability of electrical power supply.

6.3.3. Prevention of overpressure of the containment

Slow over-pressurization may occur due to steam generation by decay heating as a result of a loss of the heat sinks. Non-condensable gases, contributing to the

containment pressurization, can also be generated by thermal–chemical interactions of hot core materials.

Several sources of steam release in the containment during an unmitigated severe accident have been identified and a specific SAMG has been developed to address them, including venting strategies.

The existing design features that protect the containment integrity against over-pressurization are the large containment volume and passive condensation on reactor building structures and the local air coolers and dousing spray.

For scenarios involving containment overpressure, there are three groups of venting strategies:

- Use of filtered vent paths;
- Use of water scrubbed vent paths;
- Use of non-filtered vent paths.

The primary benefit of venting strategies is the reduced peak pressure from an uncontrolled hydrogen burn or detonation, due to the removal of gases from containment, which reduces both the energy released by a burn and the initial pressure. A secondary benefit of venting through a filtered discharge path is that containment pressure is reduced, thus reducing the potential for releases through unfiltered leakage paths.

Containment venting can be performed through a water-scrubbed pathway, either through the spent fuel bay (SFB) or through the steam generators (SG). The SFB can be boxed up and the SFB building maintained at a slight negative pressure by purging a small flow through the SFB ventilation exhaust filters. Atmosphere steam discharge valves and any other relief/isolation valve can be closed manually.

If none of the filtered or scrubbed pathways above is available for containment venting, a non-filtered path can be used. There are several containment penetrations of various sizes that could be used to vent containment in a controlled manner and their use for this purpose has been addressed in the SAMGs.

In addition to the existing systems and provisions to reduce containment pressure, the installation of an emergency filtered venting system has been contracted for both Cernavoda NPP units.

6.3.4. Prevention of re-criticality

The potential for re-criticality under severe accident conditions was evaluated for the reactor, the spent fuel bay and the fresh fuel storage room and it was concluded that this is not a concern for the CANDU-6 design.

The CANDU 6 reactor is heavy-water moderated and cooled. The reactor uses natural (non-enriched) uranium fuel in a 37-element bundle geometry. The fuel is located in horizontal channels in a square lattice of 28.5 cm lattice pitch, which is a design that is optimized for neutron economy. Approximately 2 channels per day are refueled with eight bundles of fresh fuel to maintain the reactor critical with all

reactivity control devices in their nominal positions. Any significant deviation from this geometry would result in a reduction in core reactivity and it would not be possible to maintain criticality. In a severe accident when the core might be in a molten state, the collapsing of fuel channels and fuel bundles into corium would reduce the overall core reactivity due to less neutron moderation which consequently leads to higher resonance absorption in uranium and keeps the molten core subcritical.

Also, criticality of CANDU fuel bundles in ordinary (light) water is not possible, removing a concern in severe accidents (injecting light water into the core is part of the severe accident management).

The spent fuel bays use light water to cool the spent fuel and provide shielding. The 37-element natural-uranium fuel bundles in an infinite array in a light-water medium have been demonstrated to remain subcritical.

6.3.5. Prevention of basemat melt through

Measures for retaining the core melt in the calandria vessel and subsequently in the calandria vault have been described in Sections 6.2.2-3. Due to the inherent capability of the moderator system and the calandria vault to assure calandria vessel integrity for an extended duration, the likelihood of an event progression to an ex-vessel core damage state is very low and no further mitigation measures are considered necessary, beyond the design modifications mentioned in Section 6.2.2.

Assuming conservatively that the molten debris is not retained in the calandria vault, analyses have shown that it would reach the containment basement floor after several days since the initiation of the core melt.

Prevention of basemat melt-through is based on a SAMG aimed at injecting water into the containment for cooling the core debris and limiting the molten core – concrete interaction. Preferred and alternate strategies have been identified, taking account of the availability of the systems that can be used for water injection and of the expected conditions resulting from various severe accident scenarios.

A specific SAG for injecting water into the containment can be used in case the water level on the containment basement is below 1.5 meters (this entry condition is unlikely to be met for most severe accident scenarios, since significant water inventory should be present on the containment basement from the operation of the systems used for core cooling and for containment pressure suppression). The main objectives of this specific SAMG are the prevention of further MCCI and protection of the ~4.5 meter thick base slab.

The preferred strategies identified for injecting water into the containment involve the use of the EWS water either via ECCS piping directly to containment (ECCS sump) or via the dousing tank. CANDU 6 reactor design provides sufficient floor space for debris spread and means to keep the debris on the floor submerged in water.

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

The need for electrical AC and DC power and compressed air to equipment used for protecting containment integrity has been assessed and the following provisions are in place:

- If no other power supply sources are available, reliance will be placed on the mobile diesel generators.
- Instrument air is required to supply the seals of the containment airlocks (for ensuring isolation). For this purpose, in addition to the instrument air system supplies, nitrogen bottles are provided by design as a back-up to ensure airlock seals functionality.

6.3.7. Measuring and control instrumentation needed for protecting containment integrity

As mentioned in Section 6.1.3, the existing instrumentation has been assessed for adequacy to support the implementation of the SAMGs and measures have been identified and will be implemented for the environmental qualification to severe accident conditions and for extending the measurement domain, as well as for providing additional instrumentation, such as hydrogen monitoring system inside the reactor building.

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

The human and equipment resources appointed for emergency response activities have been assessed and allocated, based on the assumption that both Cernavoda NPP Units would be affected by an accident, in conformance with the specifications for the "stress test".

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

The containment building provides the fundamental barrier protecting the public in the unlikely event of a severe accident by limiting the radioactive releases to the environment. Its effectiveness requires limiting the interior temperature and pressure following such an event. In general, the main challenges to containment integrity in the event of a severe accident are: Containment slow over-pressurization, hydrogen control, and MCCI.

The challenge to containment integrity in the event of a severe accident is slow over-pressurization due to steam generation by decay heating as a result of a loss of the heat sinks. Multiple provisions are available to avoid steaming in to the containment; highly reliable, active, post-accident heat sinks are provided for the PHT, the calandria vessel, and the shield cooling, which would stop the steaming into the containment atmosphere when available. The reactor building is provided with Local Air Coolers dedicated to cool the containment atmosphere. The dousing sprays are used for containment pressure suppression. They are initiated automatically upon detection of high pressures that may challenge containment integrity.

The containment structure designed to provide natural circulation mixing. The LAC fans, if available, promote forced air circulation for hydrogen mixing to avoid pockets of locally high concentrations. In Cernavoda Unit 1, the provision for hydrogen control is to reduce hydrogen volumetric concentration is by inerting the containment atmosphere so that in the longer term the containment integrity is not threatened. Cernavoda 2 is equipped with igniters to deliberately ignited and burned the hydrogen as soon as it reaches flammable concentration; thus avoiding its detonation at higher concentrations with consequential higher pressure within containment. For both units installation of passive autocatalytic recombiners is considered, due to the fact that these recombiners do not need any external power supply.

In the unlikely case of ex-vessel core damage, potential molten corium-concrete interactions may challenge containment integrity by pressurization from non-condensable gases and steam production. The Cernavoda NPPs provides sufficient floor space for debris spread and means to keep the debris on the floor submerged in water. SAMGs are in place to provide for water injection into the containment.

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

Several design improvements have been identified and are planned for implementation to enhance the capability to maintain containment integrity in case of severe accidents. These include the provision of water make-up to calandria vessel and calandria vault to arrest the progression and relocation of the core melt, the provision of hydrogen monitoring systems and passive autocatalytic recombiners for hydrogen management and the installation of filtered containment venting systems.

6.4. Accident management measures to restrict the radioactive releases

During the progression of a severe accident, the highest priority for severe accident management is to protect the public and environment by taking all measures necessary to prevent a fission product release. In the event that release does occur, then strategies are needed to quickly terminate the release and mitigate any further releases.

Fission product releases are monitored as part of the Diagnostic Flow Chart (DFC). Severe Accident Guideline 4, Reduce Fission Product Releases, is called upon if the radiation dose measurements exceed the SAG-4 setpoint (0.2mSv/h at the station boundary). The setpoint covers both airborne releases and direct shine from containment (the conditions should be prolonged and not triggered by a short term transient release).

The strategies to reduce fission product release from inside containment are grouped under the following four categories:
1) Stop the release (box-up the containment);
2) Slow down the release rate (reduce containment pressure);
3) Remove airborne component of fission products (time, dousing, chemicals, water, local air coolers);

The highest priority is to stop the release of radioactive material from the plant by isolating the source of the release. If it is not possible to isolate the release path, then heat sinks should be used to reduce containment pressure to reduce the driving force and minimizes the release rate. If the release path cannot be isolated and heat sinks are not available or sufficient to reduce containment pressure, the action of next highest priority is to remove the airborne portion of fission products. Containment local air coolers, for example, remove airborne aerosols and iodine whether or not they are performing a cooling function. Dousing is also a quick and effective means to wash out airborne fission products. The strategies in guideline SAG-4 do not include venting actions.

If the actions in guideline SAG-4 are successful, then the radiological consequences are controlled. However, if actions in SAG-4 are unsuccessful or if radiation levels rise to the point of imminent hazard to the public and the environment likely to require offsite protective actions, then Severe Challenge Guideline 1, Mitigate Fission Product Release is entered. The measures for restricting radioactive releases have been addressed under Section 6.3.3, in relation to the containment venting strategies.

6.4.1. Radioactive releases after loss of containment integrity

As described in Section 6.1.1, any event which involves an actual or potential release of radioactive materials to the environment that would require urgent protective actions off-site triggers the declaration of "General Emergency".

Protective measures for population and on-site personnel are devised in accordance with the Generic Intervention Levels and Operational Intervention Levels outlined in the following tables.

Table 6.2 - Protective Actions According To Generic Intervention Levels

Protective Action	Generic Intervention Level (avertable dose)
Sheltering	10 mSv
Evacuation	50 mSv
Iodine prophylaxis	100 mGy

Table 6.3 - Protective actions based on external dose rate measurements from the plume

OIL	Value	Protective actions
OIL 1	1 mSv/h [a,c]	Evacuate or provide substantial shelter [b] for this sector, the adjacent sectors and the sectors closer to the plant. Until evacuated people should be instructed to stay inside with their windows closed.
OIL 2	0.2 mSv/h [c]	Go inside, close windows and doors and monitor radio and TV for further instructions.

a) If there is no indication of core damage, OIL 1 = 10 mSv/h.
b) Substantial shelter is provided by specially designed shelters or the inside halls or basements of large masonry buildings. Shelter should be considered only for 24-

48 hours and effectiveness must be confirmed by monitoring especially in high dose rate areas.

c) Monitor evacuees and instruct the public on decontamination measures.

Table 6.4 - Protective actions based on external dose rate measurements from the deposition		
OIL	**Value**	**Protective actions**
OIL 3	1mSv/h	Evacuate or provide substantial shelter within sector.
OIL 4	0.2 mSv/h [a,b]	Consider relocating people from sector.
OIL 5	1μSv/h	Restrict immediate consumption of potentially contaminated food, and milk in area until samples are evaluated.

a) This OIL has to recalculate based on sample analysis as soon as possible.

b) For 2-7 days after the accident.

Table 6.4 - Protective actions based on I-131 air concentration measurements		
OIL	**Value**	**Protective actions**
OIL 6	70 kBq/m^3	Take thyroid blocking agent.

6.4.2. Accident management after uncovering of the top of fuel in the fuel pool

Sustained loss of fuel bay cooling represents an unlikely emergency situation, which may be induced by common mode events, like earthquake causing sustained loss of AC power. In the case of a prolonged loss of spent fuel bay cooling, make-up water is required to prevent uncovery of the spent fuel and potential hydrogen generation.

In response to the Fukushima accident, based on WANO SOER 2011-2 recommendations, an emergency operating procedure called "APOP G04 - Spent Fuel Bay cooling abnormal conditions" was developed, validated and issued in order to address prolonged/ extended loss of Spent Fuel Bay cooling capability, main goal being to prevent fuel bundles damage and H2 generation, due to overheating.

Based on calculations performed, there is sufficient time available to establish a source of water make-up into the spent fuel bay to keep the spent fuel bundles submerged. The exposure of the SFB inner concrete surface to 100°C temperature was also investigated and it was concluded that this would not result in any significant reduction of concrete properties of the structure.

The loss of Spent Fuel Bay cooling event can be managed successfully following APOP G04, given the large time frame available (based on the progression of the unmitigated accident): 9 days until radiological fields in Spent Fuel Bay rooms start to increase significantly (to 1.7 mSv/h), and 15 days until first the first row of fuel bundles become uncovered.

During the worst case scenario (loss of Class IV together with loss Class III, earthquake or Station Black-Out) when the normal cooling and normal demineralized water make-up is lost for a prolonged period of time (there are 60h available until

water in the bays would start to boil), the APOP G04 procedure guides the operators to establish another means of water make-up in the Spent Fuel Bay. This is performed using fire trucks or mobile pump via hose connections, maintaining spent fuel submerged.

The water make-up requirement is calculated to be approximately 1 kg/s with a thermal load of 2 MW, cooling of the fuel being realized by natural convection. Also in order to mitigate the steam environment in the bays, natural ventilation of the vapors and steam resulting from the evaporation is being implemented. Field set-up for monitoring H_2 concentrations is requested early in the event, using portable H_2 detectors connected to tubing lines previously installed.

If the plant fire water is not available, the ultimate water make-up source used will be fire trucks filled with water from various places (forebay, fire water tanks, demineralized water tanks or high depth wells). Another solution specified in APOP-G04 is to use water from the forebay using mobile pump and hoses directed in the Spent Fuel Bay.

Monitoring of level and temperature in the bays, hydrogen concentration and radiation dose rates in the bay room is specified in the procedure. If the harsh environment in spent fuel bay area will not allow operators to align fire hoses, a new fire water pipe seismically qualified will be used. The modification and installation of the new water pipe is in progress and will allow make-up of water in spent fuel bay without actually entering the SFB area, connections being provided outside SFB building.

APOP G04 provides direction for a wide range of initiating events. After some of those initiating events the main control room may or may not be available, so the APOP-G04 can be followed also during an event that requires APOP G02 “Secondary room area operation” or APOP G03 “Station Blackout”.

To support APOP G04 execution, the following design changes and operational measures have been implemented or are in progress:

- tubing was installed above Spent Fuel Bay for H_2 sampling;
- hard level gauge were installed in the Spent Fuel Bay and reception bay;
- a new pipe seismically qualified that has connection outside Spent Fuel Bay is installed in order to be used to add water in the bay using connections from the fire truck or from a mobile pump (in progress);
- ventilation of the area above the bay can be provided by opening SFLA doors or by designing a hatch in the roof and ventilation openings in the walls (in progress).

During the validation exercise for APOP G04 the worst case was considered, with the normal make-up water and the back-up fire water being considered unavailable. The procedure was followed and a source of make-up water was established 30 minutes after initiation of the event, using fire truck and hoses.

In conclusion, no adverse consequence is expected as a result of the loss of Spent Fuel Bay cooling and no damage to the spent fuel is expected to occur. It should also be mentioned that there is no possibility of re-criticality of the CANDU spent fuel

whether in air or in light water. Hydrogen generation is not possible as long as the spent fuel remains submerged.

The slow progression of an accident involving loss of cooling to the SFB, even for the worst case scenario, provides sufficient time for the implementation of an emergency operating procedure to ensure that fuel will remain adequately cooled and no personnel radiation doses exceeding administrative limits are expected to occur.

6.4.3. Conclusion on the adequacy of measures to restrict the radioactive releases

Design provisions and accident management measures are in place to prevent and mitigate radioactive releases for a range of accidents, including severe core damage scenarios.

Design measures for improving the containment capability in severe accident scenarios have been identified and will be implemented as described in Section 6.3.10. These include the installation of filtered containment venting systems.

Accidents involving loss of cooling to the SFB do not pose a threat in terms of radioactive releases.

6.5 Organisation of the off-site emergency response

According to the Romanian legislation, the National System for the Management of Emergencies is composed of three types of structures:

- the decisional structure – the committees for emergencies;
- the executive structure – the inspectorates for emergencies;
- the operational structure – the operative centres for emergencies.

All the decisional, executive and operational structures are established on three levels: national, county and local.

The National Committee for Emergency Situations (CNSU) represents the decisional structure at national level. The CNSU is set-up under the co-ordination of the Prime Minister and managed by the Minister of Interior and Administrative Reform (MIRA). All the ministerial, county and local committees are subordinated to CNSU. The County/Local Committees for Emergencies are directed by the county Prefect / local mayor.

The General Inspectorate for Emergency Situations (IGSU), a specialised organisation of MIRA, is established as an executive structure at national level. IGSU has the responsibility of permanent co-ordination of the prevention and management of emergency situations, at national level. At county level, there are established County Inspectorates for Emergencies, acting as public professional emergency services.

Inside each Inspectorate for Emergency Situations an Operative Centre for Emergencies is set-up, with permanent activity, ready to activate the emergency organisation in case of an accidental event. These Operative Centres for Emergencies are receiving notifications for all types of emergencies, including radiation events.

Also, the responsible organisations at national level are operating such Operative Centres for Emergencies, in accordance with the legal provisions in their field of activity. The National Operative Centre of IGSU represents the operational structure, at national level.

In order to fulfil the legal responsibilities in case of a nuclear accident or radiological emergency, CNCAN has its own Emergency Response Centre (ERC), as part of the National System for the Management of Emergencies. CNCAN – ERC is the national contact point in relation to any type of radiation emergency. As part of the National System, CNCAN-ERC is communicating with IGSU Operative Centre and with other operative centres of public authorities.

By law, the Ministry of Interior and Administrative Reform (MIRA) is responsible for the management of nuclear and radiological emergencies, IGSU and CNCAN being the national competent authorities in case of nuclear accident or radiological emergency. At local level, the intervention is coordinated by the Local Committees for Emergencies and performed by the Local Response Forces. When the emergency situation cannot be solved by the local authorities, the national level (CNSU and IGSU) is activated, in order to support the local intervention. Written agreements and protocols are in place between the responsible organizations, at local and central level, for common activities and exchange of information in emergency situations.

The response organisations have the following responsibilities:

- to elaborate and revise to date an adequate emergency plan;
- to assure by means of laws, Governmental Ordinance, decrees, the legal basis for protection of the population, environment and property, medical care, financial compensations, etc. in emergency situations;
- to establish and maintain a proper structure of the intervention sources able to: advice on nuclear safety and radiation protection, sample and analyse samples, keep in contact with police, army and fire fighting forces, keep contact and receive advice from water management bodies, agriculture produce control bodies, medical services, meteorological forecast facilities.
- to organise and maintain an emergency co-ordination centre equipped with technical means for the expertise of the emergency and sufficient communication means;
- to organise exercises and drills, to maintain the level of personnel training and equipment availability for emergency situations;
- to establish levels for the triggering of the emergency in case of transboundary emergencies.

A review of the national (off-site) emergency response strategy is currently being performed, with the aim of incorporating lessons learned from the Fukushima accident.

CHAPTER 7 - GENERAL CONCLUSIONS

The licensee has performed a safety review that adequately follows the stress test methodology and has covered in their report all the aspects required for consideration in the stress test specifications.

The stress test report submitted by the licensee has provided analyses of all the events and combinations of events required by the stress test specifications, comprehensive information on the inherent design features and engineered systems credited in the prevention and mitigation of severe accident scenarios and on the availability and performance of the heat transfer paths and on the various severe accident management strategies employed, including an assessment of the potential for cliff edge effects and the time available for operator actions.

The information provided in the report covered both the analysis of unmitigated severe accident scenarios and the mitigation strategies addressed by the plant specific SAMGs. The claims made in the report are supported by a vast set of references, most of them relating to severe accident analyses and accident management guidelines developed by COG based on more than 30 years of research.

7.1. Key provisions enhancing robustness (already implemented)

Specific emergency operating procedures have been developed and implemented to cope with Station Blackout and Loss of Spent Fuel Pool Cooling events.

Mobile diesel generators have been procured, are available on site and have been tested to enhance protection against SBO scenarios.

Station response to a loss of Primary Ultimate Heat Sink and SBO - combined event does not rely on any off-site equipment for the primary response, all the necessary equipment and resources being available on site and sufficient for coping with a prolonged SBO. For the longer term recovery phase, efforts from the “Transelectrica” National Power-Transport Company will combine with SNN efforts in order to restore off-site power supply.

After the Fukushima accident, corrective actions have been developed and implemented to consider the lessons learned from this event. The Emergency Plan and Procedures, Conventions, Protocols and Contracts in place have been reassessed and revised to better accommodate emergency response to severe accidents coincident with natural disasters. Special attention has been paid to the communication systems where actions have been taken, together with National Special Communication Services, to supplement and improve the actual communication systems in place.

A review of the national (off-site) emergency response strategy is currently being performed, with the aim of incorporating lessons learned from the Fukushima accident.

7.2. Safety issues

The potential cliff-edge effects identified based on the analysis of unmitigated accident scenarios and the time available before their occurrence have been addressed in the SAMGs, which include measures for the prevention of cliff-edge effects.

7.3. Potential safety improvements and further work forecasted

The regulatory reviews performed to date have focused on verification of the completeness and quality of the stress test report and of the supporting analyses.

In addition, a set of inspections have been performed by CNCAN staff in addition to the review of the licensee's stress test report, aimed at verifying the quality of the process implemented by the licensee in the development of plant specific SAMGs, training records from training in the implementation of the SAMGs, the availability of up-to-date emergency operation procedures at the points of use, the procedures for connecting the mobile diesel generators and the related test reports, the procedures for injecting fire water into plant cooling systems, etc.

CNCAN noted that a significant effort has been made by the licensee to respond to the lessons learned from the Fukushima accident in a timely manner. No concerns have been raised from the regulatory reviews performed to date. The conclusion of the review conducted by CNCAN is that the risk to the public from beyond design basis accidents at Cernavoda NPPs is low and is kept under control.

Confirmatory assessments have been conducted by the licensee in response to both WANO SOER and the stress tests. Potential design improvements have been identified by the licensee and are considered for implementation to further enhance the existing safety margins and reduce the risk from severe accidents.

The licensee is fully committed to implement these improvements for Cernavoda NPP Units in a reasonable timeframe and all financial resources have been already secured.

In respect with the evolution of the accident sequences two types of design changes has been identified. The first category refers to severe accident prevention measures and the second one to severe accident mitigation measures. As a general rule, the first priority is given to the first category. For each category, the relevant information related to the design change intent, implementation target timeframe and safety benefit is provided in the Tables 7.1. and 7.2.

CNCAN will continue the safety reviews and inspections related to the implementation of the severe accident management programme and of the identified improvements consisting of design changes and operational provisions.

Table 7.1 - Design Changes that enhance the prevention of the Severe Accidents			
	Design Change Intent	**Implementation Status**	**Safety Benefit**
1	Provide an additional mobile DG set and the connections to the existing EPS buses	Implemented	Improve the existing level of defense in depth for SBO
2	Provide a seismically qualified location on site for the storage of the mobile equipment required for emergency conditions	Implemented	Improve the crew response time
3	Improve the seismic robustness of the existing Class I and II batteries	Planned Q2 2012	Reduce operator burden – improve operating crew response time. Improve CSP monitoring capabilities.
4	Provide a facility to open the MSSVs after a SBO	Planned Q2 2012	Reduce operator burden – improve operating crew response time.
5	Provide connection facilities required to add water using fire fighters trucks and flexible conduits to supply the primary side of the RSW/RCW heat exchangers and SGs under emergency conditions.	Implemented	Improve the existing level of defense in depth
6	Provide permanent connection facilities required to add water from outside the S/B using fire fighters trucks to supply the Spent Fuel Bay	Planned Q2 2012	Improve environmental working condition for operator actions
7	Improve seismic robustness for the site Emergency Control Center	Q2 2012	Extend the range of events ECC remains available. Improve the working conditions for the emergency response team (actually the team will gather in MCR in the event of ECC failure during an earthquake)

Table 7.2 - Design Changes that mitigate the consequences of severe accidents and prevent progression to an ex-vessel core damage state			
	Design Change Intent	**Implementation Status**	**Safety Benefit**
1	Provide facilities to inject water in the calandria vessel from outside R/B	U1 - 2012 U2 - partially implemented, will be finalised in 2012	Increase the existing corium cooling capabilities
2	Provide facilities to inject water in the callandria vault from outside R/B	U1 - 2012 U2 - 2013	Increase the existing corium cooling capabilities
3	Install a filtered venting Containment	U1 - 2012 U2 - 2013	Improve the existing R/B Envelope protection strategies. Reduce the environmental impact of a controlled release.
4	Install Hydrogen Passive Autocatalytic Recombiners in R/B	U1 - 2012 U2 - 2013	Improve the existing R/B hydrogen Control Strategies.
5	Install a R/B H_2 concentration monitoring system	U1 - 2012 U2 - 2013	Improve the existing R/B hydrogen Control Strategies
6	Improve the existing CSP monitoring loops environmental qualification and extend the measurement domain	U1 - 2012 U2 - 2013	Improve the existing SAMGs strategies by ensuring accurate determination for the severe accidents CSP

LIST OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feed-Water (system)
APOP	Abnormal Plant Operating Procedure
ASDV	Atmospheric Steam Discharge Valve
BCW	Back-up Cooling Water (system)
BDBA	Beyond Design Basis Accident
BDBE	Beyond Design Basis Earthquake
BMW	Boiler Make-up Water (system)
CA	Computational Aid
CANDU	Canadian Deuterium Uranium
CCW	Condenser Cooling Water
CL I/II/III/IV	Class I/II/III/IV electrical power
CNCAN	National Committee for Nuclear Activities Control
COG	CANDU Owners Group
CSDV	Condenser Steam Discharge Valve
CSP	Critical Safety Parameter
CV	Calandria Vault
D_2O	Heavy Water
DBA	Design Basis Accidents
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DBSC	Danube-Black Sea Channel
DC	Direct Current
DCC	Digital Control Computer
DG	Diesel Generator
DICA	Dry Spent Fuel Storage
ECC	Emergency Core Cooling (system)
EPS	Emergency Power Supply
EQ	Environmental Qualification
EWS	Emergency Water Supply
FP	Full Power
FRS	Floor Response Spectra
FSAR	Final Safety Analysis Report
GSS	Guaranteed Shutdown State
HPECC	High Pressure Emergency Core Cooling
HCLPF	High Confidence Low Probability of Failure
IAEA	International Atomic Energy Agency
LAC	(Reactor Building) Local Air Coolers
LCDA	Limited Core Damage Accident
LOCA	Loss of Cooling Accident
LOECC	Loss of Emergency Core Cooling
LOOP	Loss Of Off-Power
LRV	Liquid Relief Valve
LZC	Liquid Zone Control system
mBSL	meters Baltic Sea Level
MCR	Main Control Room
MFW	Main Feedwater
MSSVs	Main Steam Safety Valves

MV	Motorized Valve
NPP	Nuclear Power Plant
NSP	Nuclear Steam Plant
OEP	On-Site Emergency Plan
OM	Operating Manual
OP&P	Operating Polices and Principles
PHT	Primary Heat Transport (system)
PHWR	Pressurized Heavy Water Reactor
PSHA	Probabilistic Seismic Hazard Assessment
R/B	Reactor Building
RCW	Recirculating Cooling Water (system)
RLE	Review Level Earthquake
ROH	Reactor Outlet Header
RRS	Reactor Regulating System
RSW	Raw Service Water (system)
SACRG	Severe Accident Control Room Guideline
SAEG	Severe Accident Exit Guideline
SAG	Severe Accident Guideline
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidance
SB	Service Building
SBO	Station Blackout
SCA	Secondary Control Area
SCDA	Severe Core Damage Accident
SCG	Severe Challenge Guideline
SCS	Shield Cooling System
SCST	Severe Challenge Status Tree
SDC	Shut-Down Cooling (system)
SDE	Site Design Earthquake
SDG	Stand-by Diesel Generator
SDS	Shut Down System
SDS1	Shutdown System No. 1
SDS2	Shutdown System No. 2
SFB	Spent Fuel Bay
SG	Steam Generator (boiler)
SSCs	Structures, Systems and Components
UHS	Ultimate Heat Sink
UPS	Uninterruptible power supply
WANO	World Association of Nuclear Operators