CONVENTION ON NUCLEAR SAFETY

SECOND CNS EXTRAORDINARY MEETING
ARGENTINE NATIONAL REPORT - **2012**

Autoridad Regulatoria
Nuclear
Presidencia de la Nación Argentina

CONVENTION ON NUCLEAR SAFETY

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ANNEX III. PRINCIPAL TECHNICAL CHARACTERISTICS OF ATUCHA II NUCLEAR POWER PLANT

ANNEX IV. SUMMARY TABLE

ACRONYMS

APS: Probabilistic Safety Assessment.
AR: Regulatory Authority.
ARN: Argentine Nuclear Regulatory Authority.
ASDV: Atmosphere Steam Discharge Valves.
ASECQ: Spent Fuel Dry Storage System.
BY: Emergency Diesel Generator System (CNA I)
CA: Alternating Current.
CC: Direct Current.
CCE: Emergency Control Centre.
CDFM: Conservative Deterministic Failure Margin
CDS: Core Damage Status
CIAS: Internal Advisory Safety Committee.
CICE: Internal Emergency Control Centre.
CMP: Maximum Probable Water Level.
CNA I: Atucha I Nuclear Power Plant.
CNA II: Atucha II Nuclear Power Plant.
CNE: Embalse Nuclear Power Plant.
CNEA: National Atomic Energy Commission.
COEM: Municipal Emergency Operating Centre
COG: CANDU Owners Group.
CP: Emergency Electrical Supply System (CNA II).
DBA: Design Basis Accident.
DBE: Design Basis Earthquake.
DSHA: Deterministic Seismic Hazard Analysis.
DBT: Design Basis Tornado.
DG: Diesel Generator
ECs: Fuel Elements
ECCS: Emergency Core Cooling System.
ENACE: National Energy Enterprise.
EPEC: Cordoba Provincial Energy Enterprise.
EPRI: Electric Power Research Institute.
EPS: Emergency Power Supply.
ESC: Structures, Systems and Components.
EWS: Emergency Water Supply.
FRS: Floor Response Spectrum.
FORO: Ibero-American Forum of Radiological and Nuclear Regulatory Agencies.
GV: Steam Generator.
GD: Diesel Generator
GDM: Mobile Diesel Generator.
GHC: Demineralized Water Supply System (CNA II).
GIS: Geographic Information System.
GRS: Ground Response Spectra.
HCLPF: High Confidence of Low Probability of Failure.
IDIA: Ing. Aldo Bruschi Seismic Research Institute (San Juan National University.
IFS: Final Safety Report.

INA: National Water Institute.
IXP: International Exchange Program.
JDA: Shut off Rod System (CNA II).
JDJ: Boron Injection System (CNA II).
JF: Moderator System (CNA II).
JND: Safety Injection System (CNA II).
JOEN: Operating Nuclear Emergency Chief.
JR: Reactor Protection System (CNA II).
KAG: Intermediate Cooling System (CNA II).
KBA: Volume Control System (CNA II)
LA: Feedwater System (CNA II)
LAH / LAJ: Shutdown and Start Up System (CNA II).
LCDA: Limited Core Damage Accident.
LOCA: Loss of Coolant Accident.
MCCI: Molten Corium Concrete Interaction.
MR: Refuelling Machine.
MSSV: Main Steam Safety Valves.
NA-SA: Nucleoeléctrica Argentina S.A (Argentine NPPs licensee).
NZ: Reactor Protection Signal (CNA I).
IAEA / OIEA: International Atomic Energy Agency.
OBE: Operating Basis Earthquake.
PAB: Main River Water Cooling System (CNA II).
PAC: Water Circulating System (CNA II).
PACG: Spent Fuel Storage Pool.
PAR: Passive Auto-catalitic Recombiners.
PE: River Water Assured Cooling System (CNA II)
PGA: Peak Ground Acceleration.
PGAS: Severe Accident Management Program.
PHTS: Primary Heat Transport System.
PL: Refuelling Machine System (CNA I).
POEAs: Operating Procedures for Abnormal Conditions.
PS: Tilt Bottle System (CNA I).
PSHA: Probabilistic Seismic Hazard Analysis.
QD: Pressure Control System (CNA I).
QF: Main Pumps of PHTS (CNA I).
QM: Moderator System (CNA I).
RDSI: Integrated Service Digital Network.
RL: Main Feedwater System (CNA I).
RLE: Review Level Earthquake.
RPV: Reactor Pressure Vessel.
RR: Residual Heat Removal System (CNA I).
RSMC: Regional Specialized Meteorological Centres.
RX: Second Heat Sink System –SSC- (CNA I)
SACRGs: Severe Accident Control Room Guidelines.
SAGs: Severe Accident Guidelines.
SAMG: Severe Accident Mitigation Guidelines.
SAT: Systematic Approach to Training.
SCGs: Severe Challenge Guidelines.

SAEGs: Severe Accident Exit Guidelines
SADI: Argentine Interconnection System.
SBO: Station Black Out.
SCDA: Severe Core Damage Accident.
SDCS: Shutdown Cooling System (CNE).
SDS1: Shut down # 1 system (CNE).
SDS2: Shutdown # 2 system (CNE).
SIEN: System of Intervention In Nuclear Emergencies.
SEDA: Accident Doses Evaluation System.
SHS: Second Heat Sink System (CNA I).
SL: Seismic level.
SMA: Seismic Margin Assessment.
SMN: Argentinean National Meteorological Service.
SMS: Seismic Monitoring System.
SQUG: Seismic Qualification Utility Group.
SPTC: Primary Heat Transport System.
SSEL: Safe Shutdown Equipment List.
SSC: Second Heat Sink System –RX- (CNA I).
TA: Volume Control System (CNA I).
TB: Liquid Poison Injection System (CNA I).
TJ: Emergency Core Cooling System (CNA I).
TJ-AP: High Pressure Emergency Core Cooling System (CNA I).
TJ-BP: Low Pressure Emergency Core Cooling System (CNA I).
TL: Controlled Zone Ventilation System (CNA I).
TN: Water Supply System (CNA I).
TR: Radioactive Liquid Treatment (CNA I).
TZ: Drain System (CNA I).
UA: Water Treatment System (CNA I).
UB: Hydraulic Turbine (CNA I).
UBA: Manoeuvres Building (CNA II).
UBP: Emergency Diesel Generators Building (CNA II).
UC: Main Cooling System (CNA I).
UHS: Uniform Hazard Spectra.
UJ: Potable Water Supply System (CNA I).
ULE: Fuel of low enriched uranium.
UK: River Water Assured Cooling System (CNA I).
UPD: Service Water System (CNA II).
UY: Fire Fighting System (CNA I).
WANO: World Association of Nuclear Power Plant Operators.
WMO: World Meteorological Organization.
XK: Emergency Diesel Generator System (CNA II).
YR: Shutoff Rod System (CNA I).

INTRODUCTION

The Argentine National Nuclear Safety Report to the Second Extraordinary Meeting of the Contracting Parties of the Convention on Nuclear Safety (hereinafter the Report) includes a description of the activities that have been completed and are under development, as well as the activities foreseen, for the Argentine nuclear power plants (NPPs) in response to the accident at the Fukushima NPPs. The Report includes all the actions, responses and new developments that have been initiated or influenced by the above mentioned accident.

The Report has been elaborated following the “Guidance for National Reports” suggested by the General Committee. It is structured by topic, and not by individual articles of the Convention, and does not reflect actions or issues that have developed on a “routine” basis, nor discusses the existing regulatory framework or the operational safety status of Argentine NPPs. It is therefore not intended to reflect the general nuclear safety situation in our country, but focuses on specific issues and concerns that have arisen as a consequence of the particular event occurred in Fukushima and its consequences since March 11th, 2011. When references to actions or facts outside of this timeframe are made, it is for the sake of coherence in the text involved.

In particular, and as requested in the said Guidelines, it provides ample information on the treatment of events beyond the current internationally accepted licensing basis of NPPs. It must also be noted that the understanding of the accident is still developing, and will be so for quite some time, so the appropriateness of the lessons so far been learnt or to be learned will quite possibly also be the subject of a re-assessment in due course.

It is in this vein that the Report has been written and should be understood. Its objective is to present the lessons learned from Fukushima implemented or to be implemented in the Argentine NPPs, by specifically analysing their capacity to face extreme external events and the corresponding response to the emergency caused by a severe accident occurrence.

To this end, the Argentine Regulatory Body (ARN, Autoridad Regulatoria Nuclear) and the Licensee / Responsible Entity of the Argentine NPPs under operation (NA-SA, Nucleoeléctrica Argentina S.A.) carried out activities aimed at implementing the lessons learned to date from Fukushima.

ARN performed inspections and evaluations to assess the readiness of mitigating systems including emergency preparedness, backup power sources; hydrogen mitigation systems and spent fuel storage systems.

The Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (FORO) decided to carry on for all the NPPs of its member countries, a Resistance Assessment similar to the Stress Tests required by the Western European Nuclear Regulators Association (WENRA). Its content was agreed by all the FORO members having Nuclear Power Plants. In the case of Argentina, ARN requested NA-SA a Resistance Assessment (Annex I), the main objective of which is to determine the performance to face beyond design basis initiating events occurrence which could cause the loss of the safety functions and the corresponding severe accident management activities at the Atucha I NPP (CNA I); Atucha II NPP (CNA II) and Embalse NPP (CNE).

Therefore, this requirement consists of a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth due to extreme initiating events occurrence. The requested assessment includes:

- The long term evolution of the severe accidents and the recovery capability of both the power supply, and the water supply until a stable plant condition is reached. This is to identify the most adequate recovery strategies and the components that must be available for each of the corresponding strategy implementation;
- Safety implications derived from the presence of multiple reactors in one site, identifying and implementing the corresponding measures and the procedures to use the existing resources of one unit to assist another unit;
- Spent fuel storage management strategy and spent fuel storage systems design and performance;
- Arrangement / disposal of structures, equipment and components belonging to safety systems to assure they can continue fulfilling the corresponding safety function;

- Prevention, recovery and mitigation measures: automatic and operator actions for abnormal conditions; emergencies and severe accident management;
- Availability of the NPPs resources to face on-site and off-site emergencies in severe accident conditions. In particular, from the commencement of the event occurrence until the ARN takes charge of the emergency management, including the planning and action management considering the public protection and the corresponding communication.

The results of the Resistance Assessment will be examined in a peer review by the FORO members, who will issue a final document containing an executive summary with the review results and the general conclusions. Finally, the main elements of the FORO final document will be reflected at the Second Extraordinary Meeting of the Convention on Nuclear Safety.

In addition, as a way to review the lessons learned from the Fukushima accident, NA-SA carried out a verification of the NPPs capability to mitigate conditions that result from beyond design basis accidents; station black out; internal and external flooding and; spent fuel storage systems performance.

According to what was requested by the Guidance for National Reports developed for the second extraordinary meeting of the CNS, this Report is organized by topics developed in six chapters which includes external events; design issues; severe accident management and recovery (on-site); national organizations; emergency preparedness and response and post-accident management (off-site) and international cooperation. Each one of the above chapters will give details of the activities performed by both the regulator and the operator.

Chapter 1 presents the analysis carried out as a safety reassessment considering the extreme external events occurrence being considered: earthquakes, flooding / low water level (river / lake), fires, tornadoes, lightning and wind loads. Moreover, some weak points and the corresponding cliff edge effects and some provisions to prevent these cliff edge effects or to increase robustness of the plant are indicated.

Chapter 2 analyses the foreseen actions to prevent severe damages to the reactor and the spent fuel storage systems considering the following reassessment:

- Loss of on-site and off-site alternating current (AC) electrical power supply;
- Design of the containment structures aimed to assure that it can mitigate beyond design-basis accident scenarios;
- Reliability and availability of the spent fuel storage pool makeup systems and;
- Methods to prevent or mitigate the loss of the ultimate heat sink.

In the cases where weak points or cliff-edge effects were identified, the corresponding proposals for further upgrades are included.

Chapter 3 describes the mitigation actions foreseen if severe damage in both the reactor and the spent fuel pool occurs, in order to prevent large radioactive releases. The issues considered are the development of severe accident scenarios; development and validation of procedures; equipment availability; training, the corresponding personnel resources, as well as the results of the reviews of severe accident management and on site recovery actions.

Chapter 4 describes actions aimed at strengthening the national organizations (i.e. the government, the regulator) involved in maintaining and enhancing nuclear safety.

Chapter 5 includes the review results of the activities related to preparedness for post-accidental management (off-site), including issues such as radiological evaluation, criteria and mechanisms used for decision-making, control and management of contaminated goods, resettlement, communication and information and remediation activities.

Chapter 6 includes issues such as the mechanisms for communicating with neighbouring countries and the international community; cooperation with international organizations; cooperation in the frame of international working groups; participating in international peer reviews; sharing international operating experience; and utilization of IAEA Safety Standards.

Additional information to this report is included in the following annexes:

Annex I: presents the regulatory requirement asking a stress test for each Argentine NPP according to what was decided by the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies.

Annex II: describes the technical characteristics of the NPPs in operation: CNA I and CNE.

Annex III: describes the technical characteristics of the NPP under licensing process: CNA II.

Annex IV: presents a list of the improvements and modifications as a result of the stress tests foreseen to be implemented in each NPP. This list includes the corresponding schedules and milestones to complete the operator´s planned activities.

1
EXTERNAL EVENTS

1.1. EARTHQUAKES

1.1.1. ACTIVITIES PERFORMED BY THE OPERATOR

1.1.1.1. Atucha I Nuclear Power Plant

1.1.1.1.1. Earthquake against which the plant is designed

The design of the Atucha I nuclear power plant (CNA I), consistent with the criteria and requirements established in the sixties for nuclear power plants located on sites of low seismicity (non-seismic area according the final safety report, 1973), did not include explicitly seismic loads in the design of structures, systems and components (SSC). However, since these requirements have been increased, it was decided to perform a safety assessment of CNA I against the occurrence of external events such as earthquakes, due to an update for site seismic hazard of the Atucha site (CNA I-CNA II) according to the current state of the art.

At the Atucha site, in the vicinity of the CNA I, the Atucha II Nuclear Power Plant (CNA II) began to be constructed in the 80s. The original design criteria of CNA II used by KWU / Siemens was based on a maximum acceleration of the ground of 0.05g as agreed in the Contract with the Licensee. Design principles and construction measures, valid for regions with low seismicity, were applied. The mechanical and electrical components, including their supports, were designed for a maximum horizontal acceleration of 0.15g and a maximum vertical acceleration of 0.075g.

After the beginning of the construction of CNA II in 1980, the Atomic Energy Commission (CNEA) developed a specific seismic hazard assessment for the site (GNZ, Gil-Nafa-Zamarbide) using a deterministic approach (Deterministic Seismic Hazard Analysis, DSHA), which concluded that the seismic hazard at the site of Atucha corresponds to a peak ground acceleration of 0.10g. On that basis, the Regulatory Authority in 1985 requested the Licensee (NA-SA) to develop a seismic safety evaluation of some systems of CNA II related with safety.

The above evaluation identified three seismotectonic sources and, using various attenuation relationships available at that time, calculated peak and spectral acceleration values that would be generated by these three sources. In fact, the source that eventually ruled the design was an earthquake of Richter magnitude M = 5.5, postulating its occurrence in the Paraná fault located 20 km east of the site.

The broad band standard response spectrum recommended by the U.S. NRC Regulatory Guide 1.60 (which was also presented in the appendix in the IAEA Safety Guide applicable in 1981, 50-SG-S1) was used to represent the ground motion. This response spectrum characterizes a wide frequency range of the spectrum.

It was recently decided to conduct a safety assessment of CNA I, against the occurrence of external events such as earthquakes, based on an updated seismic hazard of the Atucha site according to the current state of the art.

To perform the update of seismic hazard at the Atucha Site, the Licensee, NA-SA, contracted two international consultants: James J. Johnson & Associates (JJJ, USA) and Atomic Energy of Canada Limited (AECL, Canada) who performed the probabilistic seismic hazard assessment (PSHA) in an independent way. As will be seen later, both studies were coincident regarding the determination of peak ground acceleration and provide a solid framework for the safety assessment of CNA I, because:

- Both studies used similar geological, geophysical and seismological databases.
- Considered the various random and epistemic uncertainties adequately.
- The principles outlined in the IAEA Safety Guide SSG-9 "Seismic Hazards in Site Evaluation for Nuclear Installations", were followed.

Below, both studies are briefly presented and the conclusions of a peer review that was performed on both studies.

1.1.1.1.1.1. Study by AECL

The purpose of the study done by D'Appolonia for AECL consists on the update of the seismic hazard analysis for the site, in order to determine if the GNZ design basis of 1981 are currently valid. This update includes a review of the deterministic assumptions that were used by GNZ to define the seismic design basis, but the focus is comparing the design response spectra derived by GNZ with the Uniform Hazard Spectra (UHS) derived on the basis of a Probabilistic Seismic Hazard Assessment (PSHA), which is in accordance with the IAEA standards.

According to the IAEA Safety Guide SSG-9 , the maximum design vibratory ground motion is termed the SL-2. This level of ground motion is required to have a very low probability of being exceeded during the lifetime of the plant and represents the maximum level of ground motion to be assumed for design purposes. D'Appolonia's experience, consistent with international guidelines, is that the normal requirement for a critical facility is a 10^{-4} per annum probability (10.000 year return period).

Calculation of probabilistic seismic hazard requires basically the following input, which constitutes a model of seismicity for the region:

- Source Geometry: It is the geographic description of the seismic sources in the region around the site. A seismic source is a portion of the earth associated with a tectonic fault or, if individual faults cannot be identified, with an area of homogeneous seismicity. Source geometry determines the probability distribution of distance from the earthquake to the site.
- Seismicity: Corresponds to the rate of occurrence and the magnitude distribution of earthquakes within each seismic source.
- Attenuation Functions: Relationships that allows the estimation of ground motion parameters at the site as a function of earthquake magnitude, source to site distance and soil conditions at the site.

D'Appolonia describes the geological and tectonics of the region, and historical seismicity. The region surrounding CNA I- CNA II (where the region is considered to have a minimum radius of 320 km) has experienced minimal historical seismicity. This fact makes it impractical to define a recurrence rate; therefore a distribution of earthquakes based on considerations of the behaviour of Stable Continental Regions (SCR) throughout the World was adopted, using the methodology defined by EPRI for such cases. The model employed is shown in *Figure N°1-1*.

Regarding the seismic catalogue, a study window was taken from Latitude and Longitude 53°W to 65°W. The values were transformed in each case to moment magnitude. A detail of the catalogue is shown in *Figure N° 1-2*.

Seismic sources were characterized by recurrence relationships that describe the number and magnitude of historical and expected events at the source. The upper limit magnitude capable of being generated in a source model was also defined.

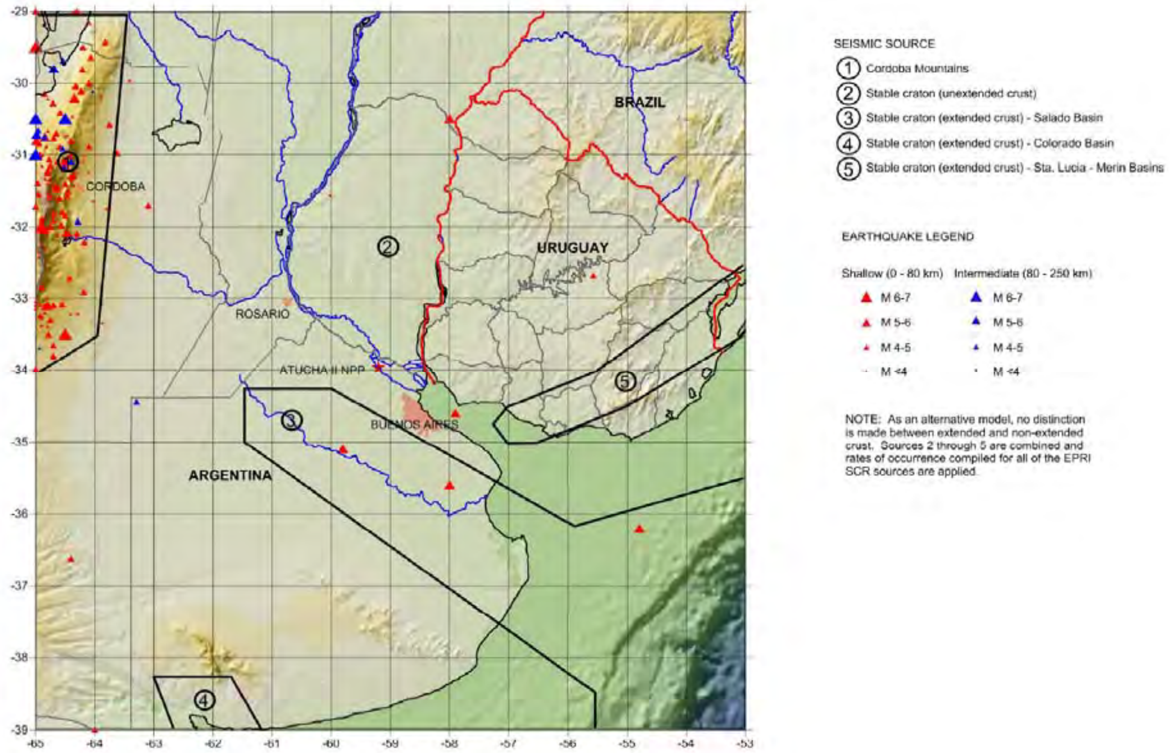


Figure N° 1-1: Seismic source model used for the study.

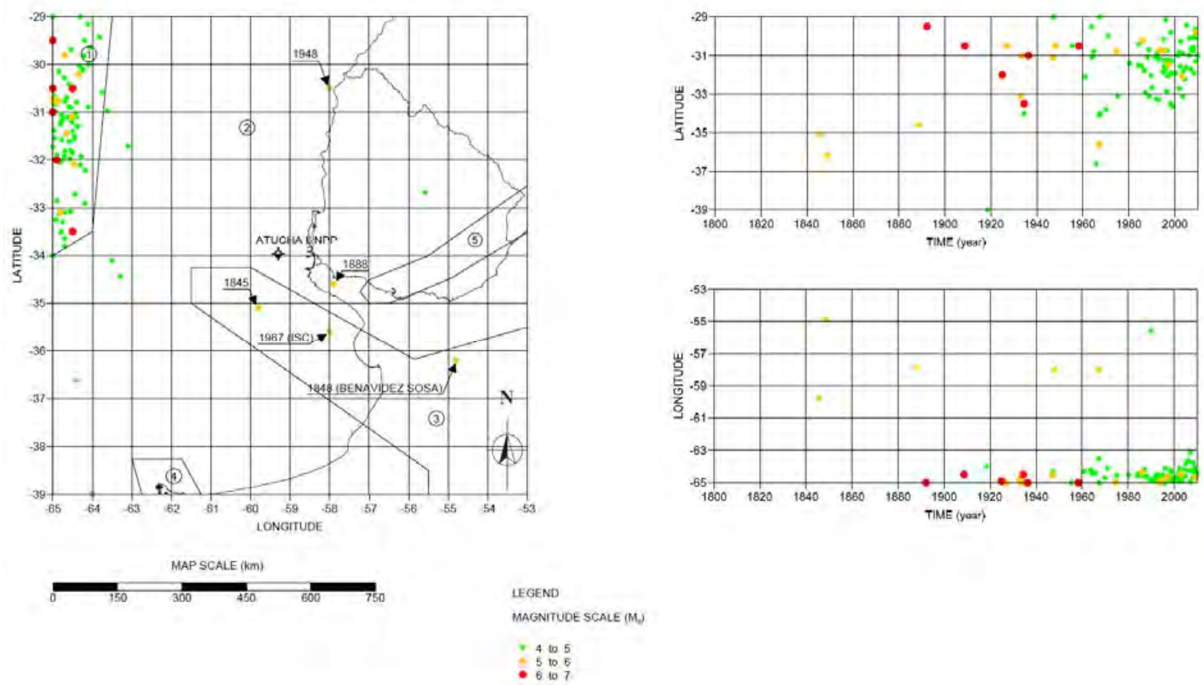


Figure N° 1-2: Catalogue of seismic events.

Because the Atucha site is located within a stable continental region, the ground motion prediction equation (GMPE), also known as ground motion models or attenuation relationships, developed for stable intra-plate regions are the most appropriate to capture the seismotectonic environment. Due to characteristics of shear wave velocity below the foundation soil, it is necessary to consider amplification effects. For this, an appropriate weighting factor was applied in the logic tree of the Atkinson and Boore (2006) GMPE, as it is the only one within the selected equations that allows predicting the ground motion both for hard rock and for specific-site conditions

The study also includes the treatment of uncertainty, distinguishing between two types:

- Random: Stochastic variability that results from natural physical processes. It cannot be reduced even with the collection of additional information.
- Epistemic: Uncertainty that results from lack of knowledge about the true state of its nature. In principle this variability can be reduced with the collection of additional data.

It is important to keep the distinction clear between these two types of uncertainties because they are treated differently in seismic hazard analyses. Epistemic uncertainty is addressed using a logic tree formulation. This implies considering various seismotectonic models, GMPE, seismicity rates and upper limits of magnitudes, using different weighting factors based on the hypothesis taken.

As for the results of the probabilistic analysis, the following stand out:

- Seismic hazard curves (namely curves relating the spectral acceleration based on the annual probability of exceedance - in rock -).
- Uniform Hazard Spectra (for a return period of 10,000 years for rock and soil with a shear wave velocity of 350 m / s). The GNZ spectrum is added for comparison.
- Disaggregation of seismic hazard.

Figure N° 1-3 shows the uniform hazard spectra (mean, median and fractals 15 and 85) compared with the spectrum of GNZ, thus D'Appolonia concludes that the floor response spectra derived from the spectrum of GNZ are conservative.

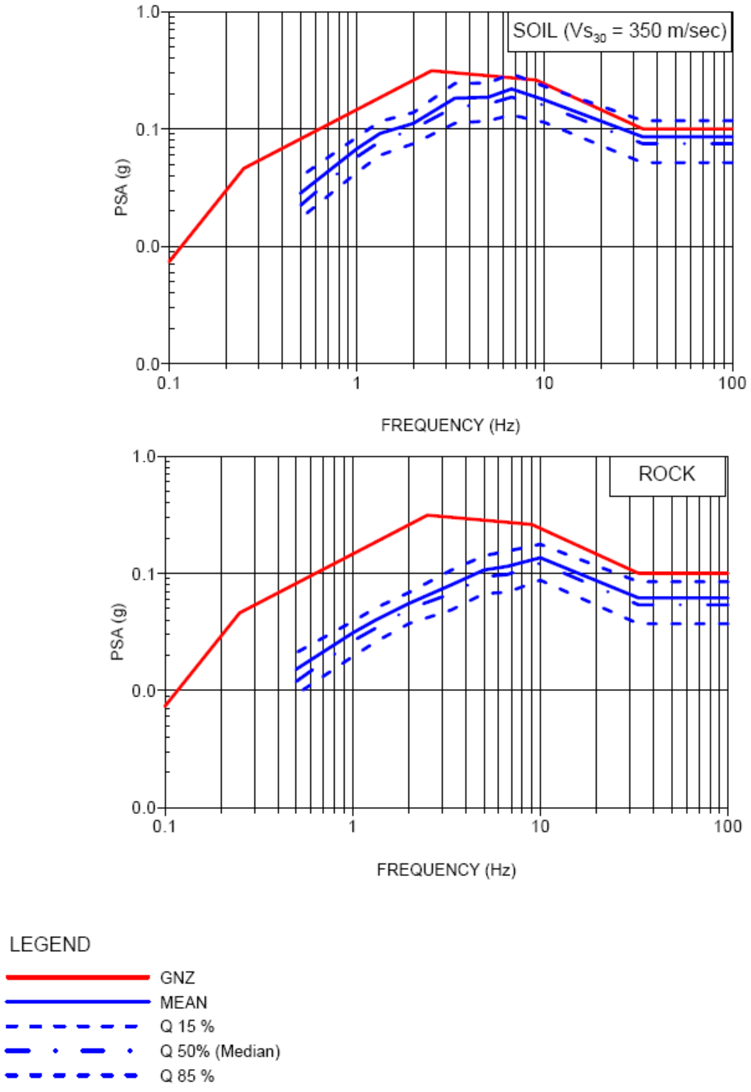


Figure N° 1-3: Comparison of the uniform hazard spectra by D'Appolonia, and the GNZ spectrum.

1.1.1.1.1.2. Study by JJJ and Associates

In addition to the discussed above, the consultant JJJ and Associates (JJJ) conducted a probabilistic assessment of seismic hazard within the framework of tasks for the assessment of the fracture mechanics in the primary system.

In that study, the following areas were considered as seismic sources:

1. Andes.
2. Pre-Andes.
3. Paraná Fault (was considered in one of the models but not in the other).
4. Background seismicity, with a maximum magnitude of 6.

Those sources are shown in *Figure N°1-4*.

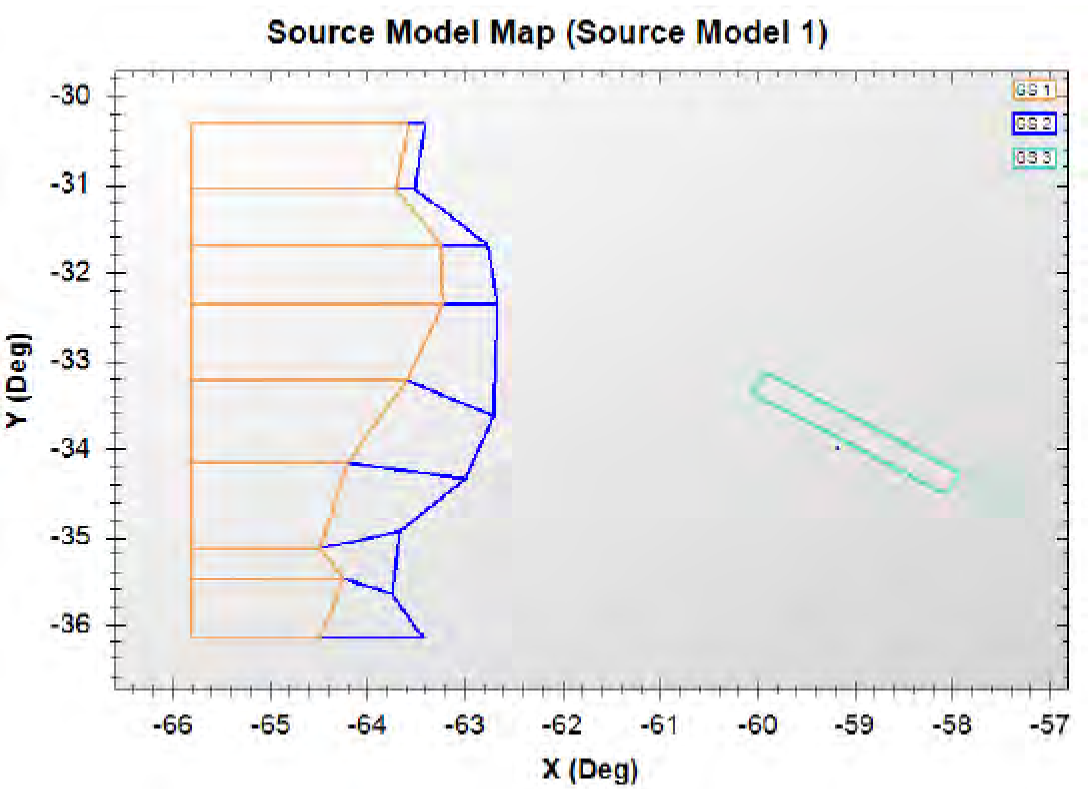


Figure N° 1-4: Model of seismic sources considered in the study.

The ground motion prediction equations were selected according to the distances to the site location and its characteristics.

The results of the study are:

- Seismic hazard curves (Peak ground acceleration, spectral acceleration for different frequencies, with a range of probability of exceedance of 10^{-6}). See *Figure N ° 1-5*.
- Uniform Hazard Spectra –UHS. See *Figure N° 1-6*.

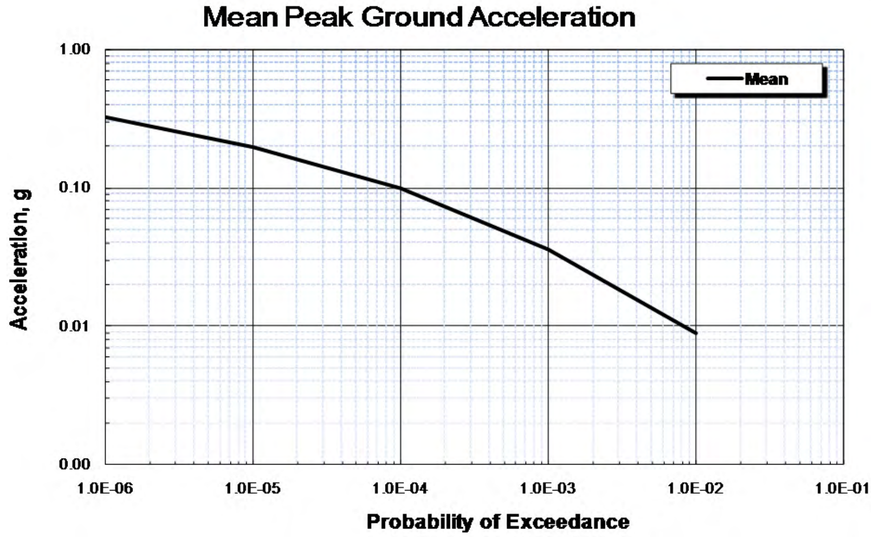


Figure N° 1-5: Seismic hazard curve for CNA I - CNA II site.

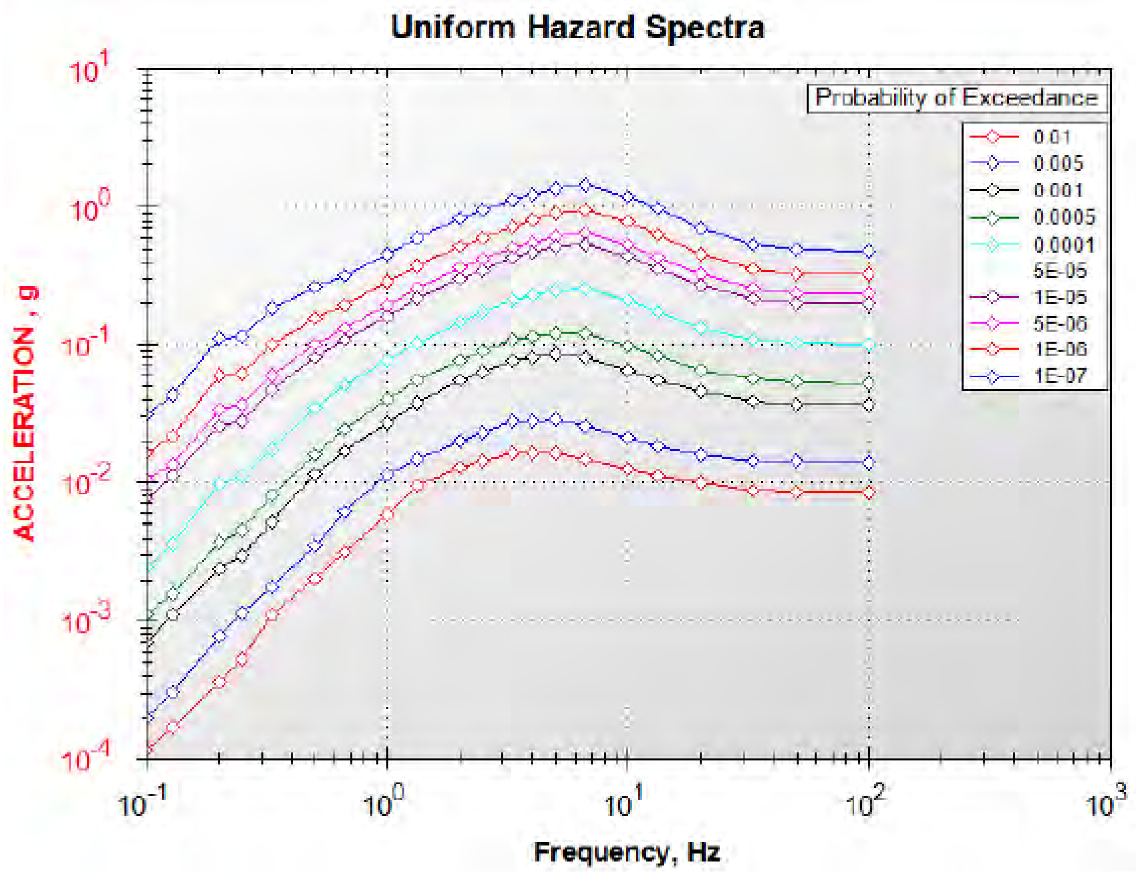


Figure N° 1-6: Uniform Hazard Spectra for different probabilities of exceedance.

1.1.1.1.1.3. Comparison between the studies of AECL and JJJ

The comparison between the studies of AECL and JJJ concluded that the seismic hazard at the CNA I – CNA II site can be based on the results of both PSHA studies. Mean uniform hazard response spectra (UHRS) corresponding to an annual probability exceedance of 10^{-4} , forms the basis for the definition of a design basis earthquake in the future for new nuclear facilities, and the definition of the review level earthquake (RLE) for seismic safety evaluation for CNA I and CNA II. The results from the JJJ study for the mean UHRS of 10^{-6} form the basis for further assessments, including the study of fracture mechanics of the Nuclear Steam Supply System (NSSS) of CNA II and other related applications with the risk.

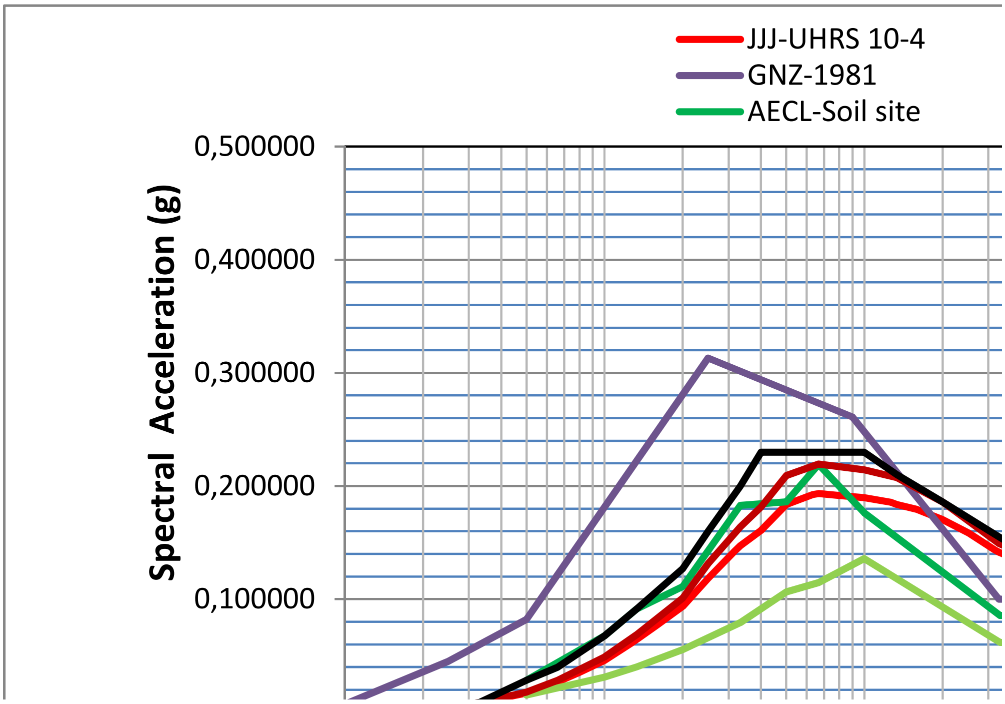


Figure N° 1-7: Envelope Spectra from Uniform Hazard Response Spectra (UHRS) from JJJ/AECL Studies, including US ASCE43-05/DF considerations

1.1.1.1.2. Provisions to protect the plant against the DBE

The CNA I was not originally designed or qualified considering the action of severe earthquakes. However, thanks to the conservative criteria that were applied in the design and the robust structures, systems and components (SSCs) typical of a nuclear power plant, it is considered that there is an inherent capacity to withstand seismic events of a certain level. In the case of CNA I, this capability will be determined by developing the methodology of "seismic margin assessment" which is to assess the status of the SSC in relation to their ability to perform its safety function before the occurrence of a specific earthquake.

This methodology aims to determine seismic capacity of "high confidence" for the nuclear power plant as a whole, called capacity of "high confidence low probability of failure" (HCLPF), which is an estimate of the earthquake's level for which fundamental safety functions could be affected. In seismic margin studies, it is considered that HCLPF capacity is the seismic level for which there is a 95% confidence that the probability of a relevant failure in the safety systems of the plant is less than 5%.

The use of this methodology, results in an appropriate focus and is sufficient for CNA I considering the low level of seismic hazard of the site and the age of the installation.

The determination of the RLE to be used in the safety assessment of CNA I is obtained from the average uniform hazard response spectrum for an annual probability exceedance of 10^{-4} and taking into account the above, it is a solid basis in accordance with the internationals objectives of safety.

The objectives of the program of seismic safety evaluation are: to demonstrate the seismic safety margin for CNA I for the RLE and confirm that there are no cliff edge effects.

1.1.1.1.2.1. Identification of SSC needed to achieve a safe shutdown condition

As discussed earlier, the CNA I is conducting a seismic margin assessment. To develop this assessment, the operator hired, JJJ, agreed with the use of the methodology of Conservative Deterministic Failure Margin (CDFM) documented in EPRI NP6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin ".

The application of this methodology requires the definition of the RLE, which in the case of CNA I was calculated according to the results of PSHA taking the uniform hazard spectra of 10^{-4}/year adjusted to be consistent in terms of risk, with the application of the methodology ASCE / SEI 43-05, "American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities", 2005.

Using CDFM methodology results in a conservative approach, because the loads used for verification of SSC are those generated by the RLE simultaneously combined with normal operating loads.

The seismic evaluation program consists of five phases, namely:

- Phase 1: Scope of the study and preliminary inspection walkdown of the plant.
- Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and Systems Walkdown.
- Phase 3: Seismic response and capacity evaluation of SSC.
- Phase 4: Inspection Walkdown of the seismic capacity and screening process.
- Phase 5: Detailed analysis and evaluation for seismic qualification.

Phase 1 was completed in December 2011. Phases 2 and 3 are necessary to proceed with phase 4, which is scheduled for May 2012. With regards to phase 5, its realization is foreseen by 2015.

1.1.1.1.2.1.1. Phase 1

As noted above, phase 1 comprises determining the scope of the preliminary assessment and a preliminary walkdown of the plant, which have been completed.

The scope that was defined for the seismic margin assessment (SMA) comprises:

- Safe shutdown procedure and number of success paths. According to the EPRI methodology, two success paths were defined with the objectives of achieving safe shutdown of the plant (hot or cold) and maintenance of the plant in that condition. The success paths include first-line and support systems, and at least one success path should mitigate a small LOCA (Loss of Coolant Accident).
- Proposed operating assumptions:
    - Loss of external power supply and no recovery in a period of 72 hours.
    - No large LOCA are postulated.
    - Small LOCA initiated by the earthquake.
    - The components should be identified in the success path.
    - External assistance is not available for 72 hours.
    - Other issues arising from phases 2, 3, 4 and 5 as appropriate. Analysis of Nuclear Steam Supply System (NSSS). Review of critical relays are identified within the SSEL.
    - Determination of High Confidence Low Probability of Failure (HCLPF) using the EPRI methodology.
    - Use of screening tables according to EPRI NP-6041.

As regards the preliminary plant walkdown, it was used to identify easy fixes as well as emergency measures aimed at reducing the vulnerabilities that are obvious: problems of interaction (impact, drop, spray, flooding), and other internal risks of the plant related with temporary equipment..

The SMA uses the concept of success path as part of the analysis. The success path consists of the set of equipment that must remain functional during the occurrence of the postulated RLE and, if operated properly, allow the safe shutdown of the plant and maintain that state for 72 hours after the earthquake. To demonstrate the seismic margin of a success path, the seismic margin of the weakest component of the success path must be evaluated. In addition, the operators' behaviour must be compatible with established procedures and training.

Following the EPRI methodology, two success paths were selected as independent as possible. For both success paths the loss of the external power supply is assumed and specifically one of them must be able to mitigate a small LOCA. In this respect, it is noted that although it is assumed that the pressure limit of the primary system resists the RLE, it considers the existence of a leak due to breakage of small pipes connected to the primary system or support systems (for example, from instrumentation and/or auxiliary systems) including possible leaks caused by the refuelling machine. It intends to cover the LOCA stage caused by the failure of the opening valve of the pressurizer. The success paths that are discussed in CNA I are:

- Small LOCA – opening a safety valve of the pressurizer.
- Loss of power supply to the bars of regular self-consumption.

1.1.1.1.2.1.2. Phase 2

As part of the work under this phase, the SSC needed to safely shutdown the reactor were identified, and the SSEL of the model of Probabilistic Safety Assessment (PSA) was made. It is emphasized that all SSC listed are necessary for the safe shutdown of the reactor and to maintain it in that condition, to remove residual heat and to avoid potential releases of radioactive substances (basic safety functions) during and after the RLE.

To the SSEL mentioned above, the following components were added because of their importance for the safe shutdown of the plant and to maintain it in that state for 72 hours after the earthquake, and to ensure the conditions of the spent fuel elements.

- Components associated with the failure of the injection of high pressure accumulators.
- Valves that must close with the containment closing signal.
- Components of the fire protection system UY.
- Piping related with the fuel elements pools.
- Important length pipes that do not have supports, located in both conventional and controlled area.
- Tanks of the volume control system (TA).
- Supply system components (TN) associated with the refuelling machine system (PL) and the tilting bottle system (PS).
- Components necessary for supplying water from pool 2, to the fuel elements pools, through pump RL 33D01 (see item for severe accident management)

1.1.1.1.2.1.3. Phase 3

This phase is in progress and its scope is as follows:

- Dynamic seismic response of building structures.
- Calculation of spectral response of the structure.

Then, using the available information for qualification, or qualification criteria based on experience, the seismic capacity of each of the components of the SSEL (above or below the RLE) is determined in a preliminary way.

Completion is expected by the second half of 2012.

1.1.1.1.2.1.4. Phase 4

The scope of this phase is a plant walkdown to determine the seismic capacity of components that comprise the SSEL and is scheduled for completion by the end of June 2012.

The objective of this phase is to screen out elements of the SSEL according to the selected SMA methodology. In this way, the components to be visited during the plant walkdown are divided into three groups:

- The elements of high-seismic aptitude which directly pass the screening and are candidates for the safe shutdown path. Its ruggedness will be established based on engineering judgment and experience.
- Those of low aptitude that should be rejected unless they are replaced or improved.
- Those elements that require the calculation of their HCLPF.

1.1.1.1.2.2. Major existing previsions

The CNA I is working to implement additional steps to prevent damage to the reactor core or spent fuel after an earthquake. In addition, as a result of the plant walkdown, corrective measures were identified (Easy Fix), in order to reduce the vulnerabilities identified and increase the ruggedness of the facility, which will be implemented by the end of the first half of 2012.

1.1.1.1.2.2.1. Mobile diesel generator

The installation of a mobile diesel generator equipment (MDG) that can be connected, if necessary, at several points of the plant is being considered. The specification of the equipment will be determined so that it can cope with worst-case scenario, which ensures its operation in lower demand scenarios.

As a first step of this study, it is considered that the new MGD shall have the capacity to feed the following components in order to keep the core cooled:

- One pump for the TA4 volume control and for the entry valves to the primary system.
- Water pump to feed the SGs (RL33D01).
- Valves and control of vent of SG.
- Pump for the potable water supply system (UJ, drinking water) and a pump that can be connected to a branch of the second heat sink system (RX) so it can cool the core, through a SG with the valves corresponding to that branch.
- A pump that takes water from the water-table and supplies the pools.

Implementation is scheduled for the first half of 2013.

1.1.1.1.2.2.2. Seismic Instrumentation

As part of the re-evaluation of the seismic hazard at the CNA I site, the installation of seismic instrumentation inside the plant and in the free field, in a radius of 20 km to 40 km around the facility is being analysed.

The installation of seismic instrumentation will allow knowing the existence of an earthquake of considerable magnitude, providing useful information for operational decisions in the event of a major earthquake that threatens equipment and plant structures. In addition, the seismotectonic model adopted for the re-evaluation of seismic hazards can be confirmed, in particular by checking the attenuation of ground in the area.

1.1.1.1.2.2.3. Easy Fixes

The modifications to be implemented during the scheduled shutdown of May-June 2012 have the following scope:

- Electrical and I & C cabinets in raised floor: an anchor will be implemented to the bottom and / or upper floor slab, reinforced in two horizontal directions.
- Batteries: additional restrictions will be installed to the racks to prevent slippage.
- Panels of the control room: the panels will be reinforced from above to the concrete wall behind.
- Emergency diesel generators: a small wall/dam of concrete is going to be built around the pit to prevent flood damage to diesel generators.

1.1.1.1.3. Plant compliance with its current licensing basis

In its original design the effect of seismic loads was not considered. However, it was decided to extend the licensing basis through the completion of SMA. In this context, as discussed in previous sections, the plant recently conducted a detailed inspection in order to meet seismic capacity and identify the need to implement improvements and / or modifications. Some of the findings of that inspection are called "Easy Fixes", which were addressed in the above point.

1.1.1.1.4. Specific compliance check already initiated following the Fukushima NPP accident

The Licensee decided to conduct a SMA evaluation to determine the plant's capacity to deal with seismic events beyond the design basis, considering a RLE of 0.1 g PGA based on the approach of the Regulatory Guide 1.208 of the US NRC.

1.1.1.1.5. Evaluation of the margins

1.1.1.1.5.1. Earthquake severity above which severe damage to the fuel becomes unavoidable

As discussed above, a reassessment of the seismic margin of the plant is being conducted using the SMA methodology, which will provide a measure of HCLPF of the robustness of the plant against severe earthquakes. The acceptance criterion is such that the SSC of the two paths that lead to safe shutdown of the plant must have a capacity greater than 0.10 g PGA with a recurrence of 10^4/y. Accordingly, safety features will cover level 3 of the defence in depth.

The ruggedness of the plant will be improved after the implementation of a set of measures including:

- Implementation of measures known as easy fixes for the next scheduled stop in May-June 2012.
- Restocking of the water of the second heat sink system in order to ensure its action within 72 hours, without outside help. It is foreseen to implement this improvement by 2013.
- Provision of an auxiliary MDG to be connected, if necessary, at several points of the plant. It is foreseen to implement this improvement by 2013.
- Completion during 2013 of the new emergency power system, which has been designed considering seismic loads resulting from seismic qualification of the current site. Implementation of an alternative system to feed the fuel storage pools that allows the monitoring of relevant parameters from outside the pool building (It is foreseen to implement this improvement by 2013).

1.1.1.1.5.2. Earthquake severity that the plant can withstand without losing confinement integrity

The integrity of the containment structure of the CNA I will be checked for an earthquake more demanding than the RLE which corresponds to an earthquake of 10^5 years recurrence. Therefore the capacity of the containment structure will be evaluated for a seismic demand corresponding to an annual frequency of exceedance of 10^{-5}. In consequence, the confinement function of containment will cover level 4 of the defence in depth.

Additionally to the results to be obtained from SMA and the improvements arising from that study, it was decided to install measures for severe accident management, such as passive auto-catalytic recombiners (PARs) - aimed at ensuring the containment function. During the scheduled shutdown of May-June 2012, the location of the recombiners will be determined, taking into account the geometry and release of hydrogen within the containment. It is estimated that the installation of 50 units be completed by the end of 2016.

1.1.1.1.5.3. Earthquakes exceeding the DBE that may cause flooding or low lake level beyond the design basis

The nearest dam to the Atucha site is Yacyretá, 1200 km upstream. An earthquake in the Yacyretá area that exceeds the design basis of the dam, capable of breaking it, could cause an extreme level of the river that is precisely the point considered in the next item. The flood wave would arrive to Atucha in the order of 30 days after its rupture. The CNA I have a second heat sink system (SSC) capable of removing heat in situations where the pump house is unavailable. CNA II has considered the breakage of the dam within its design basis so that the pump house is designed for that extreme level. As was made explicit in the corresponding point, CNA I plans to add a pump (fourth pump) to its cooling water river ensured system (UK) which will be housed in the pump house of CNA II, capable of withstanding the maximum level given by the breaking of the Yaciretá dam (facility improvement). In turn, these systems (SSC and UK) will be reviewed within the list of structures, systems and components (SSC) of the seismic re-evaluation of the plant.

Specialists are checking the scenario in which an earthquake affecting Yacyretá can coincide with one at the Atucha site. It is hoped that this information is available by the end of 2012.

1.1.1.2. Atucha II Nuclear Power Plant

1.1.1.2.1. Earthquake against which the plant is designed

The original design criteria for Atucha II nuclear power plant (CNA II) used by KWU / Siemens were based on a maximum ground acceleration of 0.05 g as agreed by the designer with the Licensee. Design principles and construction measures, valid for regions with low seismicity, were applied. The mechanical and electrical components, including their supports, were designed for a maximum horizontal acceleration of 0.15 g and a maximum vertical acceleration of 0.075 g.

After the beginning of the construction of CNA II in 1980, the Licensee developed a specific seismic hazard assessment for the site (GNZ, Gil-Nafa-Zamarbide) using a deterministic approach (Deterministic Seismic Hazard Analysis, DSHA), which concluded that the seismic hazard at the site of Atucha corresponds to a peak ground acceleration of 0.10 g. On that basis, the Regulatory Authority issued in 1985 a regulatory requirement, RQ-26, requesting the Licensee (NA-SA) to develop a seismic safety evaluation for some safety related systems of CNA II . This RQ-26 required the following:

"Dynamic analysis must be made in safety-related systems in order to obtain the behaviour of those systems when the site, where the plant is located, is affected by a seism of the characteristics which are those obtained in the specific study for the CNA-II site done by the R.E. (Gil-Nafa-Zamarbide Advisers).

The analysis to be made can be simplified, taking into account at least, e. g.: work outline: H. Shibata, 7 Smirt, K 12/10, Aug. 1983, but detailed enough to be representative of the behaviour of the systems. The systems to be analyzed will be, at least, all those related to the operation of the primary and moderator (pumps, piping, valves, etc.) and systems which must necessarily operate to take the reactor to a cold shut-down and keep it in these conditions as long as necessary"

The above evaluation identified three seismotectonic sources and, using various attenuation relationships available at that time, calculated peak and spectral acceleration values that would be generated by these three sources. In fact, the source that eventually ruled the design was an earthquake of Richter magnitude M = 5.5, postulating its occurrence in the Paraná fault located 20 km east of the site.

The broad band standard response spectrum recommended by the U.S. NRC Regulatory Guide 1.60 (which was also presented in the appendix to the IAEA Safety Guide applicable in 1981, 50-SG-S1) was used to represent the ground motion. This response spectrum characterizes a wide frequency range of the spectrum.

GNZ advised to undertake complementary studies of the soil at the Atucha site, for which aeromagnetic studies were developed by Woodward-Clyde and Associates and the Naval Hydrographic Service (Servicio de Hidrografía Naval).

The study conducted by Woodward Clyde Argentina in 1986, correlated the results from available borings and wells surrounding the CNA II plant site with the conclusion that there could be faults in the Paraná Formation (Miocene), but there is no evidence to indicate deformation in the Puelches Formation (Lower Pleistocene).

At the end of 1994 Kajima from Japan, Córdoba University and ENACE (Empresa Nacional de Energía, constructor at the beginning of the project) developed a dynamical study of the Reactor Building to determine its vibration modes and characteristic frequencies to allow the calibration of the Reactor's numerical models to be developed in the near future.

After that date and till 2006, in concordance with the delay in the construction of the plant, there was no activity on topics related to the dynamical behaviour of CNA II. At the end of that year the contacts with Atomic Energy Canada Limited (AECL) started in order to find an answer to the problems outlined by the regulatory requirement (RQ-26) imposed in 1985 by the Regulatory Body.

The re-assessment of the seismic hazard of the Atucha site was undertaken. For this purpose, NA-SA contracted two international consultants: James J Johnson & Associates (JJJ, USA) and AECL (Canada) who performed the probabilistic seismic hazard assessment (PSHA) in an independent way.

As will be shown later both studies coincided regarding the determination of peak ground acceleration and provide a solid framework for the safety assessment of CNA II, because:

- Similar geological, geophysical and seismological databases were used.
- The random and epistemic uncertainties were considered adequately.
- The principles outlined in the Safety Guide of the International Atomic Energy Agency were followed (IAEA Safety Guide SSG-9 "Seismic Hazards in Site Evaluation for Nuclear Installations").

Below, both studies are briefly presented and the conclusions of a peer review that was performed on both studies "Peer Review of Aspects of the Seismic Safety Evaluation Programme of CNA II - Argentina" JJJ & Associates, March 2012.

1.1.1.2.1.1. Study by AECL

The purpose of the study done by D'Appolonia for AECL consists on the update of the seismic hazard analyses for the site, in order to determine if the GNZ design basis from 1981 are currently valid. This update includes a review of the deterministic assumptions that were used by GNZ to define the seismic design basis, but the objective is comparing the design response spectra derived by GNZ with Uniform Hazard Spectra (UHS) derived on the basis of a Probabilistic Seismic Hazard Assessment (PSHA), which is in accordance with the IAEA standards.

According to the Specific Safety Guide SSG-9 (IAEA), the maximum design vibratory ground motion is termed the SL-2. This level of ground motion is required to have a very low probability of being exceeded during the lifetime of the plant and represents the maximum level of ground motion to be assumed for design purposes. D'Appolonia's experience, consistent with internationals guidelines, is that the normal requirement for a critical facility is a 10^{-4} per annum probability (10,000 year return period).

Calculation of probabilistic seismic hazard requires basically the following input, which constitutes a model of seismicity for the region.

- Source Geometry. It is the geographic description of the seismic sources in the region around the site. A seismic source is a portion of the earth associated with a tectonic fault or, if individual faults cannot be identified, with an area of homogeneous seismicity. Source geometry determines the probability distribution of distance from the earthquake to the site.
- Seismicity. Correspond to the rate of occurrence and the magnitude distribution of earthquakes within each seismic source.
- Attenuation Functions. Relationships that allow the estimation of ground motion parameters at the site as a function of earthquake magnitude, source to site distance and soil conditions at the site.

D'Appolonia describes the geological and tectonics of the region, and historical seismicity. The region surrounding CNA I-CNA II (where the region is considered to represent a minimum radio of 320 km) has experienced minimal historical seismicity. This fact makes it impractical to define a recurrence rate; therefore a distribution of earthquakes based on considerations of the behaviour of Stable Continental Regions (SCR) throughout the World was adopted, using the methodology defined by EPRI for this case. The model employed is shown in *Figure N° 1-8*.

Regarding the seismic catalogue, a study window was taken from Latitude 29° S to 39° S and from Longitude 53° W to 65° W. The values were transformed in each case to moment magnitude. A detail of the catalogue is shown in *Figure N° 1-9*.

Seismic sources were characterized by recurrence relationships that describe the number and magnitude of historical and expected events at the source. The upper limit magnitude capable of being generated in a source model was also defined.

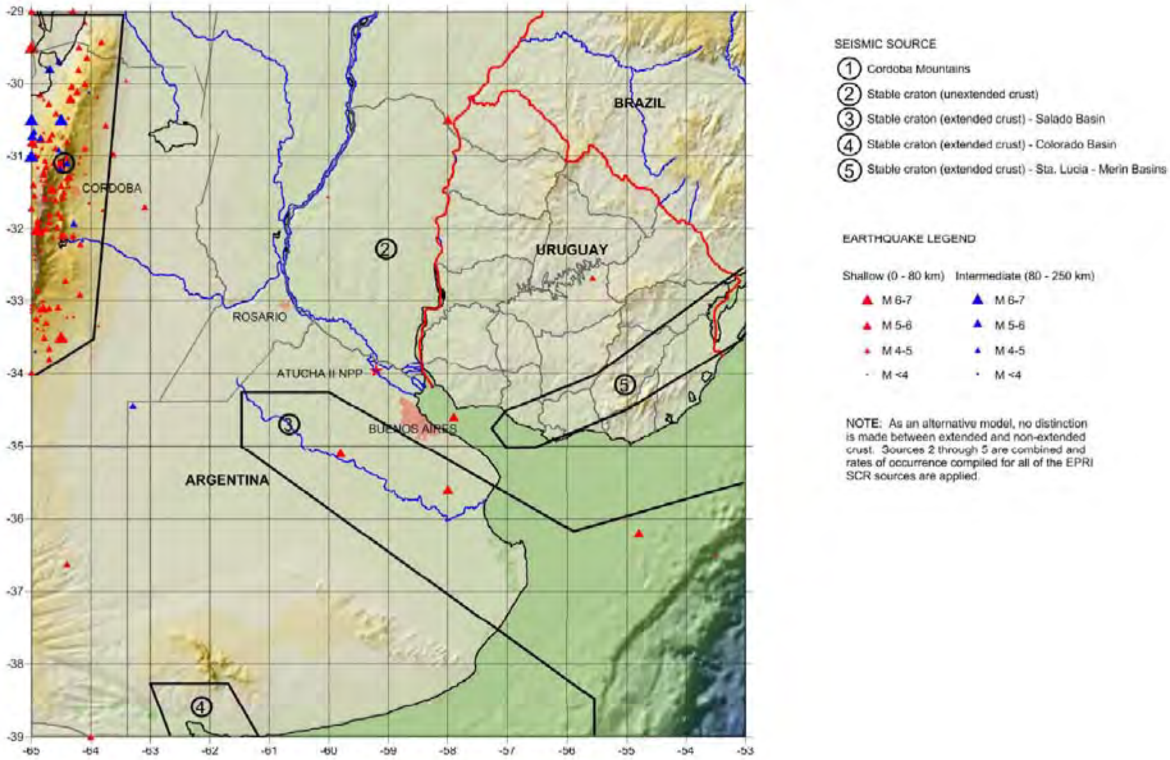


Figure N° 1-8: Seismic source model used for the study.

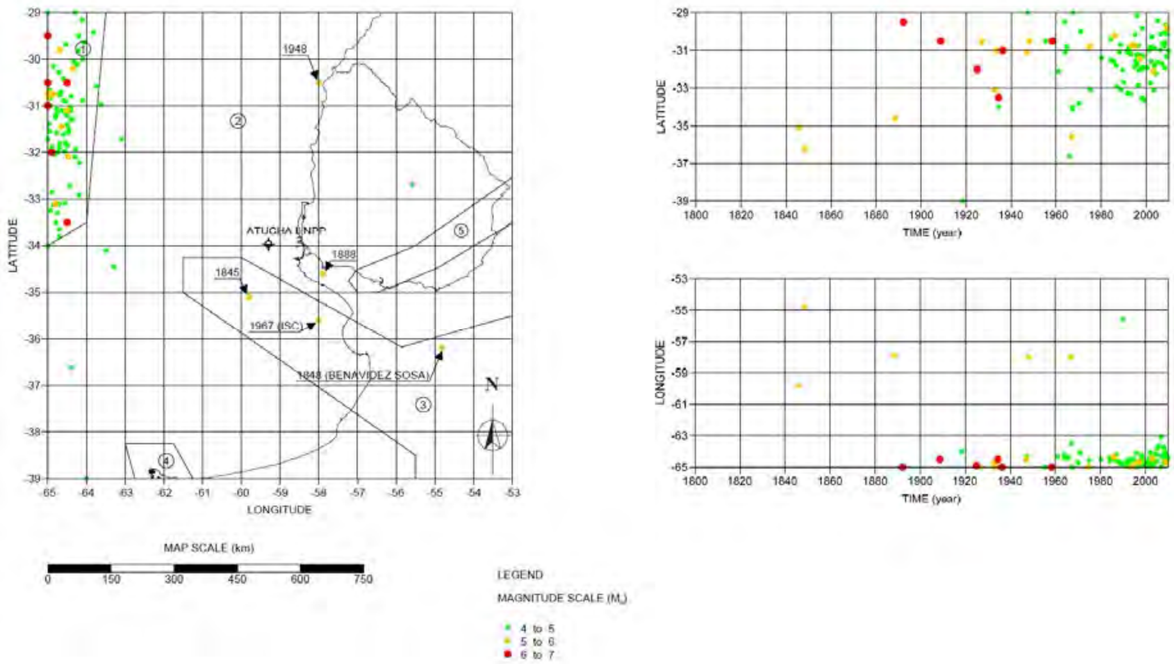


Figure N° 1-9: Catalogue of seismic events.

Because the Atucha site is located within a stable continental region, ground motion prediction equation (GMPE), also known as ground motion models or attenuation relationships, developed for stable intra-plate regions are most appropriate to capture the seismotectonic environment. Due to the characteristics of shear wave velocity below the foundation soil it is necessary to consider the amplification effects. For this, an appropriated weighting factor was applied in the logic tree to the Atkinson and Boore (2006) GMPE, as it is the only one within the selected equations that allows predicting the ground motion both for hard rock and for specific-site conditions.

The study also includes the treatment of uncertainty, distinguishing between two types:

- Random. Stochastic variability that results from natural physical processes. Cannot be reduced even with the collection of additional information.
- Epistemic. Uncertainty that results from lack of knowledge about the true state of its nature. In principle this variability can be reduced with the collection of additional data.

It is important to keep the distinction clear between these two types of uncertainties because they are treated differently in seismic hazard analyses. Epistemic uncertainty is addressed using a logic tree formulation. This implies considering various seismotectonic models, GMPE, seismicity rates and upper limits of magnitudes, using different weighting factors based on the hypothesis taken.

As for the results of the probabilistic analysis the following stand out:

- Seismic hazard curves (namely curves relating the spectral acceleration based on the annual probability of exceedance - in rock -).
- Uniform Hazard Spectra (for a return period of 10,000 years for rock and soil with shear wave velocity of 350 m/s). The GNZ spectrum is added for comparison.
- Deaggregation of seismic hazard.

Figure N° 1-10 shows the uniform hazard spectra (mean, median and fractals 15 and 85) compared with the spectrum of GNZ, thus D'Appolonia concludes that the floor response spectra derived from the spectrum of GNZ are conservative.

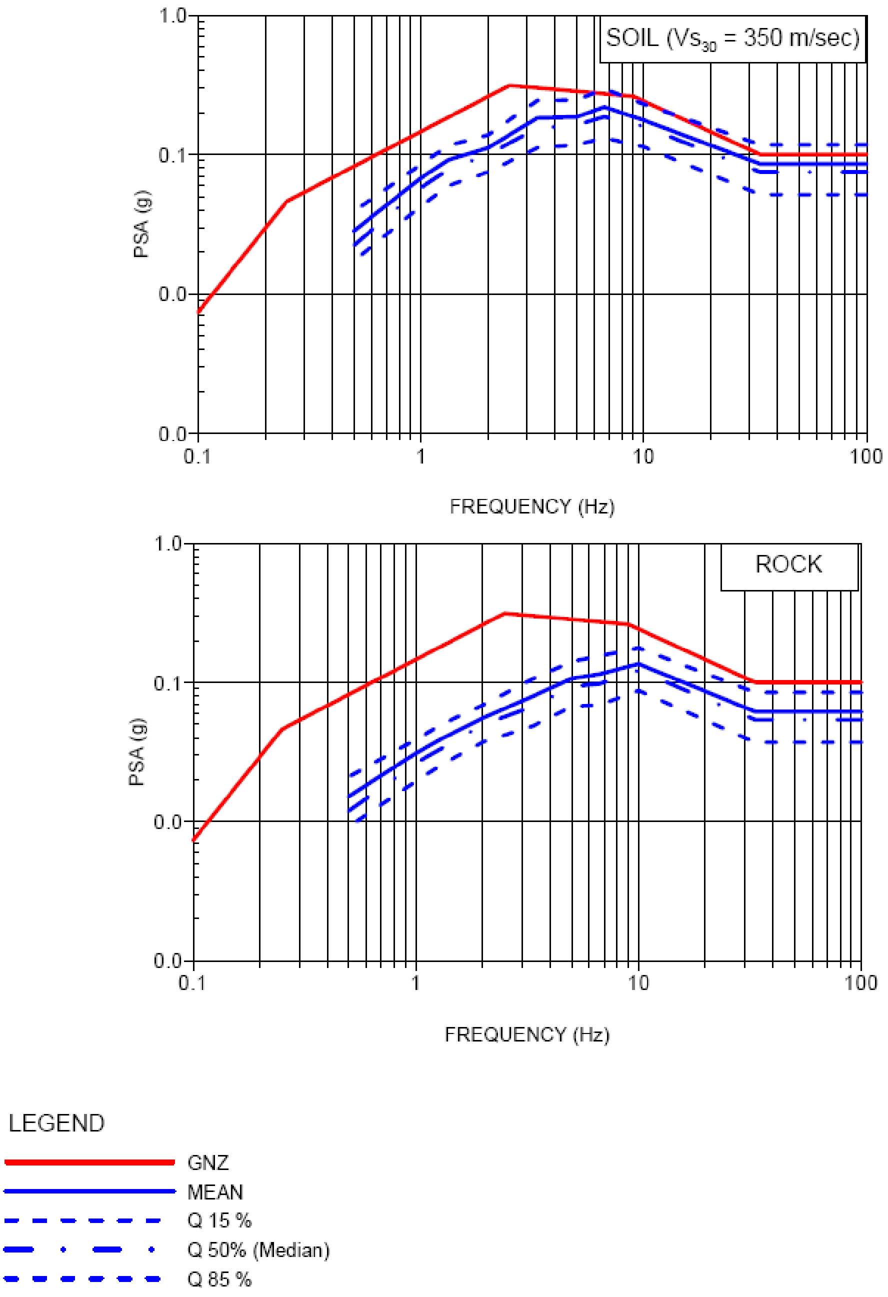


Figure N° 1-10: Comparison of the uniform hazard spectra by D'Appolonia, and the GNZ spectrum.

1.1.1.2.1.2. Study by JJJ and Associates

In addition to the discussed above, consultant JJJ and Associates (JJJ) conducted a probabilistic assessment of seismic hazard within the framework of tasks for the assessment of the fracture mechanics in the primary system.

In that study, the following areas were considered as seismic sources:

5. Andes.
6. Pre-Andes.
7. Paraná Fault (was considered in a model but not in another).
8. Background seismicity, with a maximum magnitude of 6.

Those sources are shown in *Figure N°1-11*.

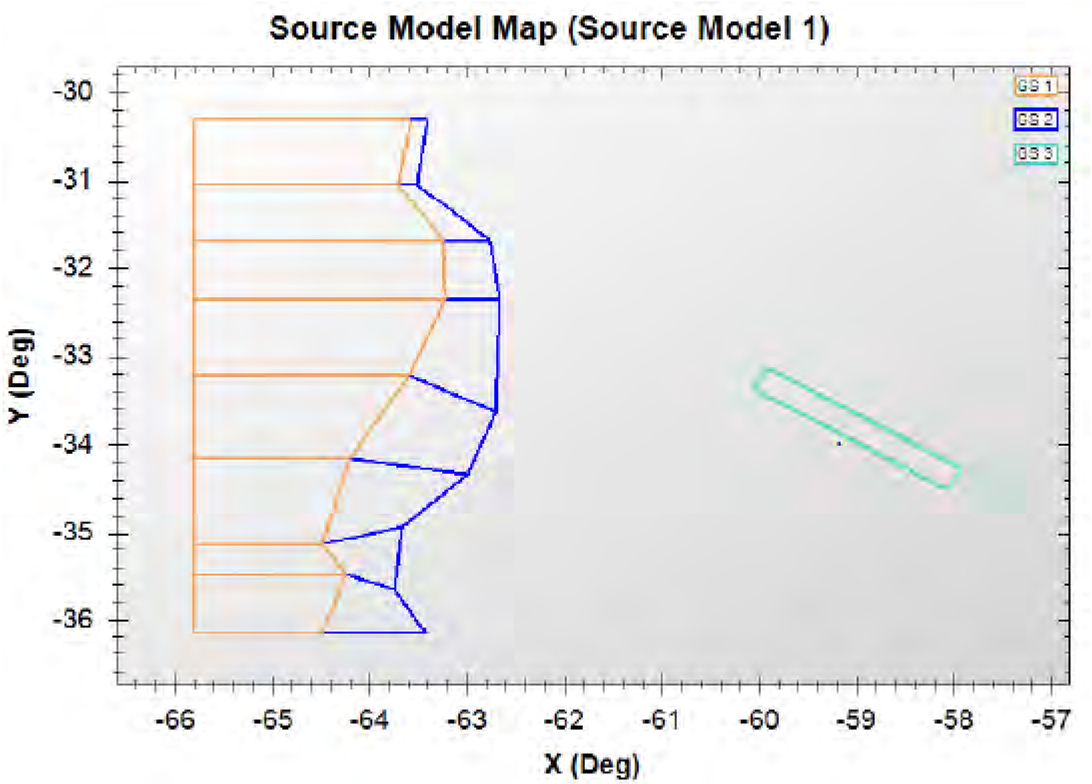


Figure N° 1-11: Model of seismic sources considered in the study.

The ground motion prediction equations were selected according to the distances to the site location and its characteristics.

The following results of the study are presented:

- Seismic hazard curves (Peak ground acceleration, spectral acceleration for different frequencies, with a range of annual probability of exceedance of 10^{-6}). See *Figure N ° 1-12*.
- Uniform Hazard Spectra –UHS. See *Figure N° 1-13*.

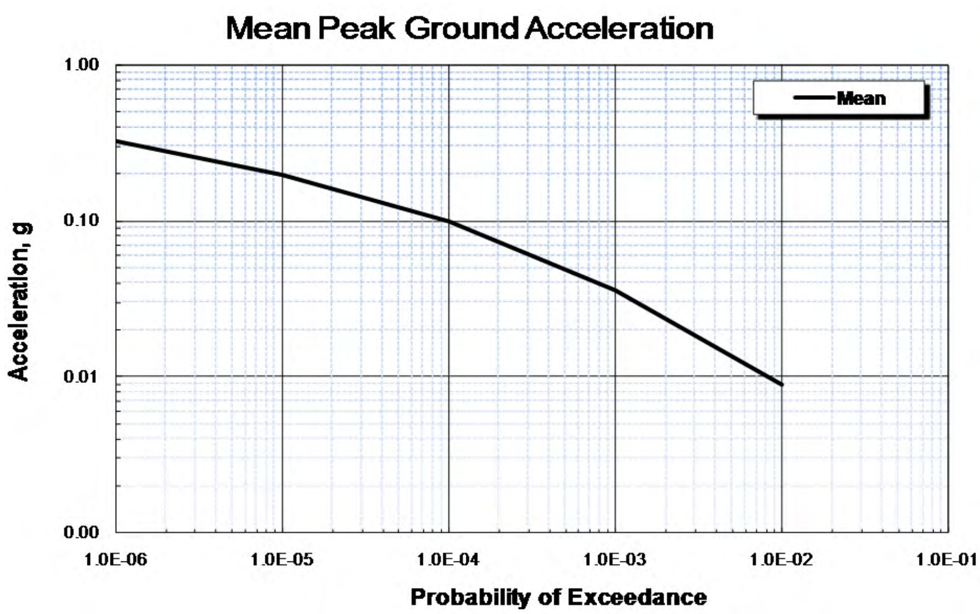


Figure N° 1-12: Seismic hazard curve for CNA I-CNA II site.

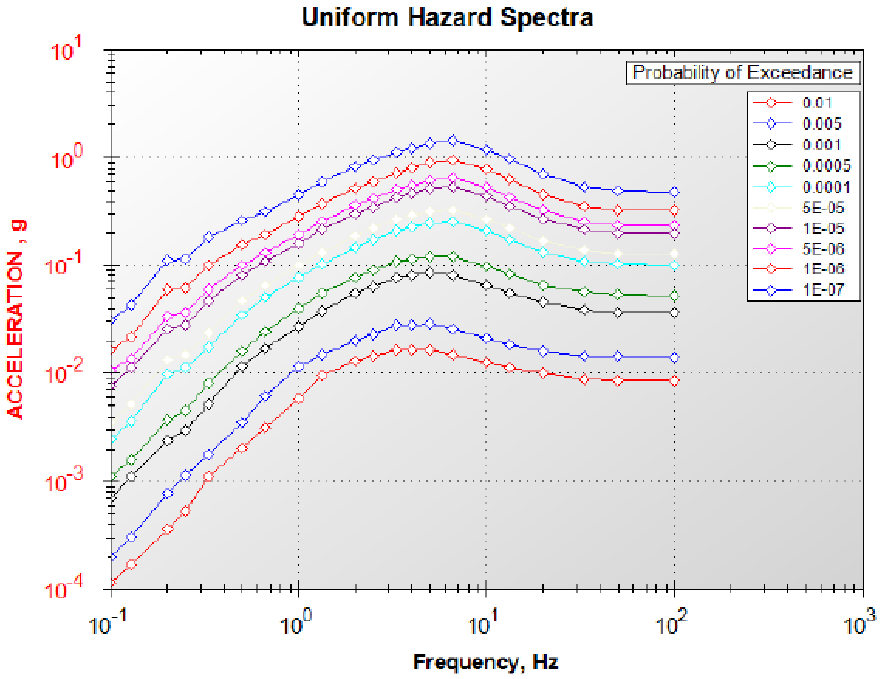


Figure N° 1-13: Uniform Hazard Spectra for different annual probability of exceedance.

1.1.1.2.1.3. Peer Review of Aspects of the Seismic Safety Evaluation Programme of CNA II

Comparison between the studies of AECL and JJJ.

The comparison between both studies concluded that the seismic hazard at the site of the CNA I-II can be based on the results of both PSHA studies, JJJ Study and the Study of AECL. Mean uniform hazard response spectra (UHRS) corresponding to an annual probability of exceedance of 10^{-4} forms the basis for the definition of a design basis earthquake in the future for new nuclear facilities, and the definition of the review level earthquake (RLE) for the seismic safety evaluation of CNA I and CNA II. The results from the JJJ study for the mean UHRS of 10^{-6} form the basis for further assessment, including the study of fracture mechanics of the Nuclear Steam Supply System (NSSS) of CNA II and other related systems at risk.

Figure N°1-14 shows the proposed Review Level Earthquake (RLE) as envelope of mean 10^{-4} UHRS from JJJ and AECL PSHA Studies and includes the Design Factor of US ASCE 43-05.

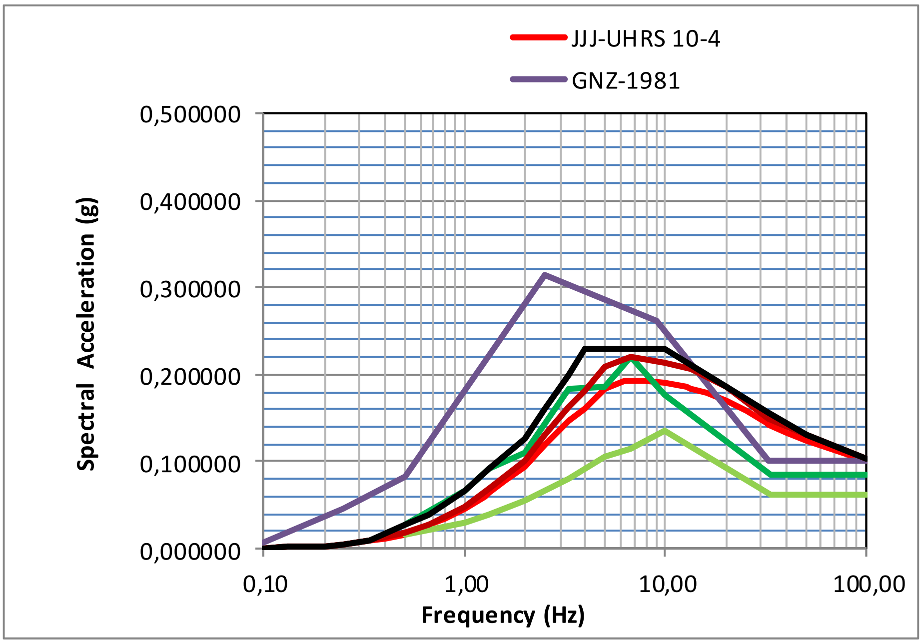


Figure N° 1-14: Envelope Spectra from Uniform Hazard Response Spectra (UHRS) from JJJ/AECL Studies, including US ASCE43-05/DF considerations

1.1.1.2.2. Provisions to protect the plant against the DBE

As mentioned above, the original design criteria of CNA II considered a maximum ground acceleration of 0.05 g. Design principles and construction measures, valid for regions with low seismicity, were applied. The mechanical and electrical components, including their supports, were designed for a maximum horizontal acceleration of 0.15 g and a maximum vertical acceleration of 0.075 g.

Later the Regulatory Authority issued in 1985 a regulatory requirement informing that the horizontal acceleration that should be considered as design basis for safety-related SSC was 0.1 g and not 0.05 g. However, since the construction had already begun, it was required to assess the impact of this difference in the constructive measures and consider the implementation of improvements arising from this assessment.

This regulatory requirement was complemented by another in which NA-SA was asked to carry out dynamic analysis to obtain the behaviour of safety related systems when they are affected by an earthquake, like the one specified for the Atucha site by GNZ. The systems to be analyzed should be at least, all those related to the operation of the primary system and moderator and those systems which must necessarily remain functional to achieve a cold shutdown of the reactor and keep it under these conditions as long as necessary.

NA-SA decided to extend the analysis discussed above to a seismic margin assessment (SMA) which consists of evaluating the status of the SSC in relation to their ability to perform its safety function before the occurrence of a specific earthquake as the Review Level Earthquake (RLE).

The RLE to be used (for Atucha site and quantified as 0.1 g PGA) is developed in Section 1.1.1.2.1.3. It was defined with a sufficient margin over the original design basis earthquake, in order to improve the safety of the plant and find weaknesses that could limit the capacity of the facility to safely bear the consequences caused by a seismic event greater than the design basis.

The use of the SMA methodology, results is an appropriate focus and is sufficient for CNA II considering the low level of seismic hazard at the site. This methodology aims to determine the seismic capacity of "high confidence" for the nuclear power plant as a whole, called capacity of "high confidence low probability of failure" (HCLPF), which estimates the earthquake’s level which could affect fundamental safety functions. In seismic margin studies, it is considered that HCLPF capacity is the seismic level for which there is a 95% confidence that the probability of relevant failure in safety systems of the plant is less than 5%.

1.1.1.2.2.1. Identification of SSC needed to achieve a safe shutdown condition

As discussed earlier, the CNA II is conducting a seismic margin assessment. To develop this assessment, NA-SA hired JJJ and AECL, and agreed with them on the use of the methodology of Conservative Deterministic Failure Margin (CDFM) documented in EPRI NP6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin ".

The application of this methodology requires the definition of the RLE, which in the case of CNA II was calculated according to the results of PSHA taking the uniform hazard spectra of 10^{-4}/year adjusted to be consistent in terms of risk, with the application of the methodology ASCE / SEI 43-05, "American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities", 2005.

Using CDFM methodology results in a conservative approach because the loads used for verification of SSC are those generated by the RLE simultaneously combined with normal operating loads.

The seismic evaluation program consists of five phases, namely:

- Phase 1: Scope of the study and preliminary inspection walkdown of the plant.
- Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and Systems Walkdown.
- Phase 3: Seismic response and capacity evaluation of SSC.
- Phase 4: Complementary Inspection Walkdown of the seismic capacity and screening process.
- Phase 5: Detailed analysis and evaluation for seismic qualification.

In relation to Phase 1, in October 2011 a walkdown was conducted whose scope was limited to the following systems:

1. FCL (Refuelling Machine),
2. JA (Reactor System),
3. JE (Main Coolant System),

4. JF (Moderator System),
5. JND (Safety Injection System),
6. KAG (Residual Heat Removal component cooling system) and
7. EPS (Emergency Power System).

Previously, the floor response spectra (FRS) had been obtained for the buildings of the reactor and Switches, Electrical Panels, I & C and control room.

The CNA II is conducting preliminary tests and finishing some constructive aspects such as mounting some supports that are still pending. Therefore, the main conclusions of the plant walkdown were focused on identifying potential issues that require verification via HCLPF capacity calculation.

The SMA uses the concept of success path as part of the analysis. The success path consists of the set of equipment that must remain functional during the occurrence of the postulated RLE and, if operated properly, allows the safe shutdown of the plant and to remain in that state for 72 hours after the earthquake. To demonstrate the seismic margin of a success path, one must evaluate the seismic margin of the weakest component of the success path.

Following the EPRI methodology two success paths have to be selected as independent as possible. Once both success paths have been selected, the safe shutdown equipment list has to be developed by September 2012.

Phases 3 to 5 are going to be developed during 2013.

1.1.1.2.2.2. Major existing previsions

As mentioned above, CNA II is working to make additional checks aimed to prevent damage to the reactor core or spent fuel after the earthquake.

1.1.1.2.3. Plant compliance with its current licensing basis

The activities currently being developed in CNA II correspond to activities under the construction license, that has been complemented by different regulatory communications (such as the RQ-26), constituting the current licensing basis.

With regards to seismic aspects, as discussed above, the plant decided to conduct a SMA. According to existing progress to date, compliance is guaranteed only by the implementation of the resulting actions.

Additionally CNA II is developing a set of mandatory documents like Inspection Service programs, repetitive tests, etc. that will allow, when the plant is in operation, to verify compliance with the licensing basis.

1.1.1.2.4. Specific compliance check already initiated following the Fukushima NPP accident

NA-SA decided to conduct a SMA evaluation to determine the plant’s capacity to deal with seismic events beyond the design basis, considering a RLE of 0.1 g PGA based on the approach of Regulatory Guide 1.208 of the US NRC.

1.1.1.2.5. Evaluation of the margins

1.1.1.2.5.1. Earthquake severity above which severe damage to the fuel becomes unavoidable

As discussed above, a reassessment of the seismic margin of the plant is being conducted using the SMA methodology, which will provide a measure of HCLPF of the robustness of the plant against severe earthquakes. The acceptance criterion is such that the SSC of the two paths that lead to safe shutdown of the plant must have a capacity greater than 0.10 g PGA with an annual probability of 10^{-4}/year.

The ruggedness of the plant will be improved after the implementation of a set of measures including:

- Implementation of measures known as easy fixes arisen from the plant walkdown of October 2011. It is foreseen to be implemented by 2013.
- Provision of an auxiliary Mobile Diesel Generator (MDG). It is foreseen to be implemented by the end of 2014.
- Implementation of an additional system to refill the fuel elements storage pools from an alternative reservoir (water inlets from the underground water-table, existing tanks, etc.) and with electrical power from the MDG. It is foreseen to be implemented by the end of 2014.
- Flooding of the reactor cavity. It is foreseen to be implemented by the end of 2015.

1.1.1.2.5.2. Maximum earthquake that the plant can withstand without losing confinement integrity

The integrity of the containment structure of CNA II was verified by NA-SA together with AECL, using an earthquake ground motion recommended by the IAEA guide (IAEA 50-SG-S2) and normalized to 0.1 g for both horizontal and vertical direction, applied at the base in a conservative form.

The purpose of that assessment was to determine the general behaviour of the structure and its resistance to seismic loading conditions as those discussed above.

The conclusions show that both the structural stability as horizontal shear resistance of concrete is adequate. Additionally, it was demonstrated that the soil bearing capacity is also suitable. The evaluation was conducted at various points of the building, in particular those considered to be critical, such as the foundation and the columns supporting the UJE building. The effect of temperature in the evaluation was not considered.

In addition to the assessment made and taking into account that the containment must be verified for events beyond the design basis, the approach to be adopted considering the two internationally recognized approaches are being discussed between NA-SA and ARN:

- The US approach requires that a plant HCLPF of 1.67 times the DBE (Design Basis Earthquake) be verified. For the Atucha site, this requirement implies an earthquake of 0.167 g PGA for the verification of the containment.
- The European Requirements approach specifies that a plant HCLPF of 1.4 times the DBE is required. For the Atucha site, this requirement implies an earthquake of 0.14 g PGA for the verification of the containment.

Additionally to the results to be obtained from SMA and improvements arising from that study, it was decided to install measures for severe accident management, such as passive auto-catalytic recombiners (PARs) - aimed at ensuring the containment function, which are going to be mounted prior to the first criticality (middle of 2013).

1.1.1.2.5.3. Earthquakes exceeding the DBE that may cause flooding or low water level beyond the design basis

The nearest dam to the Atucha site is Yacyretá, 1200 km upstream. An earthquake in the Yacyretá area that exceeds the design basis of the dam, capable of breaking it, could cause an extreme level of the river. The flood wave would arrive to Atucha in the order of 30 days after its rupture.

CNA II has considered the breakage of the dam within its design basis so that the pump house is designed for that extreme level.

It should be noted that the effect of a dam break, upstream Yacyretá on the Paraná River, is bounded by the rupture of the same Yacyretá dam.

Specialists are checking the scene in which an earthquake affecting Yacyretá coincides with one at the Atucha site. It is hoped that this information is available by the end of 2012.

1.1.1.3. Embalse Nuclear Power Plant

1.1.1.3.1. Design Basis Earthquake (DBE)

The Embalse nuclear power plant (CNE) plant was originally designed for a Peak Ground Acceleration (PGA) of 0.15 g for the Design Basis Earthquake (DBE) event. The DBE was defined as the earthquake which has an estimated probability of occurrence of not more than 10^{-3} events per year for the particular location. The Ground Response Spectra (GRS) adopted were the Housner type, as was the practice of seismic design of NPPs at the design time of the Embalse plant. This original GRS is shown in *Figure 1-15*. Appropriate adjustments for soil/structure interaction were made to take into account the characteristics of the CNE site.

Nevertheless, later knowledge about the geology of the CNE site indicated that the seismicity of the region was higher than the one assumed during the design and construction of the plant.

In 1980 the Instituto Nacional de Prevención Sísmica (INPRES) performed an evaluation of the ground motion hazard at the CNE site. Using an approach based both on magnitude and on intensity records, INPRES determined a PGA of 0.35 g for the International Atomic Energy Agency (IAEA) Seismic Level 2 (SL-2), with the spectral shape shown in *Figure 1-16*.

In view of this fact, a seismic capacity evaluation was performed in 1982 by the consulting company Structural Mechanics Associates, in order to assess the overall plant adequacy and safety margin against the SL-2 seismic event with a PGA of 0.35 g. The scope of the evaluation included Structures, Systems and Components (SSCs) like electro-mechanical components necessary for the plant´s safe shutdown. This evaluation utilized different criteria than the original design criteria, such as different damping values.

In 1983, D'Appolonia (an engineering consulting company specialized in seismic evaluations) performed a new deterministic evaluation of the seismic hazard at the CNE site, using more updated and comprehensive geologic and seismologic information. Based on an earthquake intensity approach, they obtained a PGA of 0.26 g for the IAEA SL-2 seismic event. The spectral shape in this case was taken from Appendix B of the IAEA Safety Guide 50-SG-S1 (*Figure 1-17*).

Also, a probabilistic analysis performed that same year determined that the 0.26 g PGA was associated with a return period of approximately 7,000 years (*Figure 1-18*). This SL-2 was used in the final verification of the structural design.

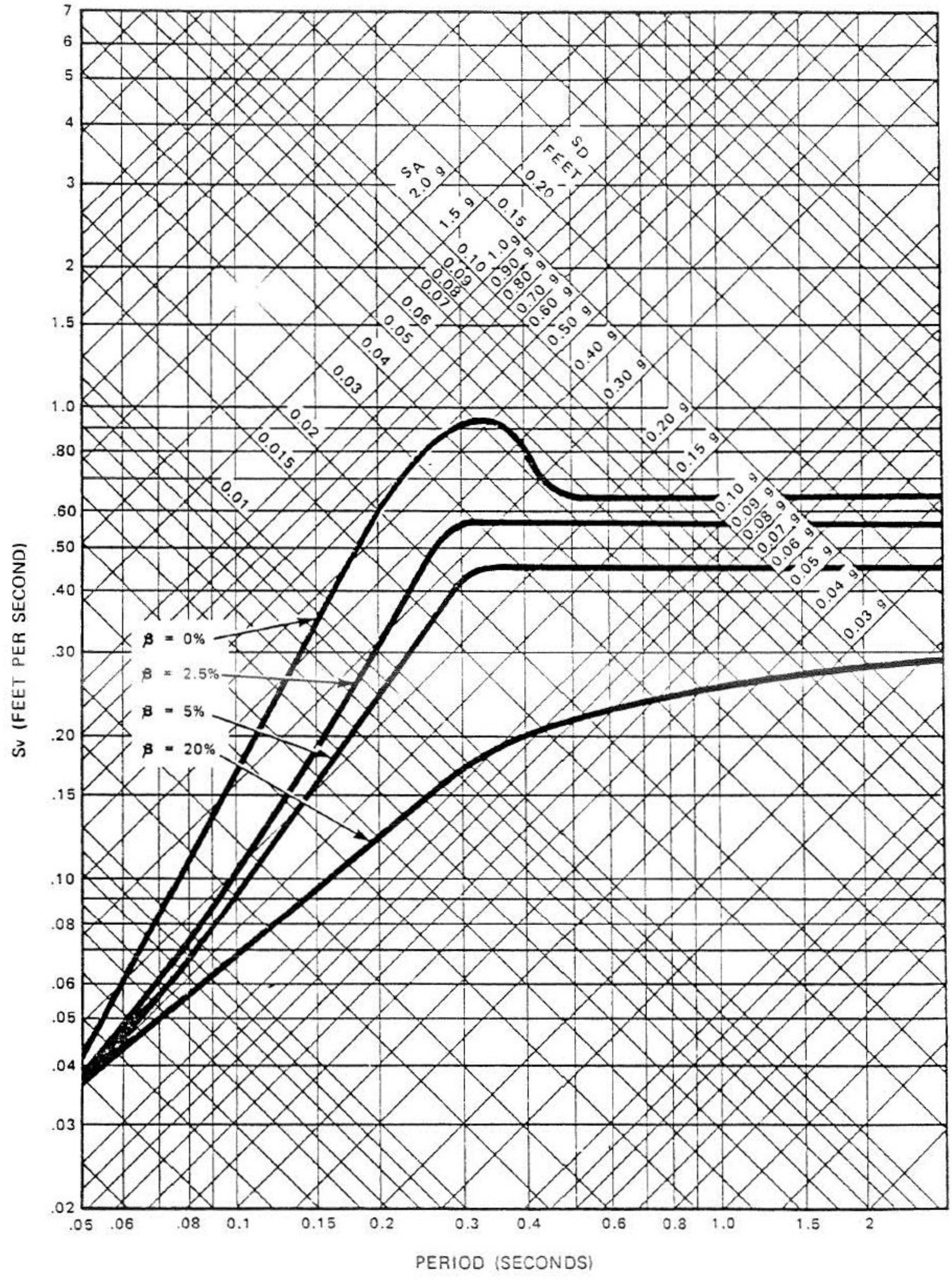


Figure 1-15: Original design horizontal GRS with a PGA of 0.15 g.

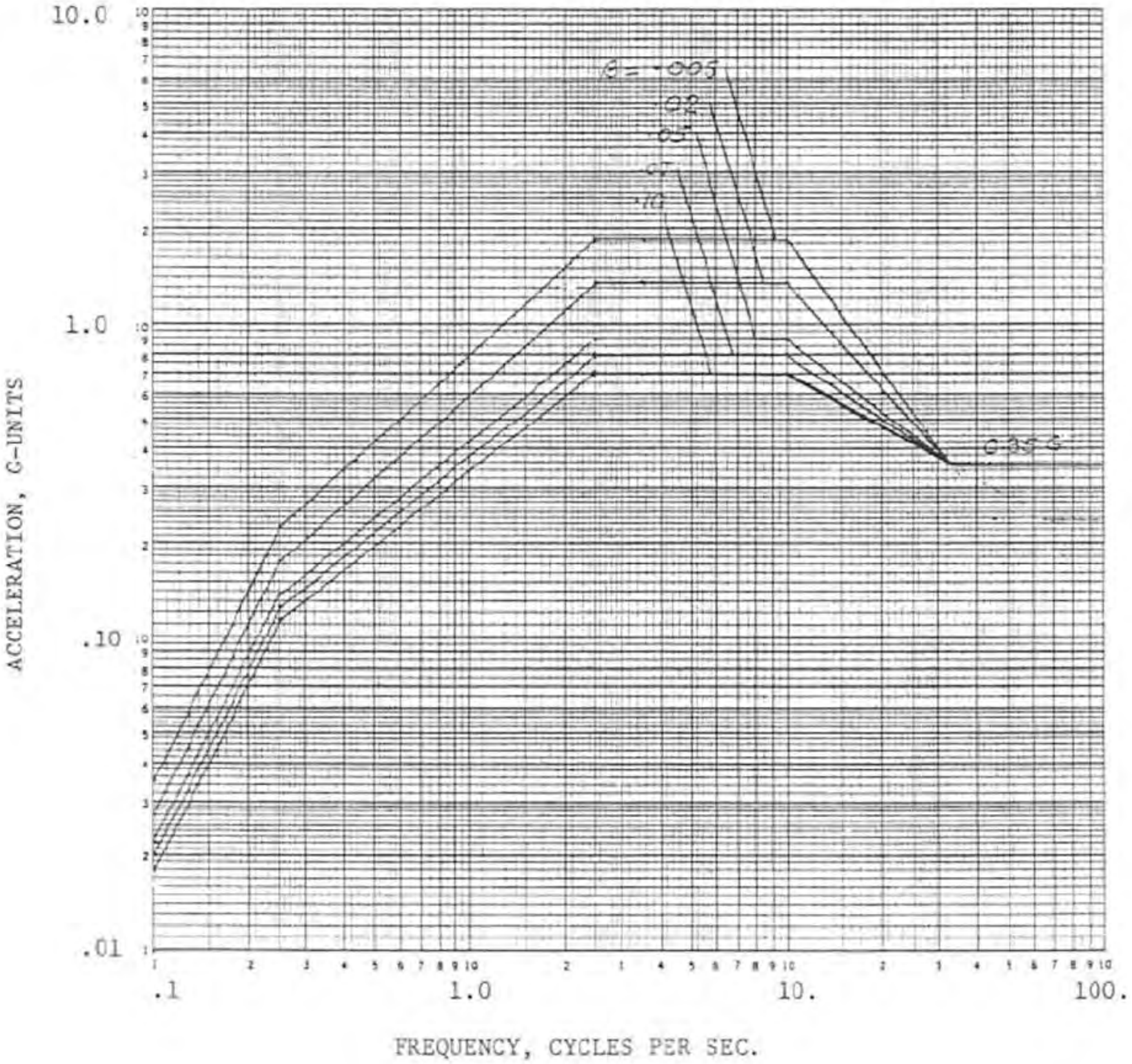


Figure 1-16: INPRES SL-2 horizontal GRS with a PGA of 0.35 g.

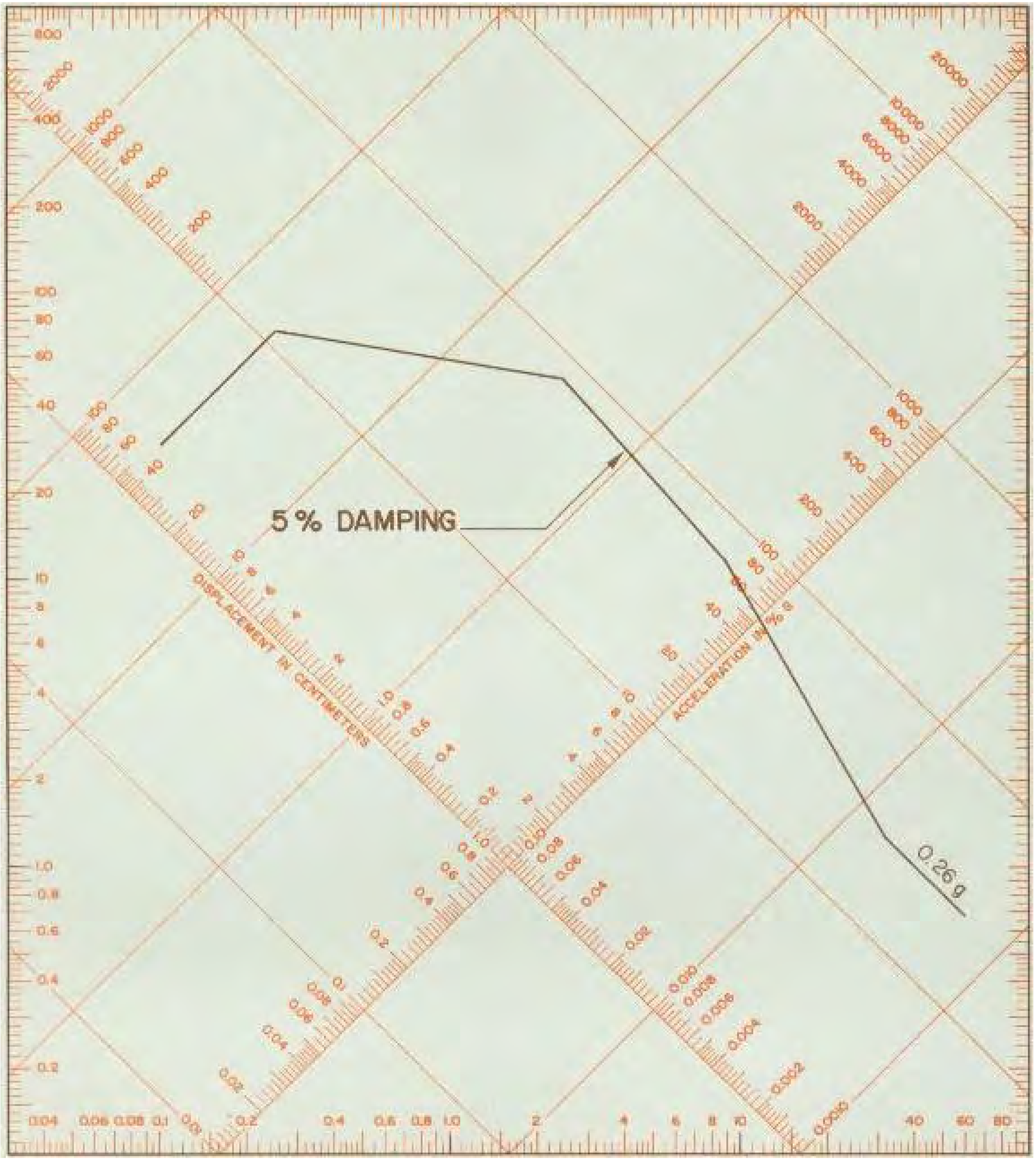


Figure 1-17: SL-2 design spectrum recommended by D'Appolonia in 1983.

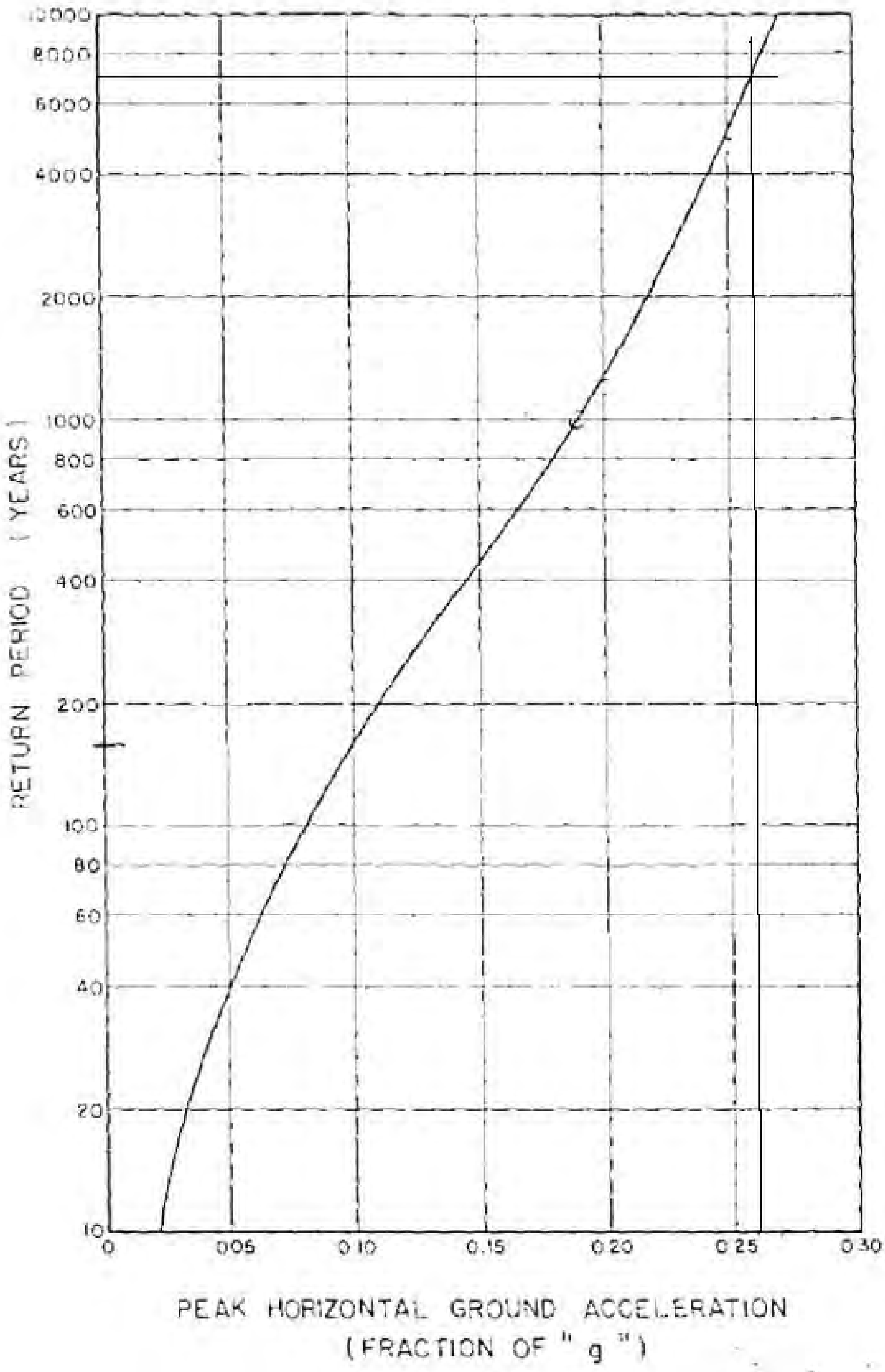


Figure 1-18: Probabilistic seismic hazard analysis and determination of the return period of the SL-2 earthquake with a PGA of 0.26 g.

In the framework of the Embalse life extension project, it was decided in 2006 to undertake the following activities:

- To perform a comprehensive review of the seismic hazard at the site,
- to perform a seismic margin assessment (based on Probabilistic Safety Assessment, PSA),
- to define new seismic demands consistent with the current seismic risk, and
- to implement the seismic upgrades necessary to qualify the plant to the new demands.

The overall activity is currently on-going, managed by NA-SA in conjunction with the plant designer (CANDU Energy Inc., formerly AECL) and its consultants. The tasks involved are briefly described below.

Field Investigations

The geological investigations performed in order to fully characterize the seismotechtonic setting of the CNE region, and quantify local fault activity, included the following:

- Flyover under low-sun-angle conditions, to identify active faults.
- Geophysical profiling, to identify where there are breaks in the bedrock surface that could be buried fault scarps.
- Geomorphic mapping and soils survey, to characterize landform surfaces and identify their relationship with soil types, mapping soils to the group and subgroup level.
- Trenching, at locations determined by the results of the previous three tasks. Soil samples were gathered at the trenches, to be used in the next task.
- Age dating of fault movement, to determine the age of the last fault movements, as well as the frequency that the faults have moved in the past.

The above tasks were conducted by D'Appolonia, with the participation of experts from both local and international universities. The seismotectonic framework model, or seismic source model, constitutes the first step in the development of the probabilistic seismic hazard analysis.

Probabilistic Seismic Hazard Analysis

The probabilistic seismic hazard analysis (PSHA) was conducted by another company contracted by the plant designer, called Klohn Crippen Berger Ltd. (KCBL) of Calgary, Alberta. The PSHA performed for the CNE site included the following steps:

1. Definition of the spatial distribution of earthquakes into source zones, either as faults or area sources.
2. Description of the frequency and intensity of earthquakes within each source zone with a magnitude-recurrence relationship.
3. Estimation of the intensity of ground motion using a set of ground motion prediction equations (attenuation relationships) appropriate for the geologic conditions.
4. Calculation of the frequency of exceeding a specified level of ground motion at the site by integrating the hazard contributions from all source zones over all magnitudes and distances. The seismic hazard curves for mean, median and 84th percentile are shown in *Figure 1-19*.
5.

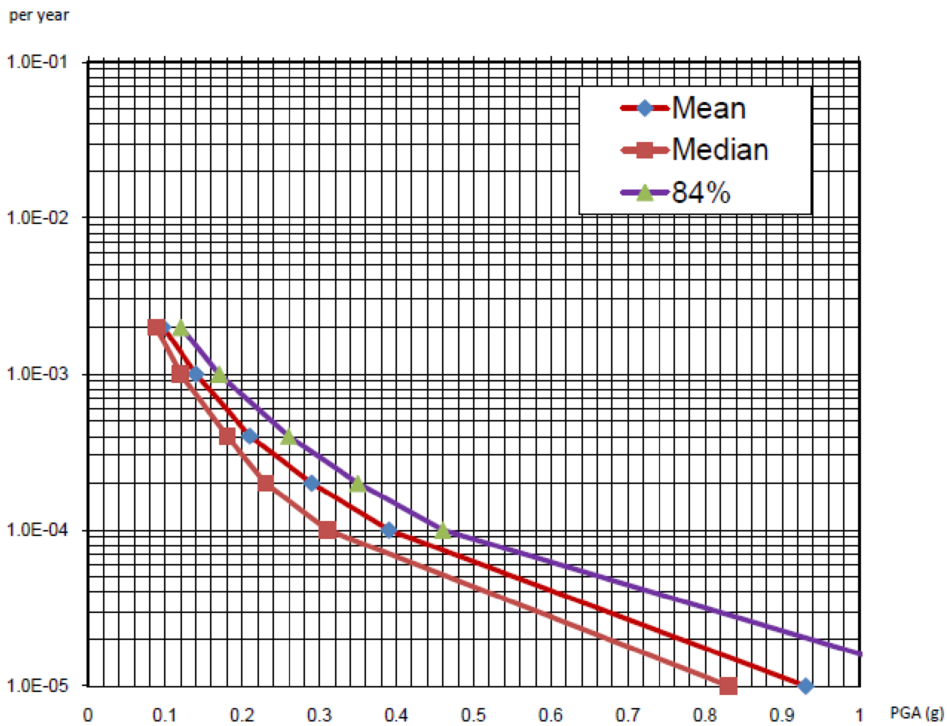


Figure 1-19: Seismic Hazard Curves for CNE.

The final result of this analysis is the seismic hazard at the site, expressed in terms of Uniform Hazard Spectra (UHS). The UHS is a plot of the ground motion acceleration for a fixed (or uniform) probability, versus spectral period. *Figure 1-20* shows the mean and median UHS obtained for CNE, at a 10,000 year return period.

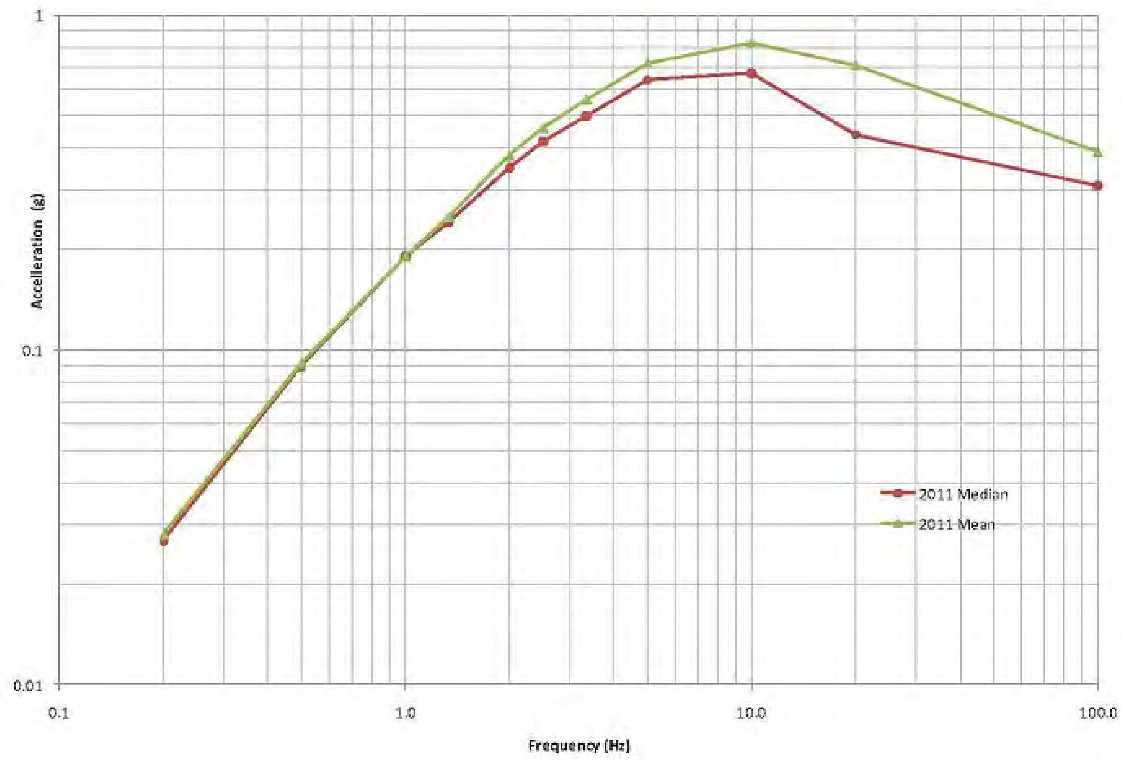


Figure 1-20: Mean and median UHS for the Embalse site, with a frequency of 10^{-4}/y and 5% damping

<u>Floor Response Spectra</u>

The Floor Response Spectrum (FRS) is the response spectrum of the motion of a particular floor or elevation in a structure when the structure is subjected to the design seismic motions.

The UHS mentioned above was used to develop FRS for several levels of the various plant buildings. These FRS are the final seismic demand, and will be used for seismic design and assessment of SSCs.

1.1.1.3.2. Provisions to protect the plant against the DBE

The approach adopted at CNE for the seismic re-qualification has been to perform a seismic margin assessment (SMA) based on probabilistic safety analysis (PSA). This means that both seismically qualified and non-seismically qualified SSCs are credited to perform safety functions in the event of an earthquake.

The SMA determines the additional seismic margin plants have, by virtue of their conservative design, to withstand earthquakes larger than the Design Basis Earthquake (DBE). The SMA measures the robustness of the plant to withstand earthquakes of a given g level. The seismic margin is defined in terms of the high confidence of low probability of failure (HCLPF) capacity of each critical SSC and the overall HCLPF of the plant.

The SMA is currently on-going at CNE, and will imply a thorough analysis of the seismic capacity of the SSCs important to safety, as well as the physical interactions and seismic induced floods and fires. The following paragraphs briefly describe the activities that will be performed in the frame of the SMA.

The main steps in the SMA are listed below:

- Collection and review of seismic design guides, design criteria, seismic analysis reports, flow sheets, arrangement drawings, and other design documents and drawings.

- Review of Internal Events PSA sequences related to severe core damage plant final states. It is to be noted that with the SMA approach both seismically qualified and non-seismically qualified SSCs are credited to perform safety functions in the event of an earthquake.
- Identification of structures/components credited above and generation of a so-called "Safe Shutdown Equipment List" (SSEL). This task is complete, and has been supplemented with observations from a preliminary seismic walkdown performed in 2007. It is to be noted that the internal events PSA fault trees do not provide a complete list of equipment for the SMA. Structural items were added to the SSEL, e.g. electrical panels and cabinets, instrument racks, masonry walls, etc.
- Performance of seismic capability walkdowns and screening out of seismically rugged SSEL. The screened out components do not require any further fragility analysis.
- Performance of fragility analysis for selected structures/equipment that remain after the screening analysis. Determination of dominant failure modes and calculation of seismic fragility curves.
- Performance of relay chatter analysis, to identify essential relays related to the PSA credited components, to estimate their seismic capacity, to determine the effects of relay chatter, and to identify potential recovery actions.
- Development of plant models, i.e. Seismic Event Trees (for seismic induced failures and for random failures).
- Quantification of accident sequences to derive cut sets leading to severe core damage and to large releases of radioactive material.
- Calculation of the HCLPF value for each seismic core damage sequence and external release sequence.
- Determination of the plant HCLPF for core damage and for large releases. The HCLPF for core damage is the minimum HCLPF value for those event sequences leading to generalized melting of the nuclear fuel. The plant HCLPF for large release is the minimum HCLPF value for those event sequences leading to large releases of radioactive material outside the containment.

The last six tasks are currently under way as part of the overall effort involved in the refurbishment of CNE. The SMA is expected to be finish by mid-2014.

Two seismic walkdowns were carried out in 2011, including one during the plant maintenance outage that allowed inspection of SSCs not accessible during normal operation. The procedure followed for these walkdowns was the one described in the SQUG Generic Implementation Procedure and the EPRI Seismic Margin Assessment Methodology Report (EPRI NP-6041-SL).

The main activities performed during these walkdowns were the following:

a) Identification of all equipment items that are expected to have sufficiently high seismic capacity.
b) Definition of failure modes for components not expected to have high seismic capacity.
c) Gathering and review of detailed information and measurements on equipment and structures to perform seismic fragility evaluations.
d) Observation and recording of deficiencies that may reduce the seismic capacity of components (e.g. missing anchor bolts, loose mounting of relays, excessive cracking of concrete).
e) Identification of spatial interactions (e.g. non seismically qualified equipment above seismically qualified equipment, potential of a pipe hitting a component, presence of unreinforced masonry wall near safety-related items, etc.)
f) Identification of seismic-fire and seismic-flood interactions, including spurious activation of fire suppression systems.

The Safe Shutdown Equipment List included over 1500 structures, systems and components. Many of these SSCs, identified as "leading items", were thoroughly inspected. Others were "wallked by" mostly due to considerations of similarity with the leading items. Comprehensive walkdown reports were produced with the results of the observations. The general conclusion was that CNE has good seismic capability, particularly for the seismically qualified SSCs. Equipment is generally well constructed and adequately anchored. Nevertheless, several recommendations were made for seismic improvements, which range from seismic housekeeping practices to seismic analysis and reinforcement of systems and components.

Some of the identified deficiencies will be addressed immediately. Others will be further analyzed and the solutions will be implemented during the long refurbishment outage (starting by 2014).

Also, it is important to mention that CNE has a seismic monitoring system (SMS) that cope with the consequences of an Operating Basis Earthquake (OBE).The SMS comprises a set of equipment which allow the plant to detect and manage a seismic event larger than the OBE. The OBE corresponds to a seismic event with a PGA of 0.14 g as determined by D'Appolonia in 1983. Due to the fact that the SMS measures the acceleration produced by seismic event, the system provides useful information related to the potential damage in the SSCs.

During the seismic event, the signal obtained by the SMS is compared against the acceleration spectra in frequency produced by the OBE (*Figure 1-21*) providing knowledge about the exceedence of the OBE.

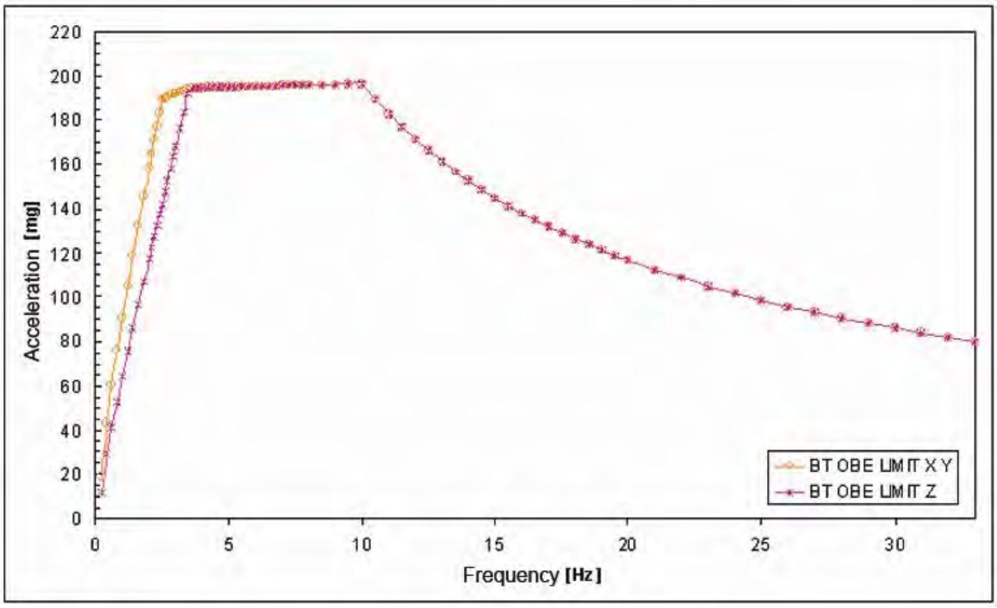


Figure 1-21: OBE spectrum for the three sensors axes.

The system has three sensors, each of them having accelerometers for high acceleration (two horizontal and one vertical). The location of the sensors are: outside of the plant (FFS: free field sensor – trip value: 5 mg), service building (SBS: service building sensor – trip value: 10 mg) and reactor building (RBS: reactor building sensor – trip value: 20 mg). The frequency range that the SMS can measure is (0.2 – 50) Hz.

CNE has developed an operating procedure, supported by the SMS, which list all the foreseen activities to be implemented after a seismic event, higher than the OBE, has taken place. The procedure includes criteria to shutdown and to return to service the reactor and inspection instructions.

The criterion for the operative reactor shutdown is that the plant must be shutdown when significant damage in SSCs jeopardizes the plant safety, the system reliability or the electrical generation. Significant damage is identified based on inspection performed by operators on concrete and steel structures, pipes, pipe supports, mechanical equipment, electrical power supply and turbine. When the result of this walkdown inspection is available, the operator can decide whether the plant must be shutdown or the operation can continue. If the decision is to shutdown the plant, there are a set of criteria to return the plant to service based on the document: EPRI NP-6695 "Guidelines for Nuclear Plant Response to an Earthquake".

1.1.1.3.3. Plant compliance with its current licensing basis

As stated in the previous section, the plant has recently gone through a thorough inspection by a team of experienced seismic capacity specialists. In general, the station was observed to have good seismic housekeeping practice. Only a few minor issues were noted, e.g. some unanchored cabinets and panels, gas containers (bottles) supported at the top with only one chain, a few missing anchor bolts etc. These deficiencies will be immediately addressed and fixed.

Apart from the issues mentioned above, no specific maintenance actions were recommended by the seismic qualification team.

1.1.1.3.4. Specific compliance check already initiated following the Fukushima NPP accident

As was mentioned before, a comprehensive effort to upgrade the seismic qualification of CNE is in progress since 2006, as part of the plant refurbishment activities. So, all the tasks to ensure compliance with current seismic standards and requirements, mentioned in this report, were initiated considerably before the Fukushima accident.

The following section discusses the seismic upgrades that have already been decided to be implemented at CNE.

1.1.1.3.5. Evaluation of the margins

1.1.1.3.5.1. Earthquake severity above which severe damage to the fuel becomes unavoidable

As mentioned, the SMA based on PSA currently under way at CNE will provide a measure (the HCLPF) of the plant's robustness for severe accidents. The UHS corresponding to a non-exceeded frequency of 10^{-4}/y was adopted as safety objective for severe core damage. Based on this UHS, the acceptance criterion is that SSCs for preventing SCD should have a HCLPF capacity higher than 0.39 g PGA.

Apart from the results of the SMA based on PSA, it has already been decided to implement the following two important design changes, in order to increase the robustness of the plant against seismic events:

- Upgrade of the Emergency Power Supply (EPS) system: replacement of the existing 50 kW / 75 kVA emergency power supply diesel generators with diesel generators of increased capacity. The new generators will be rated at approximately 1 MW and will be able to feed both the ECC pumps and the new electrical EWS pumps (see below).
- Upgrade of the Emergency Water Supply (EWS) system: replacement of the existing diesel engine pumps with two new 100% higher capacity pumps to be driven by the new EPS generators. The piping and valves from the EWS pump house to the service building will also be replaced, and the valves that feed emergency water to the steam generators and to the reactor will be duplicated. The higher capacity pumps will be able to feed the ECC heat exchanger.

Also, based on the SMA walkdown and fragility analysis findings, recommendations will be made to improve the seismic capacity of MP/LP ECCS.

With the incorporation of these design changes the seismic capacity of the CNE will be substantially improved, as it will have a seismically qualified power supply to a seismically qualified source of emergency water, and the ECCS will have both the power supply and the cooling water seismically qualified.

1.1.1.3.5.2. Earthquake severity that the plant can withstand without losing containment integrity

The SMA will also provide the HCLPF for large releases of radioactivity. The UHS corresponding to a non-exceeded frequency of 5.10^{-5}/y was adopted as a safety objective for large releases. Based on this UHS, the acceptance criterion is that SSCs for preventing large releases should have a HCLPF capacity higher than 0.55 g PGA.

Apart from the results of the SMA, it has already been decided to implement the following design changes, in order to increase the robustness of the containment function:

<u>Addition of passive auto-catalytic recombiners (PARs) in the reactor building.</u>

The PARs will provide hydrogen control following accident scenarios leading to significant release of hydrogen (LOCA/LOECC and radiolysis in the long term).

The number of PARS required will be determined by the designer (CANDU Energy Inc.) using the GOTHIC code to predict hydrogen concentration profiles inside the reactor building including mixing, distribution, and mitigation. This analysis is the basis for the determination of the required number of PARS to control the average hydrogen concentration in the containment to be below 4 % during the course of the accident progression, including the break discharge phase and the post LOCA conditions in the containment.

Two distinct phases of hydrogen generation will be considered:

- During LOCA period by Zr-steam reaction (LOECC event)
- Post LOCA condition by radiolysis of water.

According to the schedule, CANDU Energy Inc. will deliver the release of the construction document package for this design change by the end of 2014.

<u>Addition of a make-up water supply line from outside the reactor building to the calandria vault.</u>

The design change is a supply of make-up water from outside the reactor building to the calandria vault in the event of a BDBA involving severe core degradation.

The design shall include:

- A water make up line routed from outside the reactor building wall using an existing spare penetration to the calandria vault. To inject water into the calandria vault, this line will penetrate a transition plate installed on top of the existing inspection port of the calandria vault.
- Include two containment isolation manual globe valves and a check valve located on the line outside the containment wall.
- Include a rupture disk on the line near the transition plate at the calandria vault to maintain the pipe dry upstream of the rupture disk.
- Specify the required interfacing pressure and flow rate for the external supply source.

Regarding the civil engineering aspects, the design shall include a platform outside the reactor building to access the containment isolation valves for manual operation and to facilitate connection of the external water source to the pipe end.

Safety and licensing analysis will be done including fault tree analysis for make-up line and MAAP/GOTHIC analysis for input to design.

The design and supply of the external water at the required pressure and flow, as a water source to the make-up line outside the reactor building following a BDBA shall be designed by the utility. The rest of the activities mentioned above will be managed by the designer.

According to the schedule of this design change, the release of the construction document package will be available by the end of 2015.

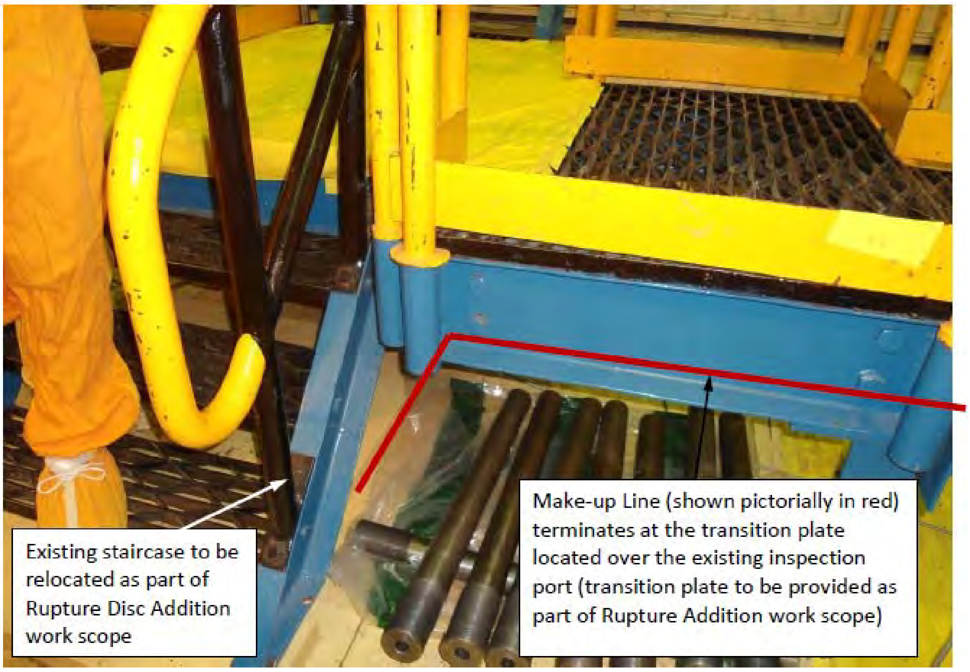


Figure 1-22: Picture showing Make-up Line proposed routing and Termination Point

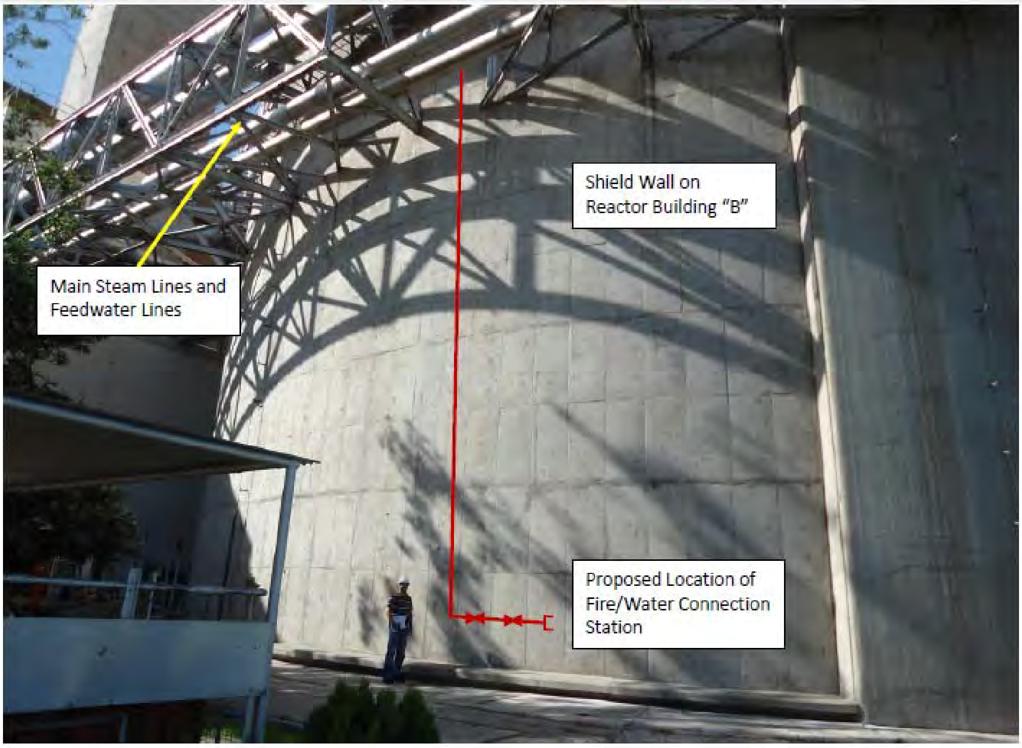


Figure 1-23: Proposed Location of Valve Connection Station

Addition of a seismically qualified rupture disc assembly to the existing inspection port of the calandria vault.

A 24-inch rupture disc (RD) assembly shall be installed on the top of the existing inspection port of the calandria vault to provide additional pressure relief to maintain the calandria vault integrity following a severe core damage accident.

The design of the RD assembly including a transition plate on top of the existing inspection port of the calandria vault shall include:

- A transition plate on top of the existing inspection port of the vault,
- A rupture disk assembly located on the top of the transition plate underneath the staircase on the reactivity mechanism deck,
- A new shielding block on the steel ledge of the vault with suitable opening for steam discharge,
- Nitrogen supply line and the vent line that were mounted on the existing shielding plug to be removed and re-routed to the rupture disc assembly,
- Modification of the steel platform on reactivity mechanism deck to make room for rupture disc assembly,
- Provision of a hole in transition plate for 3-inch calandria make-up water line.
- Addition of an emergency containment filtered venting system.

The containment venting avoids a possible threat to the containment integrity due to over-pressurization. According to the schedule, CANDU Energy Inc. will deliver the release of the construction Document package for this design change in 28 months.

1.1.1.3.5.3. Earthquakes exceeding the DBE that may cause flooding or low lake level beyond the design basis

1.1.1.3.5.3.1. Seismically Induced Internal Flood

As part of the SMA work, the structures and components whose damage can induce flooding were identified and included in the SSEL for the seismic walkdown. Seismic capacity analysis will be performed for those items that could not be screened out during the walkdowns.

Seismic induced flood scenarios will be developed, considering seismic induced impact on structures and components related to seismic flooding interaction. The results of the analysis will be the HCLPF capacity of structures or components that cause seismic induced flood or seismic induced degradation of flood protection systems. When the safe shutdown function of the plant is judged to be affected by seismic/flood interaction, recommendation will be provided to upgrade the seismic capacity of the interaction items.

1.1.1.3.5.3.2. Seismically Induced External Flood

Upstream of CNE there are two dams, one called "Cerro Pelado" and the other called "Arroyo Corto". The size of these dams is relatively small and they are about 20 km away from the Embalse site. Furthermore, the Embalse hydroelectric plant downstream of the NPP provides for a large bypass. This bypass will inherently prevent a significant rise in the water level near the nuclear plant site. Based on these facts, it is judged that flooding at the Embalse site due to seismically induced failure of the upstream dams is not a concern.

In 1982, an analysis of the consequences of earthquakes on the dam downstream of Embalse NPP was carried out. Key observations of this assessment are:

- The Dam was built in 1933; it is 50 m high with a slope of 1.2 to 1.
- The analysis was based on conservative assumptions. It demonstrated that the design criteria for the dam against postulated earthquakes were satisfied.
- The dam will resist a maximum predicted earthquake of 0.26 g with minimum damage. While some cracks in the structure may occur, no significant deformation in its upper part will occur.
- In case of still higher seismic levels (0.34 g), the dam will not collapse, but significant permanent deformation could occur. As a result, the lake water level could drop to the point such that the operation of both the EWS pumps and the service water pumps would be affected. Concerning the EWS actually different improvements are being analysed.

In view of the current updated seismic information for the CNE site, the above analysis will be reassessed as part of the plant life extension project. This analysis will help determine whether or not new provisions and/or procedures to cater to an unsafe low lake level need to be considered for the refurbished plant. It is foreseen to conclude this reassessment by 2014.

1.1.2. ACTIVITIES PERFORMED BY THE REGULATOR

The seismic design of nuclear power plants in Argentina, correspond to the licensing basis for the original design. However, considering that the seismic requirements have been consensually increased at international level, the Operating Organization (NA-SA) decided to conduct a safety assessment of NPPs for the occurrence of external events, among which include the occurrence of earthquakes, from an update of seismic hazard of the sites of Atucha (CNA I andCNA II) and Embalse (CNE).

Although the ARN agrees with the studies addressed by NA-SA, decided to conduct an independent evaluation of Atucha site, since it is an emplacement site of two nuclear reactors. This independent evaluation was conducted together with the Instituto de Investigaciones Antisísmicas "Ing Aldo Bruschi "(IDIA) of the Universidad Nacional de San Juan, Argentina (see section 1.2.2.1).

In the case of CNE NPP and framed in the Life Extension project, ARN revised and issued a decision in relation to the seismic design strategy presented by NA-SA for the refurbishment of the plant. Through this strategy, two categories of modifications were identified (minor changes and major modifications or design changes) and one for the replacement of equipment.

The ARN found acceptable adopting an acceleration value of 0.2 g PGA (similar to other CANDU plants as Quinshan and Wolgsong) as seismic requirement for minor modifications and an acceleration of 0.39 g PGA for major modifications obtained from the uniform hazard spectra (UHS) updated for the Embalse site and a recurrence of 10,000 years for design changes.

For the replacement of equipment, ARN considered that the criterion for component replacement without improving their seismic capacity should be limited only to systems whose failure or malfunction in case of earthquake, generate a negligible impact on the reactor safety.

Additionally, it was decided to conduct a Seismic Margin Assessment (SMA) in order to determine potential vulnerabilities of the reconditioned plant against extreme seismic events.

During 2011 the ARN participated of the two plant walkdowns that the CNE conducted within the SMA. The results of these walkdowns showed that the Structures, Systems and Components (SSC) of the

CNE possess adequate seismic capacity and it is only necessary to implement minor modifications reaching anchors, supports, reinforcements, etc.

As a consequence of the accident occurred in the japanese nuclear power plant of Fukushima Daiichi, ARN requested NA-SA to determine the margin of existing safety by analyzing the behavior of the nuclear power plants CNA I, CNA II and CNE, to the occurrence of extreme events that cause consequences such as loss of total power and the ultimate heat sink for a long time.

This request is framed in the mentioned regulatory requirement, RQ-NASA-38, whose ultimate goal is a Comprehensive Evaluation of Safety (Stress Tests) of the nuclear power plants in Argentina.

Finally, the ARN is working on a proposal to amend/update the Regulatory Standard AR 3.10.2, "Protection against earthquakes in Nuclear Power Reactors", in order to establish the criteria for seismic design of nuclear power plants according to the state of the art in the field, the improved IAEA regulations and include lessons learned as a result of the occurrence of seismic events in other nuclear power plants (in particular in the japanese of Fukushima Daiichi NPP).

1.1.2.1. Atucha site

The ARN entrusted the Instituto de Investigaciones Antisísmica "Ing. Aldo Bruschi " (IDIA) of the Universidad Nacional de San Juan (UNSJ), to perform a review and update of the seismic hazard for the verification of the plant and its components.

The database for the study was taken primarily from the first assessment of seismic hazard conducted in 1981 by GNZ and two additional studies (seismotectonic and seismological respectively) performed as a preparatory phase for UNSJ "Neotectonics Study for assessment of seismic hazard "and " Compilation and analysis of seismological information ".

The site of Atucha is located in the stable continental region of South America, and close to the passive margin (seismicity observed in the sea). The region has low seismic activity, however, information is available on a number of historical earthquakes (journalistic record, etc.) and instrument (through seismograph record). Geotectonic information available is insufficient to define the seismogenic sources in the vicinity of the site studied and therefore is not possible to perform a deterministic analysis of seismic hazard.

A wide study area was defined (8 degrees latitude x 16 degrees longitude), covering the coastline of southern Brazil, Uruguay and the province of Buenos Aires in Argentina, carrying out a detailed analysis of all known seismic events, both historical and instrumental.

Considering that the catalog of events and magnitude information are crucial for developing a model of seismicity of the region, the magnitudes of the known events were reviewed, and the integrity of the catalog and its time limits was analyzed.

From the analysis of seismological information, it was possible to define a recurrence law applicable throughout the study area. The definition of the recurrence law was made under two alternative scenarios: excluding the seismicity observed in the sea, i.e. the passive margin and including the seismicity of the passive margin.

Obtained the recurrence law for the whole study area it was estimated that seismic activity is distributed according to two alternative Seismotectonic models. In the first model, the seismicity is assumed uniformly distributed throughout the area. The second model assumes that the seismic activity originates in vertical planes whose position and extension was plotted based on the traces of known basement faults.

In Argentina, there are insufficient accelerographic records to generate local attenuation relationships. Thus one must recur to attenuation relationships adjusted with data from other parts of the world. The following three attenuation laws, developed for stable continental regions (Central and Eastern USA), were selected to estimate parameters of the horizontal component of movement in rock and soil:

- Atkinson & Boore (1995)
- Hwang & Huo (1997)
- Atkinson & Boore (2006)

By a logical tree, uncertainties are included in the analysis, affecting: 1) the seismogenic source model, 2) recurrence law, 3) maximum magnitude, 4) attenuation law.

Mean hazard curves were calculated for peak ground acceleration (PGA) and pseudoacceleration (PSA) with 5% damping for different periods.

Uniform Hazard Spectra (UHS) were obtained, ie pseudoacceleration response spectra for 5% damping, for a constant return period of 10000 years, corresponding to the level of verification SL-2 of the IAEA Standards.

It was estimated the seismic hazard in terms of the magnitude and distance (deaggregation), observing that it is dominated by events of low magnitude (4.5 to 5.0), which are more frequent, with focus located within 20 to 25 km, which is the minimum estimated distance to the assumed fault existing in the riverbed of the Rio Paraná. It is therefore possible to conclude that such fault controls the seismic hazard at the studied site. This can be seen in the image below, showing the deaggregation of seismic hazard for peak ground acceleration (PGA).

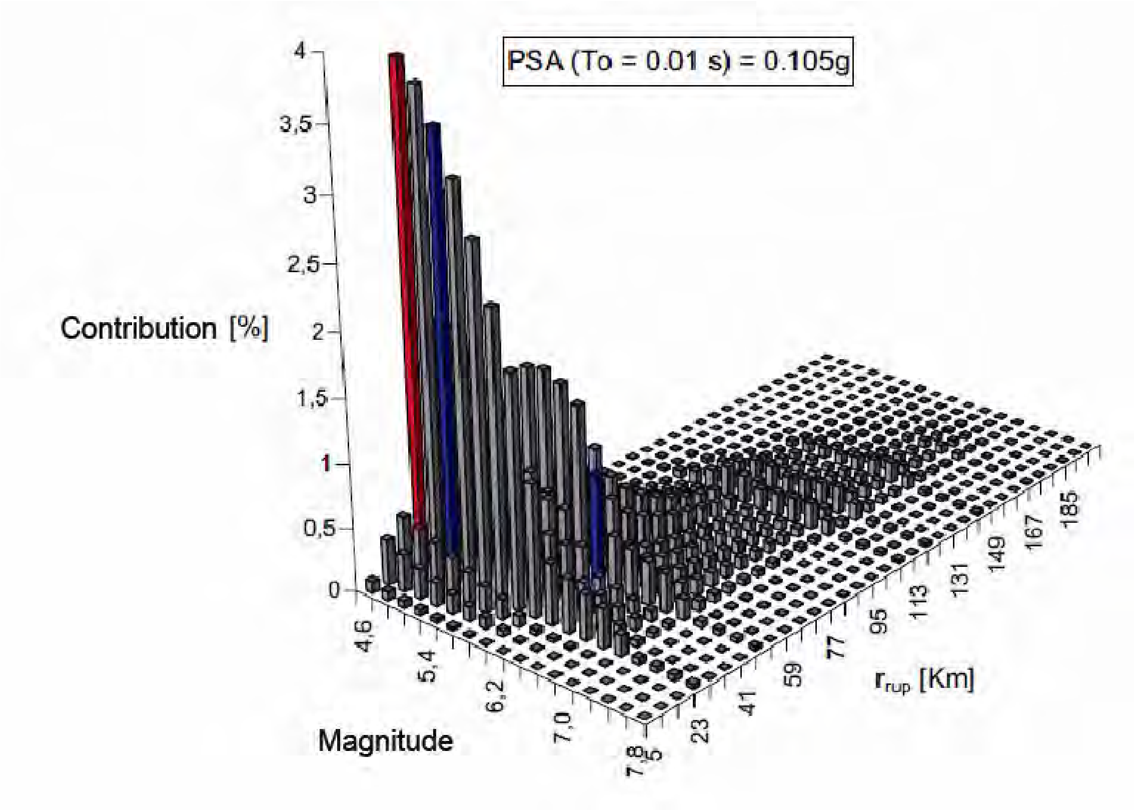


Figure 1-24: seismic hazard in terms of the magnitude and distance

The above-mentioned fault is a structure inferred from observations of the geomorphology (course change of the riverbed of the Rio Paraná) and the shape and changes in the depth of the bedrock top. To date, there are no certain data about its position, strike, dip (the angle between the plane measured with a horizontal plane), nor can one ensure their existence. However, there are records of a major seismic event (earthquake occurred on 06/04/1888) with a fairly precise location, which is located to coincide with the trace of the fault.

Comparison between the UNSJ study (ARN) and the AECL study (NA-SA)

The ARN requested the realization of a comparison between the independent study of the seismic hazard developed by the IDIA, and the corresponding one addressed by NA-SA and developed by the company D'Appolonia.

From that comparison, the following are the highlights:

- The methodology used in both studies consists of: 1) the collection and analysis geological and geophysical information published to date, 2) development of a seismotectonic model, 3) probabilistic seismic hazard assessment.
- In the study by D'Appolonia is a strong emphasis on the discussion regarding the existence of a fault along the lower course of the Rio Paraná. This discussion has great relevance for the assessment of seismic hazard in Atucha, due to the proximity to the plants site. Based on information gathered in the existing documentation D'Appolonia study concludes that, there is no active fault along the lower course of the Rio Paraná.

- On the other hand the seismotectonic model formulated by the IDIA does not discount the presence of a potentially active fault along the lower course of the Rio Paraná, taking in this regard a similar approach than the GNZ study.
- The scarce seismic activity observed does not provide enough information to define recurrence laws. Therefore, in both studies, seismotectonic model formulation is based on the global characterization of stable continental regions.
- Regarding the seismic activity, the assumptions used by both studies are similar. The major differences are found in the assumed geometry of the seismic sources. The IDIA study does not use the subdivision into extended crust and unextended crust regions. Instead, the seismicity is distributed uniformly along the basement fault lines, among which includes the supposed fault of the Río Paraná and others coincident with the basins of the Salado, Santa Lucia and lineaments in the provinces of Entre Rios, Santa Fe and south of Corrientes. In the formulation of this model is considered that although there is no evidence that these lineaments correspond to active faults, there are no conclusive evidence that they are not. In the same way as the study of D'Appolonia, the IDIA proposes a second alternative model in which the whole region is considered a uniform seismogenic source. The IDIA considers an additional possibility that is to include or not the seismicity of the passive margin to adjust the recurrence law. On the other hand the IDIA does not takes into account the seismic activity of the Sierras de Córdoba for being too far to produce significant effects on Atucha.

In conclusion one can say that D'Appolonia and IDIA carried out two studies of seismic hazard for the site of Atucha, completely independently. The base information used was very similar as well as the methodology applied. However, the assumptions made to formulate the seismotectonic model present some significant differences. Even so the results are similar and from the point of view of the structural effects that could have the specified ground motion for the verification of nuclear power plants (SL-2), one can conclude that they are equivalent.

The comparison of the spectrums can be seen in the *Figure 1-25* below.

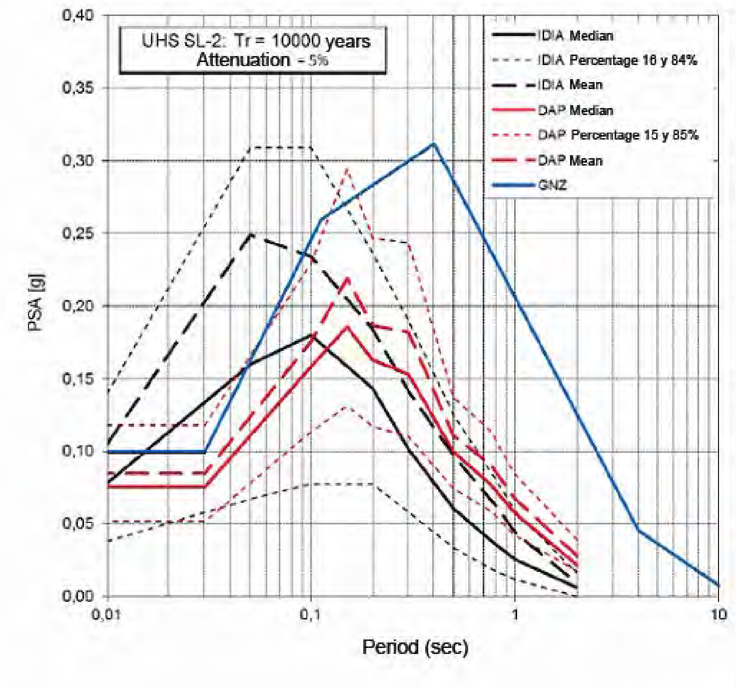


Figure 1-25: Comparison of the spectrums

1.1.2.2. Embalse Site

The plant was designed for a DBE with maximum horizontal floor acceleration of 0.15 g. Before commissioning there was a review of the design earthquake and it was established an earthquake with a horizontal acceleration of 0.26 g corresponding to a recurrence of 1/7000 years, re-qualifying the necessary components for a safe shutdown.

The ARN together with the IDIA has recently initiated an independent review of the seismic reevaluation of the site conducted by CNE. Additionally, it is being revised the floor response spectra for the reactor building and for auxiliary rooms and those provided for life extension of the plant.

1.1.2.3. Conclusions

The seismic design of nuclear power plants in Argentina is consistent with the criteria and requirements, both national and international, established at the time of design. However, since these requirements have been increasing over time, all nuclear power plants have faced a re-evaluation of seismic capacity (SMA) against the occurrence of earthquakes of a certain level, higher than the DBE.

To perform the above re-evaluation, plants have updated the seismic hazard of the sites using methodologies and databases according to the state of the art in the field.

The ARN agrees with the approach determined by NA-SA for re-evaluation of the seismic capacity of the plants, because:

- The SMA methodology is an appropriate focus and sufficient, considering the seismic hazard of the sites in question and the age of the plant.
- With the above methodology, it is possible to confirm the non existence of limit situations (cliff edge effects).
- The RLE adopted is derived from the mean response spectrum of uniform risk for a probability of exceedance of 10 -4 / year, which is consistent with the Argentinean Regulatory Standard AR 3.10.1 "Earthquake Protection in Nuclear Power Plants " and allows to verify the capacity of plants against earthquakes considered internationally as DBE.
- The evaluation of safety margins is consistent with the concept of defense in depth, since the safety functions cover the level 3 while the containment is verified at level 4 for a seismic demand corresponding to an earthquake with recurrence of 100,000 years.

Additionally, and based on assessments made to date, the ARN concludes that:

- No significant weaknesses have been identified that require to take urgent actions.
- It have been verified that NA-SA complies with both the design bases and licensing bases.
- It has been analyzed internal and external flooding caused by earthquakes and is considered that NA-SA is carrying out the appropriate actions to successfully meet these scenarios.
- For the purpose of increasing the capacity to respond against extreme conditions, NA-SA proposes to implement a set of improvements that, in the opinion of ARN, are acceptable.
- The ARN will continue to follow and evaluate the actions that are being / or will be implemented in the future to ensure that they are effective and that all necessary aspects related to the safety of plants are considered. The result of such activities will determine if it is necessary that the ARN requires complementary actions, modifications or improvements.

1.2. HIGH LEVEL WATER/LOW LEVEL WATER

1.2.1. ACTIVITIES DEVELOPED BY THE OPERATOR

1.2.1.1. High Level Water/Low Level Water for the Atucha Site

1.2.1.1.1. Introduction

The site of ATUCHA (CNA I - CNA II) is located on the right bank of the “Río Paraná de las Palmas” on the so-called “Vuelta del Pelado Inferior”, at 134,8 km in the waterway Paraguay-Paraná, near the locality of Lima, belong to Zárate area, a little over 100 km northwest from Buenos Aires City.

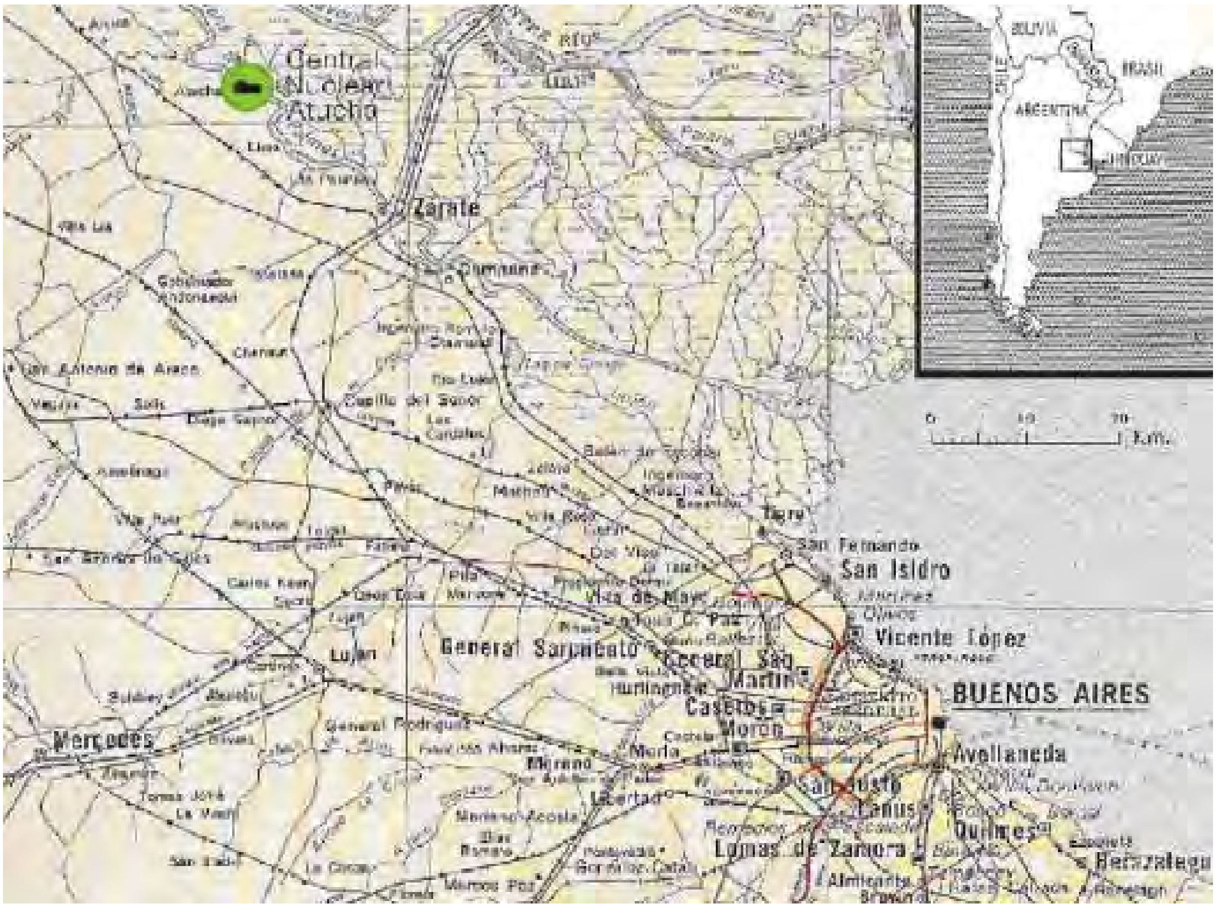


Figure N° 1-26: Geographic Location of Atucha

The Parana river is the sixth most important plain river in the world; moves an average flow rate of 16.000 m^3/s at the mouth, and is part of the basin of the “Río de la Plata”. Its level varies considerably and these variations were taken into account during the design at the Atucha Site.

Historical values of the maximum and minimum height levels of the river are 4,55 m and 0,17 m respectively, in relation with the zero of reference, corresponding to zero of the “Riachuelo” adopted as “normal zero” for all national levelling.

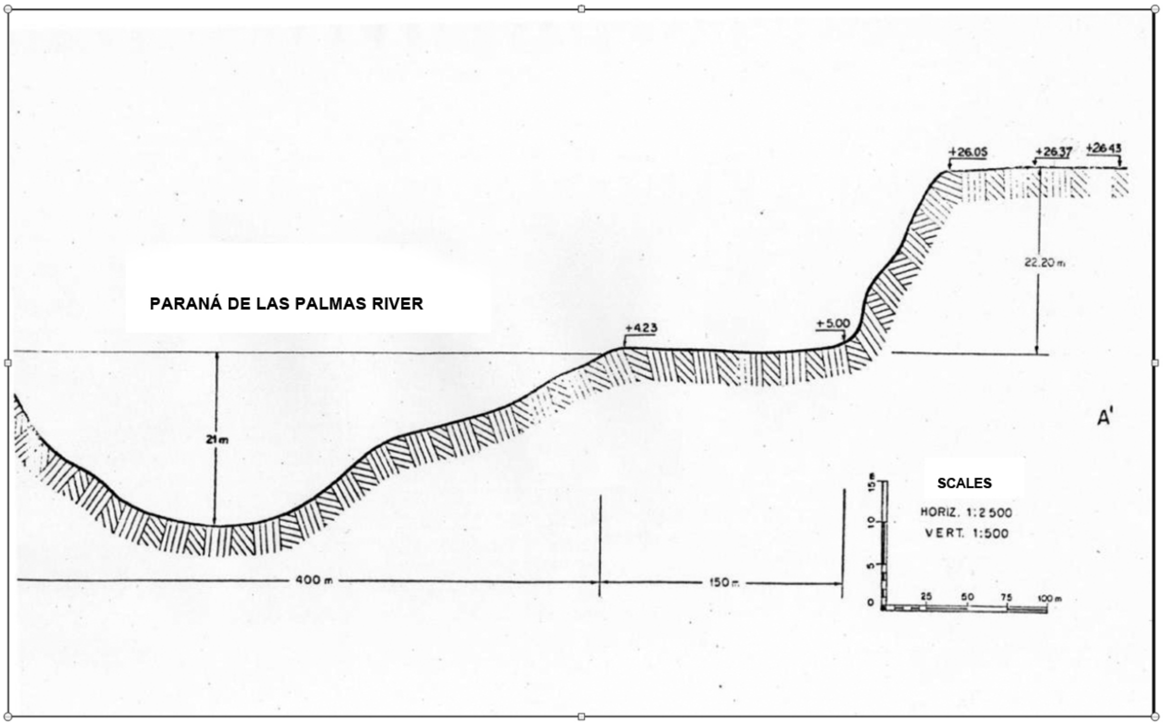


Figure N° 1-27: Profile of Paraná river in the site of Atucha

After the commissioning of the CNA I, hydrological studies were conducted about the rivers “Parana” and “Parana de las Palmas”. One of them is the "Hydrologic Study of the Atucha Nuclear Power Plant" by H. Gomez et al. which established the river levels data for the design of the second plant (CNA II) installed in the same area. In this study the historical records of water levels available at areas of interest inside the basin (“Corrientes, San Pedro, Zárate and Riachuelo”) were analysed. For the river low levels, historical data at “Zárate” and “Riachuelo” were used. For the river high levels, calculations for the CNA II were performed with a return period of 1000 years.

According to the analysis performed, the significant low levels can be originated by:

- A low flow rate at the down, middle or upper part of the Paraná river;
- Strong winds (higher than level 6 in the Beaufort scale) from the N, NNW, NW, NE and W at the Río de la Plata for long periods of time; and
- The previous two at the same time.

At the site of Atucha very low water levels of the “Río Paraná” were observed between the months of September and December. Also important water low levels, originated in the upper Paraná were registered (in the years 1925, 1944, 1949, 1951, 1968, 1969), due to a significant reduction of the rain season in Brazil.

In 1983, the maximum and minimum levels and recurrences for the site of Atucha were determined. The *Gumbel distribution* was used for high water levels and *log-normal* for low levels. The results were:

- High water level with recurrence of 1000 years: +5.2m
- High water level with recurrence of 100 years: +4.5m
- High water level with recurrence of 20 years: +4m
- High water level with recurrence of 5 years: +3.5m
- Average level: +2m
- Estimated record of low level: 0m
- Water low level with recurrence of 100 years: -0.5m
- Water low level with recurrence of 1000 years: -1m

The "maximum probable water high" (MPH) was determined in the same study by maximizing all the adverse factors associated with high water levels for the hydrodynamic model used, and considering that they occur simultaneously. The study performed for the Paraná River considered at the city of Corrientes, located at 600 km upstream of Atucha site, a water flow of 90,000 m^3/s. It is worth mentioning that the maximum historical water flow in this city was 42,000 m^3/s and from the probabilistic point of view to a return period of 1000 years it would reach the 55,000 m^3/s.

The MPH considered in the design of CNA II was 8.45 m. This value was re-valued taking into account high water levels caused by dam breaks, arriving to a similar conclusion (see below).

The existence of a series of dams upstream from the site allows the regulation to some extent, of the river level by filling the reservoirs during the rainy season and discharging them during the dry season. This adjustment is possible only during normal conditions.

The possibility of the river exceeding the level of 4 meters can be predicted 3 or 4 months in advance. This is because the high water level travels slowly, since this is a river of the plains and has a watershed that covers a large area. Also because the travel speeds of the high water level decreases on flats areas such as the ones surrounding the site. The time to travel between the cities of Corrientes and San Pedro (near the site of CNA I) of a high water level is approximately 30 days.

Through the hydrological information system of Del Plata Basin, the operator of the Atucha site obtains information on the levels of the Parana River. This system was developed in 1983 and is operated by the National Institute of Water (NIW). Since the implementation of this warning system, there have been three major high water levels in the years 1983, 1992 and 1998. The maximum level reached in CNA I was of 4.60 m in May 1998, which was slightly higher than the 4.55 m in 1983.

Based on the lessons learnt from the water level high of 1983, a retaining wall of 0.5 m was built at the pump house of CNA I. This modification permits to continue the safe operation of the plant with a water level higher than the design basis. As was mentioned before, the rise of the river level that could compromise the availability of the pump house, is of a slow evolution due to their flow and extension. For this reason, its behaviour is predictable well in advance through the records of the hydrological stations located upstream of the site.

Another possible reason for a rise in the river level on the Atucha site is a break in one of the existing dams located upstream of the site. There are currently 59 dams. The nearest is Yaciretá, located about 1,200 km. It is sufficient to consider only the rupture of this dam since it was verified that, if it maintains its integrity, it can contain the effect caused by the break of the dams previously located.

In Yaciretá dam´s design a study by Motor Columbus & Associates (1979) was used as a reference, and it determined that the MPH to be supported by the dam would come from a simultaneous confluence of the Paraguay and Paraná rivers, normally not at the same time with a resulting water flow of 95,000 m^3/s. During a bi-national experts meeting held in 1997 it was confirmed that the MPH was valid according to the existing hydrological data. The conclusions of these experts were reviewed and confirmed again 2009.

The maximum water height that would be reached on the site after the rupture of the dam would be similar to the one calculated for the simultaneous confluence of the two main tributaries due to probable maximum precipitation. For this event it was estimated that the maximum water level is 8.45 m.

Based on the above mentioned, high water levels are not expected to affect the main buildings of the NPPs in Atucha site since those are all placed at a level that provides a substantial margin from the MPH.

1.2.1.1.2. Flooding/low water level for which the plants are designed

1.2.1.1.2.1. CNA I

In the design of CNA I the mentioned early probable high and low-water-levels are considered, as a consequence of which the pump house building is the most vulnerable with respect to high water levels. Additionally, since the pumps of the assured river water intake system (UK) can only take in water up to a minimum river water height of approximately -1 m, this vulnerability was detected in regard to extreme low water levels.

The UK system fulfils the function of supplying cooling water to the following points:

- Oil cooling and hydro turbine alternator
- secondary cooling system of river water UD
- Emergency diesel generator BY

- cold water cooling system compressor
- nuclear cooling intermediate system TF
- back cooling system RR
- cooling system for pools and fuel elements transport channel
- Oil cooling pump
- Assured conventional intermediate cooling system UL

There are also the following additional supply points:

- Connection to the conventional water treatment system
- Filling connection for the river water cooling main circuit

As mentioned above through the National Water Institute (INA) is currently underway a new hydrologic and hydraulic study, which will include a review of background studies, in order to supplement and update, and also make a prospecting considering possible future scenarios. This new study reassess both high water level and low water level of the design basis and takes into account the combination of the maximum flow of the tributaries, broken dams located upstream as well as the boundary condition at the mouth of Parana river given by the levels of the Rio de la Plata. Notwithstanding the climate change occurred in the area, it is not expected that the levels change significantly, as experience gathered over the years of operation, so suggests it.

The main building of the CNA I was built on a plateau at 23 meters, being much higher than any possible river water level rise. However, the water intake where the pumps of the normal cooling system (UC) and the assured cooling system of the plant (UK), are at 6 meter level and may be vulnerable to river level rise - see Figure No. 1-2 (it is less than the "probable maximum high water level").

Therefore, to overcome the weaknesses in the design in terms of the extreme high and low water levels is being installed a fourth pump UK in the Pumps House of the CNA II. This pump will keep running even with a sharp river level rise of 8.45 m or low river levels of -2.00 m. This will allow CNA I to keep operating even when the water level exceeds the height of the pump house or when the water level is below the level of the intake.

1.2.1.1.2.2. CNA II

The main building of the CNA II was built on a plateau at 23 meters, being much higher than any possible flooding of the river.

CNA II has three pump houses. The pump house of the service water intake (UPD) contains two pumps of the assured water system PE, is situated by the river and has a water submarine intake.

Atucha II has three pump houses. The pump house of the service water intake (UPD) contains two pumps of the Assured Water System PE, is situated next to the river and has an underwater water intake.

The other two pump houses, circulation water intake of the condenser (UPC) and service intake (UQB) are located side by side 150 meters from the coast and receive water from the river through an intake channel of water. The UPC house contains three pumps from the conventional water system (PAB) and the UQB house another two pumps from the Assured Water System PE.

The houses UQB and UPD (PE system) are designed to withstand a CMP of +8.45 m and the UPC house containing the normal supply system is designed to withstand a flood of +5.20 m which is a flood with a return period of 1000 years.

The PE ensured cooling system performs the function of supplying cooling water to the following items:

- KAA System (nuclear component cooling)
- KAG System (shutdown cooling and in case of accidents)
- PJD System (Diesels)
- FAK System (spent fuel element pool)

Through the National Water Institute (INA) is currently underway a new hydrologic and hydraulic study the details of which are indicated in point 1.2.1.1.2.1.

1.2.1.1.3. Provisions to protect the plant from the design basis Flooding / low water level

1.2.1.1.3.1. CNA I

To identify critical structures, systems and components (SSCs) needed to reach and to maintain the plant in safe shutdown in case of Flooding or low water level of the Parana river, and considering that is lost the river as the heat sink, the probabilistic safety assessment (PSA) model was used.

River levels are taken daily, and this value is included in the "on-line" parameters control system of the plant. In case of abnormality is detected, it begins to follow the development of what happens upstream, maintaining the system in alert according with the detailed shown below. These measures provide in the knowledge advance of the flooding/low water level and when the level where the plant must be removed from service for safety reasons will be reach. According to records at the Parana River in CNA I, are setting the following levels of intervention:

- Alert: river level equal to or greater than 3.80 m. The CIAS (Internal Security Advisory Committee) is convened to plan preventive actions.
- Emergency: river level equal to or greater than 5.00 m. At this level, the plant operations follows the corresponding procedure (T 17).

The T17 procedure has instructions of successive actions to deal with cases where the river level rises above the design level. The aim is to ensure the removal of heat from the core and preserve the equipment that could be affected by the Flooding. The actions to be implemented depend on the level of the water and are summarized as follows:

Level> 5m

- Record the water level every 30 minutes.
- Check that there are no obstructions in the intake (sieves and rotation screen).
- Testing for leaks in the Pump House.
- Maintain inventory at maximum in the water reservoirs.

Level> 5.5m

- Take the plant to cold shutdown.
- Take out of service and de-energize the normal cooling equipment.
- Complete any fuel change operation in process, so that the fuel elements replacement machine is empty.

Level> 6.0m

- Remove from service and de-energize one of the UK pumps.
- De-energize all equipment that should not be used in the Pump House.
- In case of the flow water entering into the Pump House is higher than 100 m^3/h (the capacity of the bilge pump of the building is 100 m^3/h) or the level is greater than 15 cm, the second UK pump will be stopped and turned off.

If the water level starts to rise above the level at which the UK motors are installed, is envisaged to use the second heat sink (SHS) to ensure the continuation of the residual heat removal from the core.

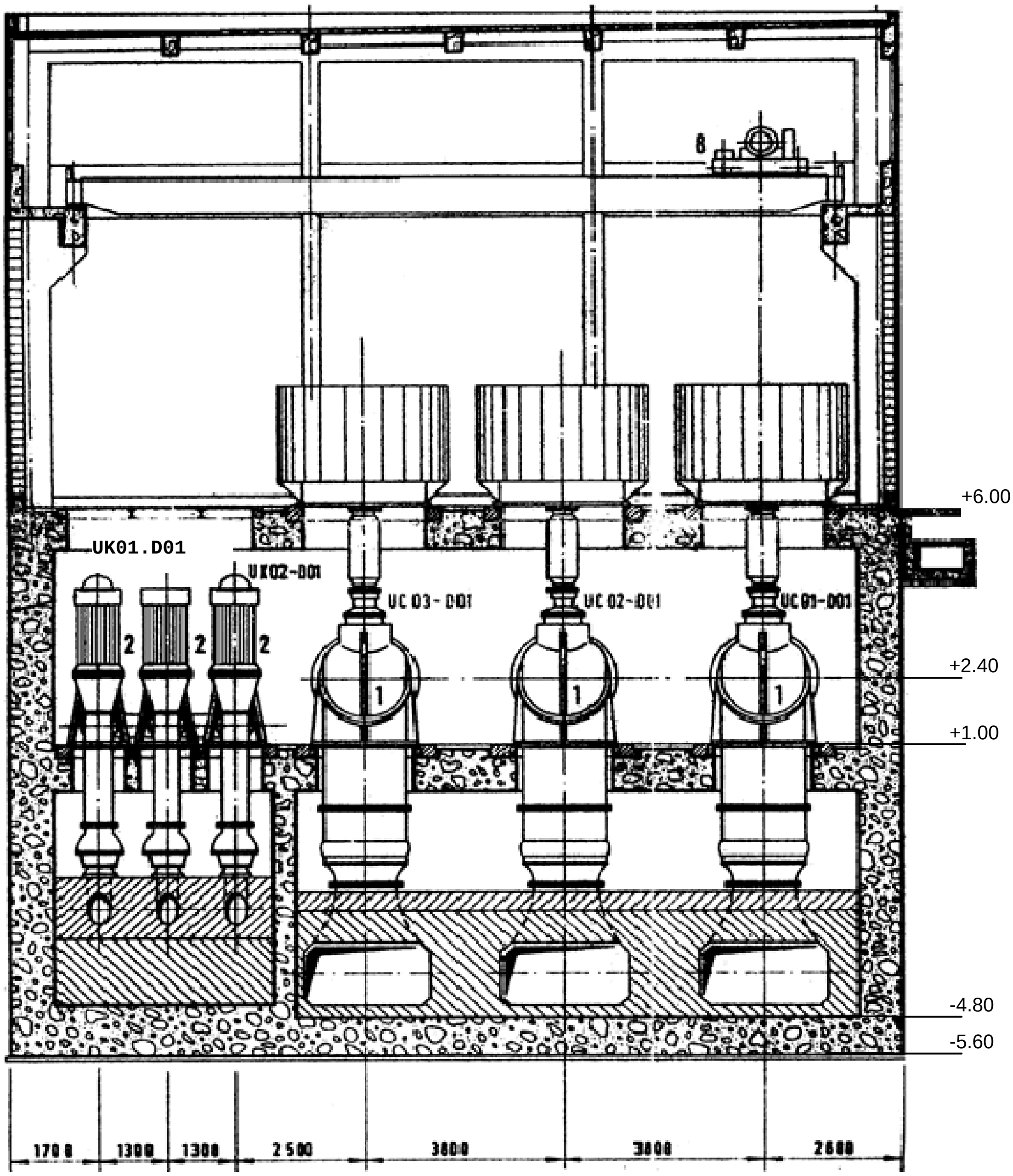


Figure N° 1-28: Pump House Diagram CNA I

On the other hand, for a same flow of water inlet to the plant, decreases in the river level generates increases of velocity in the intake channel. This effect is produced by reducing the passage area, which is a function not only the water level but also the level of deposits in the channel bottom. This determines a boundary condition of operation that depends on the water level and the height of sediment deposited at the intake channel (from which erosion occurs in this channel) and the number of pumps of primary cooling water circuit (UC) in operation.

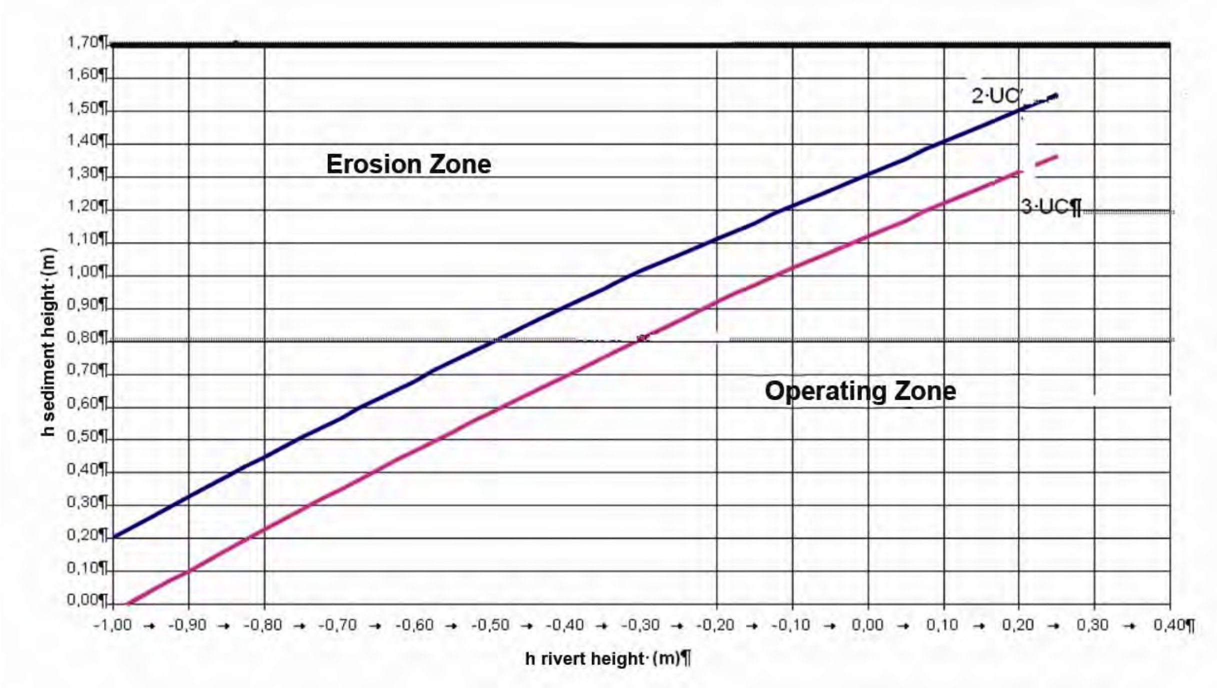


Figure 1-29: Sediment height vs river height

As shown in Figure 1-29, in the case that the operating point is located above the curve and can be verified upstream to Atucha site that the river level continues to drop, the plant must be shutdown. However, the UK pump flow represents a small percentage of increase of UC pump flow and thus the system will still be available to take to safety shutdown to the plant. In the case of persisting the decrease will be necessary to continue the cooling independently of the river through the second heat sink system (SHC). For details of the SHS see section 2.2.1.

It is planned to complete the implementation of a UK fourth bomb which will be located in the pump house of CNA II that will add diversity and redundancy, as well as will improve the system design and reduce its weaknesses considering the extreme floods and low-water-levels. This pump covers events beyond the design basis and fulfils the function of ensuring the subsequent cooling and can be used in parallel with another UK pump during normal operation even with flood or low-water-level of the river of 8.45 m or -2.00 m respectively. The CNA I has three pumps that belong to the UK system located in the pump house of CNA I. Each of these pumps supply 50% of the required flow of water under normal operating conditions and 100% under emergency conditions. Under normal conditions, two UK pumps are in operation leaving a third UK pump in reserve. The new UK pump has an auxiliary water system sealing and the associated filtration system and the electrical supply will be of medium voltage (6.6 KV) secured and will be commanded from CNA I. The implementation of this improvement is in the process of assembling pipes and it is expected to be implemented in 2013.

In addition, new modification have been undertaken to improve the SHC system response in the long term and keep the UK system in operation beyond the current design basis. These modifications include:

As it is was mentioned in 1.2.1.1.2.1 it is planned to complete the implementation of a fourth bomb UK which will be located in the pump house of CNA II that will add diversity and redundancy to improve system design and reduce their weaknesses considering the extreme Flooding and low water levels. This pump covers events beyond the design basis and fulfils the function of ensuring the posterior cooling and can be used in parallel with other UK pump of normal operation even with Flooding or low water level of the river of 8.45 m o -2.00 m respectively. The CNA I has three pumps belong to UK system located in the pump house of CNA I. Each of this pumps supply 50% of the required flow of water in normal operating conditions and 100% under emergency conditions. Under normal conditions, two UK pumps are operated leaving a third UK pump in reserve. The new UK pump has an auxiliary water system sealing and the associated filtration system and the electrical supply will be of medium voltage (6.6 KV) secured and will be commanded from the CNA I. The implementation of this improvement is in the process of pipes assemble and is expected to be implemented in 2013.

In addition, new modification have been undertaken to improve the SHS system response in the long term and keep the UK system in operation beyond the design basis. These modifications include:

1. Strategy of inventory reposition to SG

   The purpose of this modification is restoring the water inventory in the SG in the case of loss of the decay heat extraction chain and the injection of water from the SHS tank. Also, in cases where the integrity of the SHS was not affected, it is possible to restoring the inventory to the system tank. To meet this objective, the strategy for use the water from the auxiliary reservoir of the spent fuel elements pools (UA00B03/B04) and inject this water to SG depressurized, using the pumps UA10D20 and D21 and restoring water to these pools with well water using water from one of the UJ pumps (normal supply water system). See schema in Figure 1-30. This proposal also covers the possibility to feed the components involve through a mobile diesel generator MDG in case of SBO with the DG of the SHS also unavailable.

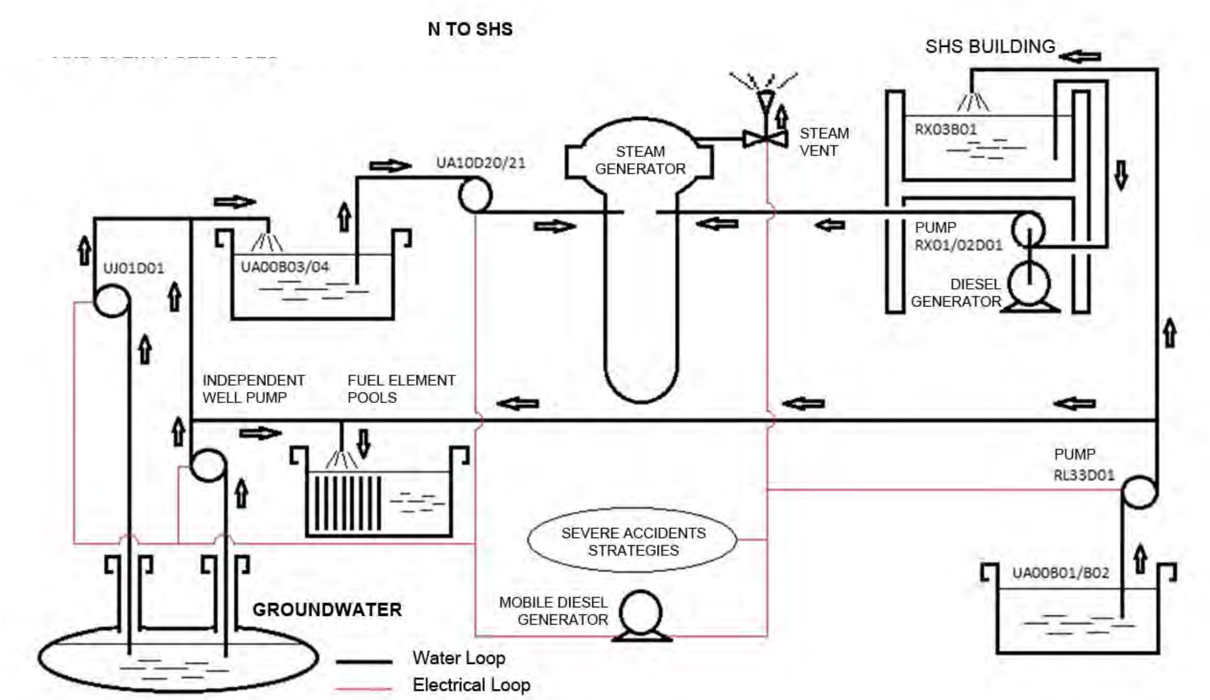


Figure N°1-30: Diagram of inventory reposition to the SHS

2. Implementation of MDG mentioned above, as detailed in the plans related to earthquakes (item 1.1). This MDG, will feed the additional pumps (UA10D20/21) to supply water to the SG through the SHS.

3. Fourth pump of the assured cooling circuit (UK) installed in the pump house of CNA II, as mentioned previously.

The before mentioned modifications will be implemented in 2013.

1.2.1.1.3.2. CNA II

For the purpose of identifying critical structures, systems and components (SSCs) needed to reach and to maintain the plant to a safe shutdown in case of Flooding or low-level water of the Parana river, and considering that the river is lost as a heat sink, the PSA model was used, assuming the loss of the assured cooling water service PE, and a SSC list to bring the plant to a safe shutdown condition.

In the case of total loss of river water supply, the loss of the ability to pump water to cool the installation from the Parana River, results in the loss of the following alternatives of heat removal:

- Steam Generator - Condenser
- Moderator heat exchanger - intermediate system of insured residual heat removal system cooling

As described in Section 2.2.3.3., if the CNA II has external power supply, the cooling of the installation is maintained through the SG venting to the atmosphere. In this condition the facility will continue to be cooled by the SG, but the secondary circuit will transfer the heat to the atmosphere operating as an open circuit venting live steam through four motorized valves with power supply provided by the secured bars of the interruptible system (ultimately powered by batteries). It should be noted that a single valve is able to remove the decay heat.

The available reserve of water to cool the core through the SGs is constituted by the initial inventory of these (120 m^3), plus the content of the feed water tank (maximum of 264 m^3 / minimum of 148 m^3) plus the demineralized water reserve from the system of demineralized water supply (GHC), 2 tanks of 785 m^3 each, with a secured reserve of 2 x 280 m^3.

As noted above, autonomy of at least two days is available in this event, in the case of having an alternative external power supply. This is so because of the availability of water in the secondary system and the tanks of the feed-water system LA. (Figure 1-31)

In this case, it is intended to incorporate improvements for extending the water supply to the SG for a longer time consisting in the mentioned MDG and the water supplied from an alternative reservoir. The possibility is being evaluating that this reservoir will be constituted by water intakes from the groundwater. The implementation of this improvement has been planned in 2014. In the case that the loss of the assured cooling system of the service water (PE) should occur due to flooding or low-water-level, it is important to mention that the main cooling water system PAB is designed to operate between 0.0 m and 5.65 m. Outside these limits, it is expected to take the plant to shutdown by cooling with the PE system, which is designed to operate between -1.0 m and 8.45 m. Taking into account the times involved in the low-water-level and flooding, which are in the order of weeks, when the PE system is not available, the plant will be shutdown and with small decay power, by what is estimated to have an autonomy well above the two days mentioned.

For the case of loss of the assured cooling system of the service water PE and as a result of the analysis performed in the PSA, the procedures for cooling the plant via the SGs if SBO are being developed. These procedures are expected to be available in 2013.

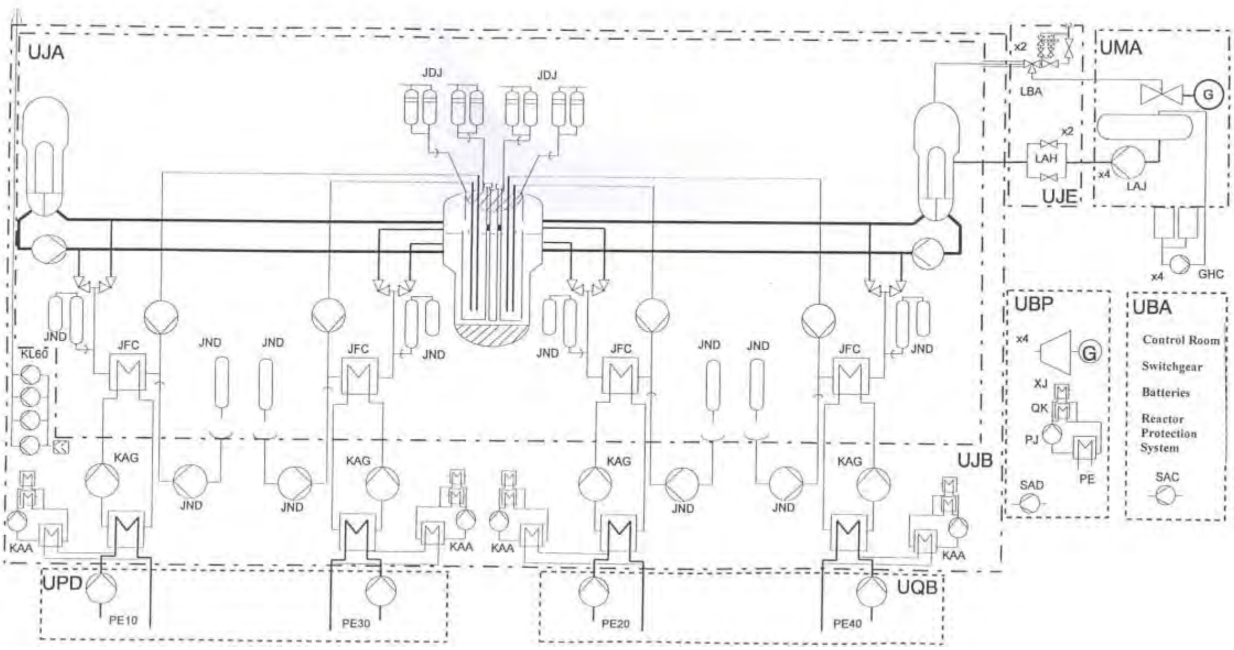


Figure 1-31: Cooling systems and residual heat removal

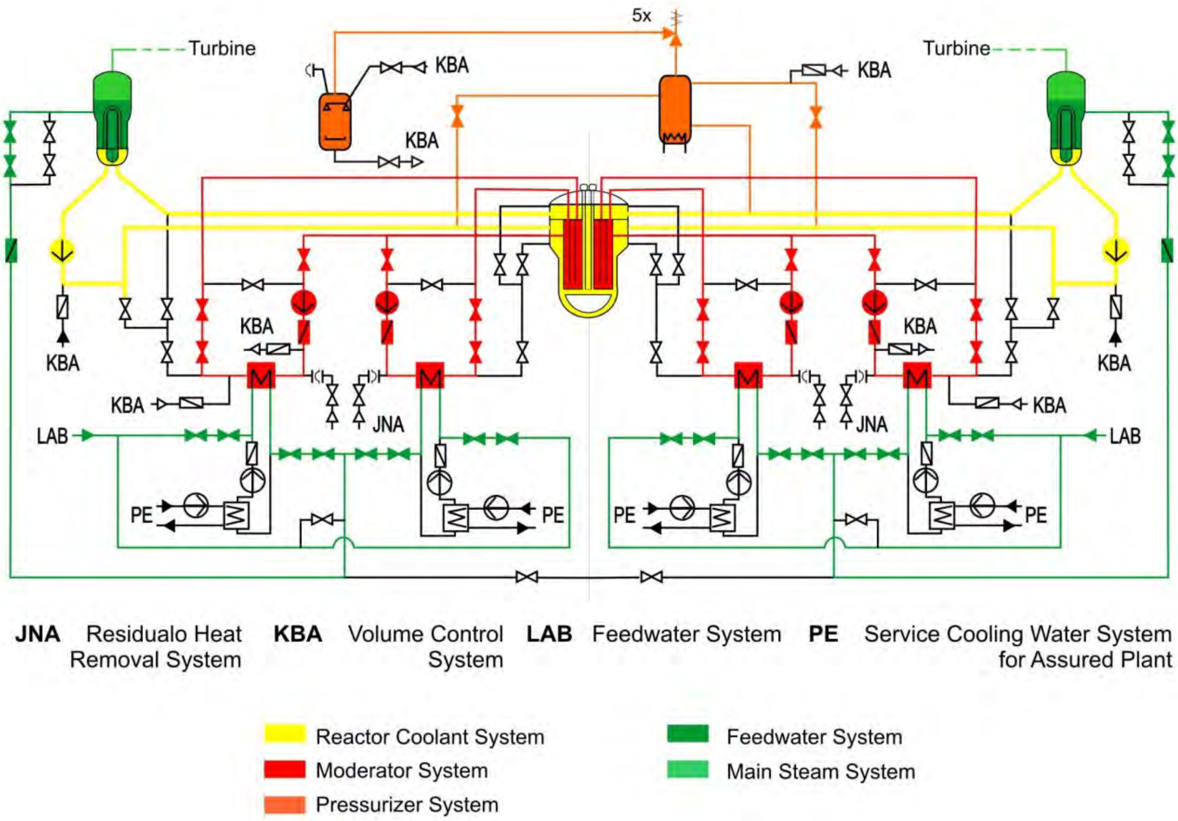


Figure 1-32: Cooling systems and residual heat removal

1.2.1.1.4. The process of ensuring compliance with the licensing basis

1.2.1.1.4.1. CNA I

The systems and components necessaries for take and keep the reactor in a safe shutdown for the cases previously considered of loss of the assured river water system (UK), have been identified in the PSA of the plant and verified its availability through repetitive testing programs and maintenance.

Loss of external power supply

In the event that an external phenomenon affects the power supply, the plant has the interconnection with emergency DGs of CNA II. These devices are capable of providing energy to the plant to achieve and maintain the plant in a safe shutdown. Also, the SHS electrical power is secured by its independent DGs, which means that it does not depend on the external electric supply or DGs of CNA II and can fulfil its function of cooling the plant through the SG.

1.2.1.1.4.2. CNA II

The CNA II has four pumps at the assured water system PE, which can operate with high water levels up to +8.50 m. In addition, a high water level of such magnitude can be predicted three to four months in advance and would take about 30 days to reach the site of the plant. Consequently, it is planned to lead the plant to a safe shutdown condition far in advance so that the decay heat is so low that the PE pumps are not needed to remove residual heat from the core.

With the plant in a safe shutdown condition for over a month, it is planned to use the inventory of water from the pool of the feed water system (LA) and from the tanks of demineralized water supply system GHC to keep the plant properly cooled for a long period.

1.2.1.1.5. Activities and specific verification test undertaken by the plant after the Fukushima accident

1.2.1.1.5.1. CNA I

After the Fukushima accident, WANO (WANO SOER 2011-2) requested the NPPs Licensees a review on the status of the SSCs and procedures for dealing with emergencies. The mentioned request requires an evaluation of the plant and its response capability against high water levels and low water levels.

In this regard, NA-SA assessed the availability of SSCs that should be working during the external flooding scenarios and developed a list of 175 necessary components to deal with such an event based on the already mentioned instruction T17. The SSCs identified were reviewed in a walkdown of the plant.

Through the National Water Institute (INA) a new hydrologic and hydraulic study is currently underway, the details of which are indicated in section 1.2.1.1.2.1.

Also other possible sources of external flooding different from Parana River (heavy and long term rains, the breaking of tanks of other nearby plants, etc.) were analysed and it was concluded that there is no other source that can cause flooding in CNA I.

Spent fuel pools

Based on was mentioned above it is estimated as unlikely, from a structural point of view, that the spent fuel storage pools are affected by high or low water levels of the Parana river. From the functional point of view, considering the loss of the UK system, the pools will lose the cooling water. To deal with this situation, a strategy for refilling the pools through a UJ pump specially installed to draw water from the groundwater (Figure 1-30) was established.

The strategy for filling the pools relies on the use of groundwater as an alternative and independent source of water which will be conducted to each of the pool buildings through a dedicated pipeline system. It is planned that the system´s design be such that it can remain available in the considered extreme conditions, and to permit the operation and monitoring of relevant parameters from outside the building.

It is planned to locate a water well for extraction of water from the groundwater adjacent to the pool building. This water well is independent of the one used in the strategy for restoring the inventory to the SHS proposed for the same scenario. The water drawn by pumps will supply each of the buildings independently. This part of the system (external part) will be static and will end on the inner wall of the building, with two couplings to allow the connection of pipe sections inside the building.

Inside the building a removable system with quick-mount by coupling is planned, because a fixed installation of pipes would obstruct circulation areas. This part of the system will be mounted when needed, coupling the sections to transport the water from the couplings located on the inner wall (mentioned above) to any of the pools of each of the two buildings.

The water refilling process will be manual, so that an operator monitoring the level and temperature of the pools will operate the intercepting valves opening or closing them, as appropriate. Both the control of the pump and the reading of the parameters are planned to be performed from an electrical panel located at the outer part of the building and both close to the water well and to the intercepting valves. This improvement will be available in 2013.

1.2.1.1.5.2. CNA II

Through the intermediary of INA a new hydrologic and hydraulic study is being undertaken, details of which are indicated in section 1.2.1.1.2.1.

The measures to be implemented for the external cooling of the RPV are being evaluated with the aim to mitigate the consequences of accident sequences with core damage. It is planned to implement these measures in 2014.

1.2.1.1.6. Margin evaluations

1.2.1.1.6.1. CNA I

It has been determined that the most significant weakness of CNA I is the pump house of the UK system. As mentioned above, in case of Flooding or low-level-water that could cause the loss of the heat sink consisting of the subsequent cooling chain due to the loss of the UK system, there will be an alternative heat sink independent of the river water formed by the SHS. The planned improvements allow the system to operate for the long term with the support of the injection of groundwater. Additionally, the support of the above mentioned MDG will be available.

Currently a procedure for the outage of the plant in case of extreme low-water-levels is being developed, based on studies of boundary conditions of operation for decreases in the river levels values. This will allow systematizing the procedure for the outage and its implementation is scheduled for the second half of 2012.

1.2.1.1.6.2. CNA II

According to what is described in section 1.2.1.1.3, CNA II has conservative design values and the time involved in flooding or low-water-levels are very long and allow the planned actions to deal with these situations to be implemented.

In addition, the plant will be out of service in case the design limits of 0.0 m and 5.65 m set for the main cooling water system (PAB) are exceeded, that is, if the flooding or low-water-level continues exceeding the design values of the assured cooling system of the service water PE -1.0 m. and 8.45 m (which would cause the loss of the PE system). In these circumstances, given that the plant would be already out of service, the residual power would be small and the supply of water to remove the heat would last much longer than those described in Section 2.2.3.3.

Currently the procedures for the removal of residual heat for these cases via SG are being developed and the procedures are expected to be available by 2013.

1.2.1.2. FLOODING / LOW-WATER-LEVEL FOR THE EMBALSE (CNE) SITE

1.2.1.2.1. Flooding / low-water-level for which the plant was designed

1.2.1.2.1.1. Flooding / Low-Water-Level External

Upstream and at a distant of about 20 km from the site of CNE, there are two relatively small dams named Cerro Pelado and Arroyo Corto. In addition, there is a hydroelectric plant located in the dam reservoir downstream from CNE, from whose spillway discharges the water when it reaches its maximum level, thus avoiding significant increases of the water level that could affect the site where the plant is located.

Consequently, the external flooding of the plant due to an increased water level of Río Tercero above the level of the spillway is not possible because it is at 657.5 m and the ground level (ground floor of the plant) is at 665 m. In addition, based on existing historical background data, it is known that the reservoir level has not exceeded about 2.0 m over the level of the spillway and the difference of 7.5 m between the levels of the spillway and the level of plant floor provides adequate protection against flooding. Based on the above, additional measures to protect the plant against a Flooding are not considered necessary.

However, as part of the activities planned for the refurbishment of the plant, a re-evaluation is planned of the consequences of the occurrence of earthquakes on the existing dam downstream of CNE. It is considered that this analysis will determine if it necessary to implement additional improvements related to a very low lake water-level. This re-evaluation is planned to be finished by 2015.

1.2.1.2.1.2. Internal Flooding

1.2.1.2.1.2.1. Turbine Building

The turbine building could become flooded as a result of leaks in the circulating water pipe of the condenser or service water pipe.

If a leak occurs in the service water pipe, it is estimated that the water will accumulate in the basement of the turbine building at a rate that would allow the operator to isolate the leak and/or stop the service water system before vital equipment for the reactor shutdown are affected.

In the event that the accumulation of water from a break in a service water pipe were to reach the feed water pumps and the leakage caused by them were isolated by the triggering of the pumps, it would still be possible to cool the heat transport system (SPTC) either by controlling the pressure of SGs or by the cooling shutdown system. In such a case it would be necessary for the operator to open the main safety steam valves to depressurize the SGs and use water from the spray tank (dousing) as a replacement therefor. The SPTC pumps should be kept in operation until the bearing temperatures indicate that damage can occur to them, at which time they must be stopped. System leakage of circulating water from the condenser may flood the turbine building and the auxiliary room until the level of the ground floor before the leak can be isolated. If water reaches the ground level, it will flow out of the building and will not accumulate.

The following equipment is located below ground level in the turbine building and is likely to fail in case of flooding:

- Compressed air for instruments,
- Coolers,
- Supply pumps,
- Condensate extraction pumps of the oil the system for the generator seals.

Under such conditions, the operator would need to start the water circulating pumps of the condenser and shutdown the reactor. To ensure the removal of heat from the core, the operator would initiate the cooling of SPTC until the level in the SG reach - 10 meters, and then the shutdown cooling system to the temperature of SPTC at that moment. The flooding in the turbine building would not affect the pump room or the services building or the reactor building, therefore, the systems necessary to maintain the cooling during the shutdown remain available.

Improvements for these events:

- An air compressor driven by a DG in the level 100 meters of the turbine building has being installed and is connected by valves to the air supply tanks of the instruments.
- The gateway to the secondary control room located in the level 100 m of the turbine building (E/T) shall be reinforced, if for any eventuality the water that floods the E/T exceeded that level. This improvement is expected to be implemented during the refurbishment shutdown of the plant (ending 2015).

1.2.1.2.1.2.2. Service Building

The flooding of the service building could occur from leaks in the main high or low pressure water service.

The equipment listed below is located in the basement of the service building, which could become flooded in the event of a break in the service water piping. The failure of this equipment could have significant effects on reactor operation:

- Pumps of the spent fuel pool,
- Shielding cooling pumps,
- High pressure service water pumps,
- Emergency core cooling pumps.

1.2.1.2.2. Provisions for the Flooding/low-water-level design basis to protect the plant

1.2.1.2.2.1. Turbine Building

In case of flooding, it is expected that the operator shutdown the circulation water pumps of the condenser and shutdown the reactor. To ensure the removal of heat from the core, the operator starts the cooling of SPTC until the level in the VG reaches 10 meters, and then the heat extraction continues through the shutdown cooling system.

The flooding in the turbine building would not affect the pump room or the service building or the reactor building, therefore the systems needed to keep operating the shutdown cooling system remain available.

The improvements and implementation timelines foreseen for these events are the same as those indicated in Section 1.3.1.2.1

1.2.1.2.2.2. Service Building

The operator could isolate the leak and stop the service water system before the water level reaches the ground level.

The equipment listed below is located in the basement of the service building, which could become flooded in case of a large rupture of the service water pipe. The failure of this equipment could have significant effects on the reactor operation:

- Pumps of the spent fuel pool cooling system,
- Shielding cooling pumps,
- High pressure service water pumps,
- Emergency core cooling pumps.

To isolate the leak it may be necessary for the operator to stop all the low pressure service water pumps. This action will cause a complete lack of service water to the plant and therefore, the reactor would be stopped and the normal cooling of the SPTC will initiate at the highest flow regime using the pressure control SGs. The SPTC pumps should be kept in operation until the temperatures of the bearings indicate that they may be damaged which could cause their outage.

Nonetheless, when the above mentioned pumps go out of service, it is estimated that the SPTC are already sufficiently cooled and the extreme shielding cooling system failure that could occur as a result of flooding would not have major implications because the heat to be extracted would be extremely low. The flooding of the system of the emergency core cooling would not be a problem, because the cooling of the SPTC would be completed before its integrity is compromised.

1.2.1.2.3. Compliance with the licensing basis

In order to meet the recommendations made by the WANO SOER 2011-2 "Verify the capability to mitigate internal and external flooding events required by station design", in middle 2011 a plant walkdown was performed where it was established that the risks identified in the design and included in the Final Safety Analysis Report (FSAR) have not increased.

There was a subsequent modification to the design, regarding the air compressor of the turbine building that was raised due to the internal flooding of this building, also a new pipe was added and a DG was provided.

The physical location of the centre of the plant lighting system (Light Centre) will be changed, since in its current location, if there is a water loss of the condenser, flooding would occur and leave the lighting system of the plant out of service.

1.2.1.2.4. Evaluations of margins

The worst situation identified in terms of lake levels correspond to a deformation of the dam as a result of an earthquake that could cause a loss of the heat sinks, simultaneously affecting the circulation systems, process water and the emergency water supply (EWS). One should focus on the emergency water supply which is qualified for earthquakes. The new EWS fed by the new higher power EPS is capable of supplying the VGs and the heat exchanger of the emergency core cooling system (ECCS).

Based on the above, it is necessary to update the seismic evaluation of the dam, either to ensure that the EWS will be available with a sufficient margin defined in terms of the High Confidence of Low Probability of Failure (HCLPF) for the most severe seismic conditions (measurement of the robustness of the plant against severe accidents) or otherwise to ensure a sinkhole via the EWS.

The seismic reassessment mentioned above will be conducted as part of the activities planned for the refurbishment of the plant to extend its life. Whether further improvements are required is expected to be determined based on the results of this re-evaluation. This re-assessment is expected to conclude in 2014.

1.2.2. ACTIVITIES UNDERTAKEN BY THE REGULATOR

In response to the events at Fukushima, the Nuclear Regulatory Authority (ARN) required (RQ-NASA-038) to NA-SA (Licensee of the NPPs CNA I, CNA II and CNE) to perform a comprehensive safety assessment (stress test) of these plants in order to detect any weaknesses and implement the corresponding improvements (see Appendix I).

The above regulatory requirements in connection with extreme external events included the evaluation of the design basis which in turn contemplates the Flooding/low-water-level for which each plant is designed for, the provisions made to protect the plants from them and compliance with the licensing basis. Furthermore, it was required the activities and specific verification tests being undertaken for each NPP after the Fukushima accident and the evaluation of existing margins for each NPP and the analysis of other extreme external events (storms, tornadoes, etc.).

The ARN has analysed the information received yet from the plants in order to judge the acceptability of the security level of each. The principal findings of this analysis are shown below.

1.2.2.1. Atucha Site

In the case of Atucha (CNA I and CNA II), the site was evaluated as a common emplacement to both plants. In this context a new hydrologic and hydraulic study is being made that includes a review of the background studies in order to supplement and update them, and make a survey considering possible future scenarios. This new study re-evaluates both the Flooding and the low-water-level design basis and considers the combination of the maximum flow of the tributaries, broken dams located upstream and the boundary condition at the mouth of the Parana river given by the river levels of the Rio de la Plata.

Many studies have been carried out after the commissioning of CNA I as a verification of its design basis and as the design basis for CNA II. Vulnerabilities were identified in the case of the pump building of CNA I which has been overcome with the implementation of modifications and new equipment (fourth bomb UK).

1.2.2.2. Embalse Site

It has been shown that an external flooding of CNE due to an overflow of Rio Tercero Reservoir is not possible due to the difference between the level of the spillway of the dam and the ground level of the NPP and historical Flooding data.

As part of the activities planned for the reconditioning of CNE, is planning a re-evaluation of the consequences of the occurrence of earthquakes on the existing dam downstream of the plant. This test will determine the need to implement additional improvements related to a very low-water-level of the lake.

1.2.2.3. Conclusions

Based on the assessments made to date, the ARN concludes the following:

- It has identified the need for regulatory actions that are not relevant weaknesses that require taking urgent action.
- It has verified that NA-SA complies with both the design and licensing basis.
- For the purpose of increasing the capacity to respond to extreme situations NA-SA proposes to implement a set of improvements, which ARN has determined to be acceptable, as well as the proposed timeline for implementing the identified improvements.
- The consideration of Flooding/low-water-level for Argentine NPPs is consistent with both domestic and international criteria and requirements established at the time of design. However, further studies were considered necessary for the Embalse and Atucha sites. For Atucha a new hydrologic and hydraulic study is being made that will include a review of background studies in order to supplement and update them, and carry out a survey considering possible future scenarios. For CNE a re-evaluation of the consequences of the occurrence of earthquakes on the existing dam downstream of the plant is expected to be made.
- Internal and external flooding situations have been analysed and it is considered that NA-SA is carrying out the appropriate action to successfully meet these scenarios.
- The ARN continue to monitor and evaluate the actions that are being and / or will be implemented in the future to ensure that they are effective and that all necessary aspects related to plant safety are considered. The results of these activities are necessary to determine whether the ARN must require complementary actions, changes or additional improvements.

1.3. OTHER EXTERNAL EVENTS

1.3.1. ACTIVITIES PERFORMED BY THE OPERATOR

1.3.1.1. Atucha I Nuclear Power Plant (CNA I)

1.3.1.1.1. Events and combination of events considered. Reasons for their choice

Other natural catastrophes are analysed: wind loads, specifically tornadoes, and lightning discharges.

These two extreme natural events were selected because:

- recently, strong winds caused damage to the a building tin roof, and
- a lightning discharge that occurred in 1977 damaged the main power transformer, determining the outage of the plant.

1.3.1.1.2. Tornadoes

1.3.1.1.2.1. Design of CNA I

The CNA I was not originally designed and qualified against the action of tornadoes. At the same site, CNA II designed in the '80s, has taken tornadoes into account as a design event. Considering that conservative criteria were applied for the design of CNA I, and taking into account the robustness of the typical structures of a nuclear power and the similarity with CNA II, we can infer that the main structures of CNA I will be able to cope with a tornado of the same type.

To calculate the basic design tornado, the methodology of McDonald has been applied to the data obtained to date for tornadoes in Argentina.

With the assumption that the ranking of the damage within the area affected by a tornado is a characteristic of the phenomenon and not of the region where it is measured, it was necessary to use databases of measurements performed in the United States to complete the information required for the analysis.

The safety-related structures have been designed considering a class F3 Tornado of the Fujita scale used to categorize the estimated strength of tornadoes by the damage they cause.

The scale has six levels ranging from 0 to 5, representing the increasing level of damage. The damage associated with an intensity corresponding to a F3 is listed:

Intensity	Wind Speed [km/h]	Damage
F_3	254-332	Severe: – Rips off ceilings and walls of prefabricated housing. – Overturns trains. – Lifts cars from the ground and displaces them. – Damages solidly built buildings.

Table N° 1-1: F3 of the Fujita scale.

According to the details of *Table 1-1*, the design parameters for a class F3 Tornado are represented in *Table 1-2*.

Maximum wind speed	245 – 332km/h	According to the improved Fujita scale
Width of affected area	170 – 450m	
Length of affected area	16 – 50km	
Maximum speed of rotation	270km/h	
Radius of maximum rotation speed	45m	Derived from USA-EC Regulatory Guide 1.76
Maximum translational speed	65km/h	
Minimum translational speed	7km/h	
Maximum pressure drop	0.1bar en 2.5seg	
Duration of maximum pressure drop	1.5seg	

Table N° 1-2: Design parameters for tornado.

The following types of objects exemplify what can be launched by tornadoes (*Table 1-3*):

Item	Object Type	Measurement (cm)	Weight (kg)	Speed relative to the air	Maximum height above ground level
1	Wood board	10x30x370	90	0.8	Without limit
2	Steel rod	2.5diam.x100	4	0.6	Without limit
3	Steel pipe	7.6diam.x300	35	0.4	Without limit
4	Steel pipe	15 diam.x450	130	0.4	Without limit
5	Steel pipe	30 diam.x450	335	0.4	Without limit
6	Wooden Post	35diam.x1000	675	0.4	9m
7	car	1.86 m^2 frontal areal	1800	0.2	9m

Table N° 1-3: Types of objects thrown by a class F3 tornado

In the design of the containment, in order to protect its integrity, the following design criteria to face the threat of objects thrown by tornadoes were used:

- The containment should not allow the leakage of any permissible quantity of radioactivity to the environment as a result of the impact of an object on it;
- Admittedly, objects launched can cause damage only to one of the redundancy systems that perform safety functions required to meet an accident.

The design of the containment of CNA II and CNAI is similar.

1.3.1.1.2.2. Structure of the main buildings in CNA I

The containment building has an outer concrete enclosure which separates it from the environment. The thickness of the walls of the enclosure is of 80cm in the cylindrical part and 60cm in the spherical part. Furthermore within this concrete enclosure is a 50m diameter and 22mm thick inner steel spherical shell.

It is worth noting that the concrete structures are appropriately armed with longitudinal and transverse rods in each direction, whereby the radial cracking propagation caused by the impact of objects will be inhibited or controlled by the steel and the concrete portion that is dislodgedby the impact to its surface is limited.

1.3.1.1.2.3. Weak points and cliff edge . Buildings and equipment that could be affected

The most important buildings to protect against a tornado are those having to do with nuclear safety, namely:

- Pool Building ;
- Pump House;
- Water cooling pumps, pipes and associated cables;
- Lines of the emergency cooling water services; and
- Second Heat Sink.

The reactor building is not considered critical because it is designed to withstand the impact of a small-sized aircraft.

The systems and main components to protect, among others, are:

- Fuel element pool and the river water ensured cooling circuit (UK).

1.3.1.1.2.4. Provisions to avoid extreme situations or to increase the robustness of the plant

The building of the new emergency power supply was designed and built taking into account the design Tornado for F3 maximum speeds (see Table 1-1) and a class 5 missile (see Table 1-3).

The new building for the Dry Storage of Spent Fuel Elements (ASECQ) adjacent to the current Pool Building N°1 is in the developmental stage of basic engineering. It shall be subject to new requirements, including of civil domain. It will be designed respecting updated documentation relating to the impact of missiles generated by tornadoes, among others, it will take into account the ACI 349-01 code, Code Requirements for Nuclear Safety Related Concrete Structures, that identifies the potential local effects on the impacted structure .

Using the criteria applied to these new buildings, the original conditions of impact of objects thrown by tornadoes shall be re-evaluated for the rest of the existing buildings at the plant, in order to avoid problems related to plant safety. It is estimated that this re-evaluation will be completed in 2013.

NA-SA has launched a re-evaluation of the risk of tornadoes for the Atucha site that is estimated to be completed by 2015.

1.3.1.1.3. Lightning

The system of lightning protection for all buildings is designed according to the original German standards and the General Regulations for Lightning Protection.

The transformers have overvoltage protection. Moreover, taking into account additional protective measures for buildings containing instrumentation and control equipment, in these, the conductors of the lightning rods are not only connected to the ground but also to the reinforced steel structures, which greatly reduces any interference field inside the buildings in the case of a lightning discharge.

1.3.1.1.3.1. Weak points and cliff edge. Buildings and equipment which might be affected

Regarding the lightning discharges, these can occur anywhere in the nuclear site. The most important buildings that may be affected are:

- Manoeuvre Building ;
- Machines House;
- Pool Building ;
- Main transformer, and
- Pump House.

The systems and main components that are protected are those related to safety leading to a safe shutdown condition of the plant in an emergency. Among other systems, the following deserve to be listed:

- Transformer plants and network transformers (Group A);
- High voltage equipment and transformers for own consumption (Group B);
- Low voltage installations, main distributions and transformers for own consumption (Group C);

- Continuous current (CC) generating facilities and emergency services. Major distributions (Group E);
- Emergency CC facilities (Group F);
- Dashboards and control panels, command post (Group G) and
- Auxiliary dashboards and cabinets for drive latching, automation, alarms and protection (Group H).

1.3.1.1.3.2. Provisions to avoid extreme situations or to increase the robustness of the plant

Based on historical data indicating the existence of an important lightning discharge that occurred in 1977 that damaged the main power transformer causing the outage of the plant, it was decided to conduct a thorough investigation of the grounding system, and the following actions were taken:

The connections linking the mesh surrounding the buildings of the plant to the pipeline of the hydraulic turbine (outside the enclosure in which is located the elastic connection of the pipe) were inspected and repaired.

To install the grounding wires 70mm2 bare copper conductors were installed in various parts of the buildings detailed in the diagrams of the nuclear plant.

The work consisted in building a sort of Faraday cage in the main buildings and, in order to distribute the atmospheric discharge fast to the ground, bare conductors were placed in a fan shape.

To further accentuate the diffuser effect other stretches of bare conductor were tended by way of a "counter-weight" with a javelin buried in each free end.

1.3.1.2. Atucha II Nuclear Power Plant (CNA II)

This Section contains a description of other external events for which CNA II is designed. At the end is summarized the protection state of Safety buildings against the followings external events.

1.3.1.2.1. Wind loads

Bases for the Determination of Wind Loadings

The plant structures are designed for the wind velocities given in the following table. The design wind velocity is transformed into statically applied design wind loads in accordance with the recommendations of DIN 1055.

The effective velocity pressure for structures and for portions thereof at various heights above the ground is in accordance with the table.

Elevation above ground [m]	Design Wind Velocity [m/s]	Effective Velocity Pressure (q) [kp/m2]
8 or less	28.3	50
8 - 20	35.8	80
20 - 100	42.0	110
more than 100	45.6	130

Table: 1-4

Calculation of Wind Loads

The total design wind load (W) on the entire building in the direction of the wind is obtained by calculating the vector sum of the resultant forces acting upon the individual elements in accordance with DIN 1055 including the effects of positive pressure on the windward wall and negative pressure on the leeward wall.

1.3.1.2.2. Tornado

Tornadoes can be characterized as vortices possessing tangential, radial and translational velocities the net effect of which is a strong wind force. The wind force varies from a small value at the centre of the vortex to a maximum at the edge and then decreases as the distance from the centre increases. Three potential effects are as follows:

- The tornado wind loading, Ww;
- A differential pressure caused by a relative rapid atmospheric pressure change, Wp;
- The impact of tornado generated missiles, Wm.

It is being presumed that the centre of the tornado passes the power plant and that sand will be whirled up.

The ventilation systems for safety related buildings are designed, where necessary, to withstand the effects of a tornado either by the closure of ventilation openings (change-over to recirculation operation) or by designing the ventilation systems for the conditions prevailing during a tornado. A tornado warning is issued by the Meteorological Station in time for the CNA II operating personnel to take the necessary measures.

It is also presumed that the outside electricity supply to and from the NPP will be disrupted and that emergency power supply from the diesel generators in the UBP building will be required subsequent to the tornado event. The ventilation system of the UBP building, the supply of air for combustion and the exhaust gas discharge system are assured.

As was mentioned in item 1.4.1.1.1, the safety structures are designed to resist the Design Basis Tornado (DBT). The DBT is taken as class F3 on the Fujita F-scale. The design parameters are:

The Design Parameters for the Design Basis Tornado are indicated in table 1-2. All buildings containing engineered safety features are protected against penetration by tornado generated missiles. In line with engineering practice in the USA, the tornado-generated missiles were considered as indicated in table 1-3.

According to table 1-3, the maximum heights of missiles above ground level considered are:

- Items 1 to 5: no limit.
- Items 6 and 7: 9 m.

The tornado-generated missiles are being considered to be capable of striking from all directions. The tornado wind and tornado differential pressure loads are applied to the structure as uniform static loads acting normal to the surfaces to which they apply.

Moreover, for the safety related ventilation systems of these buildings, the pressure difference during tornado striking was considered. This means that either the outer openings can be closed to operate the ventilation system in closed circulation or the safety function of safety related components that can be impaired by the vacuum wave of the tornado has to be avoided. It was assumed that the connection to the external grid is opened. Combustion air intake and exhaust gas outlet for emergency diesel operation is assured.

Failure of parts of structures not designed to resist tornado loads (assuming missile generation from panels, piping or non-safety related systems) will not impair the design function of seismic category 1 structures that have been designed for tornado loads. Seismic category 1 structures are designed for a tornado missile selected as being the most damaging of a wide range of postulated missiles.

NA-SA has launched a re-evaluation of the risk of tornadoes for the Atucha site that is estimated will be completed by 2015.

1.3.1.3. Embalse Nuclear Power Plant (CNE)

1.3.1.3.1. Wind loads

The maximum wind speed assumed for design purposes is 150 km / h. The building structures are designed for active loads caused by winds in accordance with the requirements of the "National Building Code of Canada" and its supplement. Subsection 4.1.8 of the Code details the methodology and criteria used to calculate the active loads caused by winds taking into account the effect of gusts. The supplement code also outlines the various approaches to determine design wind loads on buildings.

The calculated wind loads are combined with other loads to determine the stresses in the structures of buildings. The combination of charges is performed in accordance with subsection 4.1.2 of the code or according to the requirements of design guide AECB DG-18-21000-00J, as appropriate. In the final sizing of the structural members active loads are considered, either due to wind or earthquake that produce an even more unfavourable effect.

In separate structures, a system of emergency water supply (EWS) and emergency power supply (EPS) have been provided. These structures, the equipment they contain and any other equipment or component monitored or supported by this equipment are seismically qualified. The equipment contained in the structures is capable of providing alternative heat dissipation in the event of absence of the normal capability of heat removal.

1.3.1.3.2. Tornadoes (spent fuel storage system -ASECQ-)

Studies carried out on the occasion of the licensing and construction of the silos system ASECQ were added in the revision of the 1993 safety report. The data for the Embalse area is used as part of the siting study. This study is in turn based on the IAEA guide.

As recommended by the guide and taking a historical series of 50 years, the different parameters characterizing the maximum tornado were determined.

The maximum speed is V = 430 km / h which corresponds to a speed whose frequency of occurrence of a tornado with a speed that exceeds it, is lower than 10-7/year. This value is determined by considering that the frequency of a tornado with these characteristics is the product of two independent factors.

The value of the frequency of occurrence of a tornado in the vicinity of the plant is determined as the product of the frequency of tornadoes throughout the area under consideration multiplied by the ratio of the area affected by the tornado and the study area.

The tornado that was used for the design of CNE has a speed of 150 km / h which corresponds to a total frequency = 6 x 10-5/year based on the same distribution.

Based on the IAEA guide two types of projectiles have been taken into account.

1) Projectiles with a large mass and high kinetic energy whose impact produces deformations (e.g.: a car);
2) Rigid projectiles of large dimensions for which is required penetration resistance (e.g. a light pole).
3) The third type of projectile, rigid and small sized that can pass through holes of protective barriers, does not apply in this case as there are no openings in the silos.

Since the silos are cylindrical, weight 100 tons. when empty, 6 m. high, 3 m. diameter and have 85 cm. thick concrete walls, with the input of the calculation mentioned below it was concluded that an impact could affect the concrete shield partially but not the inner linner.

A calculation was performed using empirical formulas for the case of the impact of a type 2 projectile. An impact speed of 35% of the maximum horizontal speed of the maximum tornado was used and a radial direction of impact. The maximum penetration was calculated (which does not exceed 0.51 m.) and the detachment effect on the interior wall of the opposite side, concluded that even this impact does not affect the integrity of the inner linner of the silo. From the above the external event tornadoes need not be considered for the area of the silos.

NA-SA has launched a re-evaluation of the risk of tornadoes for the Embalse site estimated to be completed by 2015.

1.3.1.3.3. Intense rain

A storm of heavy rain and south wind revealed a weakness of the CNE for that external phenomenon. The impact on the plant was influenced by another climatic factor as it coincided with an intense drought, destruction of pastures due to fire and land use by humans. The material swept towards the water inlet of the NPP causing the outage of the plant. The measures taken as a consequence of this event were:

- Hiring divers to make periodic cleaning of the water inlet;
- Return the cleaning system of the intake to its design situation, and ;
- Take steps to prohibit planting and grazing livestock on land adjacent to CNE. No weeding or clearing of these areas.

1.3.2. ACTIVITIES PERFORMED BY THE REGULATOR

1.3.2.1. Conclusions

Based on the assessments made to date, the ARN concludes that:

- It has identified the need for regulatory actions that do not constitute relevant weaknesses that require taking urgent action.
- It has verified that NA-SA complies with both the design and licensing basis.
- The consideration of tornadoes and wind loads for Argentine nuclear power plants is consistent with the criteria and requirements, both domestic and international, established at the time of design. However, it was considered necessary to conduct new studies re-evaluating the risk of tornadoes for the Atucha and Embalse sites, which have already been initiated. The result of this re-evaluation will determine if it is necessary for ARN to require complementary actions, changes or additional improvements.
- Tornadoes, wind loads, lightning and heavy rains have been analysed and it is considered that NA-SA is carrying out the appropriate action to successfully meet these scenarios.
- The ARN will continue monitoring and evaluating the actions that are being and / or will be implemented in the future to ensure that they are effective and to consider all necessary aspects related to plant safety. The result of such activities will determine whether is necessary that the ARN requires complementary actions, changes or additional improvements.

2

DESIGN (DESIGN ISSUES)

2.1. INTRODUCTION

Additional actions planned and carried out by both the Licensee of Nuclear Power Plants (NPP) in Argentina (NA-SA, Nucleoeléctrica Argentina SA) and the regulator (Nuclear Regulatory Authority - ARN) are presented, in order to demonstrate the capability of the safety systems of nuclear facilities to fulfill prevention and mitigation functions against extreme scenarios posed as a consequence of the lessons learned from the Fukushima accident. Expected responses of the safety systems under conditions of successive losses of power supplies and heat sinks were specifically evaluated.

The actions include evaluations to determine the expected behavior of the safety systems of the NPP as intended from the design bases. All available resources identified as a useful alternative to maintain safety functions were taken into account.

The analysis was aimed at preventing a hypothetical severe accident or damage affecting the reactor core and spent fuel stored in pools. An estimation of the available time to prevent progression toward severe core damage is done for each case.

Additionally, the already implemented or planned actions improving the response address the weaknesses identified in each of the accident sequences analyzed.

Therefore, the above mentioned actions consisting in a reassessment of the NPPs safety margins assuming the occurrence of a sequential loss of the lines of defence in depth due to extreme initiating events occurrence. This assessment includes the long term evolution of the severe accidents and the recovery capability of both the power supply and the water supply until a stable plant condition is reached. This is to identify the most adequate recovery strategies and the components that must be available for each of the corresponding strategy implementation;

Annexes II and III describe the technical characteristics of NPP in operation CNA I and CNE, and the NPP at initial stage of commissioning (CNA II), respectively.

2.2. ACTIVITIES PERFORMED BY THE OPERATOR

The results of the assessment carried out by the Licensee of the two operating NPP (CNA I and CNE), and NPP at preoperational tests stage (CNA II) are presented. Also improvements implemented or planned in each case, taking into account the weaknesses found under extreme situations are presented (for instance, due to cliff-edge effect identified). These activities are developed in response to the regulatory requirement sent by ARN (RQ 38 – point 5 “Loss of safety functions”) in order the Licensee evaluates existing safety margins by analyzing the CNA I, CNA II and CNE NPPs behavior to the occurrence of extreme events that cause consequences such as the total loss of power and ultimate heat sink for a long time. They were also taken into account the recommendations received from international organizations like the World Association of Nuclear Operators (WANO) and experience given by the program of Significant Operating Experience Report (SOER).

2.2.1. ATUCHA I NUCLEAR POWER PLANT (CNA I)

Power Supply Systems of CNA I

The generating station of the CNA I is connected to two external electrical networks, physically separate, 220 kV and 132 kV. Additionally, if necessary by disturbances in the external network, feeding the own services from the generator is possible.

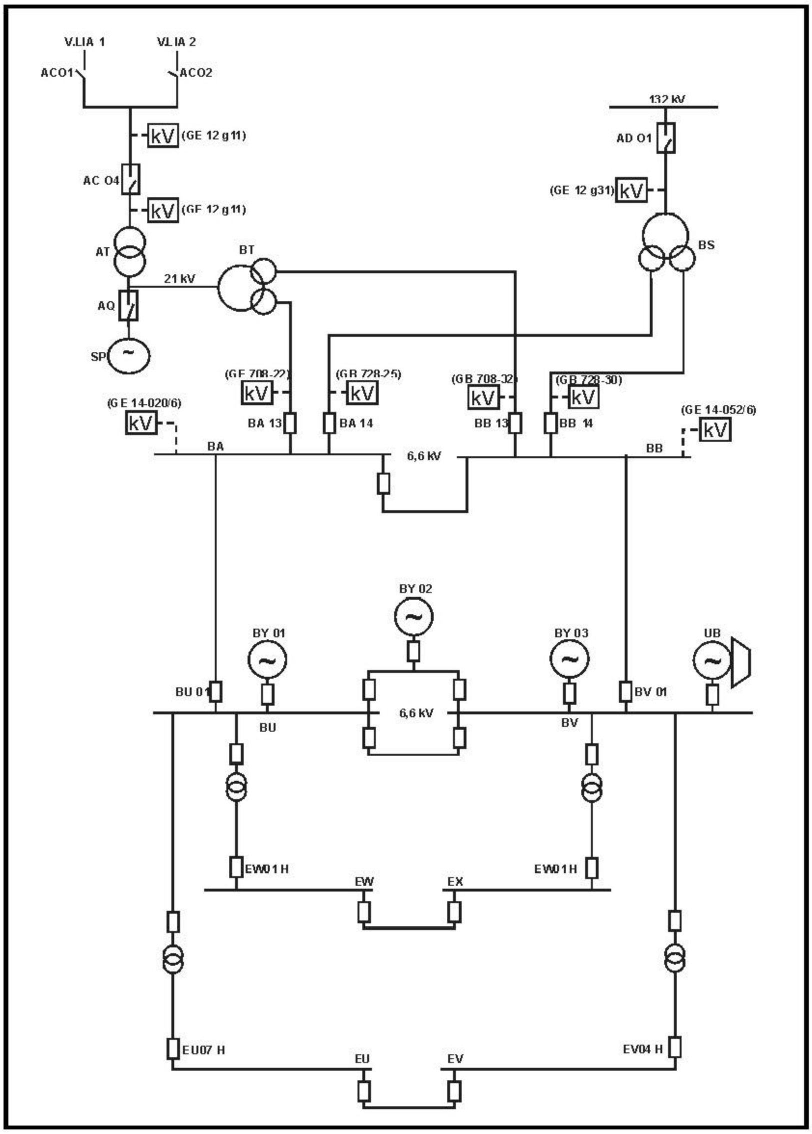
The CNA I electrical generator feeds 220 kV network through a main transformer and feeds own services through an auxiliary transformer. The output of this transformer feeds two separated bus bars of 6.6 kV, over which main loads and low-voltage loads transformers are connected.

As an alternative to these previous three supplies, the original installation design includes a set of three emergency diesel generators (DG CNA I), with 50% capacity each one of them. Additionally, the CNA I has an interconnection line to two CNA II diesel generators (DG CNA II), the NPP constructed close to CNA I. This interconnection (*Figure 2-1*) was specially built few years ago and now is available and could be used by CNA I up to CNA II NPP commissioning. These additional generators are included in the normal maintenance and testing schedules of CNA I NPP.

If the power supply to the own services fails, some specific equipment are necessary (pumps, valves, etc.) in order to remove the residual heat and brings the plant to a safe shutdown condition. This equipment must remain in operation or has to be start operation. For this reason, is intended by design base to provide secured power supply to this essential equipment by emergency diesel generators (DG). This power is provided from the secured bus bars. In normal operation the emergency supply system is divided into two bus bars connected by respective switches to each of the secured bus bars named BU/BV.

To ensure the characteristic of uninterruptible emergency power supply, both secured bus bars are fed by a water turbine up to start connecting to CNA I DG to the bars in a time less than 40 s.

Both secured bus bars are providing electrical power supply to the components with essential safety functions (moderator system pumps, volume and inventory control system pumps, secured river water cooling system pumps, safety injection system pumps and intermediate cooling system etc.).



SP - Generator
AT - Generator Transformer
BT - High-voltage Station Service Transformer
BS - Off-site System Transformer
BY - Diesel Generators
UB - Water Turbine - Driven Generator

Figure 2-1: Basic scheme of the CNA I power supply system

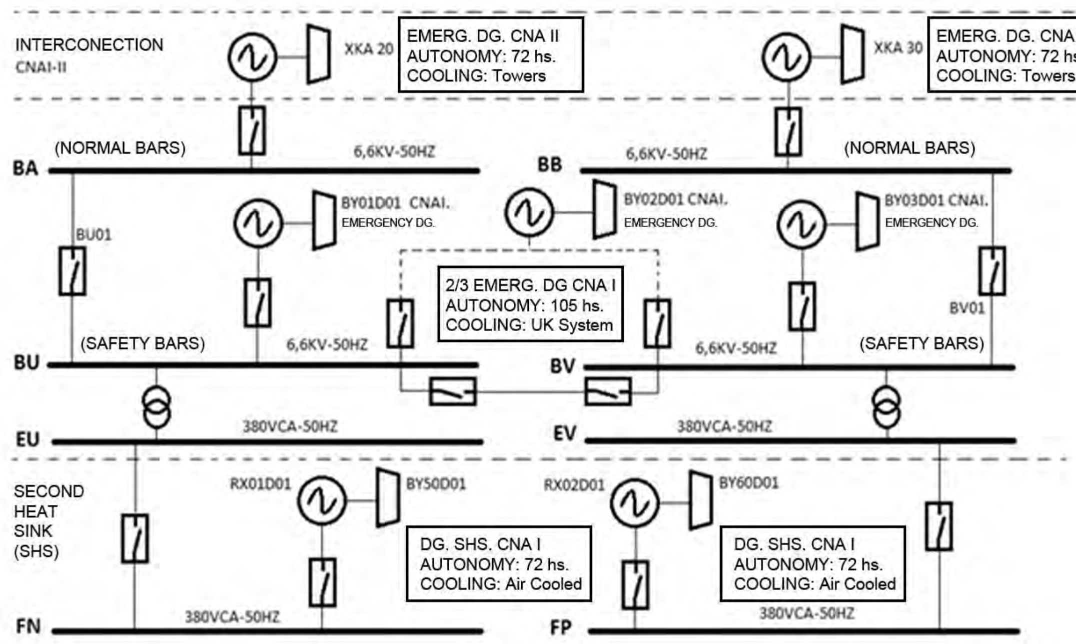


Figure 2-2: DG wiring diagram of the different systems: BY, SHS (CNA I) and XK (CNA II)

2.2.1.1 Loss of Off-Site Power - LOOP

The Loss of external power (LOOP) was originally considered as a design basis event (DBA) and the resulting transient has been analyzed in the PSA study. In that study the behaviour of the plant under hypothesis of loss of the three alternatives networks of 220 kV and 132 kV, and the failure of the generation itself was analyzed. In such condition there will be only emergency power supply provided by DG CNA I and alternately provided by DG CNA II through the interconnection line and are two different options of residual heat removal and cooling in the long term. The first is via the cooling system through the moderator system (QM) and its connection with subsequent cooling system (RR) and secured river water cooling system (UK) – named QM/RR/UK residual heat extraction chain - up to ultimate sink or river. This alternative has indeed two independent circuits. *Figure 2-3*.

The second alternative, independent of the previous one is the removal via secondary side and the steam release to the atmosphere. In this case there is an alternative heat sink system called second heat sink (SHS), whose design has two separated chains or redundancies, and each has a pump, diesel engine and its own electric generator. The baseline PSA study analyzed the potential failures of the normal heat removal system and the replacement by the SHS (*Figure 2-4*). System characteristics with important safety factors were evaluated, not only as to the effectiveness of the function but also to maintain capacity for the minimum time required.

The performance of the containment isolation, under postulated conditions for this event will not be affected due which is fed from the uninterruptible secured power supply system.

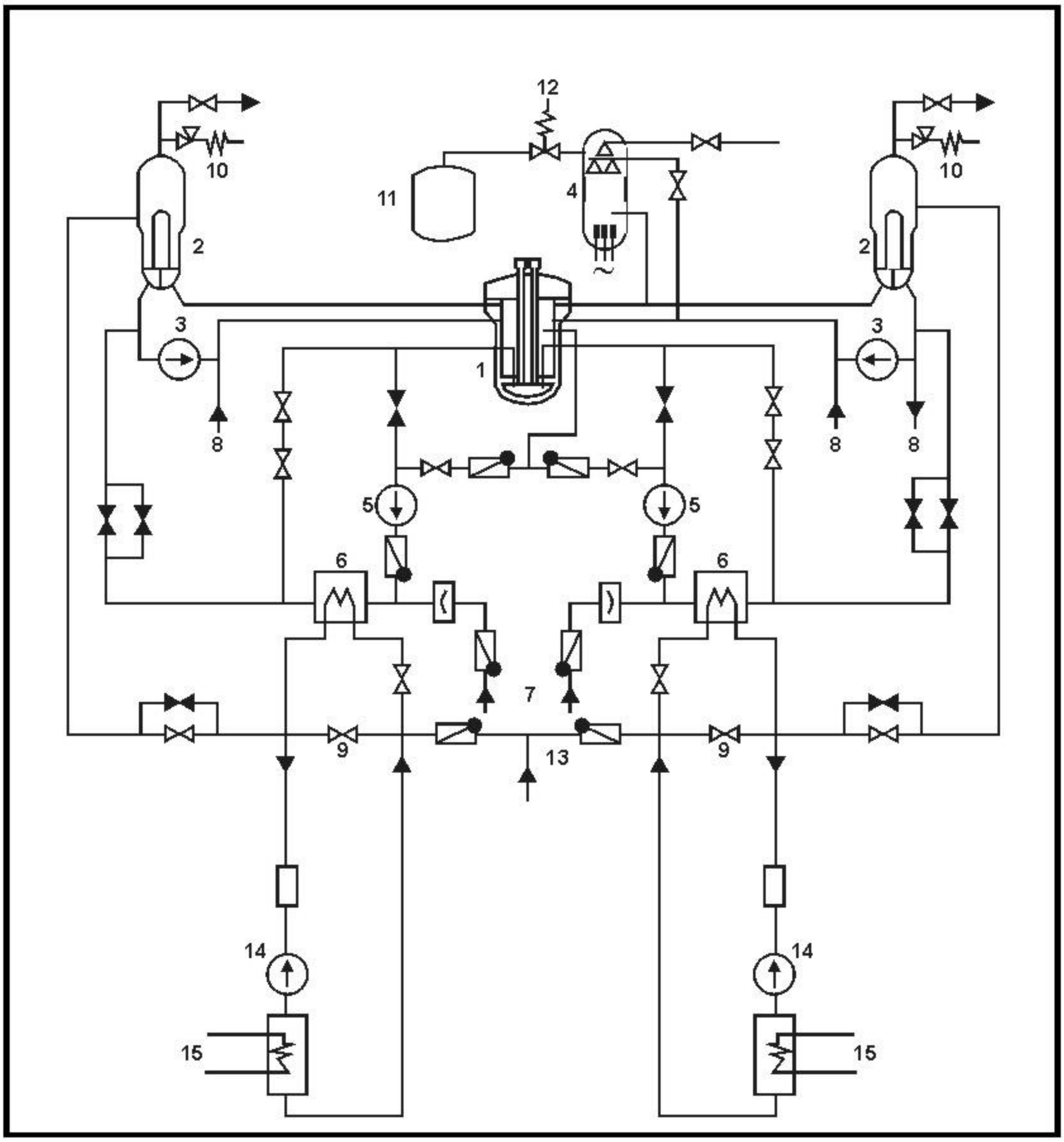
Characteristics of the Diesel Generators (DG-CNA I)

The fuel consumption is about 390 kg/h with normal demand conditions and about 430 kg/h in maximum load regime conditions. The maximum load corresponds to the situation of a small coolant accident (LOCA, Loss of Coolant Accident) coincident with loss of normal power bus bars.

The deposit for daily consumption for each DG is 1000 liters, estimated to be sufficient for approximately 2.1 hours at normal demand and approximately 1.9 hours at overload regime.

There is a common reservoir for all DG groups with a capacity of 50 m3, enough to cover service of 2 DG and a maximum emergency current of approximately 2.9 MW. This reservoir can operate two of the DG for 105 additional hours.

To ensure the proper functioning of the DG for the required time, it has the maintenance and testing procedures in place, including the support and backup systems, such as lubricants and spare parts of DG equipment.



1 - REACTOR PRESSURE VESSEL
2 - STEAM GENERATORS
3 - REACTOR COOLANT PUMPS
4 - PRESSURIZER
5 - MODERATOR PUMPS
6 - MODERATOR COOLERS
7 - EMERGENCY COOLING SYSTEM INLET
8 - PRESSURE AND INVENTORY CONTROL SYSTEM
9 - SHUTDOWN COOLING SYSTEM (MODERATOR)
10 - SECONDARY SIDE SAFETY VALVES
11 - PRESSURIZER RELIEF TANK
12 - PRIMARY SIDE SAFETY VALVES
13 - SECONDARY INLET LIGHT WATER
14 - RESIDUAL HEAT REMOVAL SYSTEM
15 - SERVICE COOLING WATER SYSTEM FOR PLANT SECURED

Figure2-3: Main cooling and moderator systems at CNA I

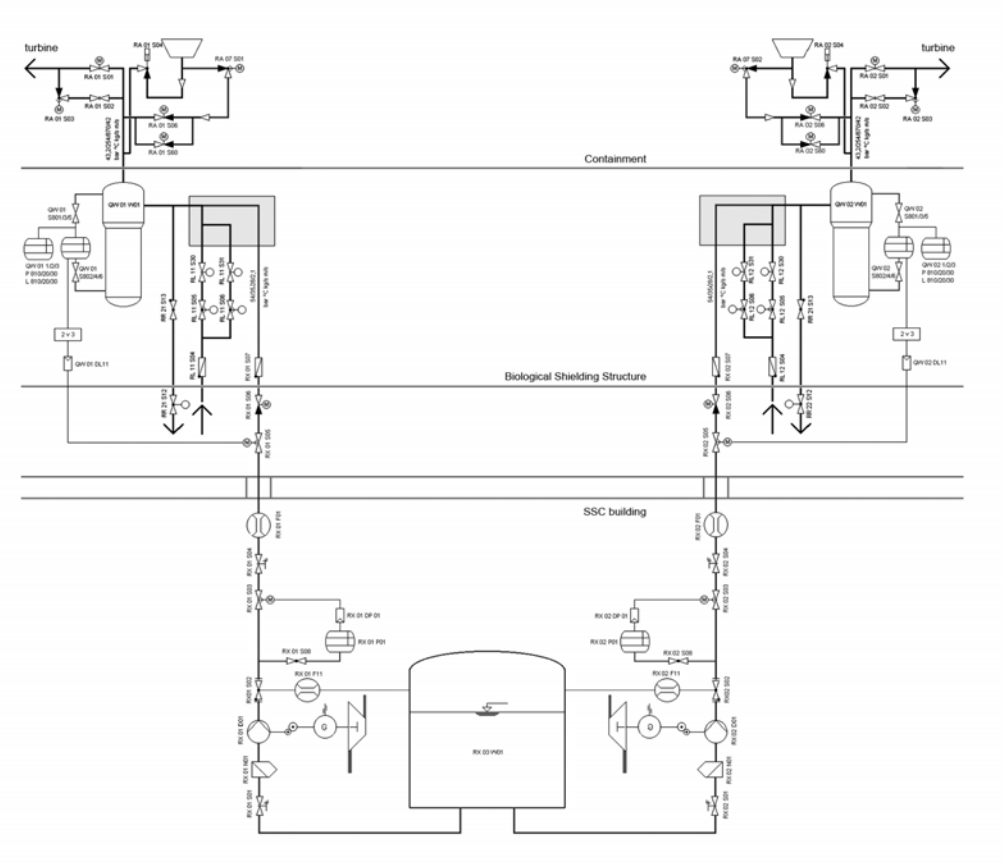


Figure 2-4: Outline of the second heat sink system (SHS).

Impact on fuel stored in pools

This case considers the emergency power supply which feeds the cooling systems, I&C of temperature, level, etc. The event discussed here does not produce degradation in the ability of these systems, due to the pools refrigeration is not affected since it is ensured through UK01/02/03D01 pumps with secured power supply.

Regarding the effect that this event would have on the refueling machine (RM) of spent fuel elements, if it occurs during use, it was estimated that the cooling function of the spent elements is ensured due to the power supply of RM is secured, allowing the completion of the replacement process or transfer the fuel elements up to the pool.

Measures already implemented or planned to improve the response of safety systems

Restoration of external power supply

For the restoration of external power supply after an eventual collapse in the network, the company responsible for the management of supply networks, Compañía Administradora del Mercado Mayorista Eléctrico Sociedad Anónima – CAMMESA, has a reset procedure for CNA I supply that imposes the priority of NPP due to safety requirements of nuclear installations.

Review and improvement of emergency procedures

The review is being made to ensure the operation of the systems that are necessary in the proposed scenarios to ensuring the proper functioning and demand of safety systems which are required in extreme events for at least the initial 72 h.

Electrical interconnection between normal bars of CNA I - CNA II

Given the presence of one electric interconnection between both NPPs CNA I and CNA II, is exploring the possibility of keeping it in the future, after commissioning of CNA II, in order that both plants have another alternative external power supply. Thus the CNA I could count on the 500 kV lines with two redundancies.

<u>New Emergency Power Supply System (EPS) in replacement of the current DG system of CNA I</u>

It is at the stage of building a new emergency power supply system (EPS). The system is housed in a new separate building, outside the CNA I, with 3 new DG with a capacity of 4,000 KVA each, feeding to 6.6 kV bus bars. The system will be set up as safety system of CNA I. This improvement is schedule to be implemented during the second half of 2013.

This system currently under construction in adjacent site to the CNA I is seismically qualified to the revaluation of CNA I earthquake (PGA 0.10 g) and the building has been designed taking into account the design bases tornado used in case of CNA II (Fujita scale F class F3).

General Description of EPS

To design the new EPS (Emergency Power Supply), was taken as a design basis accident (DBA Design Basis Accident) a small loss of coolant (Small LOCA) combined with emergency power case. This is the scenario that imposes the greatest load on bars of the EPS system.

In particular, for the design of load balancing, taking into account the possibility of having one of the DG at maintenance and the failure of another to boot. It has been taken also as a design criterion that the electrical supply to the emergency bars could be interrupted during a short period of time.

Main features of the new EPS

- Designed with two redundant trains of 100% each, independent and separated. Three autonomous DG with capacity of 100%, one for each redundancy and the third is in reserve. Each DG is located in a room physically separated and having its own fire protection (*Figure 2-5*).
- The loads connected to the interruptible "short time" bus bars and non-interruptible will be relocated to maintain reassigned so to sustain the design criteria.
- Two uninterruptible power subsystems 100% and a third unit on standby, in order to maintain the original design criteria of having a non-interruptible power supply to a set of essential loads, such as valves closing the containment in case of large LOCA and controlled steam release valves.

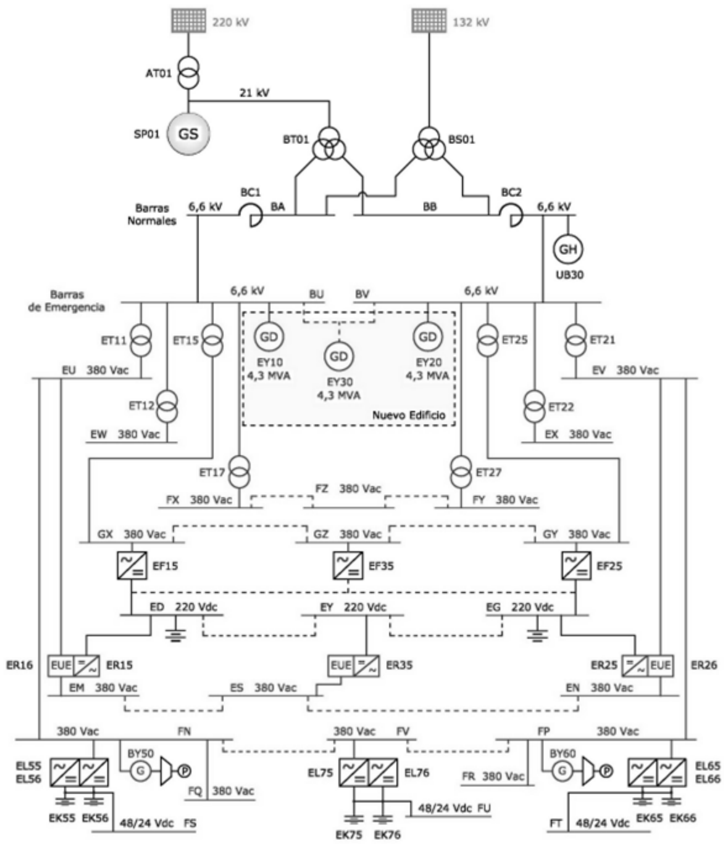


Figure2-5: Power supply system of CNA I with the new EPS system.

Impact of improvements of the new EPS

The impact of new design improvements in the behavior of the CNA I was analyzed from the safety point of view and taking into account deterministic and probabilistic approaches. The new design took into account the following changes: spatial separation of diesel groups, spatial separation between electrical bars, improvements to common cause failures, protections improvements, improved redundancy, improved ignition system, improvements in DG refrigeration.

To adapt this system to KTA 3700 standard some deterministic single failure criteria, independence between trains and redundancy were applied. The independence between trains and spatial separation can minimize coupling phenomena.

In addition, improvements are achieved in the logic of power (several signals, redundant and within the reactor protection system). For instance, the bars are connected to DG downloaded, preventing premature deterioration of the generators.

Additional improvements of EPS

The power supply to emergency power bus bars BU/BV from normal bars BA/BB will be enhanced by duplication of coupling switches between both sets of bars. This will be implemented together with the EPS.

After commissioning of the new EPS, the hydraulic turbine (UB) that currently issued to maintain the electrical load during the time needed to start up the DG and the couple of bars, will not be part of the new emergency system and will connect to a normal bar BB. This system is no longer needed to ensure the system as an uninterruptible safety system and ceased to be part of the secured supply system.

2.2.1.2. Loss of external and internal power supplies (Station Black Out - SBO)

This scenario would involve loss not only of external supplies to CNA I, networks of 220 kV and 132 kV, and power supply to the CNA I services from the generator, but also the two successive failures of emergency power supplies DG CNA I and DG CNA II. These failures can be at starting or running. On such conditions the NPP it will be at SBO stage.

The expected performance of the CNA I systems address the situation of the SBO was analyzed and identified as SBO1. In such conditions the only mechanism for cooling remaining long term is through the second heat sink (SHS), via the secondary side by the SGs.

In the SBO1 scenario the reactor shutdown occurs after main pumps being out of service. Also out of service all systems or AC loads of the facility, unless the loads powered electrically by the SHS system. The expansion of the primary is driving the pressuriser to the state of liquid phase alone (solid state). While there is a pressure relief by opening the safety valve, it is postulated rupture thereof due to exceed the specified conditions for this valve. After this transient is expected a small LOCA, activating the entrance of the SHS with a cooling ramp of 100 °C/h by automatic tripping of the reactor protection system due to the small LOCA signal (NZ52). This signal is activated by high pressure inside the containment and pressure decrease in the primary system, causes the trip of the ECCS and the high pressure system TJ (TJ-HP), but it stops about 50 minutes after when first battery depletion is foresees. The low pressure system TJ (TJ-LP) will not work from the beginning because their pumps are fed from secured bus bars from the DG. Faced with the depletion of the batteries that are powering the cabinets (JK) of the secured power supply system, needed for the instrumentation and control, trigger occurs widespread signals of the reactor protection system (NZ) and NZ51 including the signal that triggers the action of the second reactor shutdown system producing the boron injection into the primary system / moderator. Due to depleted batteries is not possible to close the injection after the entry of TB injection, avoiding the entry of compressed air to the primary system. This air will reach the top of the "U" tubes of the steam generators SG, causing deterioration in natural convection cooling and circulation originating dual phase. The primary rupture and the SHS injection in the secondary side will reduce primary temperature up to 140 °C at a pressure of 20 atm., but the absence of inventory replenishment, will inevitably reach the uncovering of active zone of the core around 2 h 50 min from the start of SBO. As estimated after 5 h would begin core degradation. At this stage will be reached core degradation and is estimated to produce the first relocation of channel coolant and fuel elements material in the bottom of the reactor pressure vessel (RPV).

Impact on the fuel stored in pools

Estimation has been carried out of the impact of SBO1 in the fuel stored in pools. These estimations took into account the expected movements according to the planned operation until 2015 and considering the total lack of refrigeration as a consequence of some SBO1 event. In that case, considering the most unfavorable conditions, the water would reach a temperature of 100 °C after 72 hours. However to get to the uncovering of the EECC it takes about 20 days, assuming that conditions remain constant.

This is a slow process allowing a margin of several days, it is expected to put water pump for feeding pools from an independent source of water, maintaining the proper water level in the storage of irradiated fuel elements, the possibility of a manual connection to an auxiliary MDG. It is estimated that this change will be implemented during 2013.

Programs of periodic inspection have been revised and updated including control of the functionality of systems of rupture vacuum / siphon pipes associated with cooling systems, and inventory control of the fuel elements storage pools. Also the frequency for the periodic inspection and testing of these systems was increased for weekly verification.

The operational instruction "Operation on disturbances and accidents" was modified in order to include the control of critical parameters of the irradiated fuel pools, among which include temperature and the level of water.

The evaluation of the integrity of the EECC inside the RM when the on-line exchange is progress, and under the conditions postulated for this event SBO is in assessment is process.

Actions implemented or planned for improving the response of safety systems

Emergency and Accident Management Procedures

In a situation such as SBO described above, the CNA I has developed a strategy proposal in the framework of the Severe Accident Management Program (PGAS). This is proposed as an alternative improvement to be implemented.

The strategy proposal is a plant operational instruction for SBO1, which proposes a manual action in a short time to inject the SHS with a cooling ramp of 100 °C/h via the secondary side. Additionally, it is necessary to manually turn off the TB to avoid the entry of air into the primary circuit in order to mitigate the accident, and taking into account that presence of air inside the primary circuit may difficult the circulation by thermo-siphon and reducing effectiveness of thermal transfer across SG's tubes. With the application of this cooling ramp of 100 °C/h is achieved quickly the lower temperature and consequently the expansion of the primary heat transport system (PHTS) is reduced, avoiding LOCA by the pipe of the pressurizer safety valve. If cooling is not achieved, according to the analysis performed for the operating conditions in liquid phase, there would be the failure of the valve and / or rupture of the pipe connecting the pressurizer. Additionally, this mentioned cooling ramp, will also maintain the primary temperature below 120 °C, avoiding deterioration of the PHTS main pumps (QF) seals. In case of this occur at temperatures above 140 °C, there would be coolant inventory losses due to failure of the main pumps seals.

The batteries duration was recalculated with realistically plant data. It was estimated as 1.5 hours for banks with highest demand and more rapidly depletion time, taking into account the corresponding loads of consumption in each case. In other cases, was estimated a duration time about 4 hours. A longer battery life keeps Instrumentation and Control indications available for longer time and facilitates control of temperature and pressure of the primary through SHS with their indications of pressure and level. However, even under conditions of total loss of the batteries, the operation is possible using the instrumentation available on the secondary SHS. This allows the operator to estimate the conditions of primary information through secondary I&C, maintained by the SHS and its own power generation system. Release valves of the SGs are maintained through the SHS, as well as the reading level and pressure of the SG. However the improved estimates of the time duration of batteries, operational experience have shown plant signals NZ triggering, about one hour after missing the AC power supply. For this reason is proposed to close TB system in the short time to prevent the air injection into the primary system. Adequate cooling time of the core in these conditions will be limited only by the water supply capacity from the SHS system. To ensure the long-term cooling of the reactor, this strategy and the facility for restocking SHS system are necessary. Implementation is expected in 2013.

In this strategy, although first indications of the instrumentation are lost after 1.5 hours, controlling temperature and pressure of the primary system are indirectly achieved by controlling the SHS with its own indications of pressure and level, once it has reached an outage between 100 and 120 °C. Thus, that prevents the opening of pressurizer safety valve and kept by thermo-siphon circulation in the primary system. This strategy would maintain the cooling within the prescribed period of more than 24 hours. To extend this function by at least 72 hours, as required by the Regulatory Body, it should expand the capacity of the water supply tanks of the SHS system, by replenishing water. It is scheduled to implement this improvement and the operating instruction by the end of 2012.

Procedure for replenishment of water of the SHS

It is in approval stage a water replacement strategy for the SHS system. This will allow refilling the two tanks, one for each independent branch of the SHS system, through feed-water line and pump system RL33D01. Considering that this pump will lose power in conditions of SBO, the power supply must be secured by an additional mobile DG (MDG). This will keep the residual heat removal for times greater than 24 h, by replacing the SGs inventory through the SHS system. *Figure 2-6*.

Additionally, in case of SHS pumps failure, this procedure will maintain the water supply, using the available water in the reservoir basins at the facility, UA00B03/B04, and injecting the SG's, when they are depressurized, using pumps UA10 D20 and D21. The water from these pools will be replenished with water well using a pump system UJ. Figure 2-6

This strategy will be implemented as an alternative improvement. Implementation scheduled in 2013.

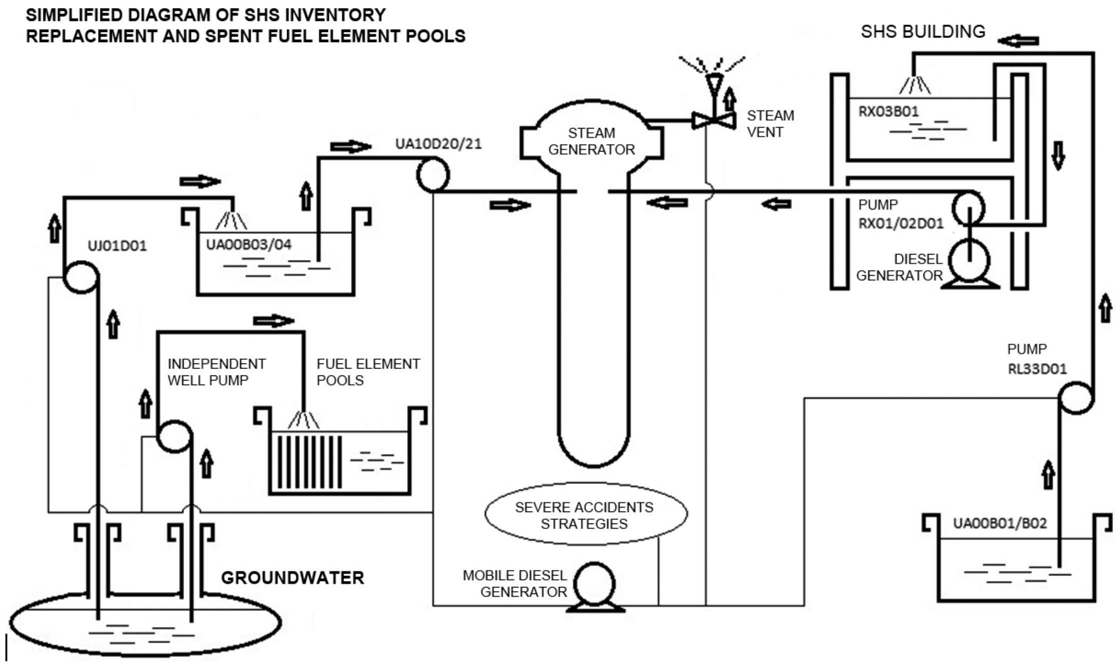


Figure 2-6: Diesel Generator Mobile and additional systems for replenishment of water

Second Heat Sink - SHS

Given the importance of this system in case of occurrence of SBO characteristics of SHS, are described (see *Figure 2-4*).

Under normal conditions the core decay heat extraction is done by feeding the secondary side of both SGs with demineralized cold water supplied by the Main Feed Water System (RL) and the Startup and Shutdown system (RR21/22). Given the need for plant shutdown in normal and / or accidental conditions, the extraction chain involving moderator, intermediate and service water systems is used as a heat sink cooling chain (QM Systems / RR / UK); at least one out of two redundancies of this system should be available for fulfill its function properly.

During fault conditions with unavailability or ineffectiveness of regular heat sinks, plant has the alternative SHS system. This system was designed to be completely independent of existing systems at the plant, and works by feeding and venting ("feed and bleed") of the SGs. The SHS system fulfills the following functions:

- Feeding on the secondary side of both SGs demineralized cold water, due to the unavailability of the SG and the startup and shutdown system.
- Removal of decay heat from fuel elements and heat stored in the primary circuit by evaporation and release to atmosphere via Steam Station Vent.

The SHS system is installed in own building consists of two identical trains, each with instrumentation, control equipment and power separate and independent. Each SHS train is connected to the common water tank and consists essentially of DG that are mechanically and directly coupled to each electrical generator, and through reduction gears to the shaft of each pump system and the I&C valves and insulation.

2.2.1.3. Loss of heat sinks

The scenario would imply the loss of both heat sinks; it means both independent alternative pathways given by chains via the primary system QM / RR / UK and via the secondary system using the SHS, in terms of its ability to supply water.

Considering that power supplies are not loss (discarding the use of DG CNA I because cooling is provided by the UK system), there will be relief valves venting the secondary system. This will be effective during the time that is available the makeup water for the SG, which has been estimated about 4 h, based on the existing inventory in the secondary systems and the feed water system under normal conditions.

Since there is not any of the mechanisms of heat removal, will cause the expansion of the moderator and primary and causing opening of the pressurizer safety valve. Under these conditions, some scenario will be reach. This scenario is close to the SBO1. To prevent the failure of both pumps seals (main primary pumps QF) as the LOCA through the pressurizer valve, or rupture of pressurizer line, the reactor should be cooled with the ramp of 100 ° C/h via secondary side. Although in this case there is no flow across of SHS branches, but once reached the temperature below 120 °C, it must use an alternative mechanism, which is designed to ensure improved long term cooling via secondary.

The time available to avoid the first pressurizer safety valve opening was estimated, based on deterministic calculations, in 15 minutes.

During the transient, according to the estimation, and with open fault condition in the pressurizer relief valve, occurs the filling of the pressurizer. Under these conditions, the plant is stabilized at high pressure of about 70 atm. The core is cooled via the SG through the mechanism of cooling and condensation of water in the primary system, inside the SG tubes (reflux condensation mechanism) while water is available in the SG secondary side. This mechanism is possible by the difference in temperature with the secondary side of the SG. However, due to loss of inventory and the unavailability of the system SHS core cooling deteriorates rapidly. The fuel elements begin to fail after about 4 hours of transient and the core begin to deteriorate after 5 hours.

The performance of the containment isolation, under postulated conditions for this event is not affected because electrical power is provided by the secured or non-interruptible power supply system.

Impact on the fuel stored in pools

There have been estimations of the impact of SBO in the fuel stored in pools. These estimations considered fuel management until 2015, taking into account the increase in the amount of power based on the scheduling of the CNA I operation. These conditions are the highest demanding of power to extract in the coming years, and assuming additionally the total lack of refrigeration as a consequence of SBO. Those results are applicable to this case of loss sinks too. In the worst case it was estimated that temperature of 100 °C is reached after 72 hours. Additionally, it has 20 days until the onset of the uncovering of the EECC stored. Although the time available would be sufficient for the implementation of countermeasures, it will be implemented an improvement in the supply of additional water, as described above, taking water from the well water reservoir. This will maintain stable conditions beyond the required time limit (Figure 2-6).

Regarding the impact on the EECC in the refueling machine (RM), if the event occurs during the use of it as a daily routine refueling exchange, it was estimated that refrigeration is not affected during transport because power supply to RM is normally provided in this event and the completion of the replacement process and / or the final deposition of EECC inside the pool is assured.

Actions implemented or planned for improving the response of safety systems

Additional equipment of water supply to the SG and controlled steam release

In the scenario referred, although is not available the SHS as a replenishment flow system for venting in the SGs, is possible to consider availability of the instrumentation and control (I&C) to control the venting. This scenario is called SBO2 that means SBO1 conditions with the additional loss of both redundancies of the SHS injection to SGs.

Handling proposed in this scenario is similar to case SBO1 discuss before, that is a manual action to apply a cooling ramp of 100 °C/h, and the manual closure of TB system. In the initial stage the procedure is using the available water inventory in the SG, and the water supply system from the condensate tank, taking into account that power supply is available in this event case. However, to ensure stability in the long term, a strategy is needed restocking water to SG's, through a pipe of the SHS. In this case, the water supply to the SG will be done using available water in the reservoir basins at the facility, UA00B03/B04, and injecting to the SG's when they are depressurized, using pumps UA10 D20 and D21. The water from these pools will be replenished with well water using a pump system UJ. (Figure 2-6).

Additional pump of the secured water river system (UK)

The CNA I has three UK system pumps; each provides 50% of the required flow of water in nominal operating conditions and 100% in the shutdown or emergency condition. Under normal operating condition two pumps are working and the third being in reserve. According to CNA I operating experience the failure rate of 2 pumps running is negligible.

To prevent the consequences of loss of pump house event the Licensee of CNA I decided to add a 4th UK pump as a reserve component. This pump performs the function of ensuring the subsequent cooling of the plant eventually needed in beyond design basis conditions. The suggested proof of this system is one day per month for testing and one month per year of continuous operation, although it could be used in parallel with a standard UK pump at any time.

This new UK pump, which is currently in the process of laying pipes, is located in the service water system building (UPD) of CNA II, and it has an auxiliary system of water seal and associated filtration system. The power supply is medium voltage (6.6 KV) and assured command of the pump is made from CNA I. This pump adds diversity and redundancy to the original system.

Regarding the river level intake, this fourth pump located in the pump house of CNA II, will take water from a source one meter below the current UK pumps intake, giving greater scope of plant operation in case of droughts and 2 meters above the level intake in the existing pumps, this could provide water even if level of Paraná river is increasing and exceeding the height of the pump house of CNA I (5.17 m) and thus keep the plant in cold shutdown through the heat chain extraction QM / RR / UK.

The implementation of this modification is foreseen in 2013.

2.2.1.4. Loss of heat sinks coincident with SBO

In this scenario both heat sinks will be loss, which means both independent cooling alternatives via the primary system by the chain extraction QM / RR / UK and via the secondary system through the SHS. Simultaneously there is the secured power supply failure to the secured bars, which means both failures of the emergency supply DG CNA I and the failure of the interconnection system to DG CNA II. In this case, the steam release will not be controlled from the SG due to the unavailability of the SCC. The remaining alternative is to open the SG safety valves of steam and the remaining heat will be extracted in this way until the SG water reservoir is consumed. In the current design of the facility, this scenario would lead to depletion of inventory in about 30 minutes. After the inability of venting as a heat removal mechanism, a severe accident progression in the core will take place.

The behavior of the plant and systems involved in this case of SBO with failure of the SHS injection was analyzed. In such case, to the event SBO2 described above the failure of controlled venting is added. This scenario has been identified as SBO3 and will result from the failure of the DG of the SHS. The SG pressure increased in the secondary and safety valves opens and the heat is extracted in this way until the water inventory runs out.

The following boundary conditions are considered after depletion of the 24 V batteries:

- Triggering of a LOCA reactor protection signals (NZ52/NZ53) with cooling by 100 °C/h ramp via SHS unavailable.

- Triggering of fast boron injection (NZ51 signal). After injection, the system must be closed to prevent air entrance to primary system from the TB system tanks. Due to depletion of batteries of 24 V, the failure to close the valves is foreseen, after the corresponding signal to close valves (NZ68).
- The pressurizer safety valve (QD01S01) would fail open after the second on demand opening.

The total time analyzed is about 5 hours. For the simulation it was considered a depletion time of 15 minutes for 24 V batteries. This time is very conservative because from new studies it has been shown a longer duration.

During the transient, it is assumed that the pressurizer relief valve fails open and consequently is is filled. The plant is stabilized at high pressure of about 70 atm. The core is cooled via the SG through the mechanism of "reflux condensation" while there is water available in the SG. Due to the drop of water lever and the unavailability of SHS, the core cooling deteriorates rapidly. The fuel elements begin to fail shortly after 4 hours of transient. The core, according to the study, begins to deteriorate after 5 hours.

Impact on the fuel stored in pools

The estimations of consequences due to SBO event in the spent fuel elements (EECC) stored in pools took into account fuel management foreseen until 2015, and considering the total lack of refrigeration as a consequence of SBO. These results are also applicable to this case. It was estimated for the worst case that temperature of 100 °C is reached after 72 h. However, under this extreme condition it takes approximately 20 days until the onset of the uncovering of the EECC. This is intended to prevent using additional water injection in the long term.

The assessment on the integrity of the EECC under the conditions postulated for this event and being in refueling or exchange process within RM is not completed yet. However, it is foreseen that the event presumably would affect the continuity of the process as well cooling the EECC inside the RM. There is no intermediate cooling system available to the RM, taking into account the conditions stated in this case of loss of sinks and SBO.

Actions implemented or planned for improving the response of safety systems

<u>Mobile diesel generator equipment (MDG); additional water supply to the SGs and pools facilities</u>

It is expected to implement a MDG. Among other charges will provide power supply to the pump restocking water to the SG. This pump must be able to inject water when the pressure in the secondary side of the SG drops up to few atmospheres. The water supply will be from auxiliary pool existing in the system (Figure 2-6). Also, this MDG will feed two pumps that take water from and alternative water reservoir and ensure replenishment flow to the spent fuel element pools.

Triggering of reactor protection signals is foreseen about an hour after start the event, including the signal to TB system actuation. Handling proposed in this scenario is similar to the case previously discuss SBO1, it means the implementation of an operation procedure to apply properly cooling ramp of 100 °C/h using release from the secondary system (manual action in this case). Additionally, this procedure requires manual closing of TB system, as explained above, to ensure the thermo-syphon in the primary system.

Strategy of SHS tank refilling is under planning using water from the reservoir basins UA00B03/B04 and injecting to the depressurized steam generator SG, using the UA10D20 and D21 pumps and replenishment with well water, using one of the water supply system (UJ) pumps. To prevent an operational error valves to be placed in the pipes must have a permanent lock system that prevents opening.

The proposal also includes the possibility of feeding the components involved through an external diesel generator mobile in case of SBO. This MDG may be connected in different parts of the installation. Must be capable of feed the charges of greatest demand foreseen in these accidental scenarios. This ensures its usefulness in lower demand. Preliminary, based on the analysis performed so far, the MDG could feed at least the following components to maintain the safety shutdown of the reactor:

- A TA4 pump and valves for the entrance to the primary.
- Pump RL33D0, which is commonly used for filling the SHS tanks.

- Control valves and controlled venting of the SG, currently secure power supplied by the SHS diesels. This feature will allow an alternative to the cooling ramp of 100 °C/h.
- UJ01D01 pump and an additional pump witch can be connected to a SHS branch allowing core cooling by a SG and valves belonging to that branch.
- A pump that takes water from the well water source and feed the spent fuel pools. This would ensure water inventory inside the pools beyond 72 hours and for all of scenarios discuss here.
- Pumps UA10D20 and UA10D21 that can inject water into the SG's from the UA00B03/B04 auxiliary pools.

The implementation of this MDG is scheduled to 2013.

2.2.2. ATUCHA II NUCLEAR POWER PLANT (CNA II)

Power Supply Systems of CNA II

External power supply

The sources of AC power outside the CNA II come from:

- Station of 500 kV (*Figure 2-7*),
- Station of 220/132 kV (*Figure 2-7*),
- CNA-I – CNA-II connection (*Figure 2-8*).

The aforementioned stations are located in the proximity of both NPPs CNA-II and CNA-I.

<u>500 kV Station</u>

500 kV station, has a link with two 500 kV overhead lines, supported by metal towers, and the prevision of a future third line.

Both the line to Ramallo station and Rodriguez station are part of the National Interconnected System (SIA), the operation and maintenance is done by the electric power transport company named Transener. Managing of the generation and transport is currently held by the company Cammesa.

The 500 kV line to Ramallo station, can be powered from the north, from large hydroelectric plants Yaciretá and Salto Grande (SGRA), and from connections to networks in neighboring countries Brazil, Paraguay and Uruguay.

The 500 kV line to Rodriguez station can receive energy from the south, from large hydro-electricity: Alicurá, Piedra del Aguila, Pichi Picún Leufú Chocón and Planicie Banderita. The 500 kV station is the type of double bus bar, working outdoors, with setting of one and a half switch, the link to CNA II is via a 500 kV overhead line of 300 meters with a double switch.

<u>Atucha Station of 220/132kV</u>

The station is the type of simple bar at 220 kV and 132 kV and is linked through a 132kV line supported by concrete posts and two 220 kV lines, supported by metal towers. The bars of 220 kV and 132 kV are connected via an autotransformer of 150 MVA.

The 132 kV line is connected to Zárate station and by this it can receive input power from 132 kV networks.

The two 220 kV lines are connected to Villa Lia station and through any of these lines may receive input power from the 220 kV network. The station is working outdoors and is connected to CNA-II by three single underground cables.

The ability of each of the lines feeding 500 kV stations as those of 220/132 kV, is sufficient to cover the energy requirements of CNA II during a safe shutdown of the reactor, removing the residual heat and preventing release of radioactivity in accident conditions with the highest demand for electricity.

<u>CNA I – CNA II Connection</u>

An additional source of external energy for CNA II, with limitations, is constituted by the possibility of connection of the BBB and BBD bars to CNA I bars BA and BB (see *Figure 2-8*).

The manual connection of two normal bars of CNA II to the normal bars of CNA I allows a power inlet, in case of unavailability of feeding on the side of 500 kV and a failure of alternative power to the CNA II from the Atucha station of 220/132 kV.

The power available in CNA I to transfer to CNA II, depends on the consumption of CNA I at the time, and if the hydraulic generator of CNA I is in service.

One can assume an availability of at least 1 MVA for CNA II in the most unfavorable conditions, that is, CNA I generating full power and the hydraulic generator out of service.

Plant Main Generator

It is the MKA01 main generator of 838 MVA and 21 kV.

Backup power supplies feeding the safety bars (diesel generators - DG)

The power required by some loads important for safety during the safe shutdown of the reactor, for example for the residual heat removal during operation under accident conditions or failures resulting from any normal supply system, are supplied by the emergency power supply system (DG).

This system is designed with four redundant independent trains, physically separate, and each one capable of supplying 50% of the power required performing the minimum required safety functions. These DG (XKA10/20/30/40) fed four buses of 6.6 kV (BDA, BDB, BDC, and BDD), *Figure 2-8*.

Other Alternative backup sources

Energy sources mentioned above correspond to the design bases of CNA II, now improvements that are expected to be implemented under conditions beyond the design basis are mentioned, in order to enhance the possibilities for power supply, through procedures and as part of accident management, allowing the removal of residual heat from the reactor and spent fuel pools in the long term:

Alternative cooling system for two emergencies DG of CNA II

Given the simultaneous unavailability of both pumps houses, the system of guaranteed river water (PE) located on the coast of Paraná River, according to the current design, the main generator of CNA II and the cooling of all engines of the emergency DG of CNA II will be loss, resulting in a total loss of emergency diesel system.

To address the above scenario, it is expected to adapt the current system of cooling by two CNA II diesel generators, through cooling towers of forced draft with the purpose of having an alternative cooling system, to have input generation of two emergencies DG (XKA20 XKA30).

The improvement includes review and modification of automatic controls and interlocks, so that by manual actions, determined in the program of severe accident management; if it will be possible to enable the operation of cooling towers and connecting the charges necessary for the extraction of heat from the core and the spent fuel pool.

The alignment of the alternative cooling system of the DG will be effected by closing and opening manual valves, whether they have a motor drive or not. This will form part of an emergency operating procedure.

The availability of power to be delivered by DG is conditioned by the cooling capacity of cooling towers, which is about 5 MW.

These improvements are expected to be implemented by 2015.

Connecting new diesel of CNA I to CNA II

Taking advantage that the new EPS of CNA I will have 3 DG of 100% and 3.4 MW each, and that they will be air cooled, it will be implemented the possibility, through manual actions, that they provide energy to CNA II, in case of unavailability of all emergency DG of CNA II.

For this system to fulfill the purpose described, it is necessary to modify the automatic control and interlocking of the new EPS system of CNA I, to allow in the program of severe accident management of CNA II, the manual interconnection in CNA I of the emergency bars to normal bars.

Taking advantage of the existing interconnection between normal bars of CNA-I and CNA-II, the CNA-I DG could feed up to two bars of the normal network of 6.6 kV and the four bars of the 6.6 kV emergency network of CNA II.

The availability of power to be delivered by DG of CNA I to CNA II, is subject to the conditions of CNA I, from a minimum value of 3.4 MW to a maximum of 6.8 MW.

These improvements are expected to be implemented by 2015.

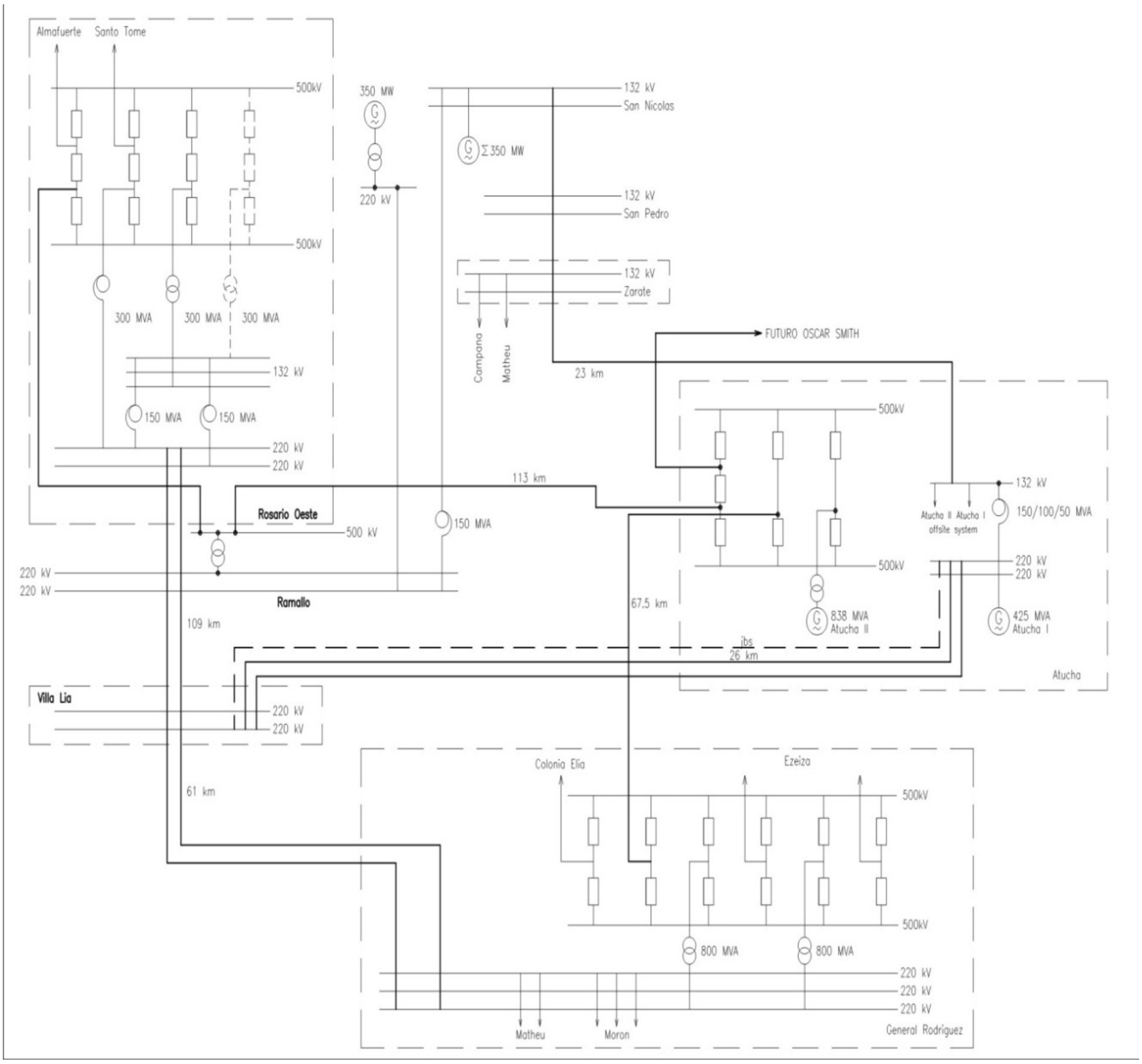


Figure 2-7: Interconnection to external networks of CNA II

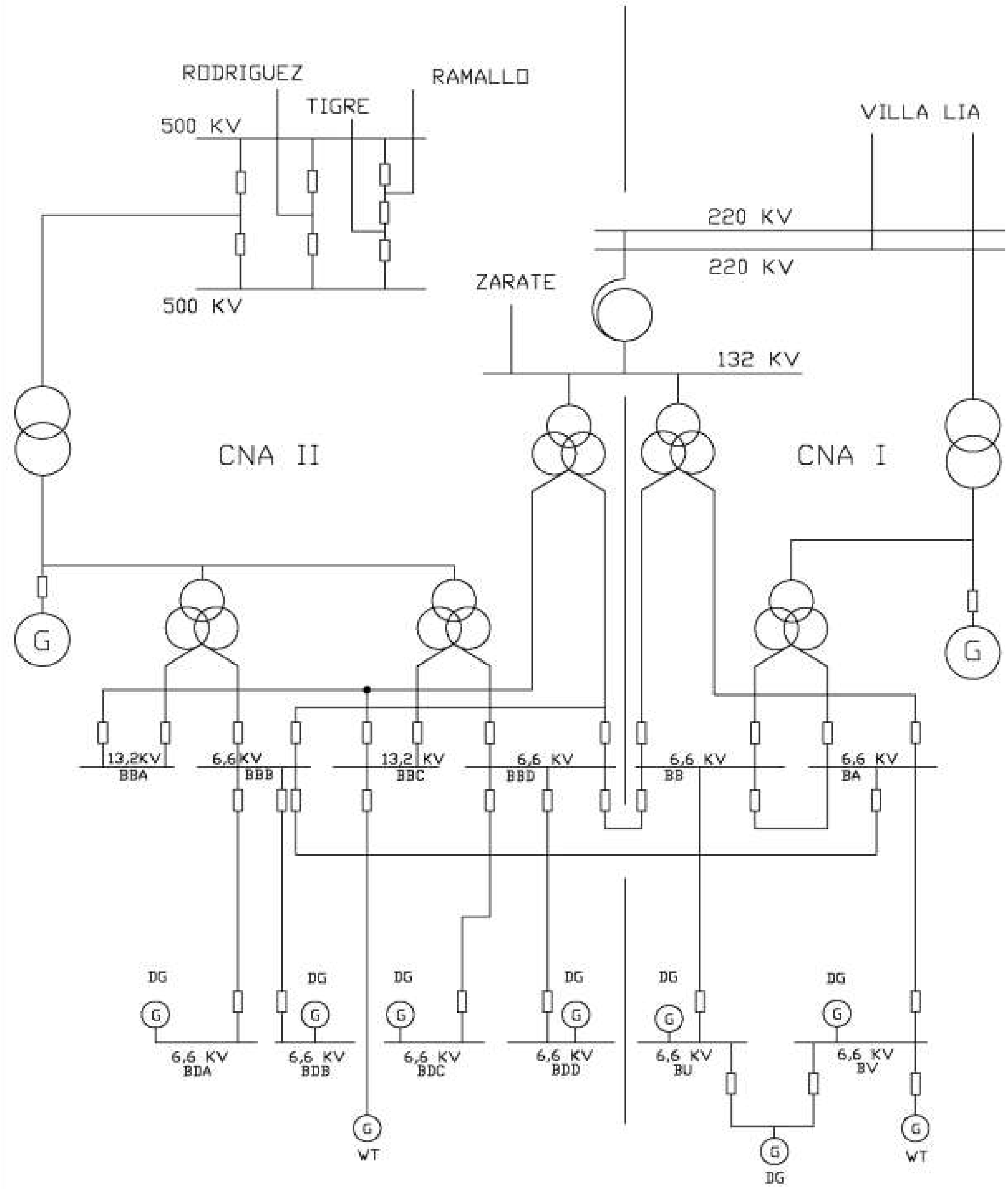


Figure 2-8: Emergency Systems Interconnection between CNA-I and CNA-II

Provision for connecting mobile 6.6 KV DG

The building of emergency DG of CNA II (UBP) is linked to UBA maneuvers building through four tunnels, each in correspondence with the bar of emergency medium voltage fed into the UBA building. In these tunnels there is an "emergency exit" that communicates with the street level between the UBA and UBP buildings. Three cables will be mounted from the entrance of each "emergency exit" to a switch on the emergency bar. In the entrance of each emergency exit, a facility for connecting a mobile diesel generator (MDG) of 6.6 kV will be mounted.

2.2.2.1. Loss of Off-Site Power (LOOP)

Assuming that the plant is generating at full power, if the two 500 kV power supply lines are loss, the load-break relay detects the load rejection, reducing the reactor power and the turbine, keeping the turbine and generator in service, supplying the needs of CNA II own consumption until it is decided to stop the generator. At design conditions, the excess reactivity of the reactor allows to operate the reactor at 80% power, evacuating the excess steam to the condenser.

If load rejection is not successful, or the line linking transformers block with the 500 kV power is out of service, feeding to the normal bars of the plant from the bars of the generator of 21 kV is loss. The lack of tension in the bar of 21 kV, after 500 ms starts the process of switching the four normal bars BBA/B/C/D, opening the switches which connect the transformer BBT01/02 and closing the switches of attachment to reserve network transformer BCT restoring power supply to the medium voltage normal bars.

Power of the BCT is sufficient to maintain the reactor subcritical, with removal of residual heat from the reactor and spent fuel pool.

In case of simultaneous loss of electrical supply lines of 132 kV and 220 kV that arrive at the corresponding station, and assuming that the CNA I is in service due to a successful load rejection, it is feasible to maintain the normal feeding of the bars of CNA II, through the backup transformer until CNA I is out of service, a circumstance that, due to poisoning of the CNA I reactor by Xenon, can occur in about 20 minutes.

In a situation of CNA I out of service, or BCT transformer failure, the normal and emergency bars of CNA II will be without tension.

After 2 seconds in this situation, the performance of the reactor protection system will initiate the opening of the switches that link emergency bars BDA/B/C/D with the bars of own consumption BBB/BBD and disconnection of all emergency system loads with a short interruption and the simultaneous startup of the four emergencies DG XKA10/20/30/40.

After a startup time of approximately 10 seconds, the switches of the DG of each train will be closed and the loads will re-connect in groups according to a preset sequence. The drive signals and release for starting the DG and the load program are generated in the reactor protection system.

The capacity of each DG is designed so that the emergency power needed to mitigate the design basis accidents can be supplied by two of the four DG (50% capacity).

The capacity of fuel stored in fuel tanks assigned to each emergency DG group, in regular and reserve tank, is sufficient for 85 hours of continuous operation, considering the accident with the highest demand for electricity. As to the reserve coolant water and lubricating oil necessary, has been estimated a range of 110 hours for each of the four DG.

Given that the operation of two emergency DG is sufficient to maintain the reactor subcritical, with removal of residual heat from the reactor and spent fuel in the pool, both in normal and design basis accidents, it is possible to extend the energy supply of two groups in about 67 hours, stopping two DG groups, and transferring the fuel from storage tanks of stopped groups into the tanks of groups in operation. This way it is possible to keep the power supply with the fuel stored in the building of emergency DG UBP for 139 hours. Following the implementation of the optimization procedure and mobile capacities to transfer between storage tanks, it is expected to increase the operating time of DG groups over 65 hours.

The transfer of fuel from the fuel storage tank will provide with another 122 additional hours of autonomy.

Consequently, the fuel reserve to cope with a prolonged emergency power failure could be prolonged by the actions of the operator up to a total of more than 272 hours of autonomy.

As for the residual heat removal, in this case there will be alternatives via primary and moderator, moderator system loops (JF), intermediate cooling system (KAG) and secured cooling system of service water (PE). Alternatively and for a limited time of water supply to the SG, secondary side venting will be available, with replacement of water by the start and stop system LAH and associated pumps LAJ, *Figure 2-9*.

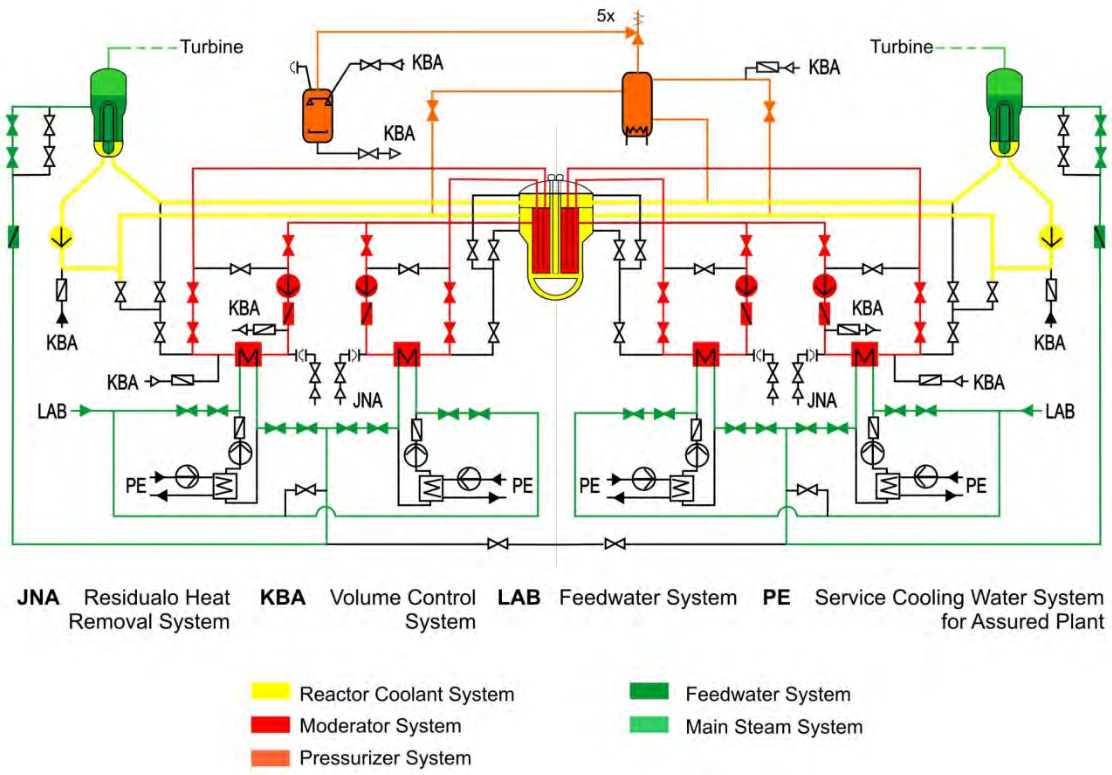


Figure 2-9: Main cooling systems, moderator and residual heat removal of the CNA II.

The actuation of the containment isolation, under the conditions postulated for this event will not be affected because it is insured by the uninterruptible power supply.

As to the effect that this event would have on the refueling machine (RM), if it occurs when it is being used, it was estimated that the cooling of the fuel element is ensured. The power for the RM is assured, allowing the completion of the replacement and / or transfer.

Among the major improvements planned in connection with the provision of additional power, with an impact on this event, are the following:

- Review of procedures to extend the use of DG using additional fuel tanks: The maintenance and testing programs will be reviewed, including verification of fuel tank level, water and lubricants. It must be ensured that the calculations presented, as to the minimum necessary provisions are maintained by the system of inspections and tests. This will be implemented in late 2013.
- An evaluation of the possibility to connect with CNA II one of the three new DG of CNA I (EPS, 3.4 MW each). Having such a connection will allow the following options for residual heat removal:
  - Primary side: main cooling loop (RHR) including auxiliary components, and/or
  - Secondary side with residual heat removal by SG connecting to a start and stop pump with the corresponding relief valves station. In this case, replenishment of water for maintaining the long term cooling must be analysed.

  This improvement will be implemented by 2015.
- Keeping current cooling towers as an alternative way of cooling two of the CNA II diesels. This improvement will be implemented by 2015.
- Turning off unnecessary loads to increase battery life. This improvement will be implemented by 2015.
- Analysis of the availability of external power supply lines including high voltage interconnection lines of 220 kV and 500 kV. This improvement will be implemented by 2013.

2.2.2.2. Station Black Out (SBO)

This scenario postulates that in addition to the LOOP event described above, the 4 DG groups of the emergency system of CNA II are unavailable. In this case, it is possible to manually connect the normal bars of 6.6 kV BBB and BBD of CNA II to the normal bars BA and BB of CNA I, restoring the power supply in CNA II.

While this power supply has limitations, as stated in above, it is sufficient to maintain the CNA II reactor cooling by steam venting. Additionally, enough power for the cooling of the spent fuel pool is ensured, because related systems are providing by the above bars.

If there is no voltage in the normal bars of CNA I, or the availability of energy is insufficient, it is possible to restore power to the emergency power system (EPS) of CNA II by the DG of the EPS system of CNA I.

In the sequence of events described, it is conservatively estimated that at least two trains of the emergency system of CNA II will reset the power supply, in a maximum time of two hours, allowing to resume operation of the rectifier equipment that supply power to the battery +24 V, -24 V and 220 V before they run out of charge. This is based on operational experience gained in CNA I.

At a minimum, the operation of the instrumentation and control of three trains are insured for functions for monitoring the status of the plant, based on the supply from the batteries.

Given the unavailability of sources of support and the mentioned measures to alternative feeds, there will be a MDG equipment of 6.6 kV with its fuel tank, as described above. The time for moving the generator set to CNA II, from the initiating event is estimated in a maximum of about 30 hours. This available time has been estimated based on the deterministic study of the transient that is generated as a result of the postulated event.

Considering the loss of all external supply alternatives, i.e., LOOP, loss of backup power supplies and loss of any alternative power supply, the only power source available will be the +24 V and -24 V and 220 V batteries, organized in four redundancies for safety functions.

The availability of AC power in 380/220 V is given by four groups of rotary converters (one for redundancy) formed by a 220 VDC motor and a generator self-excited AC (motor-generator). The four motor-generator groups are physically separated. There is a fifth group in standby, which can be aligned with any of the redundancies replacing equipment that might eventually need maintenance or in case of failure.

According to the specifications the battery life is as follows:

- +24 V, autonomy of 4 hours 24 minutes.
- -24 V, t autonomy of 4 hours 36 minutes.
- 220 V, autonomy of 3 hours 45 minutes.

It should be noted that even while the batteries are organized by redundancies, part of the loads that they feed, have double power from batteries of different redundancies, decoupled by diodes. This design allows the redundancy that has more loads to decrease faster its tension and its contribution to loads with dual feed, and then these loads will be powered by batteries that have less demand and maintain a higher voltage.

Values for autonomy of the batteries have been calculated assuming that the current demand is the design one, based on the discharge curves provided by the manufacturer.

By load disconnection procedures from 220 V batteries to be included within the severe incident management procedures potential consumers have been identified, which could lengthen the time to discharge the batteries. In a first approximation, it is estimated that the 220 V battery lives could be extended up to 7.5 hours.

It is planned to conduct a more detailed study, at the end of the start-up, to estimate reductions by the load disconnections in the +24 V - 24 V systems.

The loss of electric power for emergency own consumption of the plant generates the unavailability of the following heat removal schemes:

- SG - condenser
- Pumps and moderator heat exchanger (JF) - intermediate system of residual heat removal (KAG) - ensured cooling system (PE).

Consequently, assuming that the power failure occurs in circumstances where the level of the SG is normal (12.2 m), and the system of venting to the atmosphere is operating, the plant can be refrigerated for a limited time estimated in half an hour until the inventory gets exhausted.

From the viewpoint of cooling the core in the event SBO, it is estimated that the function of feeding the SGs would be loss in the absence of power supply (only batteries are available). If no action is taken, this scenario will lead to a situation of core damage under high pressure.

With respect to plant performance in case of SBO, there are several differences between typical PWR reactors with enriched uranium and the CNA II PHWR reactor with natural uranium, which are detailed below:

- The CNA II primary system for heat transport consists of two circuits, while the moderator system comprises four circuits. Both systems are interconnected so that the heavy water flows continuously between one and another circuit. The portion of the mass of heavy water for the moderator and coolant are similar (approximately 230 ton and 205 ton respectively). For reasons of optimizing the fuel burnup, while the power plant operates, the temperature of the moderator (average 170°C) is maintained below the temperature of the coolant circuit (average 295°C). To achieve this temperature difference, the moderator is cooled by heat exchangers and the heat exchange rate between the two circuits is kept low. In the case of events (e.g. LOCAs), moderator circuits change the configuration of the flow of moderator heavy water, in order to increase the flow exchange. Thus the relatively cold moderator water that is available, still at high pressures, is employed to cool the core. In the particular case of SBO event, there is no power to both moderator pumps and primary pumps and then the flow rate of exchange is almost zero (since they would not work). However, the more than 230 ton of moderator heavy water constitute a heat sink, through heat conduction to the moderator through the cooling channel walls. In this scenario core cooling is accomplished by heat transfer by radiation heat from the fuel to the walls of the channels and finally to moderator water, which acts as a sink.
- Through the use of natural uranium, the fuel burn-up in CNA II is significantly lower than the burn-up in reactors operating on enriched uranium, being up to 6 times lower. A direct consequence for safety analysis of this difference from the typical PWR is that the power of the medium-term decay heat is smaller for the CNA II core. This will be reflected primarily in regard to boundary conditions in the calculation for cooling the spent fuel pool, which in the case of CNA II are less stringent than in a typical PWR.

These two major differences of CNA II compared to a typical PWR make available more time to take manual action to dominate the SBO event. In the primary circuit there is a natural circulation of the coolant between the reactor and SGs. The decay heat causes boiling of the inventory of SGs whose content is vented to the atmosphere as vapor by 4 steam relief valves powered by batteries. Due to the features mentioned above, according to conservative estimates, there are about two hours before the water level in the pressurizer reach maximum values as a result of expansion and partial boiling of the primary.

Immediately after reactor shut down, in the primary circuit there is a natural circulation of the refrigerant between the reactor and SGs. The decay heat causes boiling of the inventory of SGs and venting to the atmosphere through four steam relief valves powered by batteries. After 0.75 hours, the valves are isolated when the SGs arrive at its minimum level of two meters.

From this moment, and assuming that no countermeasure was applied, there is no effective heat sink and heavy water will heat up and expand and then it is inevitable the subsequent opening of the pressurizer safety valves. Because these valves are not designed for flow of liquid refrigerant, it is desirable to take measures to cool the core before two hours and thus avoiding the possibility that the pressurizer safety valves can be locked open.

Taking into account what was stated in the preceding paragraph, studies were performed assuming two possible scenarios:

- A pressurizer safety valve would be locked open if liquid circulate: Under the conservative assumption that no action would be taken by operators and that the batteries would be discharged at two hours after the event, since there would be no input from the four accumulators of the high pressure passive injection system (light-water) to the primary, it is estimated that the first radionuclides of the fuel rods would be released into the primary after four hours after the event.

Among the improvements to be implemented to deal with this accidental scenario, it is foreseen to activate the passive accumulator injection of high pressure in the primary before the batteries run out, either by undertaking the action itself or by disconnecting unnecessary consumers to prolong the duration the battery charge. Calculations show that by this injection from the accumulators, core degradation would be delayed about five additional hours. Also worth mentioning that, although due to the increment of Xe in the fuel, it would be impossible to return the reactor to critical for a period of about 48 hours, additionally, the light water supplied by the accumulators is a neutron poison for PHWR reactors and the boron injection would not be essential for maintaining the reactor in a safe shutdown condition.

- The pressurizer safety valve would not be locked open by circulating liquid: In this scenario of high pressure, it is estimated that the first radionuclides would be released 8.5 hours after the event and then after 3.5 hours more, the connection line between the primary and the pressurizer (surge line) would fail -due to creep at high pressure and temperature-, leading the accident to a low pressure stage.

According to conservative calculations, it is estimated that due to the particular characteristics of the CNA II, there is sufficient time for the operator to take manual actions to avoid the successive opening of the pressurizer safety valve. Currently, it is being evaluated whether to implement an emergency procedure to be applied by plant operators. The analyzed alternative is to feed the SGs with demineralized water and cool the core through the steam vent to atmosphere.

To implement this strategy, preliminary studies in conservative conditions show that after 1.5 hours, the pressurizer level would reach a level of 9.5 m. Before this instance, operators should cause the depressurization of the secondary side of the SG via the main steam relief stations until it is fully emptied. By this way a partial cooling is achieved by utilizing the existing water content in the lines of main feedwater. Also before the two hours into the transient, a pump is provided to feed water (which feeds the SG). The reservation of this tank (max. 264 m^3 / minimum 148 m^3) will keep the reactor cooled within 8 to 10 hours. Then the reserves of demineralized water tanks could be used that, at most, amount to two tanks of 785 m^3 each. It should be noted that, at all times of plant operation, such tanks must have an assured supply of water of at least 2 x 280 m^3. This reserve ensures cooling for approximately 35 additional hours, bringing the total cooling time to two days.

Additionally, to facilitate the supply of electrical energy needed to maintain the DC power and equipment supplies, prolonging the operation time beyond the previously estimated, it is intended to implement a MDG. This equipment will allow extend the battery life facilitating the provision of water from an alternative reservoir, necessary to maintain feeding in the long term to the SGs and the cooling of the spent fuel pool.

The possibility of such alternative water reservoir could be formed by the water intakes from the groundwater is being evaluated. It is foreseen to implement this improvement by 2014.

Evaluation of the fuel elements in the pool

The Pool Cooling System (FAK) consists of three pumps (during normal operation two of them are running and the third is in stand-by), which are fed from the emergency water supply by three different trains. The system also has two heat exchangers to dissipate heat generated in the spent fuel pools to the secure service water cooling system (PE).

The pool cooling system and the spent fuel pools are in the building of pools (UFA).

If the pool cooling system or cooling secured water service is completely lost, it would produce an initial phase of water heating until saturation, followed by evaporation and level decrease. The times when these events would occur depend on the number of fuel elements deposited in the pools and their decay heat.

To evaluate these times it is considered that the following unfavorable conditions are present simultaneously:

- level of 16.14 m in pools (minimum),
- maximum possible number of fuel deposited in the pool,
- whole reactor core downloaded to the pool as a result of any operating condition that requires it.

This last condition is very unlikely, because the heavy water reactors perform the refueling in operation: As a result, the core discharge is not a systematic action as it is in PWR reactors and it is highly unlikely that an abnormal conditions or a maintenance operation would originate this operation.

The time sequence for a situation of total loss of cooling will be the following:

Time t [h]		Condition reached
(1) Fuel elements in pool + full reactor core downloaded	(2) Fuel elements in pool	
0 (loss of refrigeration)	0	Initial level 16,14 m, temperature (1) 47.5ºC and (2) 41°C (cooling with only one train in both cases)
54	90	Reaches boiling point
91	145	evaporation until 1 m of water, 15.14m in the pool (2,59 m of water above the level of fuel)
188	287	Upper part of the fuel gets uncovered (active part)

Analyses were performed on the worst case scenario (1) core evacuated and pools full of fuel elements. Even while this scenario is unlikely to happen, this way, it will lower time so it is possible to conservatively set times of further action to restore the cooling of the pools.

It is assumed conservatively that, before the event, only a single cooling train is available. Consequently, it is estimated that the temperature of the pool will be at 47.5 °C. The high temperature alarms (38 ºC) and very high temperature (41 °C) were initially present.

It is estimated that up to approximately 60 °C, it is possible to perform manual actions because existing conditions in the building of pools would not be still compromised. The estimated time to reach 60 °C would be about 13 hours; then, access to the building will be limited and entering will require the use of personal protective equipment.

Given that the design temperature of the spent fuel pools is 80 °C, the time without cooling is estimated at approximately 33.6 hours from the beginning of the loss of the cooling system.

It is considered highly unlikely that the active part of the fuel elements get uncovered, due to the extremely long times involved to reach this situation (188 hours as seen from above table), which would allow to restore the level and cooling in the pools.

As noted in previous sections and taking into account that the worst case scenario has very little chance of occurring, it is estimated that if it is not possible to cool the spent fuel pools (e.g. in case of SBO), given the large volume of water available and its associated thermal inertia there will be sufficient time to recover the cooling function. However, it is foreseen to add an extra set of filling water from a reservoir of alternative water and power from the MDG.

It is being evaluated the possibility of such alternative water reservoir to be formed by the water intakes from the groundwater. It is foreseen to implement this improvement by 2014.

The evaluation of the impact that this event would have on the refueling machine (RM) (if it occurs during the refueling process and affecting a fuel element in its interior) is under development.

2.2.2.3. Loss of heat sinks

In normal operation at full power, the heat produced by the reactor is transferred from the primary to the secondary system through the SGs, by the circulation of the flow driven by the main pumps (JEB). This heat is partially converted into electricity by the main turbine, with the remainder transferred to the Paraná River in the main condenser (MAN), to the main cooling system (PAB), with the water circulation system pumps (PAC).

Given the shutdown of the reactor, the residual heat is transferred to the secondary via the SG. If the main condenser maintains operation, the steam is transferred to it by the diversion station skipping the main turbine. Residual heat is removed in the condenser by the river water powered by the PAB *Figure 2-10*.

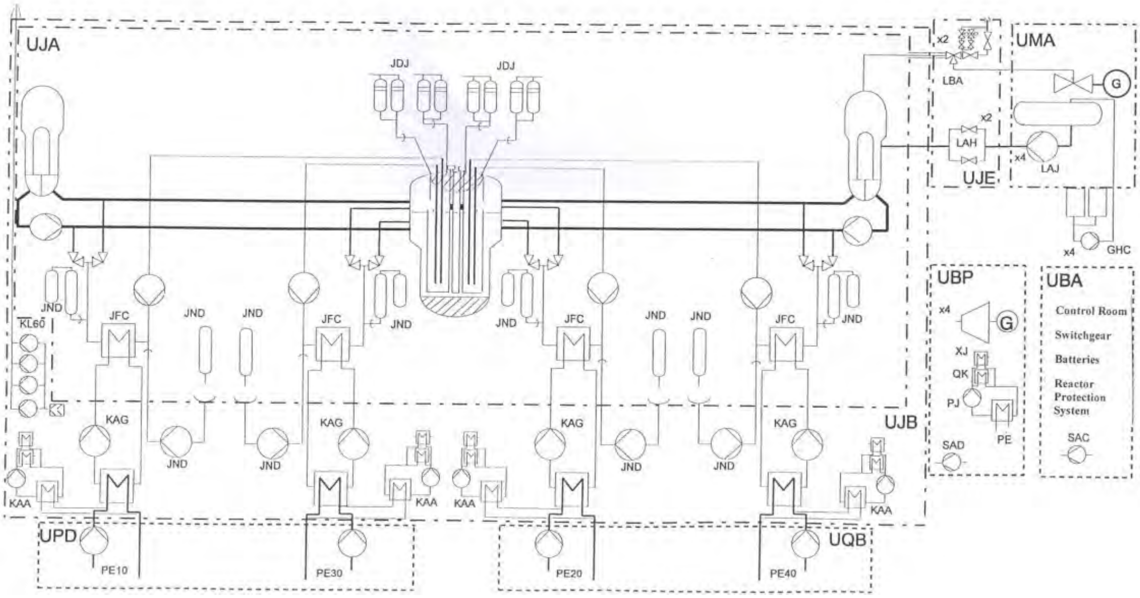


Figure 2-10: Cooling and residual heat removal systems

Loss of main condenser

The lack of external power supply or the failure of any of the services of the condenser (main cooling system, removal of non-condensable gases, etc.) disables the operation of the condenser.

In this condition the facility continues to be cooled by the SG, but the secondary circuit transfers heat to the atmosphere operating with an open circuit venting live steam to the atmosphere through four motorized valves with power provided by the insured bars of the non-interruptible system (powered ultimately by batteries), which only one single valve is capable of removing decay heat and power from the primary pump, if necessary.

If the origin of the unavailability of the condenser is the lack of external power, the feeding of the SG is no longer performed by the feed water system (normal power) being replaced by the startup and shutdown system (ensured power supply). There are four startup and shutdown pumps (LAJ), providing the water supply. A single pump is capable of extracting the decay heat and power from the primary pump, if necessary, *Figure 2-10.*

The available reserve of water to cool the core through the SGs is constituted by their initial inventory (120 m^3), plus the content of the feed water tank (maximum 264 m^3 / minimum 148 m^3) plus demineralized water reserve of the demineralized water supply system (GHC), 2 tanks of 785 m^3 each with secured reserve of 2 x 280 m^3.

It is estimated that in case of being employed this method of cooling by steam venting to the atmosphere, there would be a minimum autonomy of two days if the operator shut down SPTC pumps, which is not necessary in case of external power failure.

This is the mechanism that requires less diversity of functions. This simplicity and high redundancy which have implemented the necessary components and systems, makes this plant refrigeration highly reliable. This mechanism of heat removal can operate without external power supply as start-up and shut down pumps system (LAJ) are supplied from the emergency power system (CP).

After the two days mentioned above, for the purpose of saving available demineralized water reserves, it will be possible to manually activate the residual heat removal intermediate system (KAG) consisting of four independent loops, which only one loop is enough for removing decay heat and primary pumps power.

Unavailability of the steam generators

Lack of water supply or failure of the live steam venting station to the atmosphere, can disable the removal of heat through the SGs.

In this circumstance, as mentioned in the previous section, the facility has the KAG system. This system consists of four independent circuits and refrigerates the primary through the moderator exchanger, transferring the residual heat to the river water pumped by the secured cooling system (PE).

The control of the station for venting to atmosphere is executed by the reactor protection system (JR) that meets the demanding criteria of redundancy, train separation and electrical uncoupling. Moreover station for venting to the atmosphere is supplied by the uninterrupted emergency power system (batteries); in consequence, the venting mechanism remains available in the case of the occurrence of an event leading to the loss of the external and internal power supply (SBO).

Heat removal by: Moderator Exchanger - Intermediate Residual Heat Removal System - Insured Cooling System

This heat removal system comprises four independent and identical trains, two of them are required to effect the removal of residual heat at any operational or accident condition provided by the design of the installation. Note that most AOO events (Anticipated Operational Occurrence) can be offset by only one of the four trains available.

Since each train requires the operation of four pumps (moderator pump, recirculation pump, pressurization pump for the residual heat removal intermediate circuit and ensured cooling system pump) it is estimated that for SBO conditions (BDBA) the only possibility of using the mentioned heat removal chain, is to have the MDG mentioned above.

External events such as earthquakes, winds and tornadoes, pressure wave, chemical explosions, entering of explosive or toxic substances, electric lightning and flooding of the Parana river, have been considered in the design of buildings and structures where systems that make up the chain of heat removal are located.

All components that make up these systems are powered by the emergency power system; consequently, the cooling mechanism will remain available even if the plant is without external power supply.

All the concepts mentioned above, ensure a high availability of this mechanism of residual heat removal.

The ability to cool the plant at any accidental or incidental condition, even at the highest temperatures expected in the reactor, makes this design a unique and extraordinary resource. Typically, the PWRs lack this capacity; thereby cooling the reactor in the hot state requires the availability of the SGs.

Total loss of river water supply for cooling (loss of the pump house)

The loss of the ability to pump water to cool the installation from the Paraná River, results in the loss of the following schemes of heat removal:

- Steam Generator - Condenser
- Moderator heat exchanger - intermediate system of residual heat removal - ensured cooling system

If CNA II has external power supply, refrigeration of the installation will continue through the SG venting to the atmosphere. In this event, the loss of cooling of the DG is caused by the unavailability of emergency power. For this reason, electric power will be available only if it comes from other alternatives. Accordingly, in order to avoid the cooling loss of DG, it is foreseen to implement the above-mentioned improvement consisting in cooling towers to cool two of the DG.

As noted above, in this event the minimum autonomy is of at least two days (if there is no alternative external power supply to the CNA II). This is true taking into account the availability of water in the secondary system and system tanks LA, *Figure 2-10*.

In this scenario, it is planned to incorporate an improvement for extending the water supply to the SGs consisting in the already mentioned MDG and water supply from an alternative reservoir. The possibility that such a reservoir be constituted by taking water from the groundwater is being evaluated. The implementation of this improvement is foreseen by 2014.

2.2.2.4. Loss of heat sinks coincident with SBO

In this case, the loss of heat sinks described in the previous paragraph would be added to the loss of external power supplies and even emergency supply of DG.

In case that the plant does not have any power supply (SBO), since the lack of cooling water also disables secured emergency power generation, it is estimated that the time that the plant may be cooled would be determined by the initial inventory of the SGs and what was indicated above for the event SBO remain valid. That is, under these conditions this time is estimated in only 0.5 hours.

The projected improvement which would impact in this case is the incorporation of MDG with a water supply from the said reservoir which is alternative to the LA system, in order to ensure long-term venting at the SGs. In this case, the presented results corresponding to the event of SBO are applicable. That is, the MDG system with all the facilities of additional water supply is intended to cover the weaknesses of the SBO event described above.

As for the findings on the impact of this event in the fuel elements stored in the pool, in both the results presented and the planned improvements for SBO are the same as in this case.

2.2.3. EMBALSE NUCLEAR POWER PLANT (CNE)

Power Supply Systems of CNE

The power supplies are the following:

- Redundant external grids (500 kV and 132 kV), which provide electricity required during plant start-up and shutdown and may also provide power during normal operation.
- Turbine generator, which provides electrical power required during normal operation. Also, in case of loss of the external power supply, allows feeding operation for own consumption of the plant.
- Power sources in standby which provide the power required in cases of loss of normal power supply: Class III, four emergency DG of 50%, batteries, two emergency DG with 100% of capacity (EPS, Emergency Power Supply).

The internal distribution of energy system is divided into two groups of redundant loads (even and odd) so that the loss of one of the groups does not affect that the minimum functions can be fulfilled. Besides internal power supplies are divided into four classes ranging from uninterruptible power until other which can be interrupted with limited and acceptable impact, and are as follows:

- Class I: continuous current (DC) provides uninterruptible power supply to essential auxiliary, control, protection and safety equipment. The power supply batteries provide energy during about 8 hours.
- Class II: alternating current (AC) supplies not interruptible power to essential auxiliary, control, protection and safety equipment. The power is provided by batteries through inverters or Class III during unavailability of the inverters.
- Class III: power supply to safety-related systems. The normal supply of Class III is from the transformers, start-up transformers and the backup system are the emergency DG. There are four DG of 50% with 2.4 MWe each (two are sufficient to provide 100% of energy bars).
- Class IV: normal supply of AC to auxiliary equipment that can tolerate long interruptions without affecting nuclear safety, operating staff or safety equipment. A complete loss of both class IV bars (even and odd) produces the reactor shutdown. The loss of external power supply is a design basis accident (DBA) and the plant has backup safety systems to handle such events.

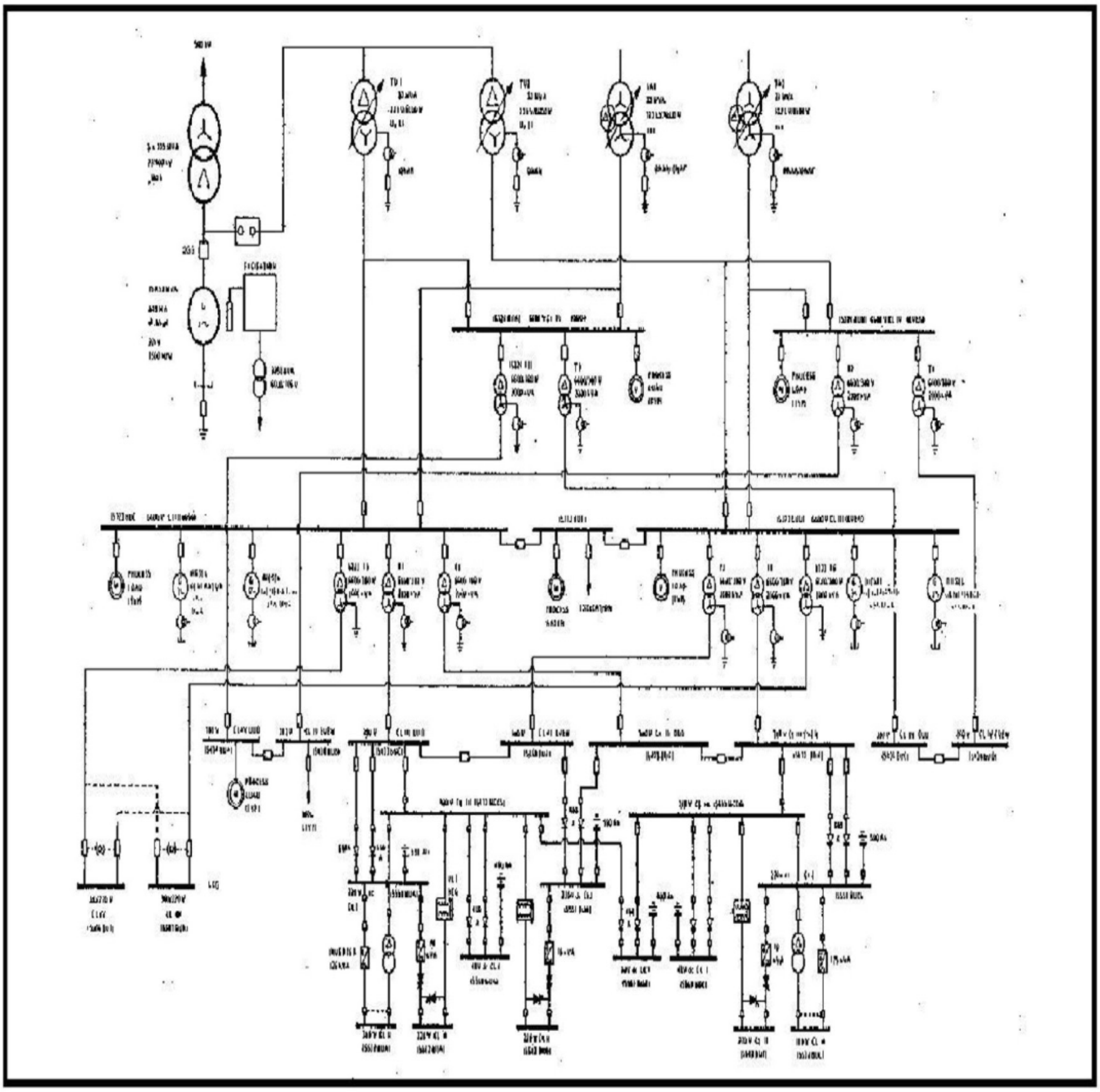


Figure 2-11: Basic diagram of the power supply system of the CNE

Recovery of the Argentine Interconnected System (SADI)

The company responsible for the electrical power distribution network in Argentina ("Compañía Administradora del Mercado Mayorista Sociedad Anónima"-CAMMESA) has a Technical Procedure - PT 07 - "Recovery of the SADI after total collapse" it is stated concerning supplies the priority to CNE for safety reasons.

Assured power supply to CNE

With the purpose of securing the auxiliary services of the CNE face to the failure of emergency DG, there will be a bus bar on the 132 kV transformers at Almafuerte station to be used as interconnection bus bar between Reolín generator (Hydraulic generator-company) and the 132 kV bus bar of CNE. If necessary, CNE would request to the Energy Company of Cordoba Province (EPEC) for this additional connection to the Reolin hydraulic generator. Thus, EPEC will request to Transener (the company responsible for power distribution through the external grid) the properly configuration at 132 kV bus bar in the Almafuerte transform station.

2.2.3.1. Loss of Off-Site Power – LOOP

This case is a basic event considered in the design bases (DBA). The CNE can operate at reduced power levels, feeding its own inputs. In case of triggering turbo-generator reactor or after loss of external energy, electric charges will not receive power from the turbo-generator. This event is called loss of Class IV in CNE (lines 500 kV and 132 kV) and it was considered as design base. The plant design ensures reactor shutdown and cooling under these conditions. The reactor shutdown state is secured by either redundant systems. Cooling is ensured by the ultimate alternative heat sink. In this case, long term cooling is performed by extracting heat from the heat exchangers of the shutdown cooling system, in the loops of the primary system to the service water system, being Embalse Lake the final heat sink (*Figure 2-12*).

Class IV loss is detected by the under voltage relays, which initiate the automatic startup of the DG Class III power supply which are connected to the bus bars affected in approximately 60 seconds. Power to the SG is reset to 4% of the nominal value. Class III also resets the feed water pump of the Pressure and Inventory Control System (SCPI) to the primary loops, so the SCPI then acts to replenish losses. After a few minutes in the Primary Heat Transport System (PHTS) is establishing a stable thermo-syphon cooling towards the SGs, which provides adequate cooling and decay heat is removed using the SGs as an intermediate sink. In the long term, with the objective of saving demineralized water, the operator can start the Shutdown Cooling System using service water and powered electrically from class III.

In the scenario presented, Class III power supply is providing power to safety-related systems. Taking into account that normal power is lost in the internal bus bars of Class III from all alternatives of Class IV, the DG system is the ultimate provision in this case. The time interruption of power to these bars will be of short duration (maximum 180 s), which are necessary for the DG system startup and get charge. Class III is also used as class I charger and backup for the loads supplied by batteries of class II.

Each DG has a daily fuel tank of 20 m^3 available and a consumption of approximately 17 m^3 per 24 hours. Also there is a reservoir of 200 m^3 of fuel to feed the daily tanks. Therefore it has autonomy of approximately 7 days of operation of two DG at full loads demanded. If necessary, the daily tank of 20 m^3 of the auxiliary boiler system would be used as an additional feeding.

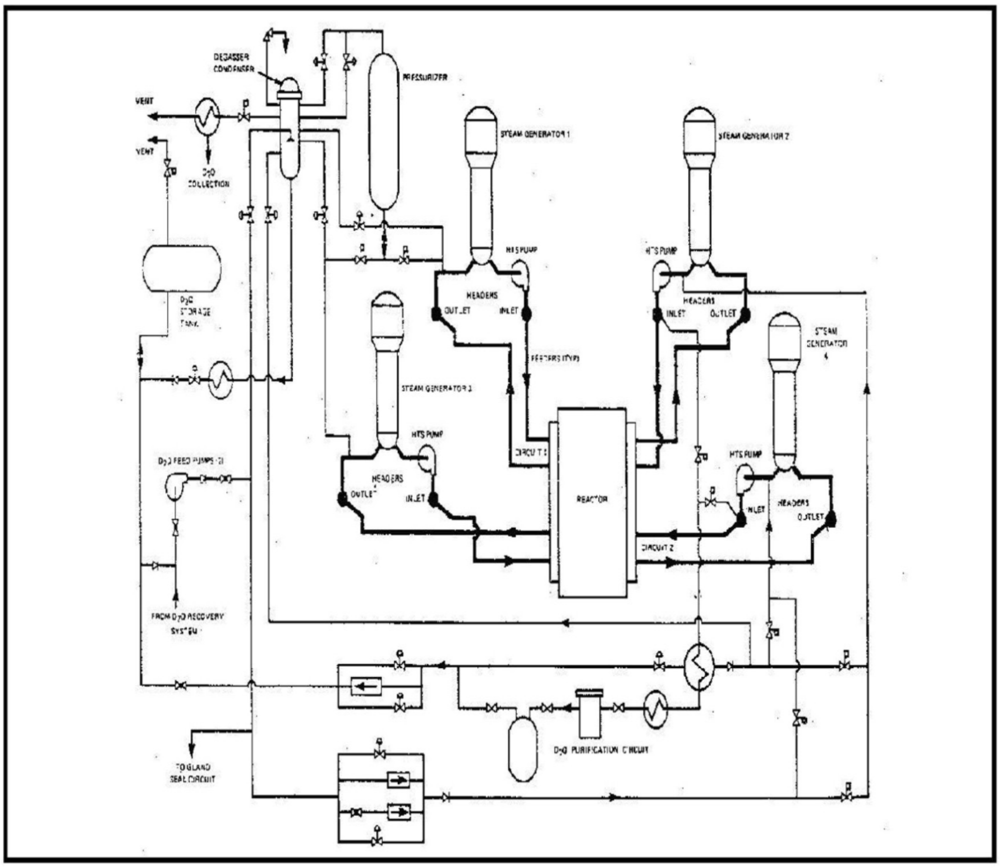


Figure 2-12: CNE main cooling system and emergency core cooling system.

In the Operational Manual is specified that when the tank has 140 m^3 of diesel fuel, it shall be reset to complete the 190 m^3.

For daily fuel tanks of the emergency diesel generators DG of Class III, there is a low level alarm in the control room, but also in this case the assistant of the operation team is recording each level shift on his routine inspection. The refill of fuel tanks is made before they reach alarm level. Also, when some periodical testing is finish the refilling of the tanks is performed.

DIESEL GENERATOR SYSTEM –DG- CL III		
PROCEDURE N°	**NAME OF THE TEST**	**FREQUENCY**
PR 52100 02	DIESEL TEST LOAD OF CLASS III	Twice a month
PR 52100 04	SEQUENCER LOAD TEST DIESEL CLASS III	Weekly
PR 52100 05	CHECK OF CLASS IV LOST IN BARS CLASS III	Monthly
PR 52100 06	TEST OF LOST CLASS IV IN BARS CLASS III	1/1,5 per year
PR 52100 07	TEST OF REJECTION LOST LOAD AND STATISM VERIFICATION	1,5 per year
PR 52100 09	TEST OF TKs AIR BLEED	Monthly
PR 52100 10	CAPACITY RUN TEST OF DIESEL	Twice a year

Table 2-1: Periodical Tests related with DG system.

The containment isolation, under the conditions postulated for this event will not be affected because it is secured by the uninterruptible power supply system.

Impact on the fuel stored in pools

In this case, the emergency power supply is available. This system feeds the cooling systems, instrumentation and control of temperature, level, etc. The event in consideration not reduces capability of these systems because power supply is ensured. Therefore, the spent fuel pool cooling is not affected under postulated conditions of this event.

The design provides three pumps, each with a capacity of recirculating 50% of the maximum required capacity of the storage pool. The pumps are arranged so as to permit operational flexibility. Normally two pumps are used for circulation in the storage pool. Initial cooling load requires only one pump to maintain the pool water temperature below 32 °C. However, approximately from the year after starting with the stable fuel exchange, it is necessary to operate both pumps to maintain the temperature of the pool water below 32 °C. The failure of a pump in this stage does not causes temperature increase above 50 °C, which is the maximum temperature used for the structural design of the pool.

In case of Class IV power loss prolonged, a pump can be connected to the supply of Class III to keep the cooling water storage basin, ensuring long-term cooling.

As to the effect that this event would have on the refuelling machine (RM) and fuel elements EECC inside, if the event occurs during the refuelling process, it was estimated that the cooling is ensured. The power supply of the RM is assured, allowing the completion of the EECC replacement and / or transfer to the pool.

Actions implemented or planned for improving the response of safety systems

During the plant shutdown for life extension or refurbishment tasks implementation (2014/2016) some improvements related with the power supply systems will be implemented. Some of them are the following:

- Protection of the 500 kV station, protection of bus bar, protection of line, circuit breaker failure protection (PFI), these elements of protection are in baying process.
- EPS: a new EPS system with greater power will be installing. The projected power is 1000 kVA (two equipment of 100% each one of them).
- DG of Class III: is scheduled for replacement, and the building where the system is located at present will be modified. Other enclosures will be added. The automation system for DG equipment will be separated in different locations; diesel one and three in one enclosure and into another will install the equipment automation of diesel two and four.

- It will change the physical location of the centre of the plant lighting system (light centre), since in the current location, in case of loss of water from the condenser would produce flooding and outage of the plant lighting system.

2.2.3.2. Loss of external and internal power supplies (SBO)

In case of occurrence of SBO event the plant shutdown is automatically ensured by any or both reactor shutdown systems SDS1 and SDS2. That is through the tripping parameters for shutdown by control rods dropping (SDS1) and/or the injection of gadolinium in the moderator SDS2 Both systems are designed to fail-safe, so that if no power supply the systems perform their function of shutdown the reactor.

The boundary condition of this event establishes unavailability of the DG for emergency power supply. I.e., there is no residual heat removal system during shutdown by the cooling service water system. In this case the cooling of the reactor is ensured by the thermo siphon effect produced in the PHTS. The heat will be transferred from the SG and discharged to atmosphere through the steam release valves (ASDV) to atmosphere. The water required for the SG will be supplied by gravity fed from the dousing tank located in the roof of the containment enclosure. The available dousing inventory is enough for cooling the reactor during at least 23 hours even under the assumption that no operator action is taken for controlling the supply flow from the dousing reservoir.

The makeup water towards the dousing and/or the SGs will be provided by the emergency water supply (EWS) system pumps which have its own diesel engines and are design with an independent water intake from Embalse Lake.

With operator actions on the valves (for instance, lock open one valve MSSV) and controlling valves that connect the ECCS and the EWS with the dousing, this inventory can be extended to 7 days. For this scenario, it is developing a specific operating procedure for abnormal events (POEA).

The system has earthquake qualification and is periodically tested. Additionally to normal maintenance and testing, during plant scheduled outages the core cooling is done using the EWS system which replaces the service water system during few days. The equipment has a fuel tank that allows autonomy of 10 days under full load.

Regarding the containment isolation valves, they will fail closed due either a power supply failure or the air supply failure. Thus, considering the SBO the containment isolation function will not be affected.

Monitoring of critical safety parameters will be assured using energy from the batteries. The batteries can provide power continuously for eight hours. Moreover, while in the scenario presented, the instrumentation is lost after batteries depletion, is foreseen an improvement with the ability for charging battery rectifiers, through a MDG.

Impact on the fuel stored in pools

Following the evaluation of the impact of this event on the fuel stored in pools, the cooling system will be lost. Consequently there will be continuous heating of water up to the boiling temperature. It has been estimated that under such conditions of SBO water temperature is lower than 100 °C at least during 72 hours (considering the pool with the EECC in unfavorable conditions is estimated that this temperature is reached in 3 days and 7 hours). However, the uncovering of the EECC would begin only after 13 days, which is sufficient time for countermeasures. These actions are taken into account in the procedure that is being prepared by the Licensee. Moreover, the problem is only residual heat removal from the spent fuel elements because of in natural uranium reactors there are no re-criticality conditions which may generate different heating conditions in presence of light water.

Regarding improvements in refrigeration systems, it is foreseen replacement of water by a pump connection from outside of the auxiliary pool building. This improvement would cover the requirements for the case of SBO and loss of heat sinks. Power supply is provided by the same strategy of MDG, to be used in the above improvements regarding water replenishment of the dousing tank. It is planned to implement before the end of 2012.

With reference to I&C, shall be installed measurements of the storage pool level and temperature parameters (regardless of which there are currently in the main control room) in the secondary control room, also with indication in main control room and electrically powered from the EPS. It is planned to implement before the end of 2012.

As for passive systems, inspection procedures were reviewed and preventive improvements were identified. These improvements have impact on the response of the related safety systems to extreme events. An important point is related to the control of passive components, such as vacuum break pipes / siphon of the spent EECC pool. Improvement of the procedures and frequency increase of testing and inspection, and it is planned to implement before the end of 2012.

The evaluation of the impact that this event would have on the refueling machine (RM), if it happens during the fuel exchange process, is under development by the Licensee.

Actions implemented or planned for improving the response of safety systems

Mobile Diesel Generator- MDG.

It will be installed a MDG of 550 kVA of power feeding the following loads considered essential in this event:

- The bus bars in the EPS system after the unavailability of two generators (100%).
- One out of two fire-fighting system pumps.
- Rectifiers for battery recharging energy bus bars of Class I.

This improvement will be implemented by 2015.

Improvements in spent fuel pool

- A facility will be installed for connection of one motorized pump providing replacement water from outside building pool. This facility will be useful in case of loss of cooling, loss of flow circulation or SBO event. It is planned to implement before the end of 2012.
- It will be written an abnormal event procedure properly to use in case of loss of cooling and/or loss of inventory of the spent fuel pool and related systems. This procedure shall include actions to monitoring parameters as coolant level and temperature from the secondary control room during a hypothetical event with the main control room and the pool enclosing inaccessible. They will include actions to replacement of water from alternative systems (for instance, from fire-fighting system or fire engine) in case of prolonged loss of cooling or inventory shrinkage. It is planned to implement before the end of 2012.
- It will be installed level and temperature measurements of the spent fuel pool parameters (regardless of which there are currently in the main control room) in the secondary control room, also with indication in main control room and electrically powered from the Emergency Power System (EPS). It is planned to implement by 2013.
- It was added in the Operational Manual "properly function of vacuum/siphon break pipes of the fuel pools" shall be controlled one time per operation shift.

2.2.3.3. Loss of heat sinks

In this scenario the simultaneous loss of service water system and EWS is considered. Under these conditions, there would be external power supply, ensuring power supply to the required systems. In this case the reactor shutdown can be activated manually from the main control room. If it is not activated manually, the automatic shutdown is guaranteed by process parameters (for instance, high temperature of the moderator, etc.).

In the first phase of the event reactor cooling is ensured by the provision of emergency water from the dousing to the SG. In this case, reactor cooling is ensured similarly to the case of SBO or loss of the ultimate heat sink or loss of service water system. But, in this case the EWS is also unavailable and SG inventory would be fed from the dousing tank because the MSSV is open. This provision of refill water from the dousing reservoir was estimated sufficient for at least 23 hours.

Taken into account that this event is not considering the loss of power supply (even the external grid would be available), the containment isolation and monitoring of critical safety parameters shall not be directly affected or degraded as consequence of this event.

Impact on the fuel stored in pools

The evaluation results about impact of SBO on spent fuel storage in pools described above are also applicable to this event. Thus, the expected behaviour and water temperature and the time available for this case are equivalent to the previous discussed case. Foreseen improvements as the additional refilling water systems are also applicable in this case.

Regarding the impact on the EECC inside the RM, if the event occurs during the use of it as a daily routine refueling exchange, it was estimated that refrigeration is not affected during transport because power supply to RM is normally provided in this event and the completion of the replacement process and / or the final deposition of EC inside the pool is assured.

Actions implemented or planned for improving the response of safety systems

- MDG - and supplementary water replenishment of the reservoir. This strategy was described above in SBO event description, but applies in this event too.

It will also include:

- Additional water supply of the dousing tank (from an independent water supply) to extend its use beyond 23 hours, estimated time is reached with the current reservoir. The modification must ensure the replenishment of the water dousing tank, taking into account that in this case there is no alternative supply from the EWS as in the case of SBO.
- It is being explored additionally the possibility of installing a motorized pump with hose connection to the lines of the ECC and the operation of the valves 3432 V 182 and 3432 V 75 to allow the addition of water to the dousing tank. This water is used as replacement feeding SG's, this enable cooling beyond the required minimum of 72 hours.

This improvement will be implemented by 2015.

2.2.3.4. Loss of heat sinks coincident with SBO

This scenario considered here is the previous one (loss of both sinks) adding the SBO case. The reactor is shutdown manually or automatically by one of the two reactor shutdown systems. The heat sink is provided for circulation and heat will be transferred from the PHTS to the secondary side of the SG and the steam will be released to the atmosphere via the discharge valves to atmosphere in the SG.

The containment and monitoring of the safety parameters is ensured in the same way as in the case of SBO.

In this scenario, the steam relief from the secondary system guarantees SG cooling. The rate of replenishment will come from the water in the reservoir of dousing, driven by gravity. The time available without any foreign intervention was estimated around 23 hours, and considering the available water of the dousing tank. In order to ensure cooling for at least 72 hours is planning to implement the same improvements foreseen for the SBO event.

Furthermore, the estimated duration for the batteries is 8 hours. After depletion, there will be no power supply necessary for the implementation and monitoring of critical parameters. Because of this, monitoring of critical safety parameters will be degraded in the event of total loss of heat sinks and SBO. The improvements foreseen as were mentioned in the preceding paragraphs will have positive effect in this case also.

Regarding the containment isolation valves, they will fail closed due either a power supply failure or the air supply failure. Thus, considering the SBO and loss of heat sink the containment isolation function will not be affected.

Impact on the fuel stored in pools

Results of the evaluation of SBO event impact on the fuel stored in pools, as described above, are applicable to this event, i.e. the expected water temperatures and times available are equivalent in this case. Planned improvements to the replenishment of water are also applicable water.

The evaluation of the impact that this event would have on the RM, if it occurs during the use of it, is under development by the Licensee.

Actions implemented or planned for improving the response of safety systems

- MDG - and complementary systems water supply. This strategy related to this event was determined based on goals to be met to face SBO event.

It will also include:

- Water supply for dousing reservoir extending its use beyond 23 hours, duration time limit estimated at present design without any secured replenishment available in the postulated conditions
- The modification should ensure the replacement of water, taking into account that in this case there is no provision from the EWS.
- Due to depletion or total loss of battery system after 8 hours, it must be supply the necessary systems to extend the availability of batteries providing recharge by the additional emergency power supply system.

These improvements will be implemented by 2015.

Improvements in the capacity and functions of the EWS and EPS systems

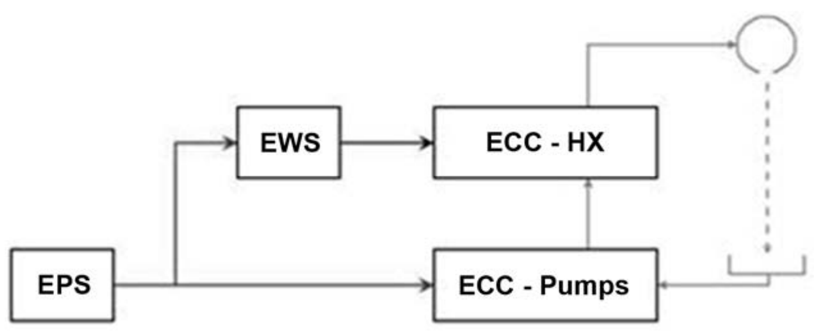
<u>EPS SYSTEM:</u>

- Replacement of existing DG (generating capacity 50 kW) for larger ones (about 1 MW). The increased capacity is intended to provide power seismically qualified to the following additional charges:
  - 3432 Pump PM1 / PM2 of ECC system.
  - 3461 Pump PM1 / PM2 of EWS system.

<u>EWS SYSTEM:</u>

- Replacement of existing diesel motorized pumps by two electric powered pumps of higher capacity and electrically fed from EPS generators.
- Doubling the valves 3461 PV7 and PV41.
- The increased capacity of the pumps is to provide cooling seismically qualified to the heat exchanger 3432 HX1 of ECC system.

These last mentioned improvements are graphically represented in the following diagram:



2.3. ACTIVITIES PERFORMED BY THE REGULATOR

As part of the assessments made so far in Argentine nuclear power plants, the Nuclear Regulatory Authority (ARN) required the Licensee a Safety Assessment taking into account the lessons learned that are applicable in the accident occurred at the units of Fukushima NPP. The aim is to detect any weaknesses and implement improvements for coping accidents in extreme conditions.

This assessment request was formalized by sending the regulatory requirement RQ-38 to the Licensee (see Annex I). The objective of the required evaluation was to determine the existing safety margins analyzing the behavior of CNA I, CNA II and CNE, given the occurrence of extreme events that cause consequences such as the total loss of power and the ultimate heat sink for a prolonged time.

2.3.1. RQ-38 REQUIREMENT: DESCRIPTION OF THE POINTS RELATED TO THE DESIGN AND SAFETY SYSTEMS

The mentioned regulatory requirement includes several aspects, among them; those included in Section 5 "Loss of Safety Functions" agreed the assessments taken as reference in this Chapter 2. Additionally, other related information is included elsewhere in the RQ-38.

In Section 5 of the RQ-38 it was required to analyze the impact of power loss and the ultimate heat sink loss on the safety features of the facility. As to the power loss it was required to evaluate the successive losses of the following supplies:

- External power supply (electric network).
- Plant main electrical generator.
- Backup power supplies that feed the safety bars (DG).
- Alternative sources of backup (DG, water/gas turbines, etc.).

Among the studies to be performed, is required to analyze the possible occurrence of the following events:

2.3.1.1. Loss of Off Site Power (LOOP)

In this case, it is required to consider conditions such as:

Total loss of the external power supply for a prolonged time and that the site remains isolated for 72 hours on the possibility of supply of heavy material by any means of transport. It is assumed that the portable light equipment could reach the site within 24 hours into the event.

Evaluating the following:

- The current design features face up to LOOP event, as well as the internal power supply systems designed to deal with it.
- The time during which the said internal power supplies can operate without any external support.
- The actions that are necessary and are planned to extend the operating time of equipment for internal power supply (e.g. filling of fuel tanks for DG, etc.).
- Possible measures to be taken to increase the robustness of the plant, such as modification of systems, modification of procedures, organizational arrangements, and so on.

2.3.1.2. Station Black Out (SBO)

The following two situations must be considered in these analyzes:

- LOOP + Loss of backup power supplies;
- LOOP + Loss of backup power supplies + loss of any other alternative power supply.

For each of these situations it is required:

- Provide information about measures envisaged in the design for these accident scenarios.
- Provide information on the battery capacity and duration. Analyze the consequences of their total loss.

- Indicate how long the plant can withstand an SBO with no external support before the occurrence of severe damage to the fuel become inevitable.
- Indicate external actions foreseen to prevent damage to the fuel.
- Identify potential extreme situations which may occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of system or procedures, organizational arrangements, etc.).

2.3.1.3. Loss of heat sinks

The ultimate heat sink is where ultimately residual heat from the reactor is transferred. It was required to consider in the evaluation the sequential loss of these sinks designed to cool the reactor and the spent fuel pool under any circumstances.

It was required to assume that the functionality of each existing heat sink is loss and that the site remains isolated for 72 hours about the possibility of heavy material supply by any means of transport, although it is assumed that light portable equipment could get to the site 24 hours after the start of the event.

Therefore, it was required to provide a description of the existing provisions in the design to avoid loss of different sinks. For example, water intakes in several places, etc.

For these scenarios, it was required:

- Indicate how long the plant could handle the situation without outside help before a severe damage to the fuel is unavoidable.
- Provide information on existing provisions in the design and internal actions to be performed for each of the above scenarios.
- Indicate external actions designed to prevent fuel damage.
- Identify potential limit situations which may occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of systems or procedures, organizational arrangements, etc.).

2.3.1.4. Loss of heat sinks coincident with SBO

It was required to assess:

- How long the plant can withstand a loss of all heat sinks coincident with SBO without any external support, before severe fuel damage is unavoidable.
- The existing provisions in the design and internal actions to be performed for each of the above scenarios.
- Indicate external actions designed to prevent fuel damage.
- Identify potential limit situations which may occur, and when these would occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (modification of systems or procedures, organizational arrangements, etc.).

2.3.2. EVALUATION OF THE RESULTS

The information generated in response to the requirement RQ-38, in relation to issues regarding the design and safety systems is presented in Section 2.2 “Activities of the Licensee”. This information was reviewed and evaluated by the ARN, and the most important issues reported by the Licensee in each case are presented.

As for the evaluation of results, first it was taken into account the characteristics specified by the current design of the plants, and deterministic results available to verify the performance expected of each safety system, as response to events postulates. This check could be performed to identify the weaknesses of the systems in each case, leading to an accident sequence to the severe accident.

In cases of identified accident sequences with weaknesses in the current design, the proposed improvements to existing systems or new systems and planned improvements are being evaluated. Also, topics on which it is required to continue the assessment could be identified.

In terms of alternatives for improvement, the focus was put on additional systems for electricity supply, covering potential failure of existing systems, and additionally, in some cases new water supply alternatives are required in order to replace the current facility sinks if necessary.

2.3.3. CONCLUSIONS

The results and conclusions that could be derived based on the assessments presented in section 2.2, for analysis of the "loss of safety functions," and for each of the different plants in question are summarized.

Loss of function events that have been assessed:

A. Loss of Off-Site Power (LOOP)
B. Station Black Out (SBO)
C. Loss of heat sinks
D. Loss of heat sinks coincident with SBO

2.3.3.1. CNA I

In the case of CNA I, weaknesses have been identified for events B, C and D. In these cases, the implementation of cooling ramp strategy from the secondary side of the SG, together with implementation of the use of a MDG, will cover the weakness found with a new alternative that may eventually replace current sinks covering the SBO scenario and/or loss of heat sinks.

As for the spent fuel that is stored in pools, estimates indicate that even in the worst conditions of SBO and/or loss of heat sink, there is a time longer than 72 hours for the start of boiling in the water pool and uncovering of the fuel elements will occur after several additional days. The strategy of including the MDG system and facilities of additional water supply is used for replacing water in the long term, in case of events beyond the design basis.

Also some issues were identified on which further assessments were required. For example, it was analyzed the possibility of having irradiated fuel elements inside RM when the event happens. In this case, a situation of SBO could result in damage to the fuel element in cases B and D. At the time of compiling this report, the Licensee has not completed such assessments. However, based on preliminary estimates, it can be inferred that the cooling of the fuel elements would not be guaranteed beyond the 2 hours after the event. For the purpose of ensuring the cooling of at least 72 hours, supplying power to the RM from said MDG is being evaluated.

2.3.3.2. CNA II

In the case of CNA II, weaknesses have been identified to events B, C and D. In these cases, the implementation of a MDG strategy for replacing at least one of the emergency´s DG is considered satisfactory for discarding core damage scenarios, identified on basis of the current design of CNA II.

In the case of events B and D, where SBO situation is considered, it is necessary to extend the supply of water to the SGs, beyond 0.5 hours, when it is estimated that existing water reservoir will be exhausted. After completing the proposed improvements, both scenarios will be adequately covered. Among the planned improvements to deal with scenarios B and D (SBO), are worth mentioning: the design change to have at least two DG cooled by a cooling tower system, and the implementation of MDG that can replace the batteries after 4 hours and feed the water supply to the SG from the LA system.

Furthermore, this strategy is also intended for use at the occurrence of event C. This strategy envisages the implementation of a procedure for emergency operation to avoid filling the pressurizer and the opening of the safety valve at the beginning of the transient. In all cases (B, C and D), the long term cooling will be ensured by MDG and supplying water from an alternative reservoir. The possibility that such a reservoir consists of water from the well water is being evaluated. It is planned to implement this improvement by 2014.

The strategy of MDG and alternative water supply would also be used to replenish the water in the spent fuel pools. In this case, the time margin was conservatively estimated in more than 30 hours.

Regarding the impact of the events of SBO, B and D, on the fuel elements that remain inside the RM, to date of issuance of this report the Licensee did not complete the assessments.

2.3.3.3. CNE

In the case of CNE weaknesses have been identified to events B, C and D. In the case of SBO event (B), even while the long term cooling would be guaranteed by supplying water to the SGs from the EWS system, it is foreseen the depletion of the batteries in 8 hours. This will affect the instrumentation and monitoring. Among the proposed improvements, the addition of a MDG will replace, among other things, this supply.

It is expected to resolve the weaknesses identified for events C and D with MDG, which also could fed to a pump to replace water from the reservoir to the spray tank (dousing). This will prolong the cooling beyond 23 hours, to meet the requirements, i.e. for 72 hours minimum.

As for the spent fuel stored in a pool, the evaluations indicate characteristic times similar to those mentioned for CNA I, in cases of SBO and loss of heat sinks, B, C and D. The modification to the design provides an additional system for compensation of water in the pool, in the extreme conditions of SBO and/or loss of sinks.

Additionally, issues were identified on which further assessments were required to Licensee, such as the possibility of irradiated fuel elements located inside the RM during the beginning of the events analyzed. The evaluation of the impact that this event would have on the integrity of the fuel elements housed in the RM is under development. However, from preliminary estimates, it can be inferred that in the fuel elements housed in the RM temperature could reach 1470 °C (corresponding to the events B and D).

3

SEVERE ACCIDENT MANAGEMENT AND RECOVERY –ON SITE–

3.1. INTRODUCTION

Regulatory activities in the Argentine nuclear power plants related to severe accidents started long before the occurrence of the Fukushima accident.

It is in this context tha the ARN required the Licensee (NA-SA) the development of Severe Accident Management Guidelines (SAMG) for the Argentine NPPs in operation (CNA I and CNE). For the particular case of CNA I, the first stage was to define the objectives and scope of the SAMG, taking as reference both international guidelines as well as particular aspects of the plantI

Regarding the available information for the development of the SAMG, probabilistic studies provided a systematic approach to the vulnerabilities of the plant. CNA I has an updated PSA (Probabilistic Safety Assessment) Level 1, in which the plant's operating experience and new deterministic studies were incorporated.

As CNA I has a unique design, there were no applicable procedures and accident management guidelines obtainable from similar plants.

Moreover, in the unlikely event of a severe accident, there wasn't any model of its evolution. Therefore, one of the most important tasks for this program was the development of this model. The model allows the analysis of the evolution of the plant in different accident scenarios after a core meltdown, and the evaluation of the mitigation strategies effectiveness (molten core cooling, containment venting, limiting emissions, etc.).

In this section it is shown the mitigation actions foreseen if severe damage in both the reactor and the spent fuel pool occurs, in order to prevent large radioactive releases. The issues considered are the development of severe accident scenarios; development and validation of procedures; equipment availability; training, the corresponding personnel resources, as well as the results of the reviews of severe accident management and on site recovery actions.

3.1.1. ACTIVITIES OF THE OPERATOR

3.1.1.1. CNA I

The plant model was performed with the MELCOR code, which is one of the codes most used to simulate accident progression based on a core meltdown.

In addition, some preventive strategies have been undertaken to date. Progress in their implementation varies from case to case. In some cases, the plant instructions have been approved and they have been included in the operators' training plan.

3.1.1.1.1. Accident management measures currently available to protect the core at various stages of a scenario of loss of cooling function of the core

Within the SAMG, different strategies were made, mainly preventive management of the accident scenarios considered. In general, these preventive strategies point to an accident sequence, or a group of accident sequences that lead to loss of a safety function, and to the applicability of a strategy. In the field of mitigation measures, scenarios are identifies for which these measures apply regardless of how this stage is reached.

3.1.1.1.1.1. Measured before the onset of damage to the fuel

The presented preventive strategies to avoid damage to the core are related to the loss of the cooling function. The following cases are analyzed:

1. Inventory replacement in the primary with the "pressure and inventory control system" (TA) in small LOCA conditions with different alternatives of design changes.
2. Water supply to the steam generators SG using the second heat sink system (SHS / RX) in different accident scenarios.
3. Strategy for a total loss of electric power scenario (Station Black-Out -SBO): Establishment of a mechanism for cooling the core, preventing at the same time air from entering the primary circuit from the boron injection system (TB).
4. Strategy for the 220 AC alternate current supply failure.
5. Strategy for the decrease of the 24 AC alternate current voltage

The strategy set out in case 1 is applicable to small LOCA scenarios, combined with the failure or inability of the low pressure water injection system (TJ-BP). The strategy raises the possibility of injecting light water with the pressure control and inventory system (TA). We have proposed two alternatives, whose application depends on the availability of normal or insured electrical supply and cover LOCA scenarios up to 10 cm^2. The second alternative, allows injecting water using the three pumps of the TA system and is effective for LOCAs up to 20 cm^2

The strategy set out in case 2, aims to restore the ability of the SHS, in a scenario of heat removal failure, and of the system of water supply / waste heat removal (RL / RR), as a result, for example, of a catastrophic malfunction of the water tank RL11B01, combined with failure or collapse of the SHS.

The SHS failures targeted with the aforementioned strategies are the unavailability of the RX03B01 tank of the SHS inventory depletion thereof, or due to failure of the SHS pumps (RX01/02D01). See *Figure 3-1*.

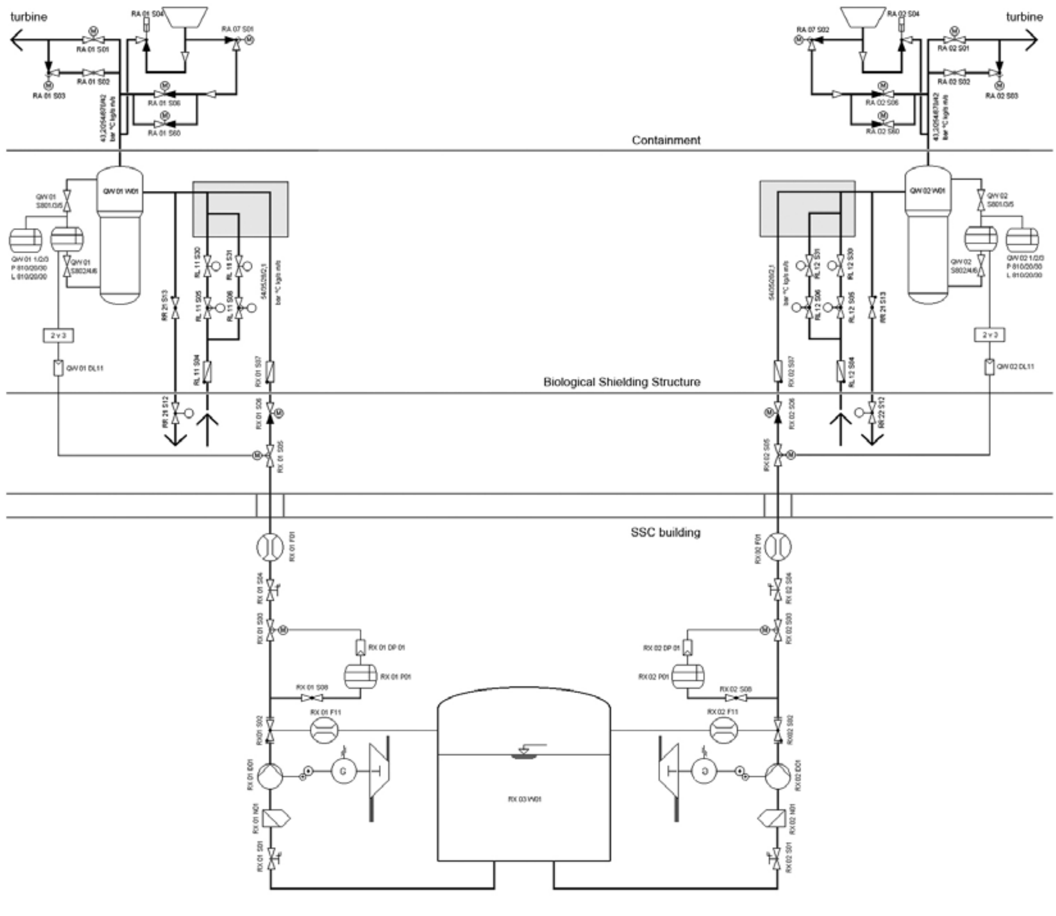


Figure N° 3-1: Diagram of the second heat sink system.

The strategy is to take water from the pools of demineralized water (UA00B03/B04). The flow will be driven by existing pumps, using a fixed installation pipe with a pump, with the possibility of supplying water into three points of the SHS system, depending on the scenario (the tank, the impulsion and aspiration). In the event that the SHS pumps are not available, the SG must be depressurized (*Figure 3-2*). The needed design changes will be implemented by the end of 2012.

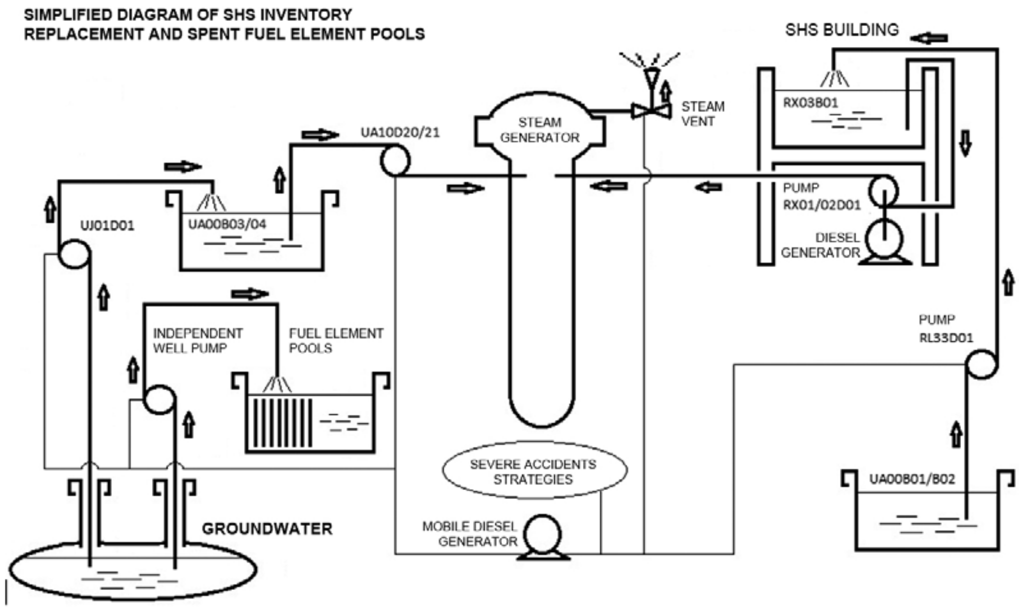


Figure N° 3-2: Mobile Diesel Generator and additional systems for water replacement.

The strategy proposed in case 3 is applied to a scenario of total loss of AC power at the plant, where the SHS pumps have their own diesel generators (DGs). In this case, the available battery-time is used to cool the plant and implemented the SHS.

The strategies proposed in cases 4 and 5, point to taking the plant to a safe shutdown in the case of a failure scenario of the alternate current rectifier. These scenarios leading to the loss of AC were identified in the PSA level 1 as important contributions to core damage.

To meet the recommendations arising from the WANO SOER 2011-2, where it was requested to check the availability of Structures, Systems and Components (SSC) available for accident scenarios, the Licensee compiled a list of 253 components that needed to be verified necessary to deal with events included in the design basis, which emerged from the walkdown and inspections performed during 2011.

The verification list gave the following information:

- Item: system, component, or training
- Repetitive Testing / Training.
- Latest test results: Satisfactory, Unsatisfactory or Satisfactory with innovations/changes.
- Notice of Pending abnormalities with its priority.
- Electrical Predictive Preventive Maintenance with its priority.
- Mechanical Predictive Preventative Maintenance with its priority.
- Instrumentation Predictive Preventive Maintenance with its priority.
- Component state at the moment of the Walkdown: available, unavailable.

3.1.1.1.1.2. Measures after the occurrence of the fuel damage

Severe accident phenomenology

As stated before, for the SBO sequence, a qualitative comparison of the evolution of the phenomenology of core degradation for CNA I was performed, based on CNA II results. The following summarizes the relevant aspects of the study.

General features

The MELCOR code model for CNA II was developed at the GRS Institute (Gesellschaft für Anlagen- und Reaktorsicherheit, Germany) with the support of the Nuclear Safety Division of the Comisión Nacional de Energía Atómica (CNEA).

The CNA I characteristics and its particular design considered relevant for the in-vessel severe accident (progress within the reactor pressure vessel RPV) are summarized below:

- **Fuel elements inside cooling channels.**
  Each one of the fuel elements of the core is housed within a vertical tube of zircaloy (or coolant channel) inside of which the primary coolant system circulates, while on the outside the water of the moderator system flows. This particularity in the design of CNA I, wherein the fuels are not arranged one after the other as in a PWR or BWR, but are located within separated vertical channels, directly influences the evolution of a severe accident. Firstly, the different geometrical arrangement modifies the normal flow patterns expected from the thermo-hydraulic standpoint. Furthermore, the presence of zircaloy coolant channels represent additional mass which can be oxidized (generating heat and hydrogen) and is involved in the core process degradation. Furthermore, the physical separation imposed by the channels between the water moderator and the primary system also influences the course of the accident. Different degradation scenarios are thus possible according to the water level in the moderator tank at the start of the event, so it is necessary to investigate which of these is most likely to occur in a real case.

- **Separated cooling channels**
  Additionally, each fuel element is located within a channel of approximately 10 cm in diameter, it is important to note that these channels are separated by a distance between centers of about 25 cm. Accordingly, the core has an array of spaced fuel elements unlike the compact cores of PWR or BWR. As a result of this, an evolution of core degradation is not expected to involve the formation of a "pool" of molten material (known as molten pools) as happened in the accident at TMI-2. This is because the area of passage in the moderator is large, which does not allow the molten material to re-solidify at a lower position forming a crust that supports the molten material. Instead a degradation process is expected in which the molten material or core debris settles at the bottom of the tank of the moderator system.

- **Volume of heavy water inside the moderator.**
  The large amount of water available in the moderator tank serves as a major heat sink, directly influencing the evolution of the postulated severe accident. As will be described later within the results obtained for CNA II, this characteristic is a decisive one for the in-vessel severe accident stage, since it imposes a slow core degradation process, with low heating rates.

- **Control rods inserted in the volume of the moderator system.**

  The control rods (29) are inserted in the moderator in CNA I at different angles with regards to the vertical to allow the movement of the replacement of fuel elements machine.

  This arrangement is quite different from that of typical PWR or BWR reactors, in which the control rods are inserted vertically and very close to the fuel rods, and play a significant role in the degradation process.

  In PWR and BWR the control rods are usually the first components to melt, and as they relocate to lower positions of the nucleus they interact with other components transferring heat and forming eutectic components which may melt at temperatures lower than the components do, there-by forming blocks which prevent the passage of fluid and retain molten material relocated from higher positions.

  Since in CNA I the control rods are immersed in the tank moderator, relatively far from the fuel elements, they are not relevant and their influence is limited in the degradation process. Moreover, they do not represent a significant amount of material with regards to the mass of the fuel elements and the cooling channels.

- **Re-criticality is not relevant in case of core re-flooding.**

  If the re-flooding of the core is taken into account as a measure of mitigation in the case of a severe accident, it is important to note that the potential core re-criticality is not a risk that should be taken into account in CNA I. This is because the plant, in addition to operating with a small reactivity excess compared to other designs, possesses slightly enriched fuel (0.85 U^{235} weight %) and require heavy water as a moderator for their operation. In case of an accident, emergency injection systems will inject light water, after which the recovering of the critical condition in the core is not possible.

  If the re-flooding of the core is performed by external means, light water will be injected, which means that even in this case core re-criticality will not be an issue to be considered.

- **Filling material.**
  The filling materials are large pieces of steel placed in the upper and lower dome of the pressure vessel, with the aim of reducing the amount of heavy water needed in the primary system. The filling materials in CNA I play an important role in the progression of a severe accident (as confirmed by the results obtained for CNA II) because the large amount of steel which is part of this material functions as an energy repository. In this sense, the lower placed filling material plays a greater role, because after the relocation of the molten material at the lower plenum when the rupture of the bottom of the moderator tank occurs, a great amount of the decay power is transferred to the filling material, significantly delaying the progression thereof, and thus delaying the failure of the pressure vessel. This design feature of CNA I represents a remarkable advantage from the point of view of a severe accident.

3.1.1.1.3. Comparison of CNA I and CNA II parameters

Making a comparison of temperatures, pressures and flow rates between CNA I and CNA II during normal operation, it is concluded that given the occurrence of an SBO, the initial conditions of temperature and pressure in the primary, moderator and secondary systems are similar for both plants.

In *Table 3-1*, important design parameters for the calculation of thermo-hydraulic severe accident progression within the pressure vessel and before its rupture are compared. Because the cooling channels and fuel elements of CNA I and CNA II are geometrically similar , and also the average power per channel and per bar, as well as the flow passage area per channel are similar, it can be concluded that the thermo-hydraulic parameters for a representative average channel are similar. Therefore, since the initial operating conditions are also similar, it is expected that the thermo-hydraulic evolution within the pressure vessel in an accident sequence such as a SBO, be similar for both CNA I and CNA II.

	CNA I	CNA II	CNA I/CNA II Relationship
Number of cooling channels.	250	451	0.55
Active length [m]	5.3	5.3	1.00
Overall power [MW]	1180	2160	0.55
Total amount of fuel rods	9106	16687	0.55
Power per bar [kW]	130	130	1.00
Average linear power [W / cm]	232	244	0.95
Power per channel [MW]	4.7	4.8	0.98
Channel diameter [mm]	120	112	1.07
Channel spacing [mm]	272	272	1.00
Moderator tank area [m2]	16.4	28.3	0.58
Moderator area per Channel [m2]	0.065	0.063	1.04
Moderator water mass [t]	119	206	0.58
Water mass in the Primary System [t]	88	172	0.51
Water mass in the SG [t]	16	46	0.34
Mass of UO_2 [t]	44	97	0.46
Mass of UO_2 per channel [kg]	180	220	0.82
Mass of Zry [t]	19	28	0.65
Steel mass in the degradation region [t]	213	616	0.35
Overall power / water mass relationship (primary + moderator + GV) [W / kg]	530	510	1.04

Table 3-1: Comparison of CNA I and CNA II characteristic parameters.

Summary of estimated characteristic parameters for CNA I

Table 3-2 shows the characteristic parameters of a severe SBO type accident at CNA I, taking as a reference the equivalent for CNA II.

SBO type accident sequence: Identical conditions to the reference sequence in CNA II		
Parameter	**CNA II Value (MELCORr)**	**CNA I Value (Estimated)**
Initial time of the core exposure	2.9 h	~ 3 h
Initial time of the core degradation	5.3 h	~ 5 h
Time to failure of the moderator tank bottom	11.3 h	~ 11 h
Time to failure of the RPV	23.8 h	~ 24 h
Core temperatura increase rate	0.1 K/s	~ 0.1 K/s
Percentage of oxidized Zry	50 %	~ 50 %
H_2 production by oxidation of Zry	600 kg	~ 390 kg
Total production of H_2	800 kg	~ 460 kg

Table 3-2. Characteristic parameters estimated for CNA I - SBO identical to that postulated in the reference sequence for CNA II

3.1.1.1.4. Conclusions

- In general terms, it is expected that the progression of a severe SBO type accident at CNA I, the interval phase that takes place in the pressure vessel is slow, with characteristic longer times. Core degradation is delayed mainly due to the existence of the moderator, which represents a major heat sink. After the relocation of the molten material to the vessel's lower dome, the vessel failure is considerably delayed due to the presence of the filling material, to which is transferred a substantial part of the power generated by decay.
- The fuel heating increase rate after its exposure is estimated to be in the order of 0.1 °K / s as the generated power per channel are similar for both plants. Note that this parameter (heating increase rate) is one of the most important to characterize the progression of an RPV in-vessel severe accident. The estimated value implies a slow heating of the fuel, and a longer time for the oxidation of the zircaloy.
- No core degradation process along with the formation of a molten pool as occurred in the TMI-2 is expected due to the presence of the cooling channels and to the spacing between the fuel elements.
- The total production of hydrogen prior to rupture of the pressure vessel in a severe accident of this type is expected to be around 460 kg, with about 50% of the total core zircaloy oxidized.

The specific model for CNA I simulated with the MELCOR code is under development and is scheduled for completion by the end of 2012.

For the same sequence SBO there is a comparison between the behavior of CNA II (PHWR) and a German KWU reactor of similar power. The conclusion of this study is that the calculated time to failure of the RPV for CNA II is five times greater than that of the German plant.

Once there is evidence of deterioration of one of the containment barriers, generally when finding the failure of the preventive strategy, the plant personnel must implement strategies to preserve the integrity of the next barrier, following the concept of defense in depth.

The following possible strategy consists in cooling the external side of the pressure vessel once significant damage to the core has been observed. This strategy and its effectiveness, including the guides for severe accident management, are currently under development and analysis for CNA II.. Effectiveness analysis for CNA II will be completed in late 2012. Finally, it is planned to analyze its applicability to CNA I. Any necessary design changes to provide the cooling water is expected to be completed by 2015.

3.1.1.1.5. Accident management measures and plant design features for the protection of the integrity of the containment function after the occurrence of fuel damage

RPV Protection

As mentioned above, in CNA II, cooling the RPV from the outside as a way to protect it is being evaluated. In addition for CNA I the progression time of the fusion till the integrity of the RPV is compromised have been evaluated. Once the results for CNA II are completed an equivalent strategy for CNA I will be analysed.

Prevention of deflagration and detonation of hydrogen, considering the real capacity of the containment venting

The CNA I Licensee has decided to incorporate passive autocatalytic recombiners (PARS) to enhance and ensure the containment function. Currently, AREVA was contracted to determine the specifications for the PARs and their location in the plant.

By the end of 2012, it is expected to have the results of the analysis in order to determine their location and quantity, since their installation depends on the geometry and the release of H_2 inside the containment.

It is expected that the detail engineering and the required procedures will be completed by 2013. While the number of apparatus will be determined by the detail engineering, through the experience acquired in CNA II, it is expected that between 30 to 40 PARS will be required to be installed. The installation of this equipment is scheduled to be completed by 2014.

Furthermore, the following technical information for determining the characteristics of the recombiners was sent to AREVA:

- Zircaloy inventory in the core.
- Free volume in the containment sphere.
- Exposed surfaces of walls and components.
- Mass of UO_2.
- Areas of openings between rooms.
- Air flow rates between rooms.

Preventing overpressure inside the containment

The following are several aspects related to the prevention of overpressure inside the containment:

a) CNA I Containment Model

A preliminary simulation model of the CNA I containment has been developed in order to study different accident scenarios under the PGAS Project. The numerical simulation model is executed using the MELCOR calculation code.

A study on the stability of the CNA I containment model was performed which concluded that all the pressures and temperatures are stable and the flows stabilize in few seconds. The flows stabilize with null flows, except those belonging to subdivisions made on the premises of the steel sphere, where the differences of heat provided by the biological cylinder produce constant streams. These flows provide more cooling to structures of the biological cylinder.

The model is being reviewed taking into account the CNA II results and it is expected to be completed in 2013.

b) Strategy

Regarding the strategy to be used in the management of the containment, once the final results for CNA II are available, they will be applied to CNA I, taking into account the similarities between both plants.

The analyses performed in the level 2 PSA for CNA II show an increase of the containment pressure that in no case exceeds the design pressure. This is mainly due to the very low power/free volume relation in the containment. This leads to the conclusion that venting of the containment is not needed. The abovementioned severe accidents are those resulting from the sequences that lead to core melting starting from design basis events. Based on the comparison between both plants made at an earlier point, a similar result is expected for CNA I.

Moreover, the following strategies to be implemented to diminish the containment pressure during severe accidents are being evaluated:

- From the outside by means of the annular space ventilation.
- From the inside through air recirculation using the existing ventilation system.

The evaluations related to the mentioned strategies were performed considering that the containment has been closed due to the protection system (NZ54 signal) and the verification of the closure has been performed by the operator in the early stages of the accident. Given a scenario in which the internal pressure increases continuously, the control of the pressure is expected to be made through the ventilation sweeping the annular space so as to facilitate the cooling of the sphere. If this action fails to prevent the pressure exceeding the design value it is planned to manually vent the containment before it fails.

Data and simulations for other similar KWU design plants are available as well as simulations with simplified models for CNA I. From this information, two possible options for reducing the containment pressure are being evaluated. Its implementation will be defined when the results for the accident progression model become available, expected in 2013.

Prevention of re-criticality

The reactor rod shut down system consists of 24 hafnium control rods (black rods) with a reactivity value of 80 mk and it is designed so that given the shutdown signal the rods drop due to gravity in 3 s when the current that energizes the coils that hold the rods is interrupted. The effectiveness of the shutdown system by rods covers all the variations of reactivity and ensures sub-criticality of the core at any state of the plant.

Moreover, in case of a large loss of coolant accident (LOCA) (guillotine rupture), due to the need of a faster shut down speed, a second shut down system is demanded, which is the boron injection into the moderator (TB), with a delay of 500 ms, that is adequate for this scenario.

Criticality evaluation studies were conducted; whose main results were the following:

- "Levels of sub criticality in the Irradiated fuel storage pools in CNA I": The study shows that in the pools there is no risk of criticality for fuel elements with 0.85% slightly enriched uranium (ULE) conservatively considering an infinite amount of fresh fuel elements.
- "Verification of criticality calculations for the CNA I pool using the MCNP Code (Monte Carlo N-Particle transport code) for selected experimental reference cases": it was concluded that the safety margins for the pools, calculated in an extremely conservative way, are greater than 200 mk under normal conditions and greater than 50 mk for accidental conditions.

Consequently, the case of re-criticality in the event of re-flooding of the core is not relevant for this type of reactor.

Prevention of meltdown passing through the slab

Regarding the prevention of penetration of the slab, as mentioned previously, once the results for CNA II become available, an equivalent strategy for cooling the RPV from the outside will be analysed, as a way to protect its integrity and stop the accident progression. The CNA I specific progression model will define its applicability. The results of the application of this model are scheduled for 2013.

Alternating and direct current supply, and compressed air supply to the equipment needed to protect the integrity of the containment

The signal that produces the containment closure depends on the assured supply of direct current and is activated by the loss of 24 VCC. The penetrations of larger diameter correspond to the TL System, which has two valves in series, one is pneumatic and the other is motorized (powered from the assured 380 VAC bar). The mentioned signal sends the closure order to both valves, the pneumatic and the motorized one, so that both will close in a normal situation, ensuring the containment insulation. In case of loss of AC, the motorized valves will not have electrical power and they will not close, causing the insulation of the containment to be produced only through the pneumatic valves.

The main hatch is electrohydraulic and in case of power failure can be maneuverer manually. The emergency hatch is very simple and its actuation is mechanical and manual.

3.1.1.1.6. Measures for accident management to deal with loss of cooling in the fuel storage pools

3.1.1.1.6.1. Before and after losing adequate protection against radiation (loss of shielding of the water column)

To deal with the situation of loss of cooling in the fuel element storage pools and avoid the exposure of the upper part of the fuel and its degradation, the following management measures have been taken:

a) The values of the decay heat in both Pool Buildings “Casas de Piletas” were calculates for two situations: the current situation (September 2011) and the prediction for May 2015, when it is assumed that all the free positions in the pool buildings “Casa de Piletas II” will be completed (see *Table 3-3*).

	2011	2015
Pool 1	31	28
Pool 2	33	26
Pool 4	688	772
Pool 5	122	71
Pool 6	61	44
Pool 7	61	56

Table 3-3: Decay heat generated in each of the pools in kW.

b) The times were estimated in which the water temperature in each pool reaches 100 °C in the case of normal loss of cooling for the two situations described, the current (September 2011) and the prediction for May 2015, discriminating by individual pools: Pools 1 and 2 in Pool House I and “Pools 4, 5, 6, 7” in Pool House II.

   In normal operation, the mean water temperature in the pools is 30 °C. Based on this situation the times were estimated in which the water temperature in each pool reaches 100 °C in the case of normal loss of cooling. *Tables 3-4* and *3-5* show the results in each Pool House.

	Año 2011	2015
Pileta 1	69.1	76.6
Pileta 2	60.6	82.7

Table 3-4: Times to reach 100 ºC in case of loss of cooling in Pool House I.

	2011	2015
Pileta 4	3.1	> 2.9 (*)
Pileta 5	17.1	28.4
Pileta 6	36.0	49.8
Pileta 7	36.1	39.3

Table 3-5: Times to reach 100 ºC in case of loss of cooling at Pool House II.

(*) This value was obtained with very conservative assumptions, both in determining the heat source in each pool, as in the estimation of the time to reach 100 °C if the cooling is interrupted. Therefore, in realistic conditions this number should be significantly higher. Even so, the source term will be reduced by performing a redistribution of the fuel elements between Pools 4 and 5 because, as shown in Table 3, the main contribution to the decay heat takes place in Pool 4.

c) Actions and contingency plans have been developed to monitor the level and water temperature in the spent fuel storage pools.

An emergency operative procedure for responding to an event of loss of cooling or inventory of the pools of spent fuel storage is under development. This procedure provides for the monitoring of the level and temperature of the pools during the emergency and the possibility to restore inventory even under the following conditions:

- Loss of Control Room
- SBO
- Earthquakes and flooding

For this purpose the adoption of the following measures are being evaluated (see Figure 3-2):

1. Installing a separate well pump to supply the Spent Fuel Element Storage Pools. Installing an electrical panel for manually connecting the pump to the auxiliary emergency mobile diesel. The additional well water pump will be oversized, in order to be utilized to feed the UA pools if necessary. It is expected to have this improvement by 2013.
2. Modification of the electrical connection of the well pump to power it from the assured bar of the new Emergency Power System (EPS). It is expected to have this improvement by 2013.
3. The proper functioning of the program for the verification of the vacuum breakdown / siphons associated with cooling was checked, finding them in perfect condition.

There are also plans to add to the periodic inspections program, the control of the functionality of the breaking vacuum / siphon system associated with the pipes of the cooling systems or the inventory control of the Fuel Element Storage Pools. The schedule for periodic monitoring to analyse the items and how to perform the control will be defined. This control will have a frequency of annual implementation. Currently, this measure is under implementation and is expected to be implemented by the end of 2012.

d) Against extreme events, the E0 statement "Operation in Perturbations and Accidents" was modified in order to include the control of critical parameters of the spent fuel storage pools. The monitoring of temperature and level in the pools has started.

3.1.1.1.6.2. Before and after the exposure of the top of the fuel

According to the measures designed to prevent fuel exposure, detailed in the previous item, it is estimated that the fuel would not be exposed due to the long times required for this. See item 2.

Table 3-6 shows the calculated times to expose the fuel elements after reaching 100 ºC.

Year	Power	Additional time to expose the fuel elements
Pool building I	[kW]	Day
2011	66.5	161.8
2015	54.2	198.5
Pool building II	[kW]	Day
2011	978.5	19.7
2015	1084.7	20.3

Table 3-6 Time to uncover the fuel elements.

3.1.1.1.6.3. Before and after degradation of the fuel (rapid oxidation of the claddings with hydrogen production)

As stated in the previous points, it is considered very unlikely to lose the heat removal function of the spent fuel pools, which would cause the degradation of the fuel, and given the long-time available to take actions in case of total loss of the various cooling systems and inventory replenishment. See item 2.

3.1.1.1.7. Additional Aspects

In addition to the Emergency Plan detailed in item 5 of this report, which reports on the organization and the means for emergency management at the site, the use of external support to the plant and the procedures and instructions for emergency situations, training and education, the CNA I has an Operations Manual which includes operational situations to address both emergency conditions and abnormal situations.

In addition, there are guidelines and strategies for the prevention of accidents that will allow the responsible for the emergency management to guide the operations group. On the other hand, provisions for the implementation of modifications to the facility designed to manage the mitigation stage between the aforementioned implementation such as hydrogen recombiners, cooling systems and alternative energy supply as detailed in item 2.

Also, as detailed in item 5, CNA I has a systematic process of training its personnel in such a way as to meet the operational needs of an emergency. In particular, the operation staff is trained in the full-scope simulator of Angra nuclear plant, Brazil; in which training takes place under normal, abnormal and emergency situations operating conditions. Retraining in Full Scope Simulator is held every 2 years and annually on issues of radiation protection, nuclear safety, industrial safety, emergency preparedness, first aid and environmental management.

The staff of the Emergency Brigade is systematically involved in retraining theoretical courses and emergency simulation exercises on the ground. Such staff annually participates in a training developed at a fire simulator.

Regarding the availability and supply management, with an emergency scenario caused by a severe accident, it is expected that the transfer of the required supplies is made through land access routes, and in case of unavailability, it can be done through fluvial and / or air, with the support of “Gendarmería Nacional”, “Prefectura Naval”, “Armada Argentina” and “Ejército Argentino”.

Management of possible radioactive emissions and provisions for mitigating them.

The CNA I has a containment building capable of withstanding an overpressure of 2 ate, which is a barrier to the uncontrolled escape of fission products in case of core damage during an accident.

Since the Containment Building is enclosed by the Cavity, possible leaks from the Containment Building are held within the Cavity. This building has a Ventilation System (TL 80) which allows a controlled release of air through a chain of activated carbon and HEPA filters, reducing the quantity of pollutants from the exhaust air, which is passed through detectors to the plant chimney. These detectors measure tritium, noble gases, aerosols and iodine activity.

The CNA I also has a calculation model (the CDE program), which starting from an assumed specific core damage allows an estimation of the possible dose to the public, depending on the weather conditions at that time. Using field measurements made by the CNA I MIU (Mobile Intervention Unit) they can calculate and adjust the core damage assumed, and thereby accurately estimates the emissions.

To support this system, the CNA I, will soon install a remote measurement of dose rate in a circle of 10 km radius in all directions is expected to be implement during 2012. The system has 13 monitoring stations to assess the field conditions, which send their information on line to both the internal centre of emergency management (ICC) and the external centre of emergency management (CECE).

Liquids leaking from containment enter the annular ring preventing the direct emission of liquid into the environment. If the leaks occur, they are channelled by the TZ system, to the liquid treatment system (TR). Release into the environment will be under the same conditions as in normal operation, allowing the release into the environment to take place in a controlled manner.

Item 5 describes the management of potential doses to workers and the previsions for limiting them.

Impediments to do certain jobs because of high local dose rate of radioactive contamination and destruction of facilities

Before the possible intervention, CICE will assess the existing radiological conditions to properly plan the intervention, optimizing staff doses, and taking into account the emergency dose limits given by the Regulatory Standard AR 10.1.1.

Given the magnitude of the problem, if its own resources are insufficient, support will be asked to response organizations such as “Armada Argentina” (the navy) and “Ejército Argentino” (the army) who have trained human resources and the equipment required.

If the CICE were exposed to high dose rates, it is planned to move it to the CECE in the town of Lima, a distance of about 5 kilometres from the plant.

Potential effects on other nearby plants

Potential effects of the CNA I over CNA II and vice versa are being analyzed. Because CNA II is at the beginning of the Licencing phase, the results of these analyses are expected at the beginning of the power rise phase in 2013.

3.1.1.2. CNA II

3.1.1.2.1.

Accident management measures currently available to protect the core during the different stages of a scenario of loss of its cooling function:

In the frame of the Probabilistic Safety Assessment for CNA II several analysis were performed regarding severe accidents.

As a result of the plant specific PSA Level 1, a number of relevant Core Damage States (CDS) have been identified and used in PSA Level 2 MELCOR code calculations in order to assess the severe accident progression and the possible radionuclide releases into the environment. The results have been used directly by PSA Level 3 study.

The dominating sequence according to PSA Level 1 analysis is the Station Black-Out (SBO), for which a low pressure scenario, a high pressure scenario, and several sensitivity cases have been assessed. In particular, a review of the Low Pressure SBO scenario will be described, in order to highlight specifics of CNA II regarding timing of the accident scenario, in comparison to PWR reactors.

Results of MELCOR Code Simulation: calculation for low pressure SBO

Starting from normal plant operation a SBO was assumed at time 0:00 h: min. The main cooling pumps and moderator pumps run down due to the loss of electrical power supply. Injection of water into the reactor cooling circuits is not available because of the loss of all pumps. The containment is successfully isolated due to the SBO condition (fail-safe principle of isolation valves).

Due to the loss of heat sinks, both the primary as well as secondary side pressure starts to increase. On the secondary side, the cool down by a rate of 100 °K/h is initiated. This cool down is available because the relief valves of the secondary side steam dump stations are supported electrically by batteries. The secondary side pressure decreases, while pressure increase on the primary side is bounded (*Figure 3-3*). The water level of the steam generators is continuously decreasing due to the cool down process and falls below 2 m at 0:43 h (*Figure 3- 4*). Thus, the isolation of both steam valves and steam dump stations occurs after steam isolation. Small steam leakages of the pilot valves are considered.

Due to the lack of heat release from the primary circuit, the water level inside the pressurizer rises. After completion of the 100 °K/h cool down, the primary pressure starts to increase again. At 0:52 h the primary pressure reaches 12.45 MPa for the first time (*Figure 3-3*). The first opening of the pressurizer safety valve occurs at 0:55 h. Subsequently, the valve opens intermittently in order to limit the primary pressure. During the first cycles of the valve only steam is blown into the relief tank. Following, a water steam mixture is released into the relief tank. The burst disk of the relief tank fails at 2:31 h because a pressure gradient over the disk of 1.4 MPa is reached. For the last three cycles only water is discharged into the relief tank. After the safety valve fails (stucks open) a continuous leak flow through the relief tank occurs. Due to the leak flow through the relief tank (*Figure 3-5*) the primary pressure starts to decrease (*Figure 3-3*).

Up to about 2:40 h a free convection flow can be observed at the two upper loops of the moderator system. For each of the lower moderator loops the flow disappears with initiation of the station black-out event, as the moderator heat exchangers are located below the core height.

The JR31 ECCS signal is triggered at 2:31 h. The containment pressure rises due to the leak flow. Thus, several burst membranes located at the steam generator boxes and pressurizer box fail. The flooding signal JR36 occurs at 2:43 h because the primary pressure has fallen below 6.6 MPa. However, the water of the flooding tanks cannot be discharged into the sump due to the lack of electrical power supply.

Owing to the permanent loss of coolant, the water level inside the cooling channels of the core decreases. At about 3:30 h the cooling channels are nearly empty. At that time the water level inside the moderator tank is still located at two thirds of the total height of the tank (*Figure 3-6*). From this time on, the decay heat is evacuated from the fuel elements to the coolant channel walls by convection and radiation, as far as they are cooled from the outside by the moderator tank water. The temperature increase shows, especially in the lower parts of the core, a plateau at about 3 h to 4:30 h of the transient (*Figure 3-9* and *Figure 3-10*) which evidences the decay heat evacuation by radiation as already stated. As a consequence, the water level inside the moderator tank continues decreasing (*Figure 3-6*).

Simultaneously, the water level of the pressurizer slowly decreases (*Figure 3-7*). Because of the exposure of the cooling channel, the core starts to heat up in the upper part (*Figure 3-8*). The heat-up of the lower parts of the core follows with a delay (*Figure 3-9* and *Figure 3-10*). The water discharged through the relief tank flows into the containment sump and is lost for core cooling, because Safety Injection pumps are not available due to the total loss of electrical power supply.

The first radionuclides - noble gases and volatile radionuclides - are released from the fuel elements gap after bursting of the cladding starting at ~3:56 h. The gap release starts in the central part of the core and migrates to the outer radial core rings. After 5:10 hr, all radial core regions have got a gap release. The radionuclides form aerosols which are released together with steam through the open relief tank into the containment. Some of the aerosols may also deposit inside the RCS.

The core heat-up accelerates when the oxidation process of the zircaloy cladding and coolant channels starts at about 3:30 h (*Figure 3-8*). The hydrogen formed by the oxidation is mainly released through the relief tank into the upper containment compartment where the relief tank is located. After 10 h some of the hydrogen generated inside the pressure vessel is stored inside the Reactor Cooling System (*Figure 3-1*). In this analysis no combustion was calculated neither in the rooms near the relief tank nor in other rooms. Hydrogen fractions during the first 10 hours are small because of the presence of Passive Autocatalytic Recombiners (PARs). The amount of hydrogen removed by the PARs installed in the containment compared to the generated amount is shown in *Figure 3-11*.

The failure of the coolant channels starts in the upper part of the four inner core rings at 5:16 h. At 5:20 h the damage of the middle part of the inner rings begins. This process opens additional flow connections between the damaged parts of the coolant channels and the moderator tank. Thus, some water flows from the moderator tank into the lower intact parts of the cooling channels where the water level is increased (*Figure 3-6*). Up to 8:20 h the lower plenum of RPV is still full of water. Then, its evaporation starts due to first small relocation of core materials. Inside the moderator tank and cooling channels the water is fully evacuated at about 9 hours after the event initiation (*Figure 3-6*).

The on-going core heat-up causes a failure of the fuel assemblies and the core failure process spreads radially to the core periphery and axially downwards. This is shown by an abrupt decrease in temperatures in *Figure 3-8* to *Figure 3-10*. Failure temperature for fuel rods is around 2300K. The reason for this is the following: in CNA II, fuel elements are hanging from a special construction located in the upper part of the channels above the core, which allows the so-called on-line refuelling. As the “fall down” process of a fuel assembly inside the coolant channel after support failure cannot be modelled by MELCOR, some assumptions have to be made about the failure of the fuel assembly. The mechanical failure of fuel rods is assumed to result from a combination of loss of intact, non-oxidized cladding material; thermal stress; molten Zircaloy, which “breaks out” from ZrO_2 shell at 2400 °K; and the collapse of standing fuel rods (that will form particulate debris) based on a cumulative damage function. Due to swelling and thermal expansion, the mechanical stresses inside the pellets increase with temperature. Therefore, a function was build modelling the fuel integrity with time vs. temperature taking into account that a long free standing largely damaged fuel column is non-realistic. As a conclusion, failure is shown to take place at a lower temperature than the one usually found in experiments, because a loss of integrity concept for fuel elements is being applied.

Together with the core degradation, significant amounts of noble gases and volatile radionuclides are released from the fuel. The failure of the control rods before the coolant channel failure is of less importance for the core melting process of CNA II compared to LWR because of the small masses of control rod material. Re-criticality is not a topic as re-flooding of a partly destroyed core is not calculated.

The upper (*Figure 3-8*) and middle (*Figure 3-9*) part of the fuel in the inner rings fail in two phases at ~5:30 h and ~6:30 h. The failure of the whole five inner core rings is completed at about 9:30 which is shown in Figure 3-10 as an abrupt decrease in temperature. At that time, the outer sixth core ring is still intact because of the low decay power level. Then, the core debris of the five inner rings accumulates on the moderator tank bottom. The first significant material relocation into the lower plenum after local failure of the moderator tank bottom of radial ring 3 happens at 11:17 h. Thereafter, the particulate debris and melt respectively are released into the lower plenum, start evaporating the remaining water and to heat-up/melt the lower core/moderator tank support grid and the filler pieces in the lower plenum from the top. The water of the lower plenum has fully evaporated at 12:14 h. As the melt spreads radially on the filler pieces it gets into contact with the RPV bottom head side wall as well, which heats-up (*Figure 3-12* and *Figure 3-13*). In the next almost 12.5 hours besides the core debris/melt, additional several tons of metallic melt are formed from the filler pieces and an additional significant amount of radionuclides are released into the RCS. After 19:44 h the outer radial core ring (6th Ring) totally collapses instantaneously due to the loss of support from below and relocates into the lower plenum. At this point, the mechanical loading to the RPV wall is very limited, determined by the mass of melt, filler pieces and the RPV bottom itself. This means that creeping effects are probably proceeding slowly, as the pressure difference across the wall is almost zero. Therefore, RPV failure is assumed in the MELCOR calculation if a segment of the outer RPV wall temperature reaches 1573 °K (1300 ºC). This consideration is also done because no pressure difference exists across the wall in the late phase of the analysed transient, which otherwise drives the 1D creep rupture model in MELCOR. The RPV failure is calculated to take place at 23:45 h after event initiation (temperature peak for segments 10 to 12 in *Figure 3-13*).

Inherent Safety Capabilities of the Plant

Several publications on SBO transients for PWR reactors can be found in the open literature. In particular, information on the expected timing of core uncover and beginning of core heat-up for a German PWR reactor of type KONVOI has been selected from PSA level 2 results of GRS to compare with the very specific characteristics of CNA II PHWR.

Table 3-7 shows some selected data comparing results of the Low Pressure SBO transient for CNA II, with a German PWR type KONVOI SBO transient considering depressurization as a result of creep rupture failure of the surge line after core melting has started. In contrast to CNA II, in the German PWR the hot leg accumulators will inject after depressurization and stop the core degradation for some time. This will happen even if the batteries are already depleted. If the accumulator injection was neglected, the RPV would probably fail up to one hour earlier.

As can be seen, the time that core uncover starts is almost one hour later for CNA II than for the Reference PWR and mainly caused by the failure of the pressurizer SV stuck open and the not functional accumulators. Two reasons can explain this. Firstly, the Primary System volume to core power ratio for CNA II doubles that of the reference PWR, which means that there is a larger mass of water to be evaporated to reach complete emptiness of the primary system. This, combined with the very low values of CNA II decay heat, results in a great advantage.

On the other hand, the existence of two separated systems, namely the coolant and the moderator system, has as a consequence a very specific means of heat evacuation of the decay heat produced in the fuel elements. As was seen in a previous section, fuel channels are nearly empty at about 3:30 h, but water level in the moderator tank is still at two thirds of the total height of the tank at that time. Although upper parts of core channels start to heat up (and eventually fail), the bottom of the fuel channels is in contact with water from the outside, which cools them mostly due to radiation heat transfer. The existence of such a heat sink derives in an increase of channel failure timing, which in the end derives into a later relocation to LP, and a larger time to implement Severe Accident Management Procedures prior to RPV breach.

Plant Characteristic	**CNA II PHWR Low Pressure SBO MELCOR**	**Ref. German PWR KONVOI Low Pressure SBO MELCOR**
Core Thermal Power [MW] (total)	2160	3765
Core Decay Power [MW] at Reactor Scram	129.6	241.9
Total volume of Primary System [m^3]	520	415
Primary System Volume to Core Power ratio [m^3/MW]	0.241	0.110
Event listed for CNA II PHWR / Event listed for PWR	time [h:m]	time [h:m]
SG dry-out	~2:00	0:57
PRZ SV stuck open failure	2:31	-
Start of uncover of core / moderator tank	3:00 / 3:00	~2:10 / -
Total uncover of core / moderator tank	3:30 / 9:00	~4:30 / -
Gap release from fuel	3:56	2:22
Beginning of Core Channel failure (PHWR) respectively Core failure & Debris Formation (PWR)	5:16	2:35
Failure of surge line due to creep rupture followed by accumulator injection	-	2:55
Fuel relocation to moderator tank bottom (PHWR) Molten pool formation in core (PWR)	> 5:30	>4 h
Failure of moderator tank bottom (PHWR) / core support plate (PWR) and debris relocation to lower head	11:17	5:02
RPV failure	23:45	6:22

Table 3-7: CNA II PHWR vs German PWR – Timing of events in an SBO

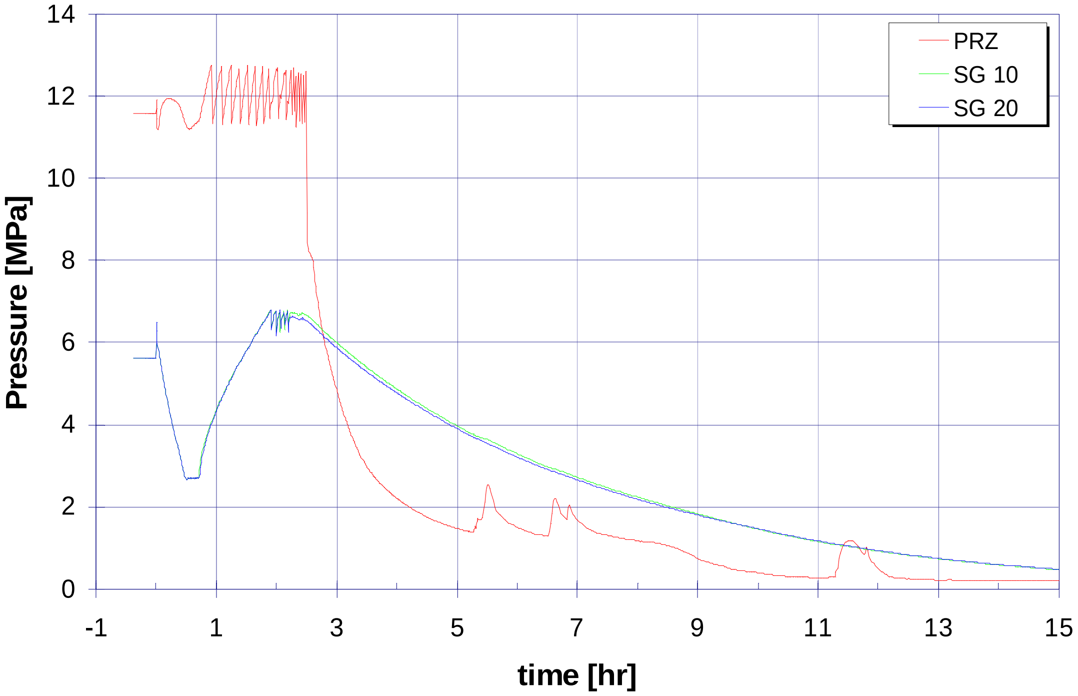


Figure N° 3-3: Pressure in pressurizer and SG10 and SG20.

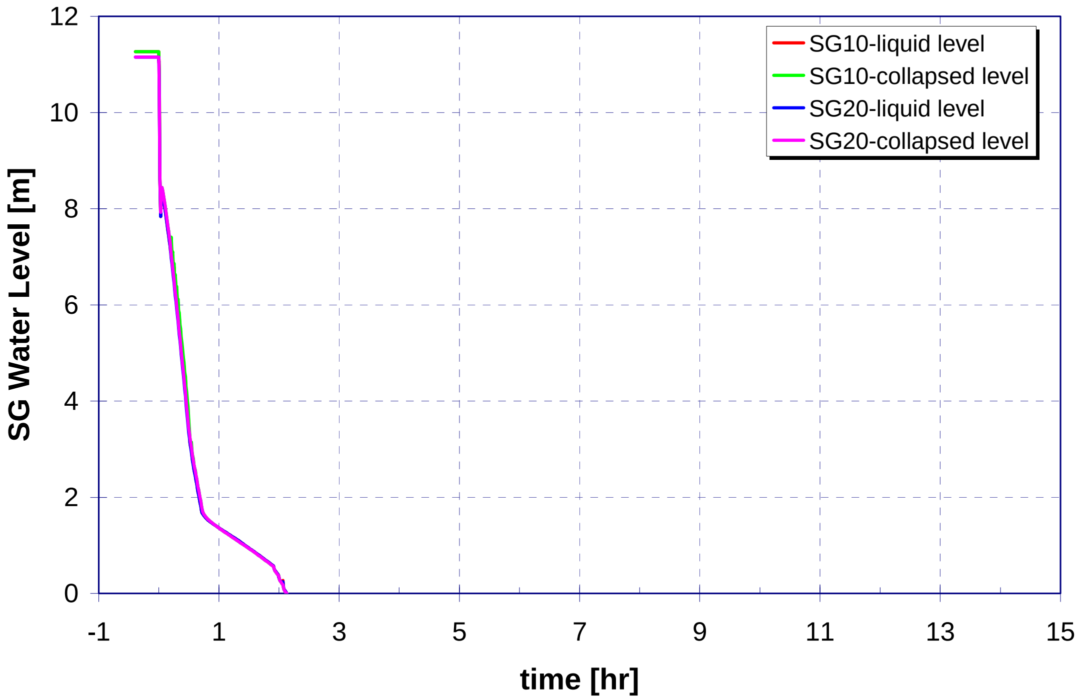


Figure N° 3-4: SG10 and SG20 water level.

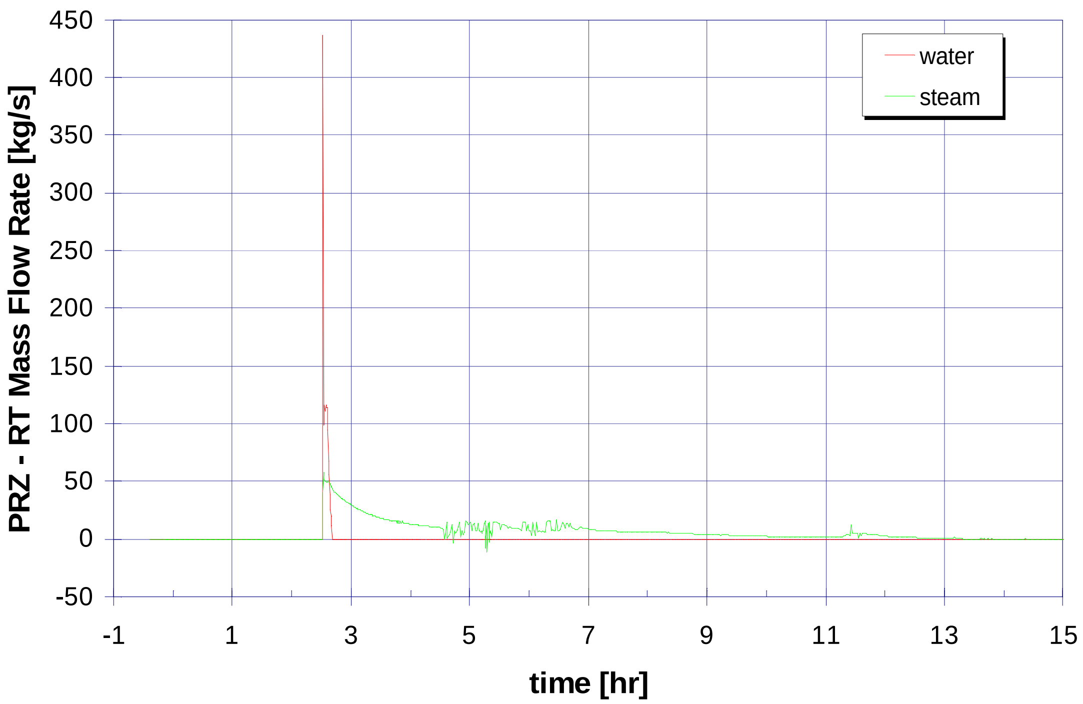


Figure N° 3-5: Flow rate through pressurizer relief tank - water and steam.

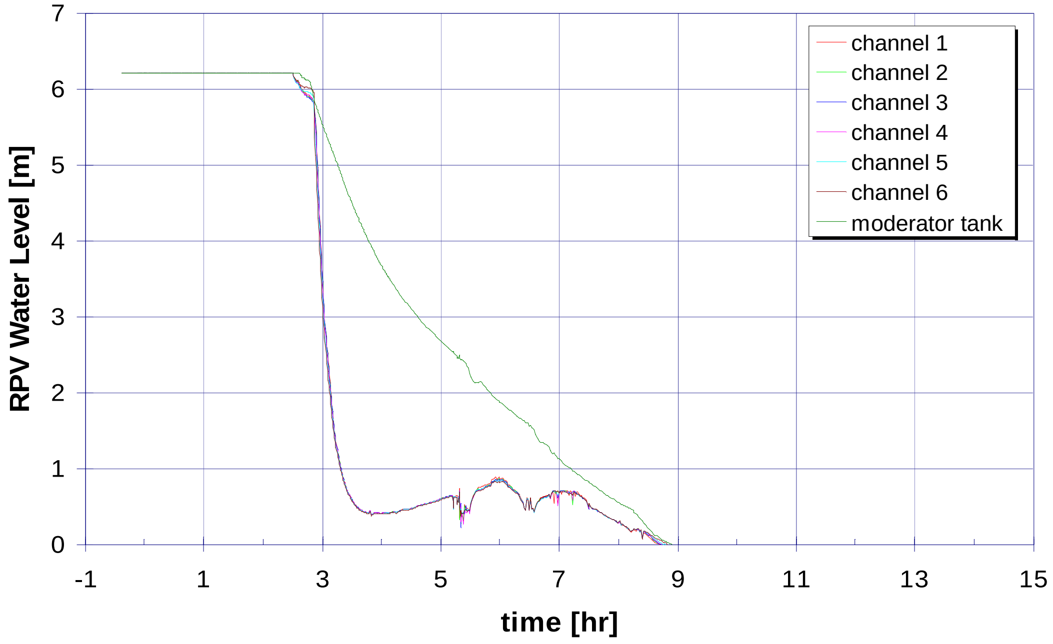


Figure N° 3-6: Water level in coolant channel rings 1 to 6 and moderator tank.

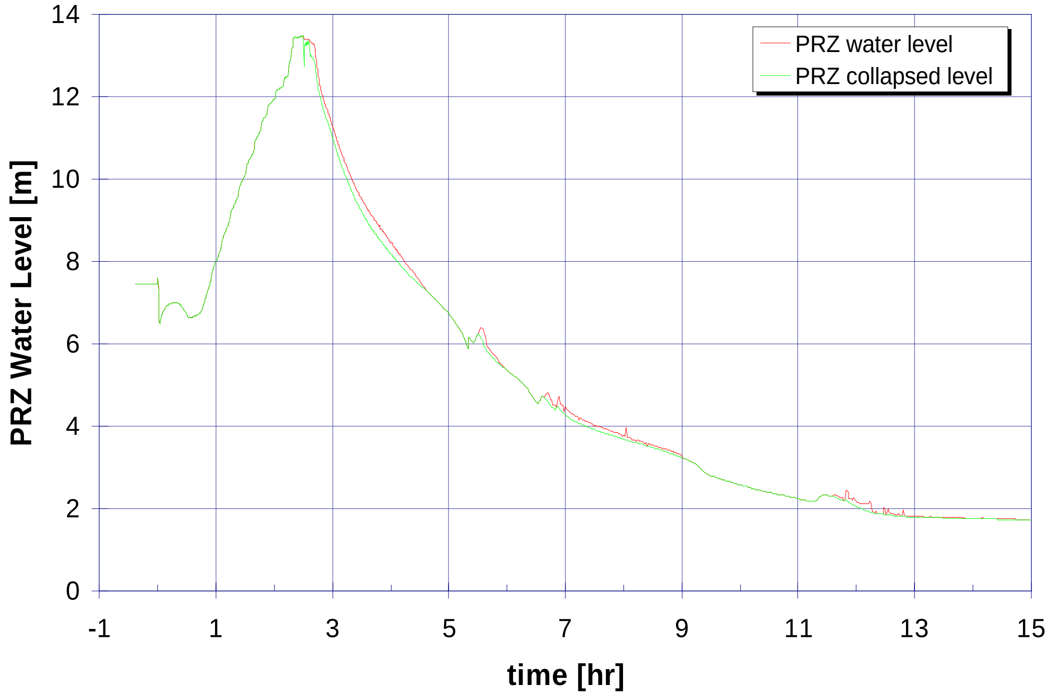


Figure N° 3-7 Pressurizer water level.

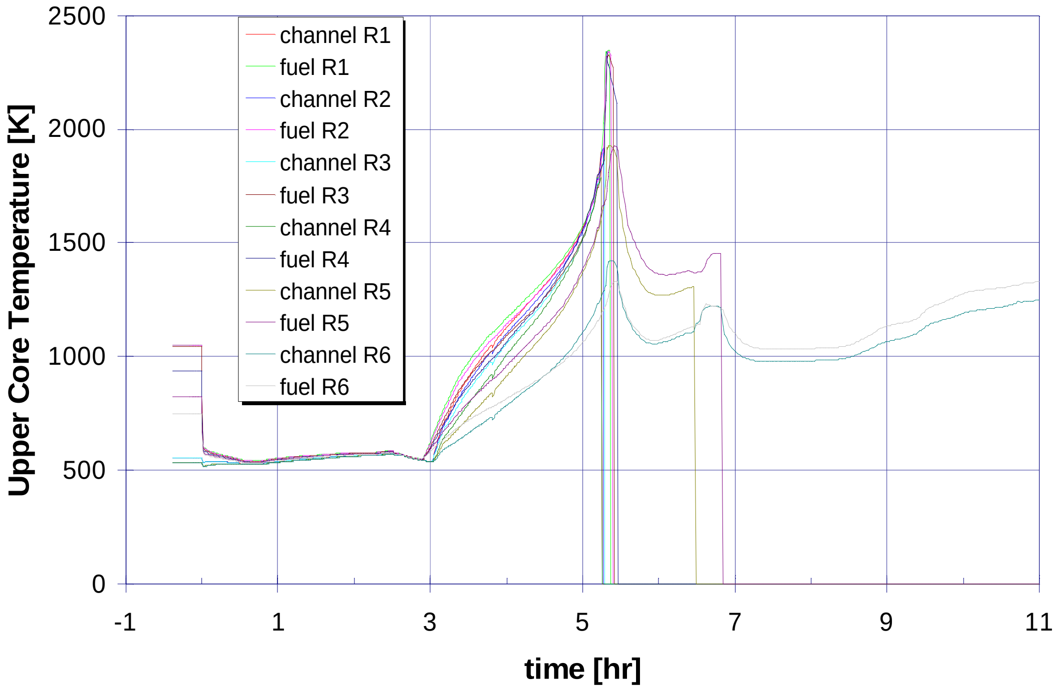


Figure N° 3-8: Temperature of fuel and coolant channels in R1 to R5 of Upper Core.

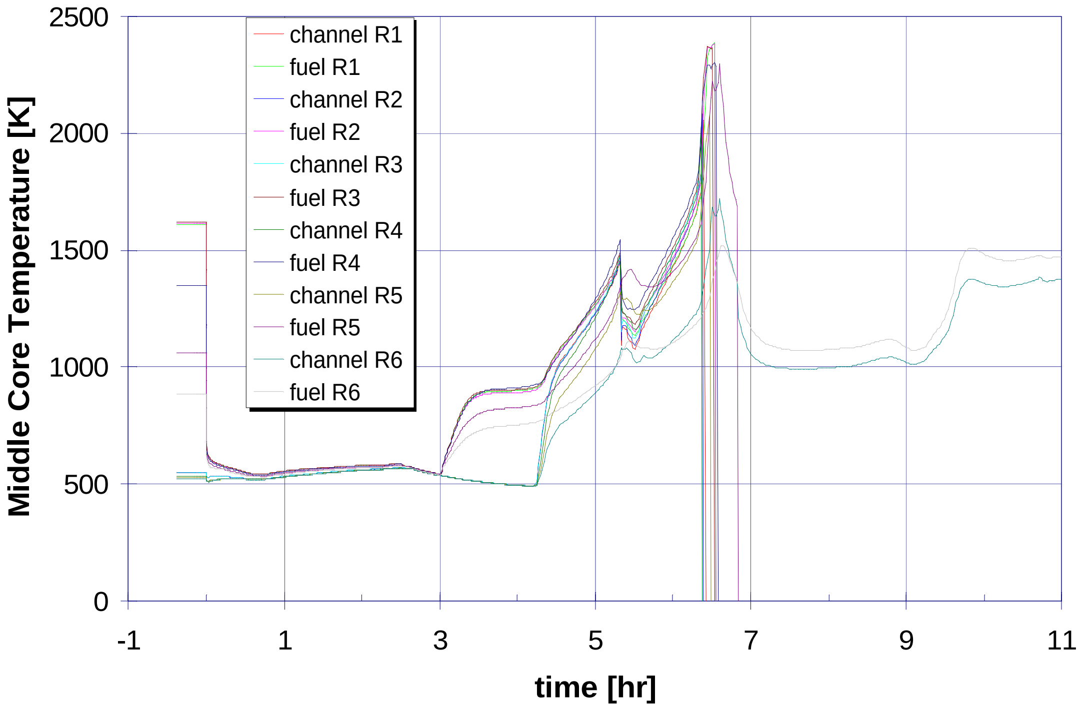


Figure N° 3-9: Temperature of fuel and coolant channels in R1 to R6 of mid core.

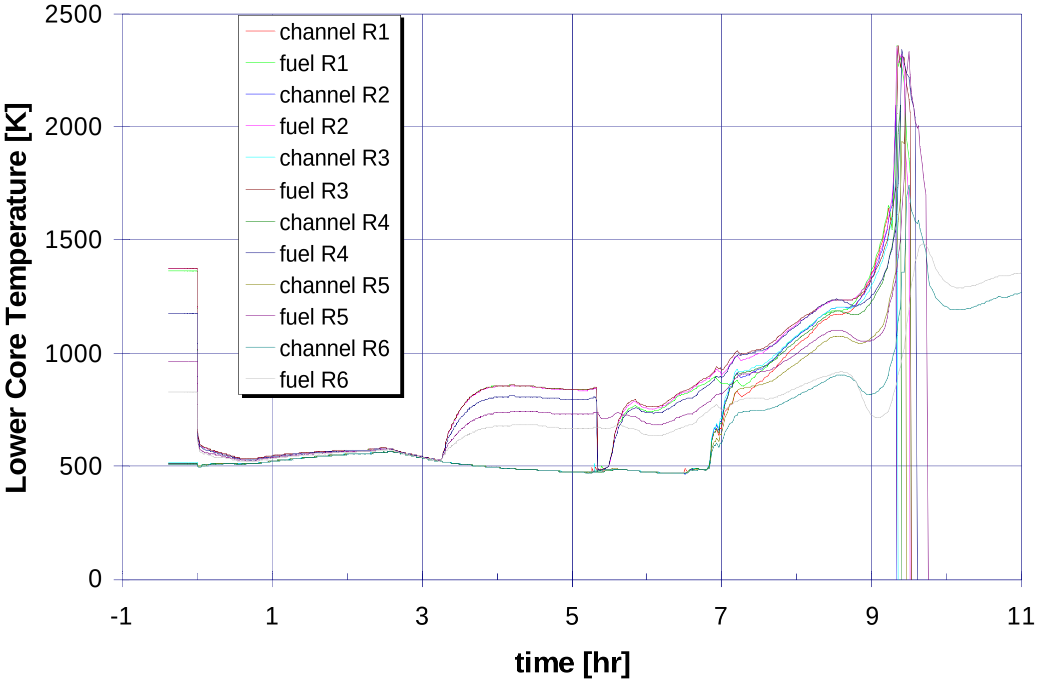


Figure N° 3-10: Temperature of fuel and coolant channels in R1 to R6 of lower core.

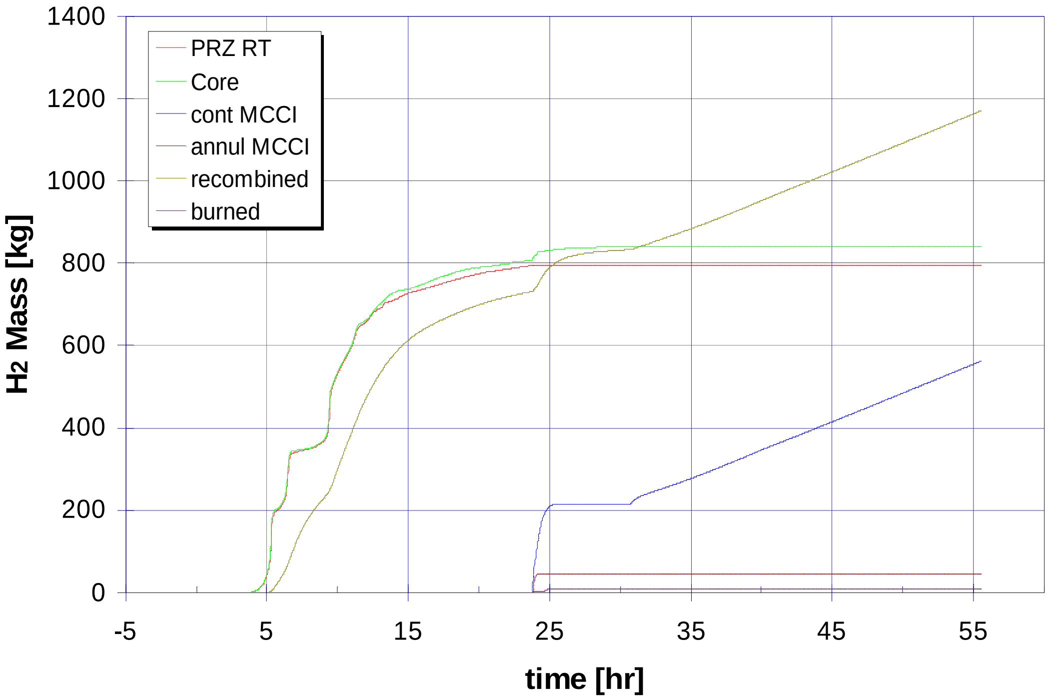


Figure N° 3-11: Hydrogen mass generated in core and from MCCI in containment, released from leak and recombined by PARs.

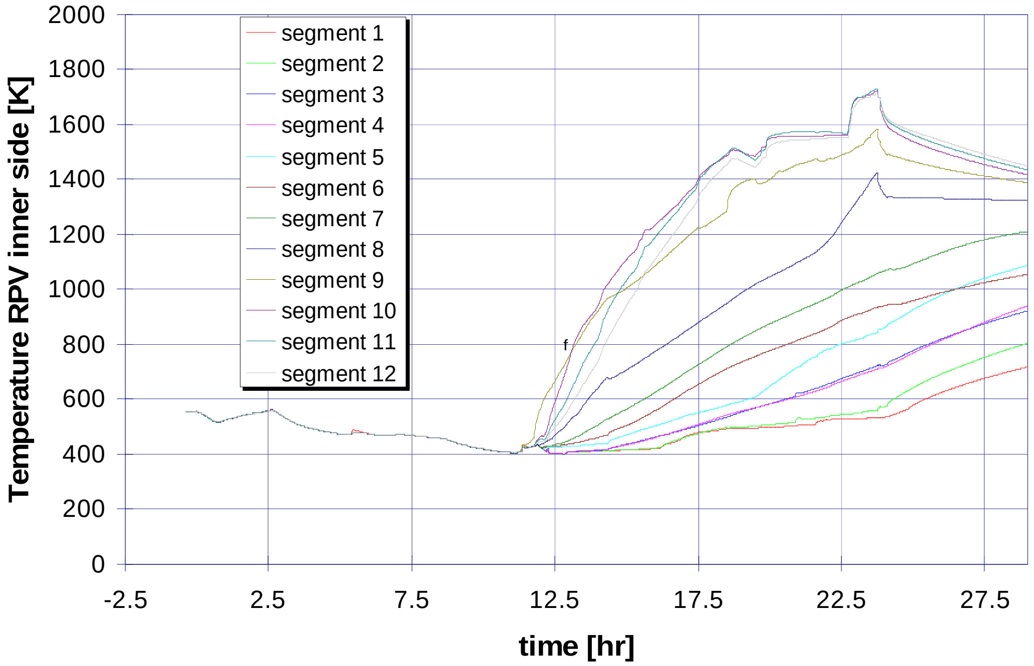


Figure N° 3-12: Temperature of RPV bottom wall at inner side.

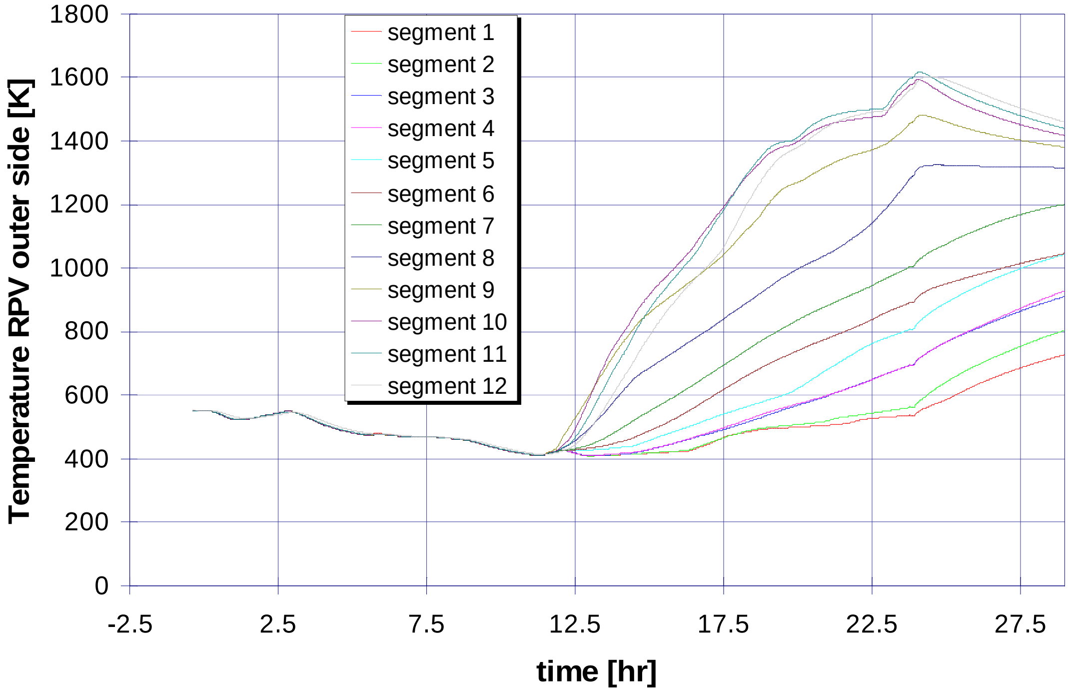


Figure N° 3-13: Temperature of RPV bottom wall at outer side.

3.1.1.2.2. Accident management measures and design features to protect the integrity of the confinement function after the occurrence of damage to the fuel

The following are the accident management measures and design features to protect the integrity of the containment after the occurrence of damage to fuel:

Hydrogen recombiners (PAR – Passive Autocatalytic Recombiners

The severe accident and source term analysis mentioned before were used to support recombiners design concepts. The current PARs arrangement is widely based on the design experience of the KONVOI NPPs.

An equal distribution of PARs in the containment taking symmetry of convection loops into account was one basis for the system design. General PAR locations are the following:

- Upper compartments (dome) for limitation of stratification,
- Equipment rooms for support of convection,
- Annular rooms for prevention of H_2 accumulation,
- Dead end / small rooms for prevention of H_2 accumulation.

An average recombination rate of 142 kg/h is reached by a total of 54 PARs.

The location of the PARs in the containment in the MELCOR input deck is characterized in *Figures 3-14 to 3-16.*

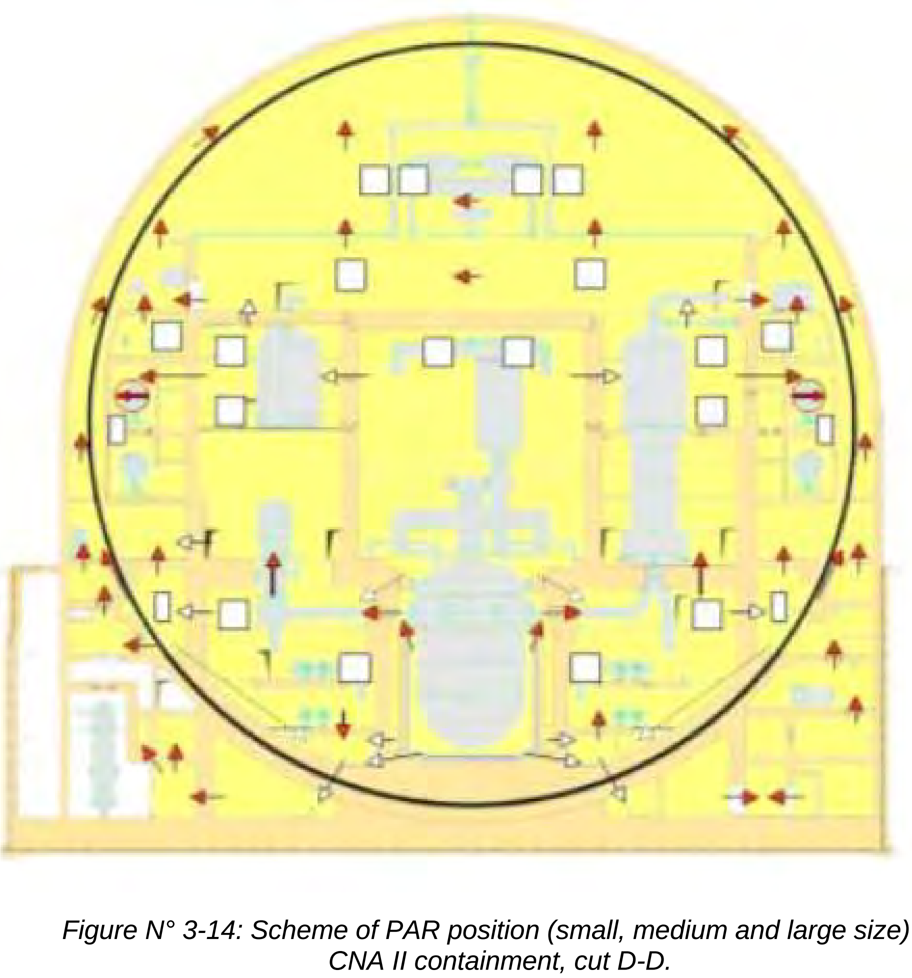


Figure N° 3-14: Scheme of PAR position (small, medium and large size) in CNA II containment, cut D-D.

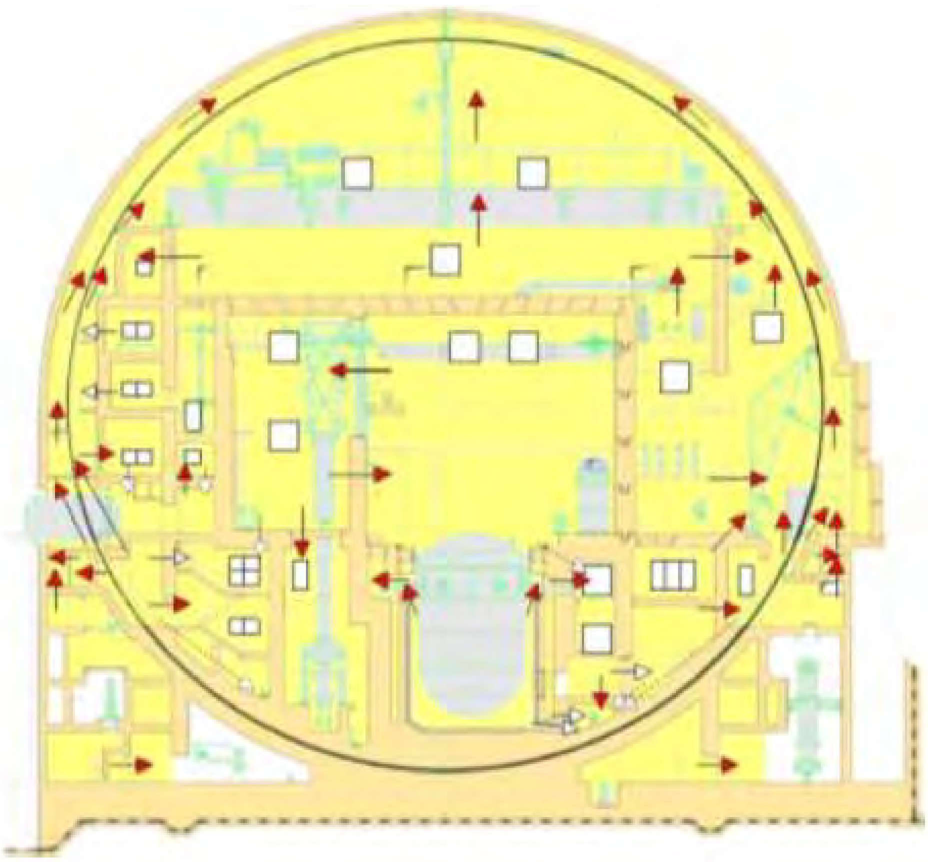


Figure N° 3-15: Scheme of PAR position (small, medium and large size) in CNA II containment, cut A-A.

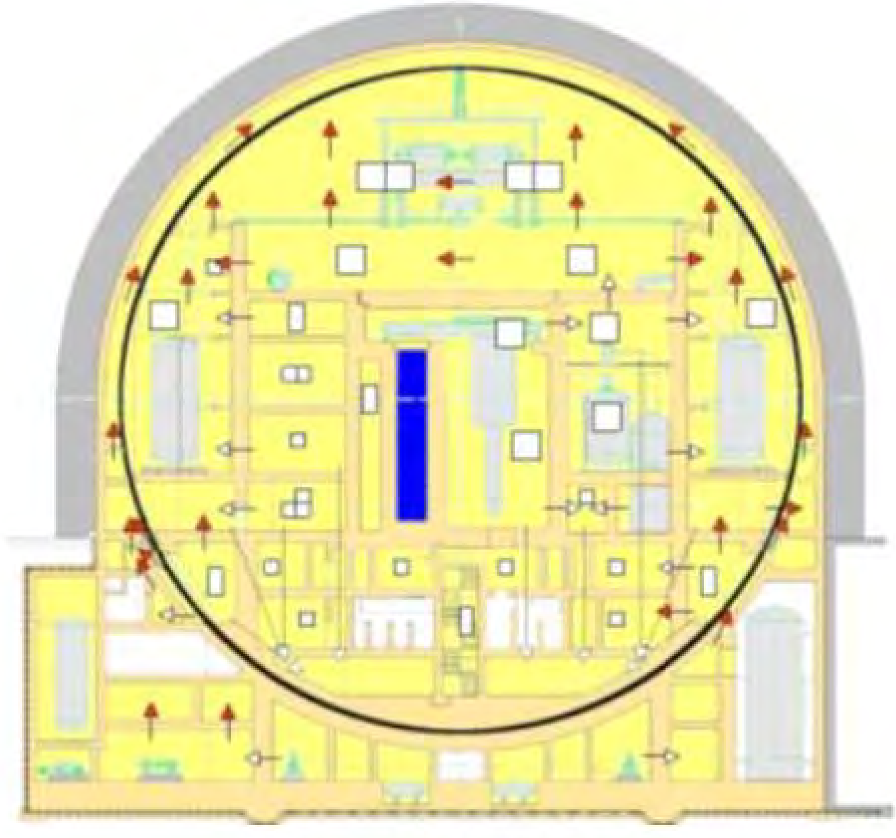


Figure N° 3-16: Scheme of PAR position (small, medium and large size PARs) in CNA II containment, cut B-B.

External Cooling of the RPV

As was mentioned before, a number of relevant CDS have been identified as a result of PSA Level 1. In the PSA Level 2, MELCOR calculations have been done for each of the relevant CDS in order to assess the severe accident progression.

As a result of the analysis, it was concluded that the external cooling of the RPV is a solution to avoid the RPV failure.

The SAMP will include the external cooling of the RPV as an additional improvement. This issue is under development and some preliminary results are shown below:

In addition, the following (old) sequences are still running with the MELCOR code:

- 0.1A LOCA (95 cm² / moderator loop 10, section T) with failure of four safety injection system pumps (JND) and external cooling of RPV only by opening the connection to the sump
- 0.1A LOCA (95 cm² / moderator loop 10, section T) with failure of four safety injection system pumps (JND) and external cooling of RPV with increased water inventory

The base case scenario without external cooling has a RPV failure after about 77,000 sec. This is the case with the old MCP modeling. Both cases with external cooling of the RPV already reached ~240,000 sec of process time w/o any RPV failure while most of the filler pieces in the lower plenum are molten. Taking into account the simplifications of the MELCOR modeling, the analyses indicates that an opening of the two water connections between the cavity and the sump early in an accident would significantly delay the RPV failure or prevent it. The results also indicate a continuously significant steam concentration and pressure increase in the whole containment, after the water saturation temperature is reached followed by a slow continuous decrease of the cavity/sump water level. In the case with an open cavity connection but no additional water supply to the containment, the upper of the RPV starts to heat up in the late phase. This indicates that a RPV failure might happen sometime; even if the remaining power in the core is very low at that time. In the other case, with increased water amount in the containment, the RPV lower head stays fully submerged in water. As the heat transfer area from the RPV head is large (50 – 100 m²), depending on the water level, a very low probability of RPV failure has to be expected.

A deeper analysis is needed on how to remove the heat released from the RPV to the evaporating water in the cavity from the containment, as the steam concentration and the pressure are continuously increasing. The steam condensation on the concrete walls and the containment itself do not seem to be enough.

First draft results are enclosed in the following *Figures 3-17 to 3-30*. The time axis is always in seconds.

Results of cases with open cavity connection to sump

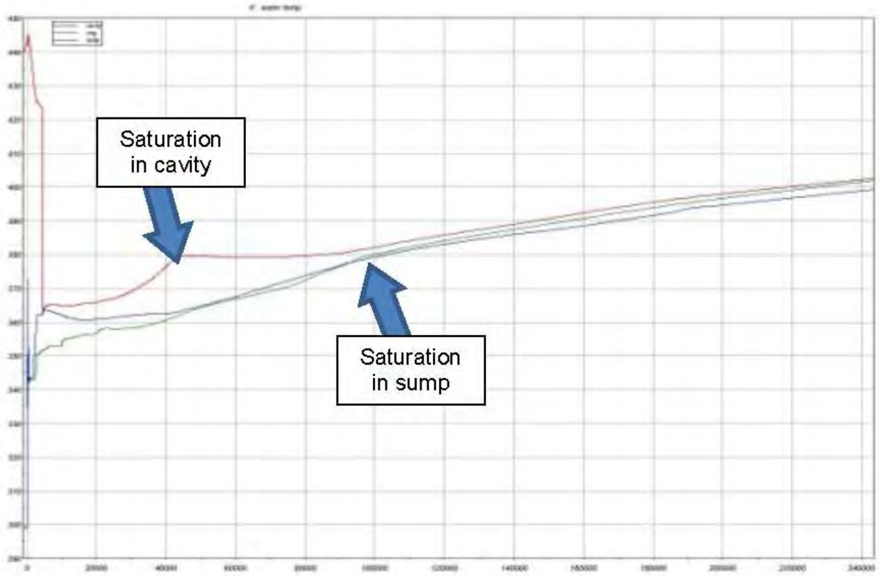


Figure N° 3-17 Water temperature [K] in cavity, sump and cavity ring; time in [s], case with open cavity connection.

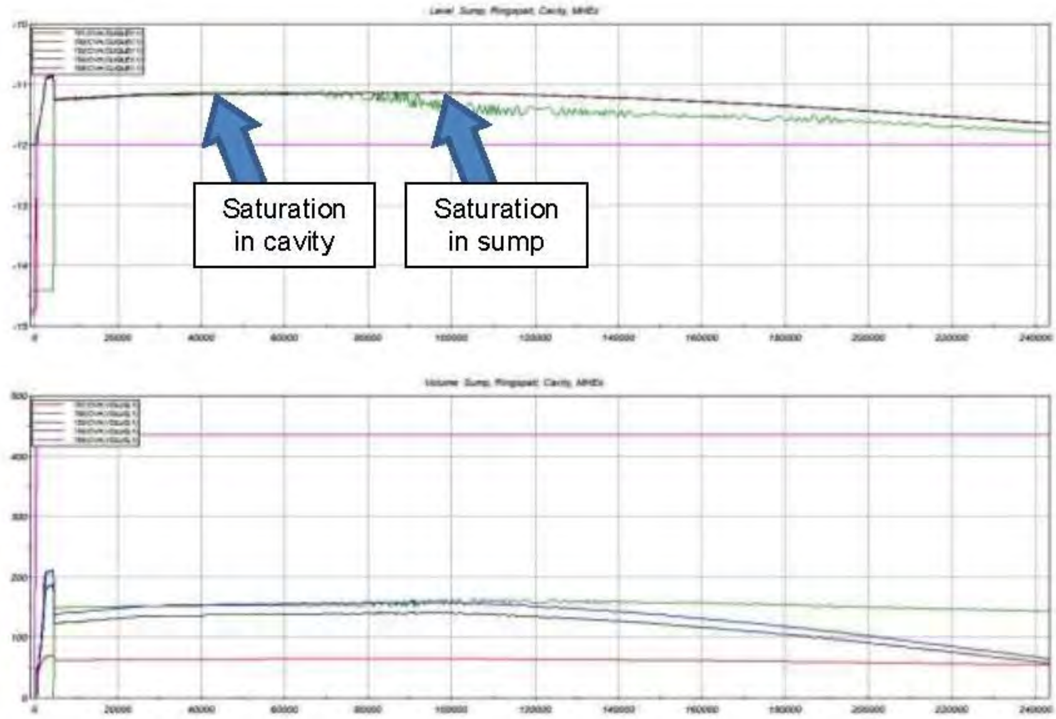


Figure N° 3-18: Water level [m] and water volume [m³] in cavity, sump, cavity ring and lower mod rooms; time in [s], case with open cavity connection.

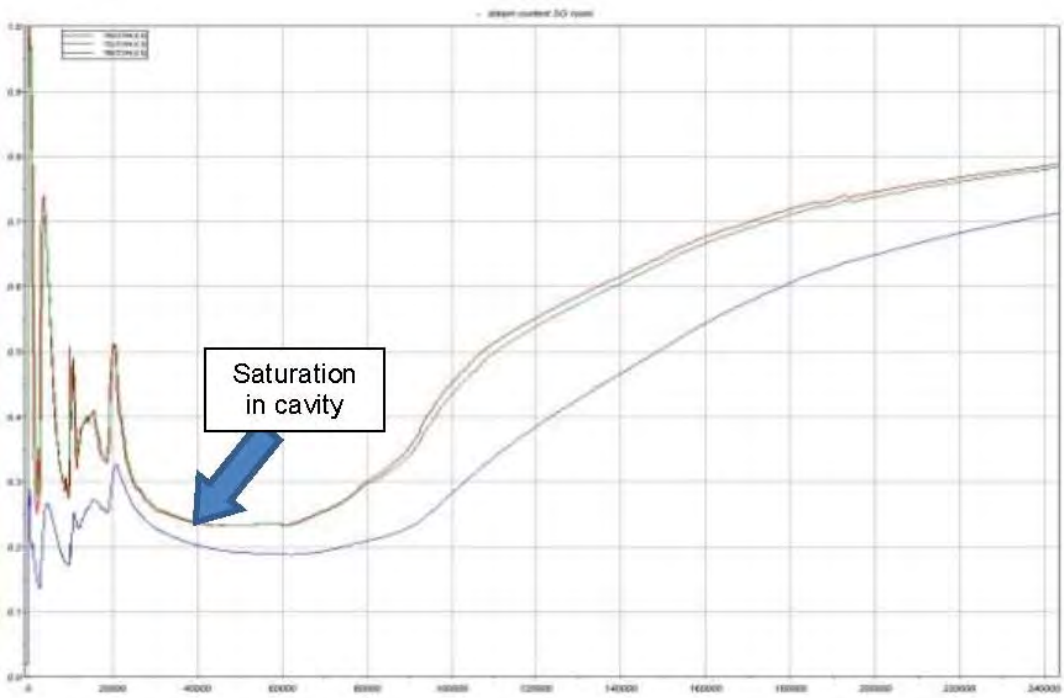


Figure N° 3-19: steam concentration [-] in on SG box (red, green) and dome (blue); time in [s], case with open cavity connection.

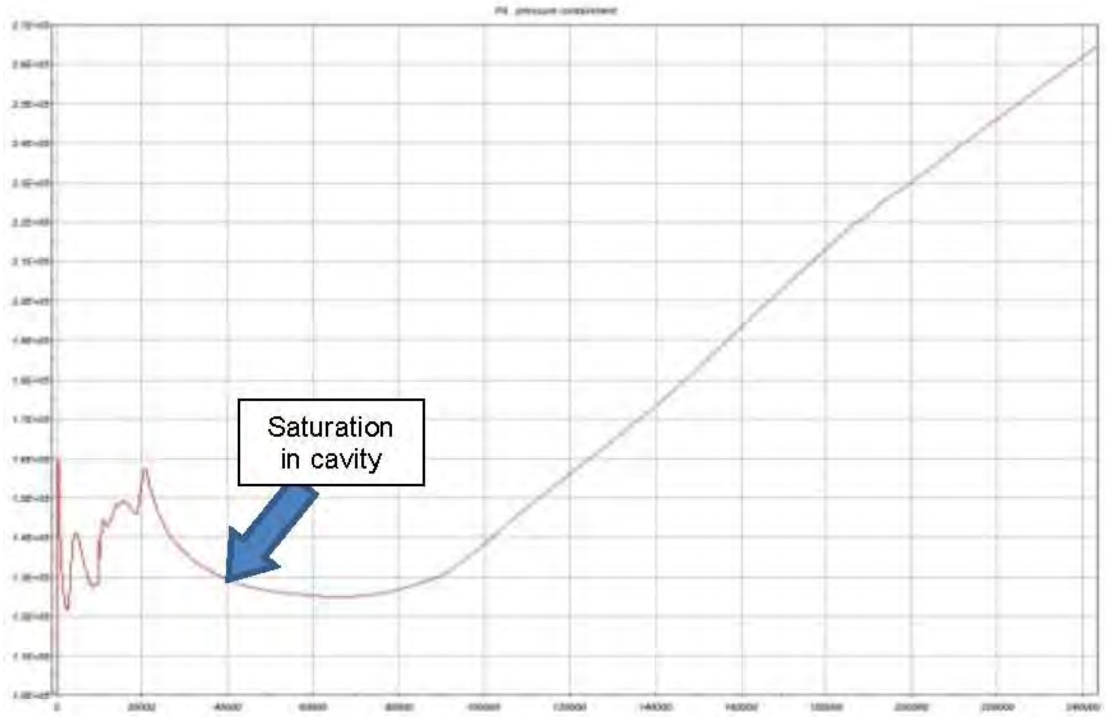


Figure N° 3-20: Containment pressure [Pa], time in [s], case with open cavity connection.

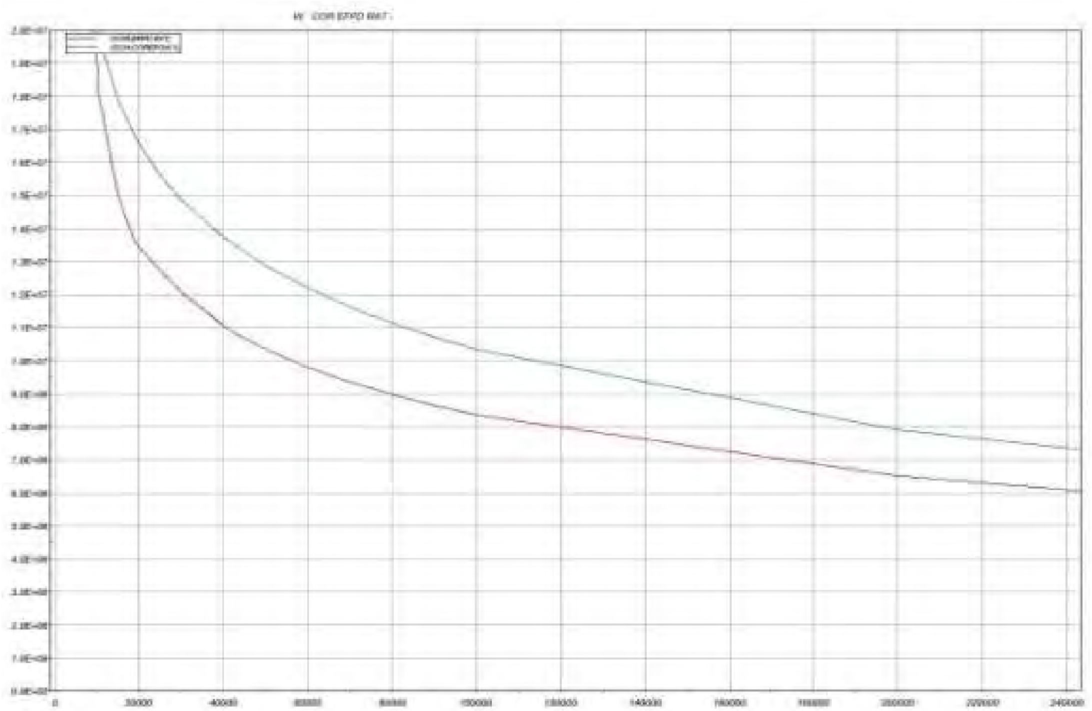


Figure N° 3-21: Total (green) and actual (red) core power [MW], time in [s], case with open cavity connection.

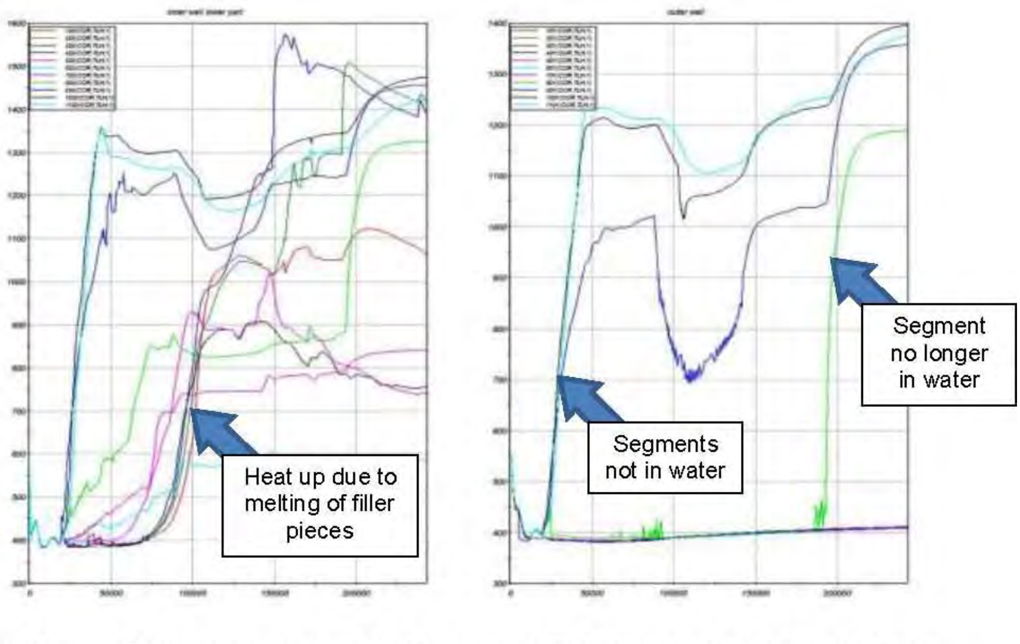


Figure N° 3-22 RPV wall temperature in 11 segments (left - inner wall; right – outer wall), case with open cavity connection.

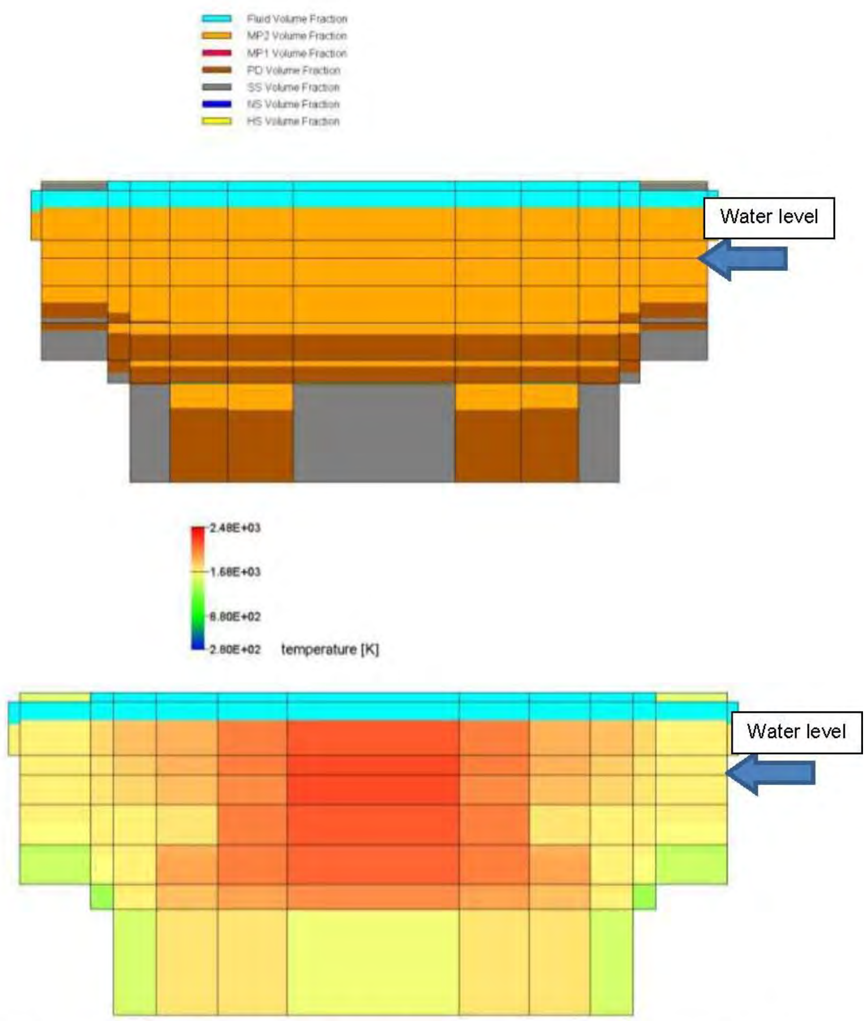


Figure N° 3-23: Melt configuration and temperature, case with open cavity connection.

Results of cases with open cavity connection and increased water amount in sump for external cooling

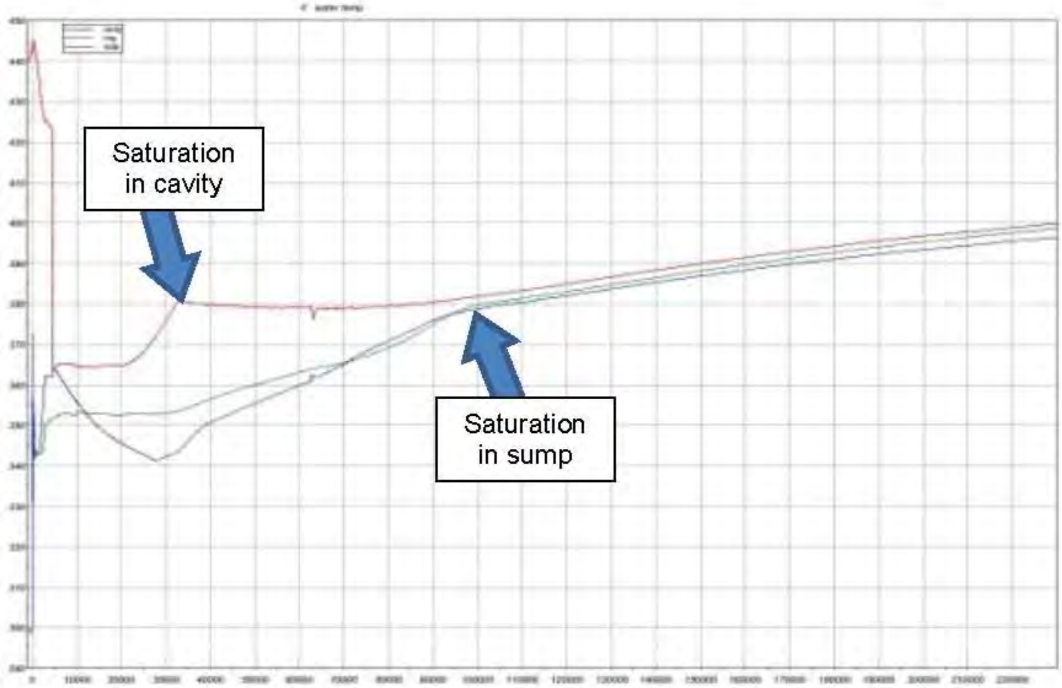


Figure N° 3-24: Water temperature [K] in cavity, sump and cavity ring; time in[s], case with open cavity connection & increased water amount.

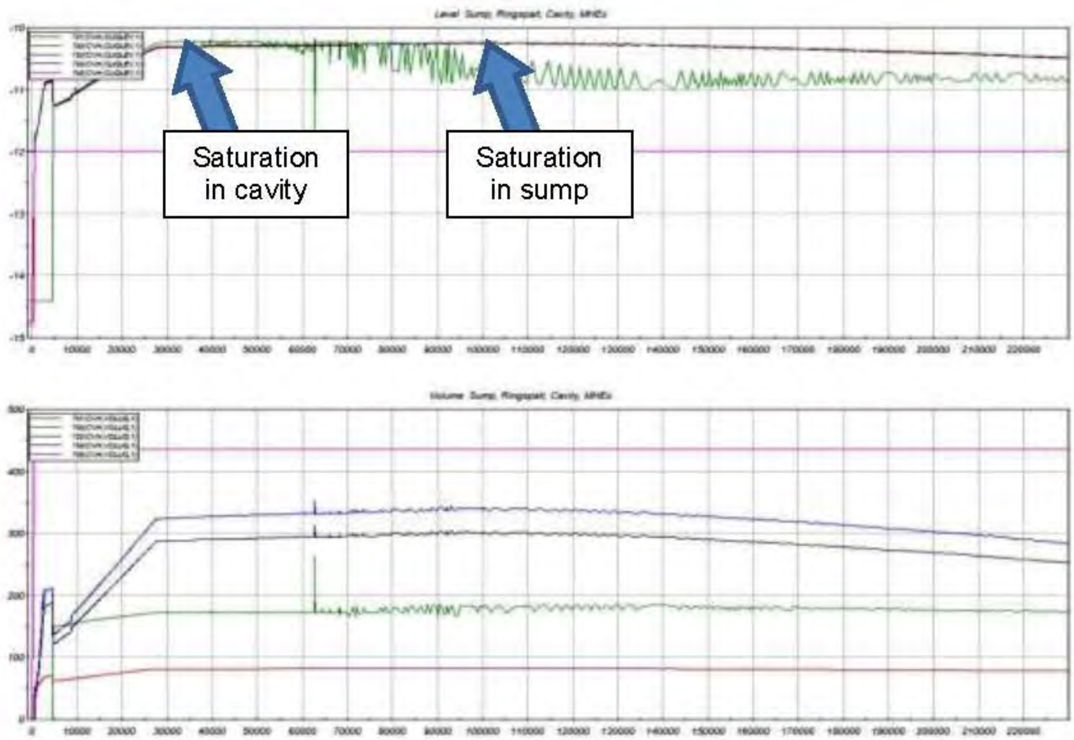


Figure N° 3-25: Water level (upper, [m]) and water volume (lower, [m³]) in cavity, sump, cavity ring and lower mod rooms; time in[s], case with open cavity connection & increased water amount.

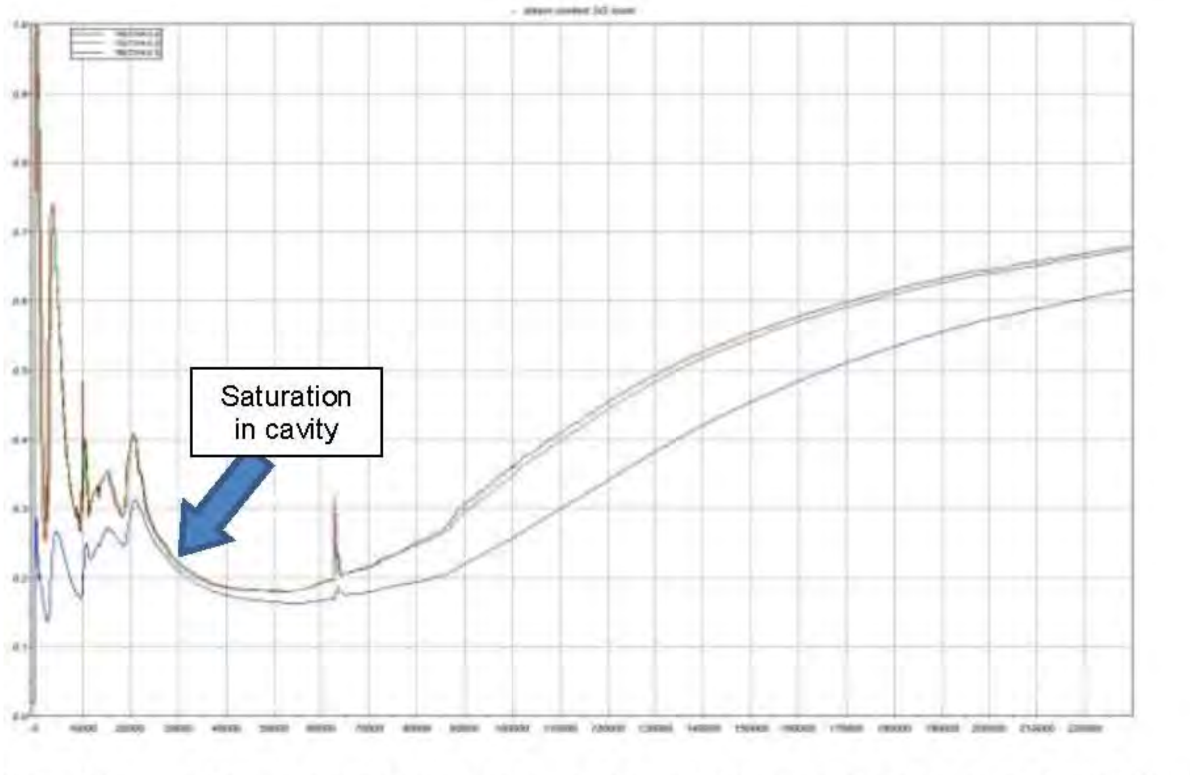


Figure N° 3-26: steam concentration [-] in on SG box (red, green) and dome (blue); time in[s], case with open cavity connection & increased water amount.

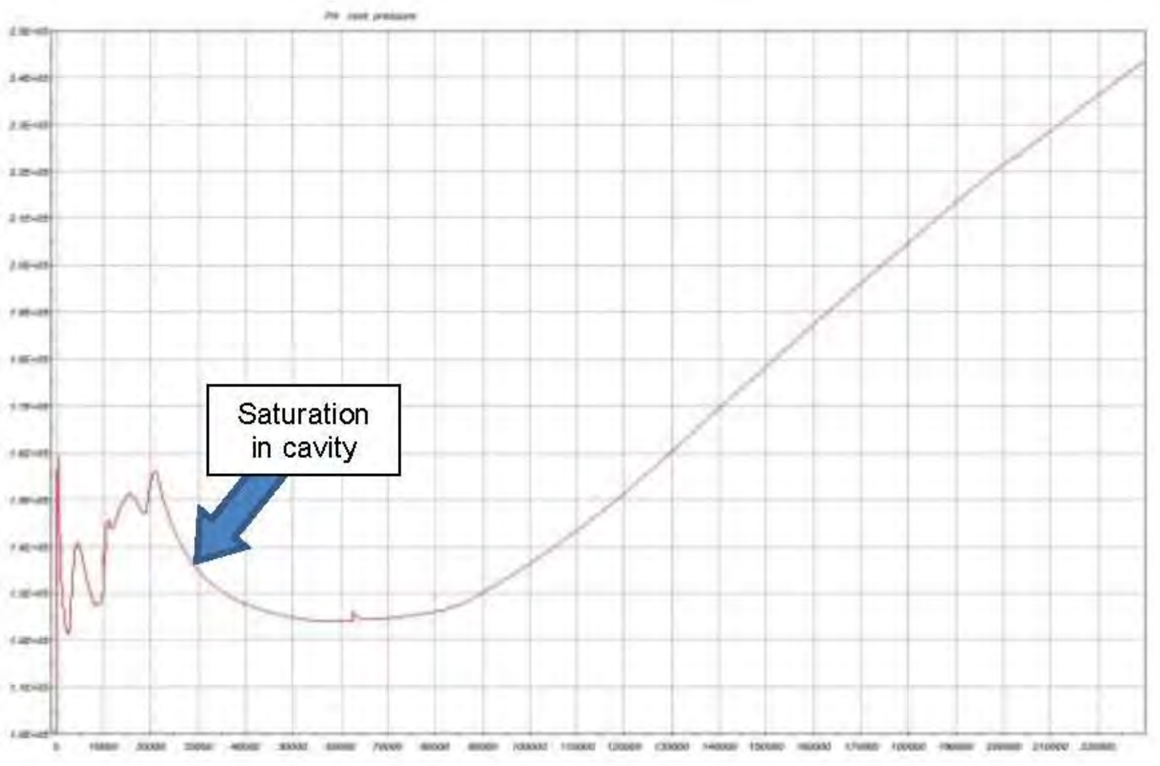


Figure N° 3-27: Containment pressure [Pa], time in[s], case with open cavity connection & increased water amount

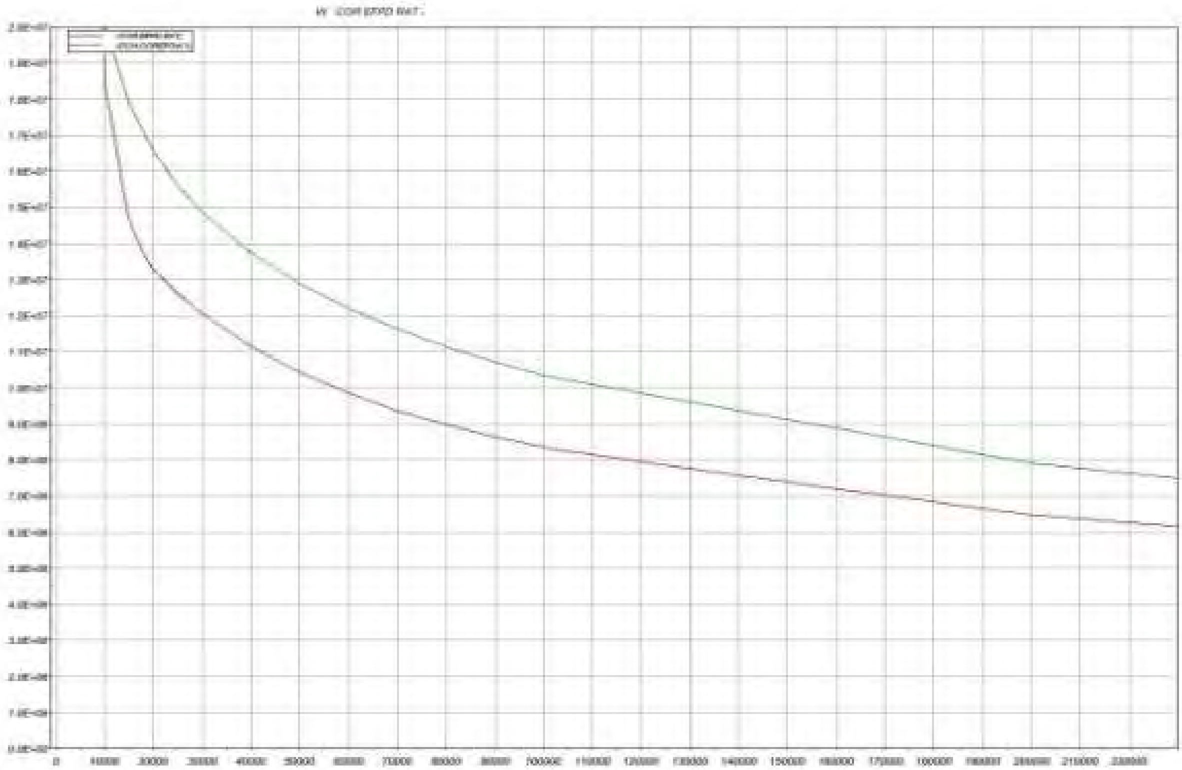


Figure N° 3-28: Total and actual core power [MW], time in[s], case with open cavity connection & increased water amount

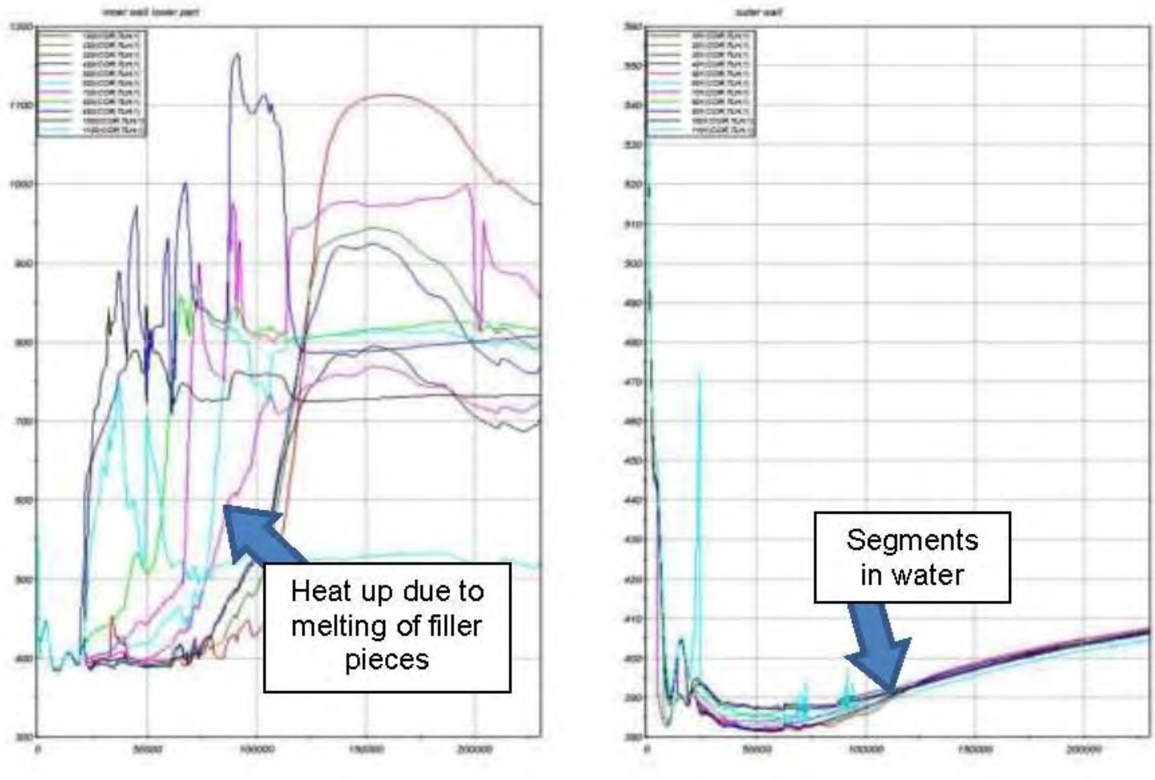


Figure N° 3-29: RPV wall temperature in 11 segments (left - inner wall; right – outer wall), case with open cavity connection & increased water amount.

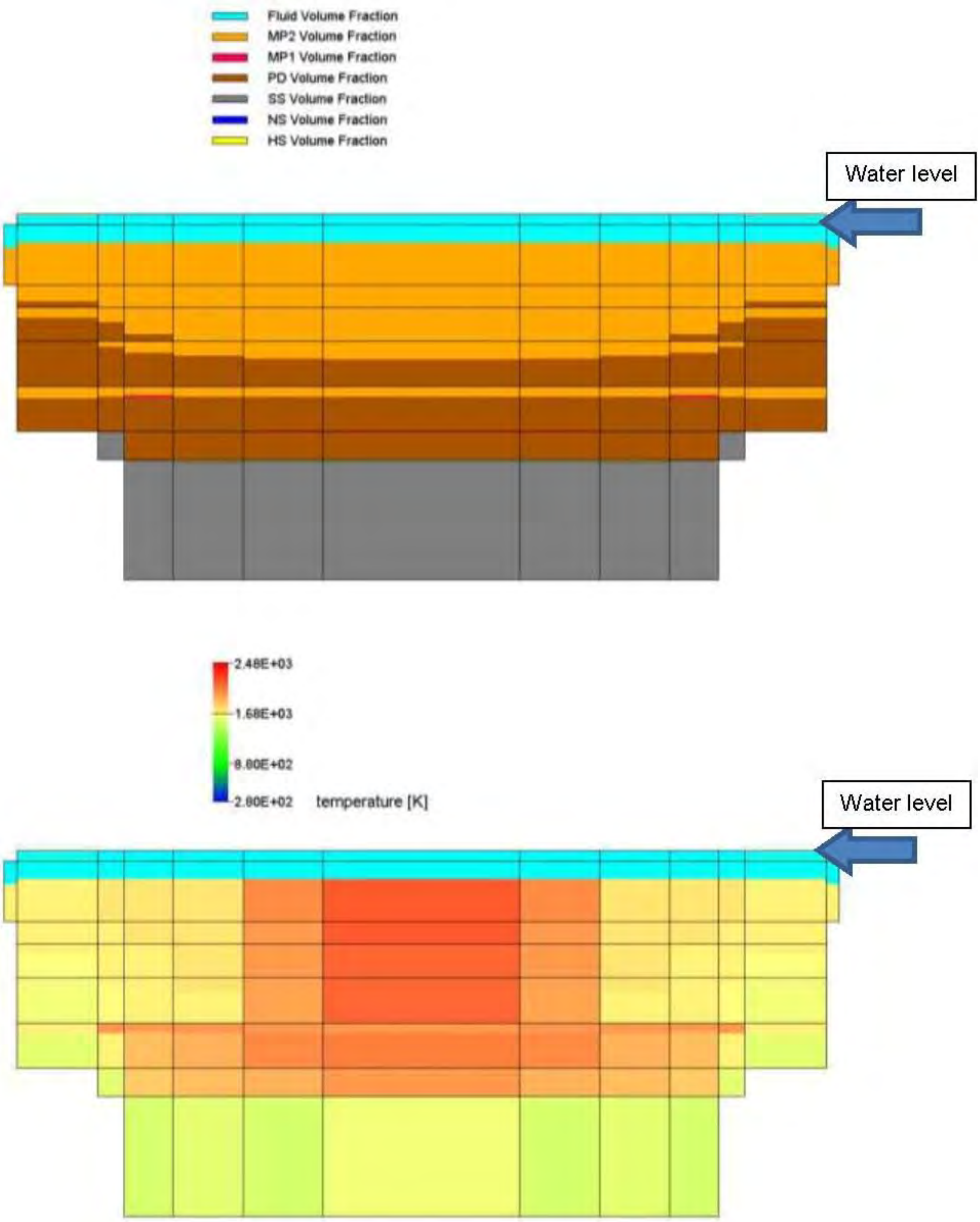


Figure N° 3-30: Melt configuration and temperature, case with open cavity connection & increased water amount.

3.1.1.2.3. Accident management measures currently available to deal with the successive stages of a scenario of loss of cooling function of the fuel storage pools:

Loss of adequate radiation protection (water column shield).

The Fuel Element Pool Cooling System (FAK) consists of three pumps (two in operation and the third is in stand-by), which are fed from the emergency network using three different trains.

The system also has two heat exchangers to dissipate heat generated from the fuel element pools using the secured cooling water service (PEB).

The pool cooling system and the spent fuel element pools are in the pool building (UFA).

If the cooling system of the pools or the secured cooling water service were completely lost, an initial phase of water heating is produced till saturation, with an initial increase in level, followed by evaporation and a subsequent decrease in level.

The time in which these events occur depend on the number of fuel elements deposited in the pools and their decay heat.

To evaluate these times, the most unfavorable conditions are considered to occur simultaneously, i.e.:

1. Level of 16.14 m in pools (minimum level),
2. Maximum number of fuel elements deposited in a pool,
3. Entire core downloaded to pool as a result of any operating condition that requires it.

This last condition is very unlikely because heavy water reactors undertake refueling with the installation in operation. As a result, the core download is not a systematic chore as in pressurized water reactors and abnormal maintenance conditions that could originate this operation are not likely.

The time sequence for a total refrigeration loss situation will be as follows:

- t = 0, initial level 16.14 m, Temperature 47,48 °C (cooling with a single train), loss of cooling
- t = 54 hours to reach boiling point.
- t = 37.5 hours + 54 hours to evaporate 1 m of water, leaving the pools with 15.14 m (2.59 m of water above the fuel elements).
- t = 134.5 hours +54 hours, for the top of fuel elements to be exposed (active part).

Considering that the core continues inside the pressure vessel, and the pools are filled with water and without cooling, the time frames are greater, since the power to dissipate is lower:

- t = 0, initial level 16.14 m, temperature 41.23 °C (cooling with a single train), loss of cooling.
- t = 90 hours to reach boiling point.
- t = 55 hours + 90 hours to evaporate 1 m of water, leaving the pools with 15.14 m (2.59 m above the water EECC).
- t = 197 hours + 90 hours, for the top of fuel elements to be exposed (active part).

Analysis

The analyses were performed on the worst case scenario (evacuated core and pools full of spent fuel elements). While this scenario has a low probability of occurrence, based on these conservative assumptions, shorter times are obtained, allowing defining conservatively the timing of actions to be taken to restore the cooling of the pools.

Heating until saturation

Before the event, a unique cooling train was assumed. Consequently, the temperature of the pool was 47.48 °C. The high temperature (38 ºC) and very high temperature (41 °C) alarms were initially present.

Until about 60 °C, manual actions are possible, because the conditions at the pool building have not yet been compromised. The estimated time till this temperature is reached is approximately 13 hours, then access to the building will be limited and will require personal protective equipment.

Given that the design temperature of the fuel element pools is 80 °C, the time without cooling needed to reach it, will be approximately 33.6 hours from the beginning of the loss of the cooling system.

Drop in the water level of the pools due to evaporation

Once the temperature reaches 100 °C, the water in the pools will start to boil. Evaporation of the water inventory will occur, producing a decrease in its level, but the cooling function of the fuel elements is expected to be maintained.

The drop in the water level in the pools will activate the low-level alarms and, in this situation, its replacement is scheduled to be performed by the pools purification system (FAL), while the restoration of the cooling trains is attempted.

As monitoring instrumentation, water temperature instrumentation is available for the fuel element pools that, as part of the post-accident instrumentation, are qualified for the conditions of design basis accident (LOCA).

Approximately 91.5 hours are available until 2.59 m of shielding is obtained, to be able to recover the water level and the cooling system.

Cliff edge effects

- In 91.5 hours the condition is reached when the biological shield is insufficient (1m below the minimum level / 2.59 m above the fuel elements).
- The time that the design temperature (80 °C) is reached in the pools (most unfavorable conditions: pools filled + fresh core) is approximately 33.6 hours.

Exposure of the top part of the fuel elements

It is considered highly unlikely that the exposure of the active part of the fuel elements may be reached, due to the extremely long times involved for this situation (188.5 hours), which allows actions leading to the recovery of the water level and the cooling of the pools to take place.

Degradation of the fuel elements

As mentioned in the previous section it is extremely unlikely that fuel element degradation be reached, because there will be 188.5 hours to take action to recover the cooling system.

Robustness

As mentioned in previous sections and taking into account that the worst case scenario (pool filled with evacuated reactor core) is very unlikely to occur, if the fuel element pools do not have the possibility of some cooling as in the case of an SBO, given the large volume of water available and its associated thermal inertia, it is estimated that there is sufficient time to recover their cooling system.

3.1.1.2.4. Planned improvements

Based on the evaluations, the following planned improvements have been identified (see item 2 for more detail):

- Additional power supply.
- Removal of residual heat via steam generator (SG): Injection of additional water to the SG.
- Accident Management Programme

Additional power supply

The major improvements planned regarding the additional power supply, are the following:

- Review of procedures for extending the use of the Diesel Generators (DG) using additional fuel tanks: the maintenance and testing programs were reviewed, including verification of fuel tank level, water and lubricants. It must be ensured that the calculations presented regarding the minimum necessary provisions, are maintained by the system of inspections and tests. This will be implemented by 2013.
- The possibility of connecting one of the three new CNA I DGs (EPS, 3.4 MW each) to CNA II is being evaluated. Having this connection available will allow the following options for residual heat removal:
    - Primary side: main cooling chain (RHR) including auxiliary components, and / or

- Secondary side with residual heat removal by steam generators (SG) connected to a startup and shutdown pump with the corresponding station relief valves. The reposition of water for maintaining the long term cooling must be analyzed in this case.

This improvement will be implemented by 2015.

- Maintain current cooling towers as an alternative mode of cooling of two of the CNA II diesel. This improvement will be implemented by 2015.
- Disconnect unnecessary electrical loads to increase battery life. This improvement will be implemented by 2015.
- Analysis of the availability of external power supply lines including the interconnection with the 220 kV and 500 kV high voltage lines. This improvement will be implemented by 2013.
- Connection to mobile diesel generator (MDG). This improvement will be implemented by 2014.
- Using the auxiliary boiler fuel to increase the autonomy of the DGs. This improvement will be implemented by 2014.

Heat removal through the steam generators (SG)

Heat removal is performed through the injection of additional water to the SG and venting via SG safety relief valves: In this case, improvements are intended to be incorporated to extend the water supply during a longer time to the SGs consisting in the use of the mentioned MDG and the supply of water from an alternative reservoir. The possibility that this reservoir consists of water intakes from the groundwater are being evaluated. It is planned to implement this improvement by 2014.

Accident Management Programme

Accident Management Programme development to beyond design basis accidents and severe accident :

For the postulated events, success paths are expected to be defined, i.e. define the systems that are needed to act to successfully meet the scenarios caused by these events. The following actions are planned:

- Reactor pit flooding from the spent fuel pool with the FAL system, with reposition of water from the demineralized water tank (GHC), from the firefighting system or from an external reservoir.
- Closing the switchgear building (UBA) and use of portable purification equipment.

These improvements will be implemented by 2015.

3.1.1.3. CNE

The CNE acquired the CANDU Owner Group (COG), generic guidelines for severe accidents developed for CANDU reactors and based on guidelines for Severe Accidents developed by company Westinghouse of U.S. Then it hired the company CANDU Energy of Canada for the development of the Specific Guides for Severe Accident applicable to CNE. The project consists of four sets of guidelines:

- SACRGs: Severe Accident Control Room Guidelines
- SAGs: Severe Accident Guidelines
- SCGs: Severe Challenge Guidelines
- SAEGs: Severe Accident Exit Guidelines

The project is being developed in two stages. The scope for the first stage corresponded to the development of the SAGs, in the following priority order:

- SAG-1: Inject into the Heat Transport System
- SAG-2: Control Moderator Conditions
- SAG-3: Control Shield Tank/Calandria Vault Conditions
- SAG-4: Reduce Fission Product Releases
- SAG-5: Reduce Containment Hydrogen
- SAG-6: Control Containment Conditions
- SAG-7: Inject into Containment

This phase has been completed and the guidelines are currently being analyzed by staff of the Life Extension Project of the CNE.

3.1.1.3.1. Accident management measures currently available for the reactor core protection at different stages of a scenario of a loss of core cooling function

The reactor of the CNE has a robust design for the prevention of severe accident, which includes both preventive and mitigation features. The engineering provisions are focused to: prevent core damage, terminate the progress of core damage, retain the core inside the calandria vessel, maintain containment integrity, and minimize radioactive releases.

In the CANDU reactors, the accident progression with or without core damage is strongly influenced by the design of the core. In particular, the low pressure moderator in the calandria vessel surrounding the pressure tubes, and the large volume of light water in the calandria vault, which acts as a passive heat sink and adds a significant delay in the progression of the severe accident sequence. Such delay is beneficial for decision-making and implementation of the corresponding mitigation actions.

There are basically two distinct categories of accidents:

- Accidents within the design basis: those for which the core geometry is preserved (the fuel remains inside the pressure tubes, intact) and the core cooling is maintained. Some accidents that would be considered severe accidents, e.g. loss of primary coolant with failure in the coolant injection from the Emergency Core Cooling System (ECCS), are included as part of the design basis events in CANDU reactors. In this case, the moderator can remove heat from the reactor preventing fusion and maintaining the integrity of the fuel channels. This type of accidents within the design basis is called "Limited Core Damage Accidents" (LCDA).

  In LCDA accidents, the fuel material remains within the limits of the Primary Heat Transport System (PHTS) and the core cooling system is maintained while the moderator system is available.

  Events involving limited fuel damage will not require the use of "Severe Accident Management Guidelines (SAMGs), since these events have been anticipated and considered in the Operating Procedures for Abnormal Events.
- Severe core damage accident: is possible only under the following additional conditions: loss of coolant of PHTS with loss of the ECCS and an additional loss of the moderator as a heat sink. In those accidents a large number of fuel channels fails and collapses to the bottom of the calandria vessel. This type of severe accidents beyond the design basis is called "Severe Core Damage Accidents" (SCDAs) and its management is based on SAMGs.

In the design of CANDU-6 reactors, the fusion at high pressure and direct heating of the containment are excluded as explained below. Reactivity induced accidents are avoided by two fast and highly reliable shutdown systems and the important inventory of water surrounding the fuel and the core fully acts as a heat sink, removing decay heat after reactor shutdown.

Even if all decay heat removal systems fail, the large amount of water available provides sufficient time for implementation of severe accident management actions and since heat removal is by passive evaporation, it is not needed the operation of a valve or pump.

As explained below, accident prevention as an objective of the SAMGs for CANDU design is not feasible. The guide for the prevention of severe accident is provided by the Emergency Operating Procedures (EOPs) and the SAMG correspond to the severe accident mitigation guidelines and are for operating staff after an event evolves to severe core damage. Thus, the fission product release barriers (containment, steam generator tubes, isolation system, etc.) must be protected and for this, the potential challenges of these barriers must be identified and treated.

Severe accidents in CANDU reactors progress through a finite number of "Core Damage States" (CDS) described in *Table N° 3-8.*

	Core damages state	Challenge Criteria
CDS1	The fuel is heated inside the channel due to loss of primary coolant.	PHTS devoid of coolant
CDS2	The hot fuel channels are disassembled and release their contents within the calandria vessel.	Rupture disks burst in the calandria vessel and the first row of fuel channels is exposed.
CDS3	Collapsed and disassembled segments of the channels evaporate the remaining water in the calandria.	The moderator inventory in the calandria is completely evaporated by core debris.
CDS4	The bottom of the calandria vessel failure due to loading of the core debris and waste released by the calandria vault. Wastes are initially made and cooled by the evaporation of accumulated water in the	Calandria vault failure.
CDS5	The shield tank/ calandria vault are dry, penetrate the floor of the shield tank/ calandria vault and the corium is relocated in the basement of the reactor building.	Core debris penetrates the floor of the shield tank / calandria vault.

Table N° 3-8: Core damages states.

Severe accident management guidelines (SAMG) correspond to mitigation, and the input conditions are such that fuel damage has already occurred or is imminent. The output condition of each guide corresponds to having reached a steady state specified in the same directory. This steady state corresponds to the confinement of radioactive material preserving at least the last barrier and long-term heat removal.

The analysis of PSA consequences identifies three prerequisites for the release of fission products into the environment during a severe accident:

- The release of a significant amount of radioactivity from the fuel.
- The existence of an open pathway to the environment.
- A force which facilitates the transport of fission products from their point of initial release through the way open to the environment.

As for the mitigation strategy, considered in the development of the SAMGs to stop the progression of the accident by keeping the cooling of the calandria, it consist on filling the calandria vessel or injecting water into the calandria vault and keeping it flooded all the time to maintain its integrity. However, in the next level of protection, the calandria extra vault phenomena have been considered and the design previsions have been reevaluated, so that the use of SAMGs point to protect the containment functions.

3.1.1.3.2. Accident management measures and design features of the plant to protect the integrity of the confinement function after the occurrence of fuel damage

The containment building provides a barrier containment of radioactive emissions to the environment and to the public, which is critical in the unlikely event of a severe accident. The function of protecting the containment requires the limitation of the inner temperature and pressure of the containment. The severe accident analysis for the generic design of CANDU-6 reactors has been revised to identify scenarios that could present a challenge to the containment integrity. The main conclusion of this review is as follows:

Molten core ejection scenarios do not represent a challenge to containment for CANDU reactors. Depressurization of SPTC (either directly through a system loss or indirectly via the automatic depressurization of the secondary side by opening the safety steam valves) occurs long before the potential development of core meltdown conditions. Even if the depressurization mechanisms fail, overheating of the fuel would cause the failure of a limited number of fuel channels, which would depressurize the SPTC. In this way, the fuel channels of CANDU-6 act as a "fuse pressure relief" that prevent progress of the accident, thus avoiding the high pressure and temperature in the SPTC.

Bypass of the containment through the breakage of the tubes of the steam generators is not possible during a severe accident in the CANDU reactors. In the severe accident scenarios, it is estimated that the pressure tubes will break before the tubes of the steam generators fail due to SPTC overpressure. Relief valves of the SPTC will limit the SPTC pressure to values below the values of rupture of the SG tubes. This avoids the consequence that the breakage of SG tubes leads to the containment bypass.

Steam explosions (fuel/coolant energetic interaction) are considered unlikely during the progression of a severe accident in CANDU reactors because the geometry of the core sets the mode by which the hot material of the core comes into contact with water. In the case of a very high pressure scenario in the SPTC, one or more pressure tubes would break, relieving the pressure of the SPTC. If the tubes are broken, hot material from the core would pass to the Moderator, which is at low pressure (0.26 kg/cm^2) and low temperature (71°C), and would act as a heat sink for this material.

The main challenges identified are due to hydrogen production, slow over-pressurization of the containment and interaction of molten core with concrete of the calandria enclosure (molten corium-concrete interaction MCCI).

One of the management strategies of hydrogen is to dilute or remove hydrogen from containment atmosphere to prevent its combustion and thus maintain the integrity of the containment in the long term. The dilution of hydrogen with air in the containment is possible using the compressed air tanks of the reactor enclosure (the set outlet pressure of the compressors must be raised to overcome the opening pressure of the safety valves of such tanks) to inject air and thus reduce the concentration of hydrogen in air inside the containment (the concentration of H_2 in air must be kept at less than 4% to avoid the explosive mixture).

Another strategy is based on the homogenization of the gases in the atmosphere of the containment, taking into account the containment design that favor the natural circulation, and through the forced air circulation if the Local Air Coolers (LAC) were available.

A specific guide has been developed based on pressurizing the containment to dilute the concentration of hydrogen (Reduce Containment Hydrogen). The various alternatives favor the increase of steam concentration within the containment.

Strategy #2 - Pressurize Containment with Steam to dilute Hydrogen Concentration

TABLE D-1: Lineup Availability

LINEUP	Lineup 2a	Lineup 2b	Lineup 2c	Lineup 2d	Lineup 2e	Lineup 2f	Lineup 2g
DESCRIPTION	Pressurize Containment with steam by turning off cooling to LACs	Pressurize Containment with steam by turning off Dousing	Pressurize Containment with steam by turning off cooling to ECC	Pressurize Containment with steam by turning off Moderator cooling	Pressurize Containment with steam by turning off Shield Cooling	Pressurize Containment with steam by turning off Shutdown cooling	Pressurize Containment with steam by turning off external water addition to RB
ELECTRICAL POWER REQUIRED (ON/OFF)(*)							
Power	On[1]	Off[3]	Off[2]	Off[2]	Off[2]	Off[2]	Off[4]
COOLING SOURCE REQUIRED (ON/OFF)(*)							
Cooling source	Off	Off	Off	Off	Off	Off	N/A
WATER LEVEL REQUIRED							
Water level[5]	N/A	N/A	Adequate	Adequate	Adequate	N/A	N/A
Conditions that may limit use of strategy	• If ECC pump running with cooling water to ECC HX off, then pump seal will heat up. Cooling to HX may need to be turned on intermittently to keep sump water temperature within acceptable limits for ECC pump operation.						

* On / Off determines if that item is available in said state. Circle as applicable for each lineup option.

NOTE:

(1) Local air cooler fan motors should be kept running to help with hydrogen dispersion.
(2) If necessary for accident progression mitigation, the pump(s) may be allowed to operate to provide circulation/cooling.
(3) Dousing valves need to be closed - instrument air to pneumatic control loops of dousing valves may need to be isolated.
(4) Discontinue all other sources of external water addition for containment cooling purposes. Minimal water addition for steam generation purposes should continue.
(5) Will need to maintain adequate water level in location where core debris is located to assist in steam generation and minimizing hydrogen production.

Table N° 3-9: Description of alternatives to increase the content of steam

Specific SAMG for the control of hydrogen in severe accident situations are being developed, including those situations such as venting and not venting of the containment. In addition, as a second line of defense it is expected to install passive autocatalytic recombiners (PAR) to increase safety margins and ensure control of the hydrogen concentration without needing electric power.

The slow over-pressurization may occur due to the steam production by decay heat as a result from the loss of heat sink. Non-condensable gases, which would contribute to the pressurization of the containment, can also be generated by thermo-chemical interactions of hot core materials. The existing design features that protect the integrity of the containment against over-pressurization are the large volume of the containment (48000 m^3 volume net) and passive condensation in building structures of the containment, local air coolers and dousing spraying. A specific SAMG is being developed to address this phenomenology, and it is expected to perform the installation of filtered vent system from containment atmosphere in the extension of the life of the plant.

In addition to changing the filtered vent installation scheduled for the new operation stage (post refurbishment), the current design allows for two venting possibilities. One is the venting of the containment from the dousing and through the SG, so that the scrubbing of what is vented is by passage through the water of the SG. Another way to vent, which the current design allows, is by opening the valves of the spent fuel port from the maintenance room of the fueling machine to the fueling machine room (R 001) and from it down by the transfer channel to the reception pool and the atmosphere of the building making the scrubbing in the water of R 001/pool. Both venting possibilities are planned in one of the SAMG.

The first option assumes that at that stage of the severe accident, the dousing is empty, so that the evacuation would be performed from the pipe that comes down from dousing to PV7 and PV41 and then to the SGs, crossing through the water in them and being evacuated through the main steam safety valve.

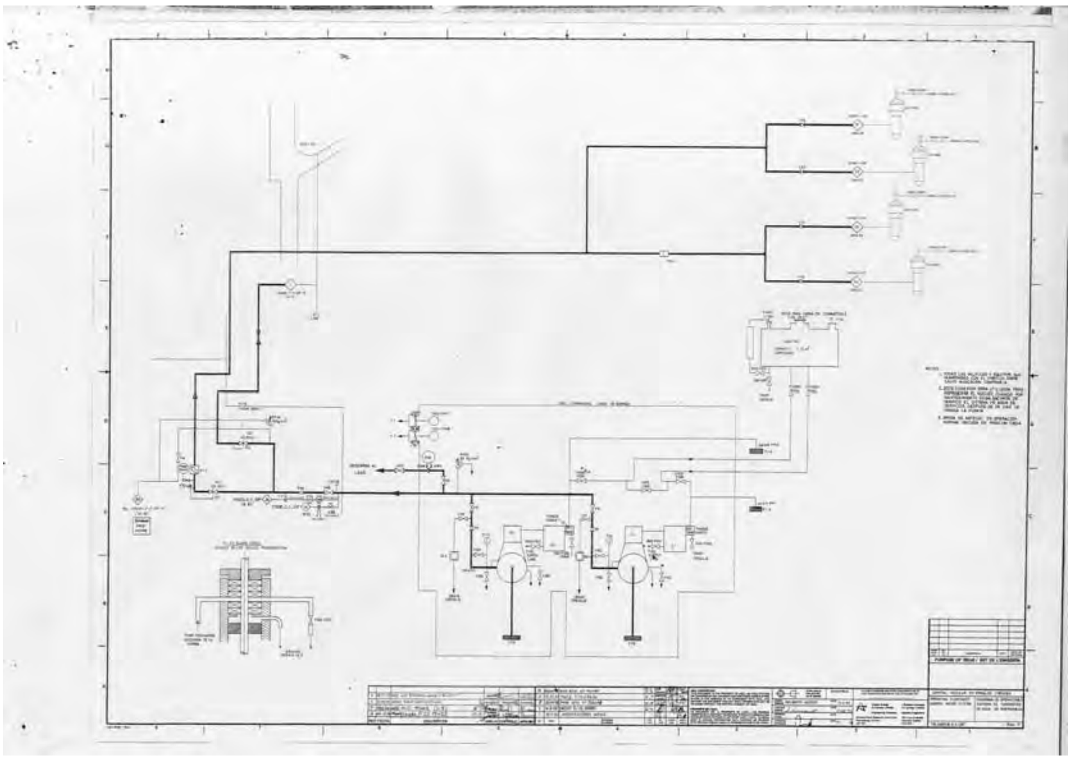


Figure N° 3-31: Venting through SG.

The second option is: with the fueling machine decoupled from the burned fuel port, open the two valves in series of the ports side A and side C, thus pressurizing the room

R 001. The water level in that room would drop to 3.5 m, which is the level where the beginning of the transfer channel is, allowing the flow to bubble in the reception pool towards the atmosphere of the Service Building.

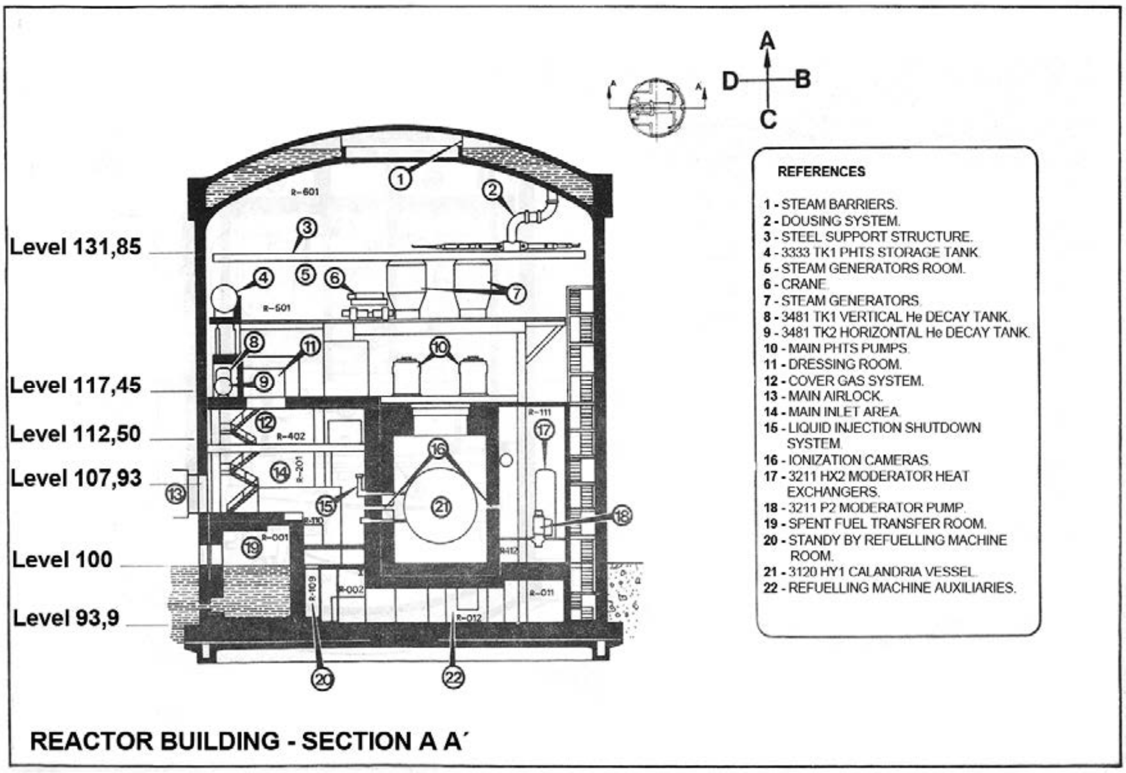


Figure N° 3-32: Venting of the room R 001.

Preventing the molten core to pass through the concrete of the foundation of the reactor building is based on a SAMG pointing to the injection of water into the containment to limit the MCCI. In the generic CANDU-6 design assessments, preferred and alternative strategies were identified, taking into account the availability of systems that can be used for water injection and expected conditions resulting from various severe accident scenarios. These generic assessments show that, if the part of the core that passed the boundaries of the moderator vessel is not submerged in water and cooled, MCCI will not occur at least until two (2) days after the initiation of the accident, and only after four (4) days from the accident, it is expected concrete ablation. This condition provides enough time to implement mitigation actions to bring the plant to a controlled and stable state. This consideration was taken into account in the development of guidelines for severe accident management of CNE.

If the re-flooding of the degraded core is to be considered as a mitigation measure of the severe accident, it is important to note that the potential core re-criticality is not a risk that would arise in CNE. This is because the plant, in addition to operating with a small reactivity excess compared to other designs, possesses natural uranium fuel and requires heavy water as moderator for its operation. In case of accident, emergency injection systems inject light water, so that is not possible to recover the critical condition in the core. If the core re-flooding is to be performed by external means, light water would also be injected, which is why even in this case the core re-criticality would not be an issue to consider when evaluating the possible implementation of this measure. A study was made where criticality risks are calculated for infinite hexagonal arrays of CANDU assemblies immersed in light water, under conservative conditions (fresh fuel or plutonium peak). This study demonstrates that re-criticality cannot be reached. Even in the most critical case analyzed, with an infinite hexagonal array of fuel assemblies in the plutonium peak separated by approximately 4.5 mm, the criticality margin is about 80 mk, higher than usual margins of 50 mk.

Plant conditions that require a transition to the SAMG

The severe structural damage to the core begins when the channels start to fail due to thermal elongation caused by the failure of moderator to act as a heat sink, sometimes referred to as "dismemberment of the core." The events and indications that lead to these plant conditions involve the failure of the primary fuel cooling (low refrigerant flow, low inter-collectors pressure, loss of sub-cooling) or decreasing level of moderator (loss of moderator inventory). However, the SAMG will also be invoked if the symptoms are a very large release of fission products or the occurrence of a hydrogen generation as a result of the severe accident, even if the moderator cooling is available.

Although not normally considered credible, a special case could exist for reactivity initiated events if it was postulated that the shutdown systems failed to prevent a power excursion in time, leading to a multiple failure of fuel channels (pressure tubes). Such events would progress very rapidly, so that the main core transients would happen in few seconds.

Another category of events that could result in the input to the SAMG are those associated with a prolonged loss of the secondary system and the shutdown cooling system (heat sinks). The Operating Procedure for Abnormal Events do not include events beyond the point at which critical security parameters or pressure of the SG and its level cannot be restored, except to try to ensure that the moderator is available as an ultimate heat sink. If this cannot be performed, the result may be a transfer condition to SAMG, especially if the state of the plant involves a generalized loss of power.

So that any of these sequences would result in severe damage to the fuel, the temperature in the affected channels should reach a value over the failure criteria of fuel cladding (between 600 °C and 1000 °C). The preferred approach for the transition to SAMG would be a simple unambiguous temperature measurement. The advantage of using this parameter is that the SAMG can be invoked before there is generalized damage to the fuel, giving a final opportunity to use alternative methods of generalized core damage prevention. It is also entirely independent of the system function, hypothesis or analysis and is a direct indication of the core conditions.

However, given the range of possible accident conditions in CANDU reactors, it is unlikely that a single parameter can fill all requirements. Existing measurements of the temperature of the feeders are considered an unreliable indication of the fuels temperatures under conditions of degraded water flow within the core and also has a measurement limit of 325 °C.

SAMG diagnosis is directed to the symptoms associated with challenges to the barriers of fission products. The current state or imminent occurrence of an event, are diagnosed directly through the observed behavior of the plant. The diagnosis should be based on the monitoring of those parameters (pressure, temperature, etc.) that represent challenges to the fission products barriers so that actions to prevent their failure can be identified.

The time scale of the challenges to the release of fission products barriers is dependent on the event sequence.

In general, the challenges to the containment structure of CANDU plants are not expected before over 24 hours after initiation of the accident. However, as shown in Table N° *3-1*, this can vary substantially depending on the failure modes and the type of challenge to containment.

Figure N° 3-3 illustrates the response of the containment pressure to the sequence of loss of all heat sinks. It should be noted that the results of severe accident analyzes are subject to many assumptions and uncertainties associated with the progression of core damage accident, thus the time scale is approximate.

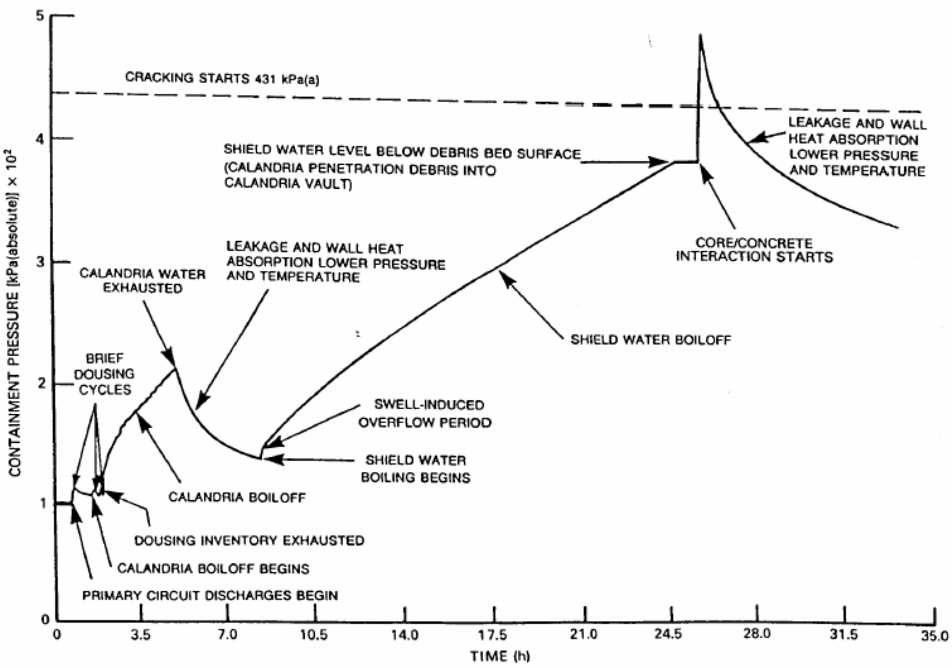


Figure N° 3-33: Typical response of the pressure containment for CANDU 6 plants.

	Typical time scale for challenge to containment				
Event	**Immediate**	**0 – 6 hr**	**6 – 24 hr**	**+24 hr**	**Comments**
Failure to stop	Over - pressure				Potential fissure for pressure relief
Isolation failure	Open way				Use of controlled venting limited by steam caused by the severe accident
Single failure of SG tube				Bypass	Effective release way when the first channel fails at the end of the CDS1
Multiple failure of SG tubes		Bypass			
Small LOCA bypassing the containment			Bypass		
Big LOCA grande bypassing the containment		Bypass			
More severe accidents			Hydrogen	Over-pressure, Combustion of non-condensable gases, filtered venting	Rapid hydrogen generation occurs mainly at CDS1 CDS2, radiolysis may be an issue in the long term (this phenomenon is known as H_2 generation source for long-term radiation decomposition of the water molecule)
LOCA and ECCS failure		Hydrogen			The accident sequence ends in CDS1
LOCA with failure of the isolation of the containment		Lower - pressure			Only occurs if the failure is recovered after the purge.

Table N° 3-10: Challenges to the containment

The ability to assess the status of the plant is an essential component of the SAMGs. To make decisions that are appropriate and effective to control and mitigate the severe accident, it is important to be as clear as possible about the state of the plant and how the accident progresses. There are two main reasons to use the plant instrumentation:

- To diagnose the plant condition for the selection of appropriate strategies.
- To confirm/monitor the success of the implementation of strategies

Survival of the instrumentation for severe accident conditions is not evaluated as part of the development of SAMG. The generic analysis have shown that environmentally qualified instrumentation for design basis accidents in a conservative manner can be expected to exhibit a significant capacity to remain operable in severe accident conditions, especially given the needs of accuracy of the measurements required. For example, it has been shown that the environment of contention for the first hours of severe accident is not likely to exceed the environmental qualification. The best estimative judgments suggest that proper instrumentation performance margin will exist at least for 24 hours to less severe accidents, and 2-3 hours to more severe accidents.

The necessary information for specific parameters of environmental conditions in the containment can be extracted from the instruments and other equipment in a degraded or even failed condition, from the collective state of systems (e.g. the observation of which system has failed and which is still operating, from temporary or portable equipment and from direct observation). The identification of redundancies and alternative media to obtain information of the key parameters can increase the confidence in the capabilities of existing instrumentation in severe accident conditions. When multiple sensors measure the same parameter, it is easier to identify which instrument failed. It is also often possible to obtain indirect information of a given parameter. As in the case of generic design, for the CNE has been developed some visual aids to facilitate the interpretation of some indications of trends in specific parameters. These visual aids are called "Computational Aids (CAs)" and are analytical tools that facilitate the timely assessment of plant conditions, when a direct measurement is not available. They help the staff of the Technical Support Group (TSG) in the implementation of the SAG or SCG and in the formulation of severe accident management guidelines. CAs are always called specifically from the guides (SAG and/or SCG) and are not provided simply to get information but they allow to take decision on the action to be implemented.

Summary of computational aids:

CA-1: Individual dose to a member of the public from a containment vent.

Objective: It is used to estimate the potential dose to an individual member of the public at various distances from the plant, from a proposed containment vent.

CA-2: Rate of Water Addition for Decay Heat Removal by Vaporization

Objective: CA-2 is used to determine the rate of water addition required to remove decay heat by evaporation with respect to time.

CA-3: Rate of Water Addition to Maintain or Increase Moderator Level

Objective: CA-3 is used to provide a simple means to estimate the rate of moderator makeup required to compensate for the observed rate of moderator depletion.

CA-4: Hydrogen Flammability in Containment

Objective: CA-4 is used to determine the conditions in containment related to the flammability of hydrogen, in particular whether input conditions for SCG-3 and SAG-6 are met.

CA-5: Containment Water Level

Objective: CA-5 relates the volume of water released to the containment to water depth in containment. Contributions from important sources of water are indicated (SEEN, HTS, Moderator, Shield tank). This CA provides support for SAG-5 (pressure seen by low elevation airlocks) and SAG-7 (to determine whether any core debris can be quenched).

CA-6: Determination of magnitude of core damage from measured dose rates

Objective: CA-6 is used to estimate the degree of core damage (i.e., fraction of core inventory of volatile fission products released to containment) using dose rates measured at specified locations outside containment.

On the other hand, to obtain a complete picture of the accident and its progress, it would be necessary to measure a large number of parameters. However, it has been shown that a detailed picture is not necessary to define a management plan for severe accident and that only a few key parameters are sufficient for this purpose, thus reducing the need for instrumentation. These considerations are also taken into account in the development of severe accident guidelines, i.e. they anticipate that if there is not any specific measurement available, the parameter will be verified taking into account conditions and measurements from some other system, laptop, etc.

While it is known that the range of instrumentation and its ability to survive can be improved, in the development of SAMG for the CNE, existing instrumentation is used for severe accident management.

Considering the operative experience of other NPPs, the Licensee is analyzing ti implement changes in the following instrumentation:

- PHTS Subcooling Margin
- Moderator Level
- Calandria Vault Water Level
- Containment pressure
- Plant Radiation measurement
- Containment hydrogen flammability
- R/B Basement water level

Some examples are:

- The maximum current range of measurement of the containment pressure is 150 kPa and the CAs input for its venting is of 200 KPa. The extension of the range is not being evaluated; the pressure will be used to enter the guide in 150 kPa, implying vent in a scenario where perhaps the pressure of 200 kPa will never be reached.
- The maximum range of water level in the containment is 5 meters and the maximum level to which water can be injected in it is about 20 meters.
- The evacuation towards the containment of the whole inventory of HTPS water, moderator, shielding, high pressure and low pressure ECCS and dousing will mean a level of E/R of 2.92 m. With the current instrumentation it can be measured up to 5 m, therefore water may be added until it reaches that level. From then on there is no way of being certain about the level.
- Currently there is no measurement of H_2 in the containment, but a calculation of it is expected in a CA according to the CDS.
- One CA guide is composed of containment pressure curves vs. H_2 volume in the containment determined by calculations made by AECL. This CA is included in the SAMG package.

As an example we present the following graph:

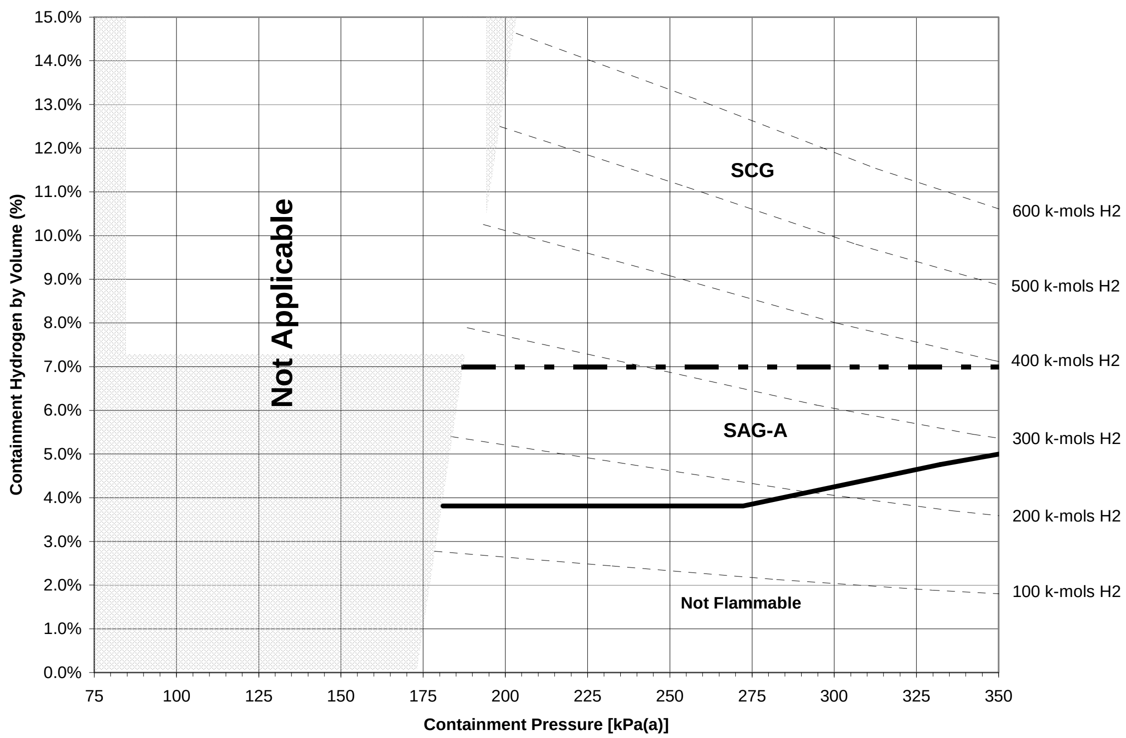


Figure N° 3-34: Potential for hydrogen burning based on wet hydrogen measurement (initial non-condensable gas 3500 K-mol).

3.1.1.3.3. Accident management measures currently available to handle the successive stages of a loss of cooling scenario in the spent fuel storage pools

At the present, there are 39,544 FE stored in the spent fuel pool.

The sustained loss of cooling in the spent fuel pool constitutes an unlikely emergency situation which would usually be prompted by a common cause failure, such as an earthquake causing a persistent blackout. In this scenario it would be necessary to replenish water to the pool in order to prevent the FE from being discovered and avoid the potential hydrogen generation. According to the calculations performed, there would be enough time to set a water supply of 1kg/s so as to keep the bundles submerged. It would 3.26 days for the water in the pool to reach boiling and 16.27 days for the first row of FE to be exposed. Therefore, a loss of cooling event should be successfully handled by the Standard Operating Procedure for Abnormal Event, which is currently being revised.

There are no adverse effects expected from a loss of cooling in the spent fuel pools. The fuel would remain adequately cooled and staff doses would not exceed operational limits. There is no chance of re-criticality in the pool of CANDU reactors, whether the FE are submerged or exposed.

As a result of the previous analysis, it was decided to incorporate the following improvements:

- It will be installed a facility to connect a fire-truck from outside the pool building, which will replenish water to the pools in the events of loss of cooling, circulation or SBO. Implementation is scheduled by end of 2012.
- A new Operating Procedure for Abnormal Events that covers response to loss of cooling in the spent fuel pool and / or loss of inventory will be developed. This procedure shall include actions to verify the coolant level and temperature of the pool from the secondary control room in the event that the main control room and the pool room are unavailable. It shall include actions to replenish water from alternative systems (eg. fire hydrant or fire engine) in the event of sustained loss of cooling or loss of inventory. Implementation is scheduled by the end of 2012.
- Instrumentation will be installed to monitor water level and temperature in the storage pool from the secondary control room (independent from the nowadays existing in the main control room) with a repeater in the main control room. The system will be connected to the Emergency Power Supply. Implementation is scheduled by the end of 2012.

A new item will be incorporated to the Operations Manual: "Check once per shift the functionality of break vacuum pipes / siphon belonging to fuel pools ". Such control is planned to be included in the walkdown form that is filled by the operations assistant when he monitors the pools. Implementation is scheduled by the end of 2012.

Effects of a LOCA in Room R-001 (Burned Fuel Discharge Pool):

The room where the spent fuel is discharged is located outside the containment. The walls between it and the Reactor Building are part of the containment boundary, and are subject to a pressure of 1.28 kg/cm^2.

On the handling of fuel, the Operation Manual states:

"Effects of a LOCA in Room R-001 (Burned Fuel Discharge Pool)"

"The room where the spent fuel is discharged is located outside the containment. The walls between it and the Reactor Building are part of the containment boundary, and are subject to a pressure of 1.28 kg/cm^2. An exception occurs when the fueling machine is coupled to the spent fuel discharge port and both of the port's ball valves are open. In this situation, the wall between the discharge room and the Service Building also becomes part of the containment boundary but is subject to a pressure of only 0.21 kg/cm^2. This estimate is conservative and was calculated considering the gas flow through the fixed orifices of the FM during a pressure peak. The limited pressure inside the discharge room allows the existence of the opening in the wall between the discharge room and the Services Building. (This opening constitutes a channel through which travels the spent fuel tray transferor when it leaves the fueling machine and is moved from R 001 to the spent fuel pool in the Pools Building.) This opening is below 3.5 m of water, which causes a sealing of 0.35 kg/cm^2, and is used to transfer burned fuel. All penetrations through this wall (compressed air, ventilation) are seismically qualified for the design basis earthquake. The pressure during the earthquake is kept under 0.35 kg/cm^2 by the Pressure Suppression System of the Reactor Building. The water seal above the opening between the discharge room and the Service Building can withstand this pressure and thus permit its opening." (see Figure N° *3*).

3.1.1.3.4. Additional considerations: Safety related design changes to be implemented during the refurbishments required for the CNE life extension

As a result of the analyzes and evaluations carried out in the framework of the CNE, life extension project activities of the former CNE designer Atomic Energy Canada Limited-(AECL), CANDU Energy has identified numerous safety-related design changes that NA-SA decided to implement during the refurbishment of the plant which will be completed by 2016. Among the listed improvements worth mentioning the following:

Amongst the analysis and assessments conducted in the Project CNE Extension of Life, the former supplier for CNE-AECL (Atomic Energy Canada Limited), CANDU Energy, has identified many design changes considered safety related that the Licensee decide to implement during the refurbishment of CNE (2015-2016). It is possible to mention the following improvements:

- New fuel channels
- Safety systems trip coverage
- ECC reliability
- PHTS main pumps protection (in order to protect associated piping)
- Robustness of the plant against earthquakes (increased EPS / EWS capacity and functions)
- Robustness of the plant against severe accidents
- Robustness of the plant against loss of service water

New fuel channels

It is committed the replacement of all the 380 fuel channels (pressure tubes, calandria tubes and feeders). The new fuel channels are not identical to the existing ones. AECL has introduced several changes in their technical specifications, among which are the following:

- Pressure tube installation reversal.
- Modifications to provide more room for fuel string expansion during LOCA.
- Improved P/T Rolled Joint.
- Purge Ring for Fuel Channel Feeder Connection Installation.
- Matte surface on outside of calandria tubes.

Safety systems trip coverage

- Current Canadian standards state that for each event that requires action from the reactor shutdown systems, there must be at least two different triggering parameters for each system. Both these standards (which are followed by AECL as designer) and the new trip coverage are addressed below.
- Various improvements in the safety systems trip coverage trigger in order to approach the implementation of these standards.

What is stated above entails adding new trips and modifying some of the existing ones, so as to improve defense in depth against accidents already covered by current trips.

- Addition of new trips modifications to the design basis.

Trip coverage: Modifications the existing scheme

Shutdown system # 1:

- Addition of a new trip due to low-pressure in the PHTS.
- Change of instrumentation in steam generators #2 and 3 to permit the addition of a new low level trip.
- Addition of new trips due to high and low level of the moderator.
- Increase of setpoint for the trip due to low PHTS flow.
- Reduction of setpoint for the trip due to high PHTS pressure.

Shutdown system #2:

- Addition of a new trip due to a PHTS main pump failure.
- Change of instrumentation in steam generators #1 and 4 to permit the addition of a new low level trip.
- Change of instrumentation in #3 to 7 outlet collectors to permit the addition of a new PHTS high pressure trip.
- Addition of trips due to high and low level of the moderator.
- Reduction of setpoint high logarithmic rate neutron flux trip.
- Reduction of setpoint for the PHTS high pressure trip.

Trip coverage: improvement in local overpower protection

- Shutdown system #1:
    - o Increase in the number of in-core platinum detectors from 13 to 34.
- Shutdown system #2:
    - o Increase in the number of horizontal rods from 4 to 7.
    - o Increase in the number of in-core platinum detectors from 8 to 24.

- Both systems:
    - o Replacement of all measuring chain electronics.
    - o Addition of dynamic compensators.

ECCS reliability

- Changes designed to ensure injection initiation.
- Changes designed to increase the system reliability.
- Changes designed to prevent leakage from the primary to the ECCS (containment by-pass).

ECCS reliability: initiation

- Setpoint increase for the main LOCA signal (low pressure in the PHTS).
- Setponit decrease for the conditioning of the reactor building high pressure signal.
- Automatic launch of ECCS to sustained low-pressure in the reactor collectors.

ECCS reliability: running

- Automation of transfer from the medium pressure stage to low pressure stage. Includes tripling of level measurement in the spray tank.
- Duplication of spray tank isolation valves (3432 PV10 and PV11).
- Replacement of 16” manual valve 3432 V75 with a 6” motorized valve operable from both the MCR and the SCA.
- Establishment of an alternative path for the injection of medium pressure. The valves currently existing are PV23/24, and PV10/11 are their duplicates. PV163/164 will be added to duplicate PV11/12.
- Addition of alarms in the MCR.

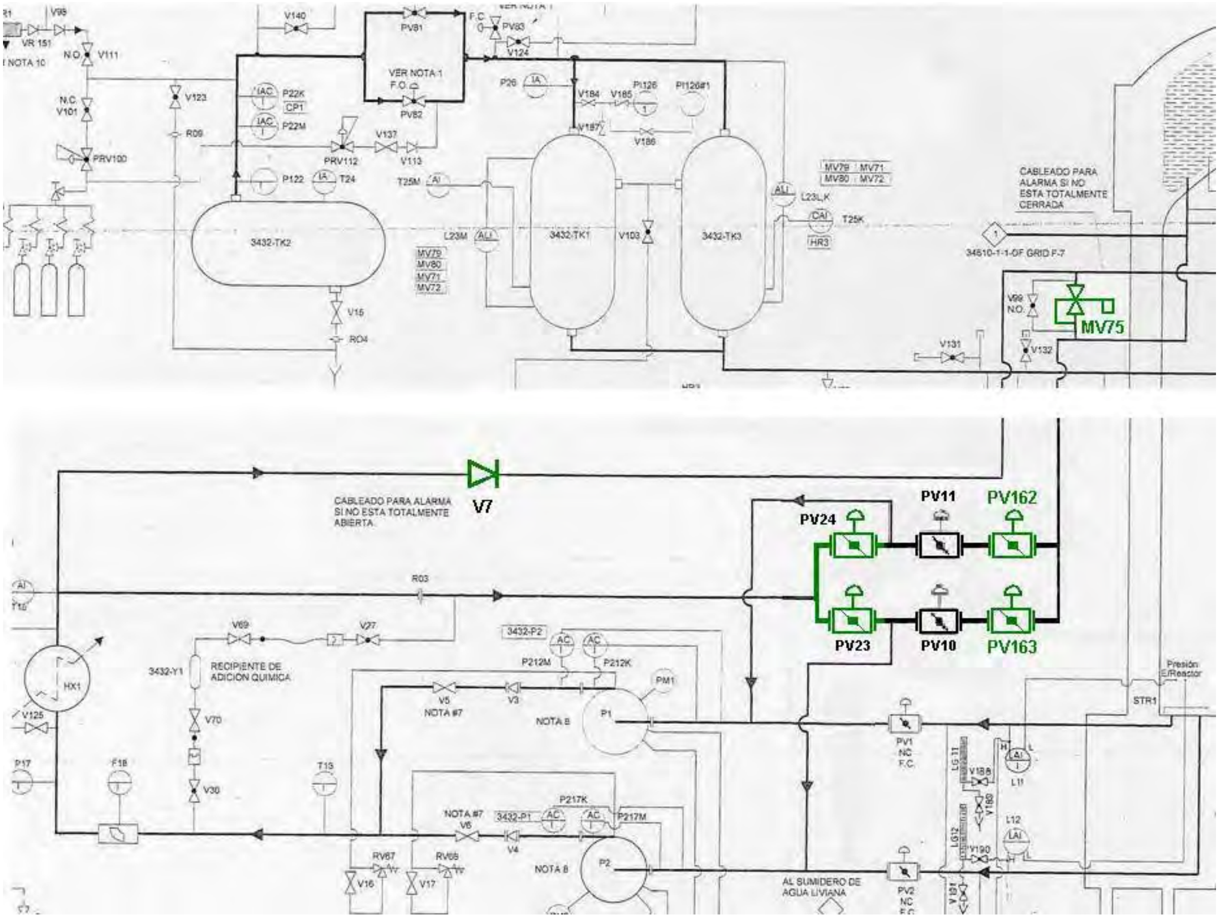


Figure N° 3-35: ECCS with design changes.

- Additional alarms in MCR:
    - o Low pressure in the ECCS's high pressure water tanks (3432 TK1 and TK3).
    - o Conditional alarm of the R/B sump recovery valves state (3432 PV1 and PV2). PV1/2 are intake valves for the 43432 P1/P2 sump pumps during ECCS low pressure stage.
    - o Warning in case those valves are open when the system is set and the low pressure stage is not working.
- Relocation of level gauges bubbling tubes in the R/B sump (63432 L11 and L12)

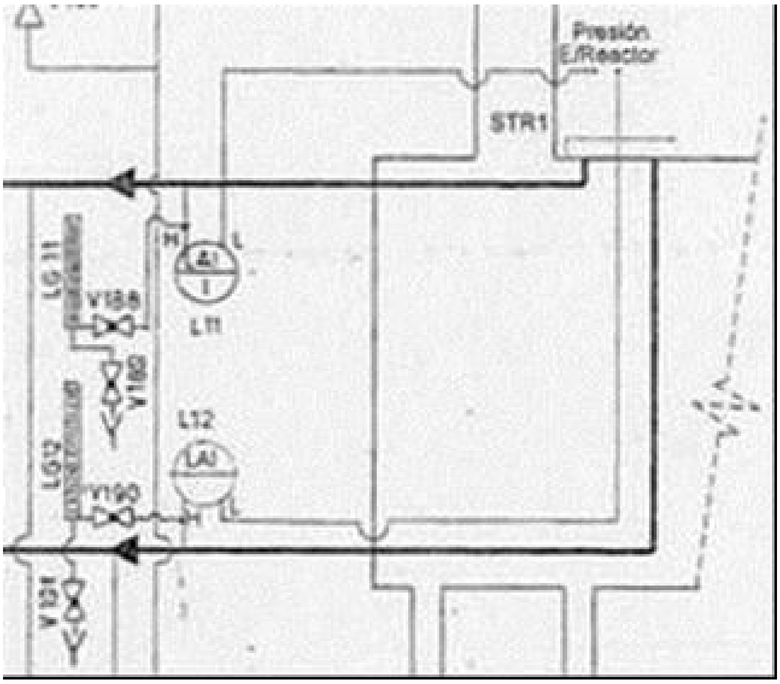


Figure N° 3-36: Level measurement in the R/B sump.

ECCS reliability: leakage prevention from PHTS

- Replacement of 3432 V7 valve with a check valve (testable and with position indicator in the MCR).
- Removal of pneumatic actuators from valves 3432 PV33, PV34, PV47 and PV48. Addition of manual actuators and positioners in the MCR.

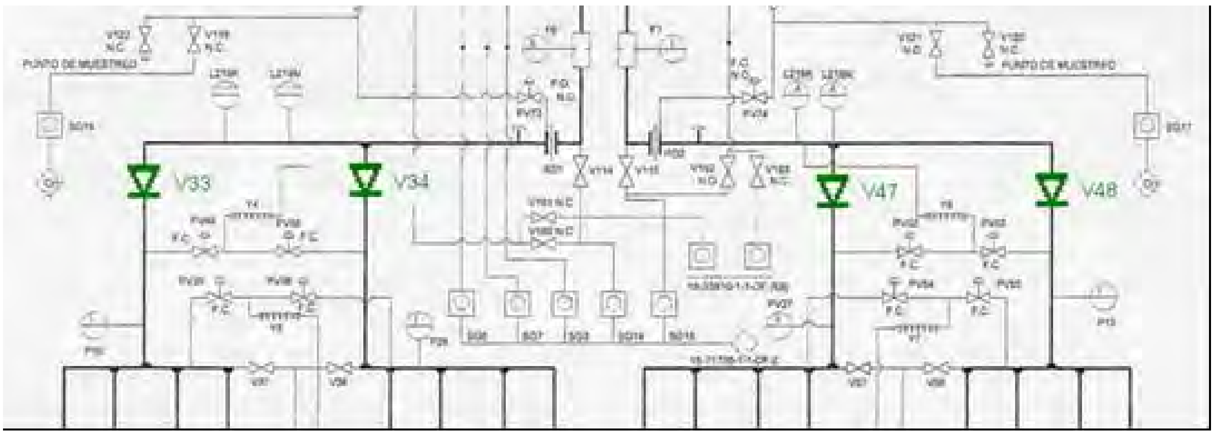


Figure N° 3-37: ECCS with design changes – Leakage prevention.

PHTS main pumps protection (in order to protect associated piping)

- Addition of a PHTS pumps trip due to sustained low pressure in the water outlet collector. (This is meant to avoid cavitation of the pumps in the event of LOCA).
- Addition of a PHTS pumps trip due to high temperature in the bearings (with a delay to avoid spurious trips). (This is mean to avoid failure of the pumps due to loss of cooling in the bearings.)

Robustness of the plant against earthquakes (increased EPS / EWS capacity and functions)

- **EPS System:**
  - o Replacement of existing 50kW diesel generators (DG) with new 1MW DGs. The increased capacity is meant to provide seismically qualified power supply to the following additional charges:
    - ECCS pumps 3432 PM1 / PM2.
    - EWS pumps 3461 PM1 / PM2.
- **EWS System:**
  - o Replacement of existing diesel pumps with two larger electric pumps, as indicated below (actual flow plus flow needed to feed 3432 HX 1, which is the ECCS heat exchanger). The new pump will be fed from the EPS.
  - o Duplication valves 3461 PV41 and PV7.
  - o The increased pump capacity is meant to supply seismically qualified cooling to the ECCS's heat exchanger 3432 HX1.

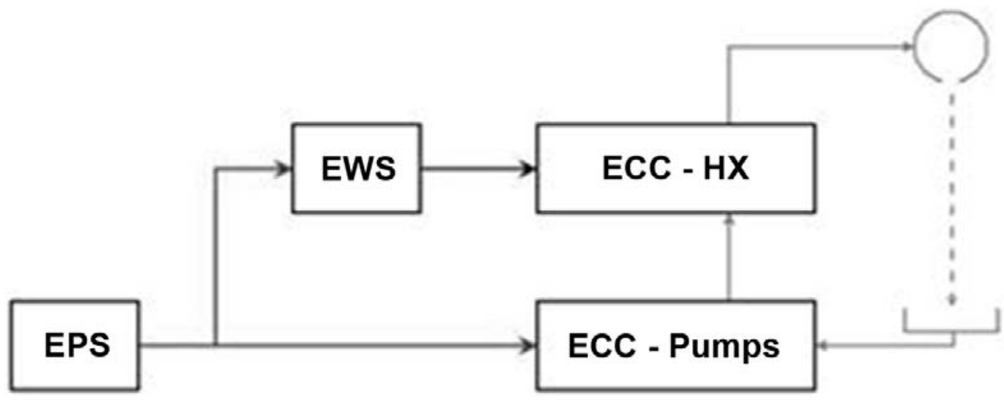


Figure N° 3-38: EPS/EWS functionality against seismic events.

Robustness of the plant against severe accidents

- Installation of hydrogen recombiners
- Passive autocatalytic in the R/B.
- Addition of a water supply line from outside the R/B, to the calandria vault.
- Addition of a rupture disk to the calandria vault (increasing the relief capacity).
- Implementation of a system of filtered venting for the containment**.**

Robustness of the plant against loss of service water

At the moment, it is being automated the switch from normal feeding to alternate feeding (which has been done manually up to now).

- Automated change to backup cooling pump auxiliary feedwater 4323 P107 (fire water).
- Automated change to backup cooling pump auxiliary feedwater 4323 P107A (drinking water).
- Automated change to the backup cooling GD class III (fire water).

3.1.2. ACTIVITIES PERFORMED BY THE REGULATOR

The Regulatory Body issued a requirement to NA-SA who is the Licensee of CNA I, CNA II and CNE to perform an integral safety evaluation (stress test) of all plants in order to identify weaknesses and to propose upgrading improvements, as well as to assess the information provided by the plants in order to determine the level of safety in view of lessons learned after Fukushima accident. Such assessment will determine the safety margins the plants have in order to face scenarios generated by extreme events with consequences like losing for long periods all electrical systems (SBO) and/or ultimate heat sinks.

The evaluation also requires considering prevention and mitigation actions designed to deal with such scenarios and it includes the following aspects:

3.1.2.1. Describe the accident management measures currently available to protect the core at various stages of a scenario of loss of cooling function

- Before initiation of fuel damage in the reactor core indicating:
    - If there is a mean to prevent fuel damage in sequences where there is high pressure in the primary system and failures in safety systems that prevent primary transport system of being depressurized.
    - Any extra action identified within accident management program capable of preventing fuel damage.
- After fuel damage has started.
- After pressure vessel has failed.

3.1.2.2. Describe accident management actions and plant design characteristics to protect the confinement function integrity after fuel damage has begun.

The accident management actions foreseen are the following:

- Prevent hydrogen explosion and deflagration (inertization, recombiners, ignitors) considering containment actual venting capacity.
- Overpressure prevention. If necessary you should vent the containment in order to protect containment integrity. In this last case the available means should be described in order to estimate the quantities of radioactive material that will be released to the environment.
- Re-criticality prevention
- Containment flooding to different levels to prevent pressure vessel failure or to limit the molten core-concrete interaction (MCCI).
- Need of alternating electric power, dc power and compressed air supply in order to feed necessary equipment to prevent containment integrity.

3.1.2.3. Describe accident management measures presently available to face the successive steps in an scenario of failure of spent fuel pool cooling

Among the main accident management measures have to be considered:

- Radiation protection (loss of water column shielding).
- Top of fuel uncover.
- Fuel degradation (fast cladding oxidation with hydrogen production).

3.1.2.4. Additional remarks

- Regarding points 3.1.2,1, 3..1.2.2 and 3.1.2.3 should be:
    - Identify any possible limit situation that could occur and evaluate how much time is available till that limits is reached.
    - Evaluate the adequacy of accident management strategies, including guides and procedures developed to face a severe accident and analyze the possibilities of additional actions. In particular some other aspects should be taken into account:
    - Regarding the installation:
        - Adequacy and availability the required instrumentation.
        - Habitability and accessibility of essential areas (control room, emergency response centers, local controls, test points, reparability possibilities).
        - Hydrogen accumulations in buildings different for the containment.

- Organization:
    - Staffing, resources and management shifts.
    - Use of external technical support and the place from where the accident management including contingencies is implemented, if it no longer available.
    - Procedures, capacitation/training and exercises.

- Availability to use existing equipment.
- Forecasts for the use of mobile equipment. Availability, appropriate connectors for coupling, the time needed to be available at the place and in operation, and accessibility to the site.
- Availability and supply management (fuel for diesel generators, water, etc.).
- Management of radioactive emissions and forecasts possible to limit them.
- Management of potential doses to workers and provisions to limit them.
- Systems of communication and information (both external and internal).
- Activities planned for the long term (after the accident).

The expected accident management should be analyzed taking into account the situation that could cause the initiating event and possible evolutions of the accident:

- The possible destruction of existing infrastructure around the plant, including communications (making it difficult the technical support and staff from outside).
- Impediments to do the work, including the impact on accessibility and habitability of the main control room and secondary and centers of emergency / crisis of the plant, due to high dose rates, the radioactive contamination and possible destruction of installations.
- The need to analyze the feasibility and effectiveness of the measures of accident management in the conditions of extreme external events (earthquakes, floods, etc.).
- Unavailability of external power supply.
- Unavailability of instrumentation.
- Potential effects on other nuclear power plants nearby.

Furthermore, it should identify scenarios which would preclude or hamper the work of personnel operating in the main control room and/or emergency control room, and centers of emergency and crisis of the plant, the site and the measures to implement that could avoid such scenarios.

Based on the evaluation of information submitted by the plants, the ARN considers that no significant weaknesses have been identified requiring urgent action taken and has verified that NA-SA complies with both the design basis as on the basis of licensing. In addition, after the proposed improvements are implemented, these plants will be able to face with greater safety margin, the situations resulting from lessons learned from Fukushima taking account the extreme external events possible at sites.

The times taken to implement improvements at each plant are considered adequate to complete the studies, the necessary engineering tasks and their implementation. The ARN will follow the development of tasks through inspections and assessments of reports that the plant must sent.

It may be noted that nuclear power plants in operation develop severe accident management programs in response to specific regulatory requirements issued prior to the accident of Fukushima. In the case of CNA I, the scope of the specific requirement was determined considering that:

- Implementation of a program of management of accidents, for accidents beyond the design basis is a component, considered essential to the principle of defense in depth and can extend the safety margins.
- This plant, with a design unique in the world, has specific characteristics, which required a significant effort to develop the accident progression model, the fundamental basis for evaluating the applicability of mitigation strategies for severe accidents.

The ARN in early before the issuance of these requirements began his familiarity with modeling with MELCOR code through the support of Sandia N.L. USA to acquire knowledge and experience.

Given the complexity of the accident progression model, in principle there was a major breakthrough in the development of prevention strategies which were detailed by the plant in the reporting.

Subsequently, as part of the response of the CNA I to the requirements of WANO SOER 2011-2 the plant expand and deepened corresponding assessments, expanding the scope of safety verifications.

The assessment by the ARN to the present has not found significant vulnerabilities. It has been validated model accident progression with MELCOR code by comparison with a SBO sequence with the RELAP code until the beginning of core uncovers. In addition, the Licensee performed a qualitative comparison of the evolution of phenomenology for CNA I and CNA II.

In the case of CNE, Severe Accident Management Program is developed as part of the documents necessary to grant the licensing activities of conditioning for life extension.

Being in this case of CANDU design, the ARN considered adequate Licensee's proposal to adapt the generic guidelines for severe accidents for CANDU reactors developed by the CANDU Owner Group (COG). In addition, Specific Guidelines for Severe Accident applicable to CNE are being developed with the advice of CANDU Energy Company of Canada.

In the case of CNA II, one of the requirements to fulfill in the licensing process, is the Argentine design standard AR 3.1.3, that requires the full-range development of probabilistic safety analysis (PSA) Level 1 and 2 and a partially PSA level 3 focused on obtaining doses to the public. The reviewed by the ARN is made with the support of GRS-Germany Company (PSA Level 1 and integration among the three levels of PSA) and SANDIA NL-USA (PSA Levels 2 and 3).

Meanwhile, the ARN has realized anticipatory deterministic assessments focus on assessing:

- The ability to flood the CNA II reactor cavity, during the progression of a severe accident in order to maintain the integrity of RPV.
- The evolution of the internal pressure and the generation of hydrogen to determine the necessary measures to ensure the integrity of the containment to a severe accident scenario.

For each task we used a simplified model (stand-alone) developed by Sandia from MELCOR code model developed by GRS / NA-SA. Both models, for its rapid implementation allowed for studies of sensitivity to parameter variation and behavior of plant associated with the phenomenology of core degradation in a case and containment in the other.

The results showed that for CNA II is sufficient time to implement any mitigation measure prior to the rupture of RPV. Another result was that at temperatures up to the bottom of RPV (lower head), from about 1500 °K, the RPV will maintain its integrity.

The effect of water in the cavity was analyzed for modeling various conditions by varying the conditions of flooding the reactor cavity or when the temperature in the lower plenum reaches a certain value (700 or 1500 K). The assessments showed that the effect of water under the conditions considered, significantly delayed the loss of integrity of the RPV.

As regards the evaluation of the dependencies of the response of the containment pressure to variations in the degradation behavior of the fuel assembly which affect the production of hydrogen, the calculations are initiated by considering a modeling SBO varying conditions on the values used in the model of the GRS / NASA.

Overall conclusion

In the framework of severe accident management, the ARN considers that once the proposed improvements are implemented, will be improved safety margins.

The ARN considers acceptable the deadlines for implementing planned strategies and for complete the studies of progression of the accidents in the MELCOR code CNA I. The above regulatory position is based on that data from the characteristic of CNA I and CNA II in a scenario of SBO accident progression in the stage in-vessel will be slow, and slower in CNA I than CNA II.

In the case of CNE, the deadlines proposed by the plant in accordance with the schedule of activities planned for refurbishment to extend the life.

In the case of the CNA II improvements have been identified, generally its implementation require further studies being carried out as required by specific regulatory requirements.

In general, the ARN has not identified significant vulnerabilities requiring urgent regulatory action and has verified that NA-SA complies with both the bases of design and licensing bases. For the purpose of increasing the capacity to respond to extreme conditions, NA-SA proposes to implement a set of improvements, as determined by the ARN, are acceptable.

The ARN continue to monitor and assessing the actions that are being and / or will be implemented in the future to ensure that they are effective and to consider all necessary aspects related to plant safety. The result of such activities is necessary to determine whether the ARN requires complementary actions, modifications or improvements.

4

NATIONAL ORGANIZATIONS

Since technical information and outcomes of the revisions performed by organizations involved in nuclear safety are provided in other parts of this report, we devoted this section to describe actions undertaken by the regulatory body in the aftermath of Fukushima and its interaction with some non-nuclear organizations.

One of the issues that the events in Fukushima clearly underlined is the importance of the existence of adequate arrangements in the area of emergency preparedness and response. Through the succesive national reports presented in compliance with the Nuclear Safety Convention, Argentina has informed in detail of its approach to and arrangements related to the matter, including the array of agreements with intervening organizations, and the fact that such agreements are part of the emergency plans.

Emergency exercises in the NPPs are carried out annually (this is an explicit demand contained in the operating licence). In this connection, and having in mind some elements of the experience during the Fukushima events, special attention was given to review the links between the different organizations involved in the regular emergency response exercise at Atucha I NPP, which took place in 2011.

As an outcome of the review, the need of engaging with the highest levels of decision of each organization involved was determined, in order to ensure that in the event of a crisis not only the operational levels are aware of the role of each institution, but also the direction levels.

A series of meetings were conducted to raise awareness of the role each organization during a crisis and during the preparation and planning stages.

An important result of those meetings was the identification of procedures that may conceivably be improved. For example, it was proposed making formal agreements with the highest levels of each involved institution, with respect to the roles, staff and materials needed during a nuclear emergency.

Since the last quarter of 2011, ARN has been negotiating the mentioned agreements. Among these it is worth mentioning an agreement between the ARN and the Joint Chief of Staff (Estado Mayor Conjunto), in addition to the existing agreements with the Army and the Navy for the training, preparation and intervention during a nuclear emergency. It will permit a more structured and comprehensive planning for the response of those institutions in a nuclear emergency and also a more focused training of the members of those forces.

Also, in addition to the agreements already in place with different hospitals for preparation, training and medical attention of eventual patients originating as a consequence of a nuclear accident, the ARN is negotiating an arrangement with the Ministry of Health of the City of Buenos Aires including all its hospitals. The agreement is expected to increase the coordination among the different hospitals during the stages of preparation and intervention in an emergency.

Furthermore, an agreement with the National Bureau of Civil Defence (Dirección Nacional de Protección Civil, of the Ministry of the Interior) is being negotiated apart from the existing ones with the National Gendarmerie, the Coast Guard and the Civil Defence. The National Bureau of Civil Defence is a federal institution with valuable experience in planning and responding to large natural disasters, which coordinates the response of local and regional organizations, in the event of natural disasters.

Although Argentina is geographically very distant from Fukushima, the need to monitor the situation of the Argentine citizens in Japan and to coordinate the repatriation of those who were in a critical individual situation (due to the earthquake) and desired to do so, arose during the first days following the Fukushima accident. A Crisis Committee integrated by the Ministry of Foreign Affairs, the Ministry of Interior and the ARN was established to manage and solve those needs. This Committee held regular meetings on different matters related to the natural disaster and the situation at Fukushima Daiichi NPPs. The joint interaction between dissimilar areas of the government, each one with a different approach and views on the problems, proved to be challenging, but very valuable.

Recognizing the relevance of a clear understanding of the roles and responsibilities of relevant stakeholders, the ARN delivered in 2011 tailored-made courses for the Security Forces and the Armed Forces, which have specific roles in emergency response activities. These courses currently include a detailed reference to the lessons learned so far from the Fukushima accident.

In the spirit of stressing ARN's interaction with other governmental organizations, representatives from the direction level of ARN gave lectures to relevant stakeholders, including educational institutions, the medical community, as well as government agencies related with the funding of the nuclear activity. With respect to the latter, the objective was to ensure adequate awareness of the importance of maintaining the level of human and financial resources of the agencies involved in the nuclear field and those related to or participating in nuclear emergencies.

Every year, the Senior Staff Association of the Ministry of Economy organizes a series of conferences in which specialists from different disciplines of science, technology and management give lectures on their field of expertise and daily work. In this frame, a lecture on ARN's tasks and responsibilities, in line with ARN's commitment to safety and bearing in mind the impact of Fukushima was given.

It became clear that all of these activities have redounded in a much better understanding of the nuclear regulatory activity and ARN's roles and responsibilities.

Regarding the Centre of Studies on the Energy Regulatory Activity (Centro de Estudios de la Actividad Regulatoria Energética - CEARE) at the University of Buenos Aires, ARN's officers continue coordinating and giving lectures on regulation and nuclear energy, including in 2011 a particular chapter on Fukushima. ARN also collaborated in developing and designing a subject on nuclear activities for the second year of the Interdisciplinary Master of the above-mentioned Centre.

Finally, it is worth mentioning that ARN has undertaken a dialogue with existing Argentine entities aimed at establishing a national Technical Support Organization that may work in areas related to nuclear safety.

5

EMERGENCY PREPAREDNESS AND RESPONSE AND POST-ACCIDENT MANAGEMENT -OFF SITE-

5.1. ACTIVITIES PERFORMED BY THE OPERATOR

In case of a nuclear emergency the Licensee (Nucleoeléctrica Argentina S.A., NA-SA), who is represented by the Plant Manager has the responsibility to implement the urgent and immediate protective actions off-site of the installation, referred to the correspondent Emergency Plan. Initially, the plant manager must to manage the actions until ARN arrives and assumes responsibility for the actions.

5.1.1. ATUCHA I NUCLEAR POWER PLANT

5.1.1.1. Direction and Control

5.1.1.1.1. Procedures to redistribute and relocate in safety areas the personnel that must stay in the plant

The Emergency Plan of Atucha I Nuclear Power Plant (CNAI) is in force. It describes the strategy, methodology, organization and necessary ways to deal with emergencies that may occur in the plant. The Emergency Plan is supported by procedures that define the way to apply the defined strategy and the particular responsibilities. The main procedures are:

- Procedure PS–104: Definition of Alert/Alarm – Notification in case of Emergency (External aspects).
- Procedure PS-116 Emission of acoustic alarms in CNA I.
- Procedure PS-120 Emergencies Brigade.
- Procedure PS-04: Performance in Medical Emergencies.
- Procedure PS-29: People Decontamination.

The Local Government of Zarate has also a “Local Emergency Plan” that defines the organization to deal with nuclear emergencies.

5.1.1.1.2. Procedures referred to agreements with external organizations and their activation protocols

CNA I has assistance agreements with external organization in order to ensure an appropriate collaboration in case of emergency.

- Agreement with Fire Fighter of Lima: This agreement provides the cooperation from the Fire Fighter in case of emergency and its involvement in nuclear emergency drills that are performed regularly, as well as the personnel training of the Fire Fighter Brigade.
- Agreement with National Guard: In normal conditions this agreement provides the control tasks and defence from the National Guard in strategic spaces of the installations and their normal operation. In case of emergencies National Guard distributes the stable iodine pills to the people involved within 10 km around the NPP. It also evacuates the people located 3 km around the NPP.
- Agreement Atomic Energy National Commission (CNEA) – NA-SA. “Emergency Management in Atucha Nuclear Site”. This agreement provides the formal cooperation between NA-SA and CNEA, organization in charge of the CAREM reactor development, located in the boundary site next to CNA I, allowing the coordination of the application of measures in case of nuclear emergency.
- Agreement with the Staff Transportation Company: In case of emergency it allows the evacuation of CNA I personnel not required for these circumstances, using the buses located in the surrounding area.

CNA-I is also part of the Safety Zone Committee Zarate-Campana – Community Alert Plan of Industrial Emergencies (PACEI) which main objectives are:

- To inform the members of the community about the industrial operations in the zone, as well as the risk involved and the measures taken to reduce them.
- To revise, update or establish response plans in case of industrial emergencies, within the cities of Zarate and Campana.
- To integrate the industrial emergency plans with those that belong to the local Governments, obtaining as result a global plan that allows dealing contingences derived from the industrial activity.
- To incorporate all the members from local community in the development, practice and implementation of the global plan in case of industrial emergency.

5.1.1.1.3. Evacuation plans and personnel assistance

In case of evacuation and assistance, CNA I has currently the following documents:

- Emergency Plan: it defines the evacuation of the personnel not specifically involved in the emergency response.
- Procedure PS-04 "Performance in Conventional Medical Emergencies": It defines the necessary human and material resources as well as the provisions to deal with eventual medical emergencies.
- Procedure PS-120 "Emergencies brigades": It defines the mechanisms for which the Organization has the trained staff to response in a nuclear emergency.
- Procedure PS-116 "Acoustic alarms emission in Atucha I Nuclear Power Plant": It defines the alarm codes that alert and notify the emergency situations on site to the personnel and emergency response groups.
- Procedure PS-29 "Decontamination of Personnel"

5.1.1.1.4. Procedures for the preparedness and coordination for the reception of additional material, equipment and personnel

CNA I has performed a preliminary procedure that includes the receptions of additional materials, equipment and personnel from Embalse NPP (CNE). Discussions and meetings with the staff involved about the emergency management are scheduled for 2012. It is expected to conclude the final version of this procedure by 2013.

5.1.1.1.5. Resources of necessary professional staff

CNA I is conformed with 6 groups integrated by Operation Personnel, Radioprotection, Physical Protection, Maintenance y Laboratory in the following functions:

- 1 Shift Manager
- 1 Shift Manager Assistant
- 1 Primary System Operator
- 1 Secondary System Operator
- 1 Refuelling System Operator
- 2 Primary Technical Assistance
- 2 Pool Technical Assistance
- 4 Operations Multiple Auxiliaries
- 1 Shift Electrician
- 1 Shift Electronic
- 1 Shift Chemical
- 1 Radioprotection Officer
- 1 Radioprotection Officer Assistant
- 2 Physical Protection Assistant
- 2 Physical Protection Auxiliary
- 1 Lodge Auxiliary
- 1 Nurse

Furthermore, there is a scheduled operational team in CNA I as available personnel (212 agents) to support or replace the service personnel when necessary, for normal or emergency situations.

5.1.1.1.6. Identification of organizations and external installations that may dispose the knowledge and appropriate resources to assist in an emergency management

Other organizations have been identified by CNA I, those who actively assist during nuclear emergency drills such as

- Local government of Zarate
- Civil Defence of Zarate
- National Guard. (Atucha Safety Troop)
- Coast Guard of Zarate
- Argentinean Navy (Base Naval Zárate)
- Buenos Aires Province Police (Lima and Zárate)
- Fire Fighter Brigade (Lima and Zárate).
- Local Hospital Virgen del Carmen (Zárate)
- Local radio stations (Lima)
- CNE (Embalse, Cordoba Province)

5.1.1.1.7. Procedure including the possibility of sharing personnel on site with more than one unit, as well as the necessary resources for the emergency management in all the units of the site

According to the joint operations of CNA I and CNA II, a joint training program is being elaborated in order to allow the involvement of the staff in the emergency response in any of the nuclear power plants of the site. This program will be implemented by the end of the CNA II commissioning (2013).

5.1.1.1.7.1.

It is expected to modify during 2013 the Procedure PS 101 "Conformation and Operation of the Internal Emergency Control Centre (CICE)" in order to include staff from the other Unit.

5.1.1.1.8. Safety areas to store the necessary material and equipment to implement the response plan

For emergency conditions, three cabinets and three storages for conventional emergencies and six cabinets for nuclear emergencies are available, destined to store materials and equipment for emergencies, located in strategic places in-site and off-site. These are located in:

- Switchgear Building
- Lodge
- Medical Service Building
- Buffet room.
- Off- site meeting point (Off site CNA I).
- External Emergency Control Centre, CECE (CNA I Neighbourhood– Lima).
- Meeting point CAREM project building.

5.1.1.1.9. Establishment of safety areas against eventual fires and radiological exposures

The personnel that must stay in the plant is provided by safety places to be distributed if necessary. These areas are:

Main Control Room (MCR). The MCR has the necessary equipment and instruments for the operation and control of the plant, and for the evaluation of the radiological consequences of the accident. Its design allows the habitability in case of accident and is provided by documentation and personal protection equipment according to the tasks to be performed by the personnel. It also has redundant communication systems with involved Authorities, the emergency control centres and the exterior.

Internal Emergency Control Centre (CICE). It is located +20 m level from Switchgear Building. Here are the personnel in charge of the emergency management. CICE is equipped with personal protection equipment and redundant communication media (telephones, radios) with the exterior to ensure the information with the authorities involved and to coordinate the external assistance if required. It has computers to access to the information and necessary documentation, including data from meteorological towers and the Perimeter System for Measuring Environmental Dose Rate.

External Emergency Control Centre (CECE). It is a building similar to CICE located in Atucha I neighbourhood, 9 km away from the installation. It has redundant communication equipments (telephones, radios) with the exterior to ensure the information to the authorities involved and to coordinate the external assistance, required in case to evacuate CICE. It has computers to access to the information and necessary documentation, including data from meteorological towers and the Perimeter System for Measuring Environmental Dose Rate.

5.1.1.1.9.1

It is expected to re-locate CICE in a new common lodge for CNA I and CNA II. This building will be provided with assured power supply, a ventilation system with positive pressure and adequate filters, so that the personnel in charge of the emergency may stay for a long time.

A new Medical Service will be provided with a meeting point for 1,300 people. It will be provided with assured power supply, a ventilation system with positive pressure and adequate filters.

The conceptual engineering stage of the project was finalized and it is expected to complete these improvements by 2015.

5.1.1.1.10. Appropriated zones, free of obstacles, to facilitate the arrival of external assistance by light aerial vehicles

In case of a long term nuclear emergency, CNA I has places free of obstacles, especially provided in order to facilitate the arrival of external assistance by land vehicles and suitable areas for helicopter landing.

5.1.1.1.11. Procedures to ensure the effective communication of the emergency direction with the control room, the safety areas, the operation shift, the recuperation equipment, the fire equipment, other external organizations, ARN, etc.

The Procedure PS-104 associated to Emergency Plan has specific instructions about communications during emergencies. An alarm code is defined to alert and notify the personnel on site, the response organizations in emergencies and involved people within 10 km radius around the plant.

To announce the emergency situations to the on-site personnel, four acoustic signals are transmitted through speakers of the plant. These signals are used to indicate General Alarm, Fire Alarm, Leak Alarm and Evacuation Alarm.

The communication of the emergency situation to the Nuclear Regulatory Authority (ARN) and to the external organizations before mentioned is conducted by telephone, fax, VHF radio or the radio system of the National Guard, which cover all the country. In some situations the Police or emissaries are also destined to this task. The personnel not on duty is notified by telephone or VHF radio.

The Emergency Plan of CNA I define two levels of alert / alarm, Green Alert to indicate that the plant is in emergency state or that may derivate in a nuclear emergency, but without a release of nuclear material to the exterior. Red Alarm is used to indicate that the plant is releasing radioactive material or that the release is imminent.

5.1.1.1.12. Analyses of the availability and compatibility of necessary communication equipments, necessities of additional emergency equipments, batteries and chargers of such equipments, etc.

CNA I is provided by landline and mobile communication equipments that integrate all the involved organizations in the communication network of emergency, and ensure an effective communication in any accidental scenario. These equipments are monthly tested and their integral use is annually performed in the framework of the Emergency Plan drills.

Furthermore, CNA I has a system of radial link in VHF band (with a fixed frequency for emergencies) with the Police of Lima (Police of Buenos Aires Province), which has also communication with Zarate Police Station. A VHF link in emergency frequencies provides communication to Zarate Local Government, where the Nuclear Emergency Operative Centre is located and also to FM radios in Lima.

5.1.1.1.13. Availability of portable fire fighting, ventilation equipments, breathing equipments, personal protection equipments, etc. conveniently located in accessible and specific places

To control possible fires, CNA I has an adequate number of portable fire fighting equipments, ventilation equipments, breathing equipments and personal protection equipments, some of them are detailed below:

Equipments	Quantity
Fire engine	1
PQS 100 kg Extinguishers	2
PQS 25 kg Extinguishers	33
PQS 10 kg Extinguishers	34
PQS 5 kg Extinguishers	175
PQS 2,5 kg Extinguishers	207
PQS 1 kg Extinguishers	20
HFC 2,5 kg Extinguishers	37
CO2 10 kg Extinguishers	33
CO2 3,5 kg Extinguishers	476
CO2 1 kg Extinguishers	10
FM 200 Extinction Fixed Equipments	4
Portable foam generating equipments	7
Breathing equipments	42
Structural clothes for fire brigade	25
Venturi Ventilation equipments	5

The equipments described above are conveniently distributed on site according to the fire risk evaluated in each area, considering for this the fire charge, kind of fire and criteria from IRAM Argentinean Standards for fire protection.

5.1.1.1.14. Planning and availability of adequate emergency lighting equipments that help to perform the actions in- site and outside the buildings of the plant

Furthermore, to perform the tasks in case of station blackout (SBO) CNA I has an emergency lighting system inside the buildings powered by batteries. For the lighting outside the buildings, the plant has two portable lighting equipments, 9 portable electrical generators and a variety of flashlights.

For emergency lightening in buildings, CNA has a emergency lightening circuit powered by 400 volt emergency electric rods. The installation has also a lightening panic system distributed in stairs, floors, and doors powered by 220 VCC battery banks, as well as portable lightening equipments located in Manoeuvre Building.

For off-site lightening, there is an emergency vehicle with a portable generator and luminaries, three 6000 W electrical generators and a 380 V-110 KVA generator trailer.

5.1.1.1.15. Availability of cables, prefabricated connections, adapters, fire hoses, etc, necessary for the performance of cooling and ventilation system, and also to ensure the operability of the instrumentation and the necessary critical elements according to the mitigation strategy

The plant is provided by cables, prefabricated connections, adapters, hoses, etc. for the equipments to be used in emergency situations and mitigation strategy.

5.1.1.1.16. Availability of applicable procedures and drawings

Emergency drawings and procedures are located in the following redundant locations,

- Main Control Room.
- CICE Room.
- CECE Room of CNA I neighbourhood.
- Engineering Archives in Switchgear Building, where Support Group of CICE works.
- Computer system through Plant network.

5.1.1.1.17. Aspects related to the radiological protection of the personnel and to the use of adequate models of dispersion to calculate the required dose and try to reduce it fully

CNA I has meteorological information and a mobile monitoring unit to measure the radioactive isotopes concentration that support the use of atmospheric dispersion models. The model for the calculation of the atmospheric dispersion allows:

- the atmospheric dispersion.
- the source term.
- the dose and aerosol concentration along the central line of the dispersion plume.
- the maximum concentration of aerosols at ground level and the maximum dose.

5.1.1.2. Reduction of Radioactive Releases

5.1.1.2.1. Availability of an appropriate water supply

The availability of an appropriate water supply was identified as fixed source associated to the water system against fires. Its primary objective is to provide water from the river to the plant in order to be used against fires. The system is sub-divided in 4 parts: an annular circuit off site of the buildings, branch lines for the buildings, sprinklers for transformers and an installation for the second heat sink building.

Complementary a fire engine is provided in order to be constituted firstly as a support equipment to the fire water network, and it is appropriate for the intervention in external installations where the mentioned network does not reach.

Finally as external support of the plant, there are fire engines that belong to the Fire Fighter Brigade of Lima and Zarate and the CAREM Project as a fixed network alternative to the CAREM project fire network (1.000 m^3).

5.1.1.2.2. Use of fire fighting equipments (water and foam) for “washing” possible radioactive emissions

As provided in Procedure PS-116 “Emergency Brigade”, the fire extinction using water is complemented by foam generating equipments that may contribute to the washing process of radioactive material deposited in areas in order to recover their accessibility. For the purpose indicated, three low expansion foam generating equipments and two high expansion foam generating equipments are provided.

Besides, the fire engine of the plant may be used for the washing of radioactive deposits.

5.1.1.2.2.1.

CNA I is also developing the specification for a foam and air fire fighting system (canyon for attack at distance) in order to mitigate fires in the heliport zone. This system is settled on a mobile basis and it could eventually assist in washing tasks or retention of particulate radioactive material. This improvement will be completed by the end of 2012.

5.1.1.2.3. Availability of temporary tanks to store large volumes of radioactive water; sand bags to facilitate the construction of temporary dikes or stabilizing materials to control and retain the flow of used water, etc.

To store large volumes of radioactive water, a 100 m^3 tank (old fuel oil deposit) and two 100 m^3 and 20 m^3 cutwaters were identified.

In order to facilitate to the Brigades the availability of materials and equipments, 5 areas for materials and equipments for emergencies are distributed in the main buildings of the plant, which can be transferred when required.

The areas have personal protection equipments, rescue equipments, extinguishers, foam liquids, accessories against fire, absorbing materials and spill over containers.

Complementary, kits of spill over containers were installed in different points of the plant. These are provided by Tyvek suites, gloves, absorbent blankets, absorbent pads, absorbent sleeves and waste bags.

CNA I storages are disposed with 200 lt. drums in order to collect the washing radioactive material and to proceed with its conditioning and temporary storage.

5.1.1.3. Revision of procedures

5.1.1.3.1. Scope of the procedures applicable to the strategy of emergency management, including fire fighting and the corresponding recovery actions

CNA I initiated a re-evaluation process of all the valid procedures applicable to the strategy of emergency management, including the evaluation of the necessity to elaborate other procedures.

The scope of the revision included the following aspects:

1) Formal approval.
2) Existence of a valid system for the revision and modification of procedures.
3) Understanding and acceptance of these procedures by the personnel (all levels).
4) Evidences that the procedures are monitored.
5) Adequacy of the procedures to the good practices.
6) Clearness of the procedures taking into account the principles of human factors.
7) Fulfilment and updating of these procedures with the hypotheses and results from the safety analyses, plant design and operative experience (including the lesson of Fukushima)

The revision of the following procedures related to the emergency management has been performed:

PS-101 - Conformation and Operation of CICE.
PS-104 - Definition of alert/notification alarm in case of emergency.
PS-108 - Distribution of stable iodine tables.
PS-110 - Access control of affected zones.
PS-115 - Meeting points.
PS-116 - Emergency brigades.
PS-118 - Emission of acoustic internal alarms.
PS-120 - Conventional emergencies.

At the end of 2012 the revision of the following emergency management procedures will be concluded:

PS-102 - Evaluation of the state of the plant.
PS-103 - Evaluation of the meteorological condition.
PS-105 - Declaration of Emergency.
PS-106 - Environmental Monitoring in emergency situation
PS-107 - Sending of representatives to COEM.
PS-109 - Coordination of the sheltering.
PS-111 - Evaluation of the external radiological consequences.
PS-112 - Preventive evacuation selective of inhabitants.
PS-113 - Emergency communications and periodic controls protocol.
PS-117 - Emergency elements
PS-119 - Expected abnormal Situations and Emergencies

5.1.1.3.2. Interferences between the components of security and the necessary mobility and facility of access during emergency situations.

The interferences that may affect the response actions during an emergency are associated with three control points integrated to the Physical Protection System of the plant:

- Turnstiles of staff access.
- Lock gate of vehicles access.
- Gate Nº 3 of dock access.

Once the emergency situation was declared, the instructions to be executed by the monitoring staff in charge of Physical Protection are:

- Unblock the turnstiles in order to allow the free access or departure of the staff.
- Open the lock gates for vehicles.
- Open the gate Nº 3 in order to allow the provision of stable iodine tablets to the Coast Guard.

5.1.1.4. Personnel training

5.1.1.4.1. Exercises and Practices

CNA I has a training program through a systematic training process especially for the Operation personnel, Emergency Brigade, members of the Safety Internal Committee who are part of the CICE and support personnel of the emergency organization (radiological control, evaluation, intervention and logistic).

According to its functions and licensing requirements, the operating personnel is subject to a theoretical – practice training program in a classroom and in a full scope simulator. The training includes emergency situations, abnormal situations, relevant design modifications and changes in operative procedures. The training in simulator is implemented every two years and is placed in the full scope simulator of Angra II Nuclear Power Plant (Brazil). It is focused on the normal operation, emergency and abnormal situation operation and comprehension of physical events that rule the behaviour of the installation.

The annual training program of the Emergency Brigades is theoretical-practice and is implemented theoretically, on field and in the specialized institutions. Firstly theoretical courses are given, secondly drills on field are implemented oriented to exercise and evaluate the performance of the Emergency Brigades, necessities of equipments and updating of procedures. The program is completed with the annual involvement of the Emergency Brigades in training in a fire drill of the Fire Argentinean Centre, member of the National Fire Protection Association (NFPA).

Training of the personnel from CICE and support areas (given its hierarchy and managing responsibilities in response organization) is regulated by the document “Training requirements for the staff performing specific functions”. The staff is annually involved in re-training courses about Radiological Protection, Industrial, Safety, Preparation for Emergency, First Aids, Environmental Management as well as Application Exercises in the Emergency Plan.

Finally the staffs before mentioned complement its training being part of the application exercises in the External Emergency Plan.

5.1.1.4.2. Familiarization with the plant

The familiarization plan provides theoretical courses and walks around the plant, in order to ensure that all CNA I workers receive the necessary information to enter, stay, move, perform the tasks and leave the plant with safety, during the normal operation as well as during emergency situations. The topics include:

- Description and operation of the plant. Distribution and communication among the buildings. Location of main components and access control.
- General rules of health, industrial safety, environmental management and radiological protection
- Specific work rules and personal protection in the operations to be implemented.
- Biological effects from ionizing radiations and risks identifications in the plant.
- Procedures of access, behaviour and controlled zone work.
- Good practices.
- Intervention in emergency situations.

5.1.1.4.3. Coordinated exercises between external organizations and personnel from operations, maintenance, recovery actions, etc.

An Emergency Exercise is annually performed. This exercise (drill) are performed in an alternate manner, one year an internal drill is carried out and the next year the drill includes the off-site emergency with participation of the people involved in the interest zone. The external organizations and ARN participate in these exercises.

5.1.1.4.4. Joint training with external organizations (protocol of assistance)

The joint training is implemented as part of the preparation for Emergency Exercises in the established framework. This training is carried out according to the cooperation agreements and existent services as mentioned in 7.1 (b).

In this training the decision making is practiced, as well as the communications and application of automatic actions (preventive evacuation of the people within 3 km radius, sheltering, prophylaxis with stable iodine and access control).

The involved organizations are:

- Local Civil Defence of Zarate,
- National Guard (Atucha safety Troop),
- Argentinean Coast Guard (Zárate Port),
- Argentinean Army (Zárate Naval Base),
- Police from Buenos Aires Province (Lima and Zárate),
- Local Hospital Virgen del Carmen (Zárate),
- Fire Brigade (Lima and Zárate) and,
- Local Radio broadcasters (Lima).
- CAREM Project Staff (CNEA)

5.1.1.4.5. Procedure for the use of equipment and special materials

The use of equipments and specific materials destined to the emergency situations management is regulated by Procedure PS-117 “Emergency elements”. Furthermore, each external organization has procedures about the equipment to use.

5.1.1.4.6. Training program of personnel on site with more than one unit

The Emergency Plan correspondent to the joint operation of CNA I and CNA II will be available before the first criticality of CNA II and it has the procedures that regulate the common actions, among them the joint training and response in emergency situations.

In the current situation, personnel from CNA II Project are considered as public members so it is subject of the preventive evacuation measures in 3km radius.

5.1.1.4.7. Training program of the external organization staff from the plant or other similar nuclear power plants

In the scope of the joint operations to CNA I and CNA II, a joint training will be available according to the Specific Agreement N°5 “Emergencies Management in Atucha Site”.

5.1.1.4.8. Training Program in equipment coupling and alternative instrumentation to power supplies and water, as well as device performance in critical or degraded situations

Training on equipment coupling and alternative instrumentation to power supplies and water, as well as device performance in critical o degraded situations is implemented through the Training Program of CNA I Staff.

5.1.1.4.9. Training program in the use of devices, accessories and special clothing

According to the Training Program, CNA I staff receives training in the use of devices, accessories and special clothing to be used during the normal operation of the plant as well as emergency situations.

5.1.1.5. Equipments

5.1.1.5.1. Availability of connectors for the coupling of the necessary auxiliary equipments, as well as the procedures for its effective alignment

Connectors for the coupling of the necessary auxiliary equipments are available, for which there are operation procedures. The personnel in charge of its utilization perform periodic practices. This is also contemplated for the new installations and equipment detailed in this document.

5.1.1.5.2. Capacity of power supply and cooling required to maintain the plant safety functions, the availability of equipments and components to interconnect easily the power supplies and auxiliary cooling circuits, and to power the instrumentation and the devices required

5.1.1.5.2.1.

On the basis of the analyses, the electrical power to be supplied by a mobile diesel generator and the electrical installations necessary for its operation was determined. This mobile diesel generator has the capacity to electrically feed:(see Figure 2-6 / item 2).

- A feed and bleed TA4 pump and associated valves connected to the primary circuit,
- RL33D01 pump of the feedwater system,
- Valves and vent control regulated by steam generators,
- UJ pump of the potable water system,
- A pump to be connected to UX branch line of the Second Heat Sink system (SSC) for core cooling through a steam generator and associated valves (see Figure 2-4 / item 2),
- A pump to replace water to the spent fuel pools provided by underground water.

According to the above detailed, a 680 KVA mobile electrical generator was decided to be incorporated by the end of 2013.

5.1.1.5.2.2.

An operative emergency procedure is being developed to response in case of loss of cooling or inventory spent fuel storing pools. This procedure includes the monitoring of level and temperature of the pools during emergency and the possibility to replace inventory even in the following conditions:

- Loss of control room
- SBO
- Earthquake and flood.

The following measures will be accordingly adopted:

- The installation of a pump to provide underground water, independent from the destined to the fulfilment of other functions during the event, which will provide the necessary reposition flow to the pools.
- The installation of an electrical board, in order to allow the electrical power of the pump, from a safety rod and mobile diesel generator.

Its implementation is estimated by the end of 2013.

5.1.1.5.2.3.

A modification is proposed to replace the inventory of water to the steam generator through the SSC, in case of loss of feedwater tank and the posterior cooling and safety system of assured water injection to the steam generators. Furthermore, in cases where the integrity of SSC is not affected, inventory to SSC tank could be refilled. Water will be injected in UA00B03/B04 water pools to the depressurized steam generator, using both pumps UA10D20 and D21 and adding water to the pools with underground water using UJ pumps (see Figure 2-6 / item 2).

The proposal contemplates also the possibility of feeding the involved components with an external generator in case of SBO so that the diesel generators from SSC are not available. Its implementation is foreseen for 2013.

5.1.1.5.2.4.

CNA I has three pumps from UK system, each one provides 50% from the necessary flow of water in normal conditions and 100% in emergency conditions. In normal conditions two UK pumps are in operation, and the third is reserved.

For the loss of the CNA I pump house, a forth pump was decided to be incorporated to ensure the cooling of the plant. This forth pump will be located in pump house of CNA II and will provide water from 1m underground and 2m above the level of the existing UK pumps, so it will improved the operation management of the plant in case of low / high water level of Parana River. In conditions of high water level this pump will provide water, even when the river has overpass the altitude of Atucha I pump house (5,17m).

Its implementation is expected to be performed in 2013.

5.1.1.5.2.5.

The new system of emergency power (EPS), under construction, incorporates three diesel generators, each one is able to provide 100% of the power demanded to EPS, in order to keep the safety conditions of the plant. These generators will be located in a building specially constructed to this goal. This modification will be completed in 2013.

5.1.1.5.3. Impact in other relevant areas of the plant (control room, switch rooms, cabling, relays, etc.) that may cause the loss of electrical supply

A study is being performed in order to verify the impact on relevant areas from CNA I in a SBO scenario. Particularly, it is pretended to ensure the electrical feeding to the instrumentation correspondent to representative signals of the variables necessary to control the core or at least, to know its state and evolution. The possible reduction of charges powered by batteries is also contemplated, in order to extend its availability until the connection of alternative electrical feeding is completed.

The definition of necessary instrumentation for monitoring the core and the containment was already performed, and it is expected to conclude this modification during 2013.

5.1.1.5.4. Availability of additional recovery equipments in safety areas

Additional recovery equipments are installed or stored in many safety areas. The equipments that are part of the proposed improvements will be also stored considering the safety and diversity.

5.1.1.5.5. Equipments of the mitigation strategy

The equipments related to the mitigation strategy are subjected to an appropriate program of maintenance, tests and inspections. Furthermore, it is expected for the new equipments to be installed an elaboration of maintenance, test and inspection program on the basis of the experience and recommendations of the manufacturers.

5.1.2. ATUCHA II NUCLEAR POWER PLANT

Because Atucha II Nuclear Power Plant (CNA II) has recently initiated the commissioning activities, the revision and verification of the topics related to emergency preparedness and management has not finalized yet. Particularly, the correspondent topics related to direction and control, damage mitigation to the fuel, reduction of radioactive emissions, revision of procedures, personnel training and equipments. Because of this situation, the results of this review are not available for the moment.

5.1.3. EMBALSE NUCLEAR POWER PLANT

5.1.3.1. Direction and Control

The verification of the situation of the following topics was performed, in order to detect any weaknesses and to define the correspondent actions that are necessary to implement:

5.1.3.1.1. Valid procedures to redistribute and relocate in safety areas the personnel that must stay in the plant (including operation personnel)

The Embalse Nuclear Power Plant (CNE) Emergency Plan defines the strategy, methodology, organization and necessary ways to deal with the emergencies that may occur. The Emergency Plan is supported by procedures that define the strategy and particular responsibilities.

The Procedure PS-103 “Preparation for Emergencies- Alerts – Notification in case of Nuclear Emergency” and the Operations Procedure POP-009 establish the guidelines to follow in order to evacuate the personnel.

Those procedures include the instructions that are applicable to the redistribution and relocation in safety areas, both for the personnel that must stay in the plant and the personnel no necessary to deal with the emergency, and also the different ways to accede to the Secondary Control Room in case of a seism or a disqualification of the Main Control Room and the priorities for its choice.

The personnel that must stay in the plant is provided by safety areas which can be redistributed. The main areas are:

- Main Control Room (MCR)
- Secondary Control Room (SCR).
- Drawing Room of the Main Control Room.
- Alternative Centre for the Intern Emergency Control Centre (IECC).

As result of a review of the mentioned procedures some weaknesses were detected, so it was decided to implement the following improvements:

5.1.3.1.1.1.

It was detected that the Secondary Control Room does not have a filtered air recirculation system that allows its habitability after the presence of smoke or emission of radioactive material to the atmosphere during a nuclear accident.

Accordingly, design changes in the ventilation system are expected. This includes the installation of a 100% air recirculation system through activated carbon filters and absolute filters in this room. The fans of this system will be electrically powered by the emergency power supply system (EWS) that also supplies this room. This improvement is expected to be performed during the reconditioning stage for the life extension of CNE, estimated by the end of 2016.

5.1.3.1.1.2.

IECC Building has deficiencies in seismic and ventilation aspects. A modification in seismic aspects was decided, so a ventilation system with HEPA filters / activated carbon and emergency power supply will be installed. Besides, the communication system will be improved to support severe accident conditions for a long term. This is expected to be performed during the life extension of CNE estimated by the end of 2016.

5.1.3.1.2. Procedures referred to agreements with external organizations and their activation protocols

CNE has the following assistance agreements with external organizations, in order to ensure an appropriate collaboration in case of emergency:

- **Agreement with Embalse Fire Brigade:** This agreement facilitates the cooperation from the local Fire fighters in case of emergency and their involvement in nuclear emergency drills that are regularly carried out, and also the personnel training of the Fire fighters body.
- **Agreement with National Guard:** In normal conditions this agreement facilitates the control tasks and defence from the National Guard in strategic places of the installations and their normal operation. In case of emergencies the National Guard distributes the Iodine pills to the population involved within 10 km. It also evacuates the population located 3 km around CNE.
- **Agreement for Medical Assistance with the Local Government of Embalse:** it allows to co-ordinate the medical assistance of first aids.
- **Agreement with Personnel Transportation Company:** In case of emergency it allows the evacuation of CNE personnel not required for these circumstances, using the buses located in the surrounding area.

5.1.3.1.3. Evacuation plans and personnel assistance

In case of evacuation and assistance, CNE has currently the following plans and procedures:

- **Emergency plan:** In case of internal evacuation, the personnel that do not have a specific task in the management of the emergency situation must go to the meeting points determined for the evacuation.
- **Procedure PSM-04 "Organization and Performance of Medical Service in Emergency":** It defines the necessary human, material resources and provisions to deal with eventual medical emergencies.

- **Procedure PS-060 “Superficial Decontamination of Human Body”:** It defines the conditions and methods to carry out with the decontamination, in normal conditions and emergencies.
- **Procedure PS-104 “Automatic Countermeasures”:** It defines the conditions and activities required to implement the automatic countermeasures in case of nuclear emergency.

5.1.3.1.4. Procedures to the preparation and coordination for the reception of material, equipments and additional personnel

When the present report was performed, CNE was still reviewing and improving the mentioned procedures.

5.1.3.1.5. Necessary professional personnel resources

The organization of an emergency on site is constituted progressively from the beginning of the incident / accident, being integrated by the service personnel on charge, formed by minimum 20 persons distributed as follows:

- 1 Shift Manager
- 1 Shift Manager Assistant
- 1 General Operator
- 1 Control Room Assistant
- 1 or 2 Technical Assistants
- 4 Assistants
- 1 Chemist
- 1 Refuelling System Operator
- 1 Shift Mechanic
- 1 Shift Electronic
- 1 Shift Electrician
- 1 Radioprotection Officer
- 1 Nurse
- 1 Physical Safety Operations Supervisor
- 1 Physical Safety Operator Assistant
- 1 or 2 Training Personnel

Furthermore, there is permanently an operation team as available personnel to support or replace the service personnel when necessary, for normal or emergency cases.

CNE has a scheduled guard system, with 197 available agents in 51 groups. The relevant areas involved in this guard system are:

- CNE Management
- Operations Management Assistant
- Safety and Radioprotection Management Assistant
- Engineering Management Assistant
- Maintenance Management Assistant
- Quality Assurance Management Assistant
- Production Management Assistant
- Engineering Department
- Nuclear Safety Department
- Technical Support Department :
- Mechanical Maintenance Department
- Electrical Instrumentation and Control Maintenance Department
- Civil Work Division:
- Medical Assistance Division
- Public Relationships Division

5.1.3.1.6. Identification of organizations and external installations that may provide adequate knowledge and resources to assist in emergency management

Other organizations have been identified by CNE, those who assist actively during nuclear emergency drills, such as:

- Civil Defence of Cordoba Province
- Local Boards of Civil Defence of Embalse, La Cruz, Villa del Dique and Villa Rumipal
- Atucha I Nuclear Power Plant
- Nucleoeléctrica Argentina Sociedad Anónima (NA-SA)

5.1.3.1.7. Procedure that considers the possibility of sharing personnel on site, with more than one unit, and also the necessary resources for the accident management in all units of the site

No applicable, given that CNE has only one unit and there are no plans to install a second one.

5.1.3.1.8. Safety areas to store the material and the necessary equipments to implement the response plan

For emergency conditions in CNE 12 emergency storages are provided, accordingly identified and located in strategic places of the site, that are destined to store materials, equipments and documentation for emergencies. These areas are located in:

- Main Control Room Storage
- Secondary Control Room Storage
- Storage in second floor. (entrance to SCR)
- Office from Radioprotection Chief (Controlled area)
- Emergency Power System Room (EPS)
- Auxiliary nave
- Class III Diesel Generator Buildings
- Emergency Water System Building (EWS)
- Emergency Core Cooling System Building (ECCS)
- Water Treatment Plant Building (Storage Room)
- IECC Room
- Medical Assistance

5.1.3.1.9. Setting up of safety areas from eventual fires and radiological exposures

In case of eventual fires and radiological exposures, the safety areas on site are:

- **Main Control Room (MCR):** the MCR has the necessary equipments and instruments for the operation and control of the plant, and for the evaluation of the radiological consequences of the accident. Its design allows the habitability in case of accident and is provided by documentation and personal protection equipments according to the personnel located. It also has redundant communication systems with the involved Authorities, the emergency control centres and the exterior.
- **Drawing Room of the Main Control Room:** here is the support personnel for the emergency that is not designed to IECC or to Control Room. It is provided by personal protection equipments, technical documentation and internal telephones.
- **Internal Emergency Control Centre (IECC):** here is the personnel in charge of the emergency management in the internal aspects and the provision of the relevant information to the Operative Centre of Local Emergency. IECC is being equipped in order to allow its limited habitability in case of accident. It has also personal protection equipments and communication media redundant with the exterior.
- **Alternative Centre for IECC:** It is a room similar to IECC located in CNE Hostel, 5 km away from the plant in Embalse city.

As result of the review, some weaknesses were detected so it was decided to implement the following improvements:

5.1.3.1.9.1.

IECC Building has some deficiencies in seismic and ventilation aspects. So the building will be modified in seismic aspects, and a ventilation system with HEPA filters and activated carbon and emergency power supply will be installed. Also, the communication system will be improved to support severe accident conditions for a long term. This is expected to be performed during the life extension of CNE, estimated by the end of 2016.

5.1.3.1.10. Appropriated zones, free of obstacles, to facilitate the arrival of external aid by light aerial vehicles

In case of a long term nuclear emergency, CNE has places free of obstacles, specially provided in order to facilitate the arrival of external assist by land vehicles and suitable areas for helicopter landing.

5.1.3.1.11. Procedures to ensure the effective communication of the emergency direction with the control room, the safety areas, the operation shift, the recuperation equipment, the fire equipment, other external organizations, NRA, etc.

The procedure PS-103 “Preparation for Emergencies – Alerts – Notifications in case of Nuclear Emergency”, associated to the Emergency Plan includes specific instructions about communications during emergencies. In this procedure an alarm code is define to alert and notify the on-site personnel, the organizations involved in the emergency response and the population within the 10 km around the power plant.

To announce the emergency situation to the on-site personnel, five acoustic signals are transmitted through speakers of the plant. These signals are used to indicate General Alarm, Fire Alarm, Leak Alarm and Evacuation Alarm.

The communication of the emergency situation to the Nuclear Regulatory Authority (NRA) and to the external organizations before mentioned is conducted by telephone, fax, VHF radio or radio system of the National Guard, which reaches all the country. In some situations the Police or emissaries are also destined to this task. The personnel not on duty is called by telephone or VHF radio.

CNE has also an emergency siren system available in the four cities within the 10 km around the Power Plant (Embalse, Villa del Dique, Villa Rumipal and La Cruz). Its function is to alert the population at the first time of the event and it is expected that the communication with the population continue through local radios, which also have communication with the plant and the COEM through VHF radios.

5.1.3.1.12. Analyses of the availability and compatibility of necessary communication equipments, necessities of additional emergency equipments, batteries and chargers of such equipments, etc.

CNE is provided by landline and mobile communication equipments that integrate all the involved organizations in the communication network of emergency, and ensure an effective communication in any accidental scenario. These equipments are monthly tested and their integral use is annually tested in the framework of exercises of application of the Emergency Plan.

Furthermore, CNE has an access to the national / international telephone system through lines of landline telephones hired by different providers, a system of radial link in VHF band (with a fixed frequency for emergencies) with the Provincial Police of all the neighbouring cities, civil defences of Embalse, Villa del Dique, Villa Rumipal and La Cruz and with local FM radios. Alternatively, the radial link system of National Guard may be used.

Nevertheless, as result of the review some weaknesses were detected so it was decided to implement the following improvements:

5.1.3.1.12.1.

Satellite landline and mobile phones for communications are in a purchase process, to be used as back-up for existing communication systems. It is expected to be in operation by the end of 2012.

5.1.3.1.13. Availability of portable fire fighting, ventilation equipments, breathing equipments, personal protection equipments, etc. conveniently located in accessible and specific places

To control possible fires, CNE has an adequate number of portable fire fighting equipments, ventilation equipments, breathing equipments and personal protection equipments, some of them are detailed below:

Equipments	Quantity
Fire engine	1
Forest fire extinction module	2
PQS 100 kg Extinguishers	3
PQS 50 kg Extinguishers	28
PQS 10 kg Extinguishers	48
PQS 5 kg Extinguishers	187
PQS 2,5 kg Extinguishers	87
HFC 5 kg Extinguishers	11
HF 2,5 kg Extinguishers	81
Foam generating equipments	6
Emulsifier liquid tanks, 1500 litres	4
Breathing equipments	25
Structural clothes for fire brigade	25
Integral equipments for chemical emergencies	6
Venturi Ventilation equipments	6

To perform the tasks in case of blackout (SBO) CNE has an emergency lighting system inside the buildings, powered by batteries. For the lighting outside the buildings, the plant has two portable lighting equipments, 9 portable electrical generators and a variety of flashlights.

As result of the review, some weaknesses were detected so it was decided to implement the following improvements:

5.1.3.1.13.1.

It was noted the necessity of strengthening the supply to the battery chargers that supply the emergency lightning. According to this, it is expected to install a 550 KW emergency diesel equipment available by December 2014.

5.1.3.1.14. Planning and availability of adequate emergency lighting equipments, that help to perform the actions in- and outside the buildings of the plant

In case of SBO scenario with loss of EPS system, this system is expected to provide the required electrical energy to the battery chargers of Class I rods (continual current secured with batteries) and Class II (powered from Class I rods). Also in this situation this system will feed the fire fighting pumps. The electrical feeding to the emergency lightings of the buildings is performed from Class I rods (batteries).

As result of the review, some weaknesses were detected so it was decided to implement the following improvements:

5.1.3.1.14.1.

An electrical panel is expected to be built outside the Service Building with connection to the lightening system. After this modification is implemented, the Personnel of Operation and the Electrical Maintenance Personnel will be trained for the connection and operation of this system. This improvement is expected to be concluded by the end of 2014.

5.1.3.1.14.2.

For the long term performing tasks outside the buildings, it was decided to incorporate:

- Three 5.5 KW portable electrical generators with lightening columns.
- Three 6 KW mobile diesel generators equipments fitted on trailers with mobile lightening columns and associated cables.

This improvement is expected to be concluded by the end of 2013.

5.1.3.1.15. Availability of cables, prefabricated connections, adapters, fire hoses, etc.; necessary for the performance of cooling and ventilation system, and also to ensure the operability of the instrumentation and the necessary critical elements according to the mitigation strategy

The plant is provided by cables, prefabricated connections, adapters, hoses, etc. for the equipments and the current mitigation strategy to be used in emergency situations. The equipments to be incorporated in the improvements will be taken into account to ensure the compatibility of the connections.

5.1.3.1.16. Availability of applicable procedures and drawings

The emergency drawings and procedures are in the following locations:

- Main Control Room
- Secondary Control Room
- Office from Radioprotection Chief
- Emergency Power System Room (Service Building)
- Class III Diesel Generators Buildings
- Emergency Water System Building
- Emergency Core Cooling System Building
- IECC Room
- Emergency Mobile
- Environment Mobile Laboratory
- IECC Room from CNE Hostel
- Administration and Bunker Archive
- Computer network of the plant

5.1.3.1.17. Aspects related to the radiological protection of the personnel, and also the use of adequate models of dispersion to calculate the required dose and try to reduce it fully

The Procedure PS-031 “Radiological calculation in early phase of nuclear accident“ describes the mechanisms for:

- The survey of meteorological data;
- The evaluation of source term;
- The estimation of atmospheric dispersion;
- The calculation of the dose and aerosol concentration along the central line of the dispersion plume;
- Determination of the maximum concentration of aerosols at ground level and the maximum dose.

5.1.3.2. Reduction of Radioactive Emissions

5.1.3.2.1. Availability of an adequate water supply

The availability of an adequate water supply was analyzed to reduce the radioactive emissions, particularly in order to improve the implementation of alternative cooling strategies of the spent fuel pool and to increase its reliability. As result of the review, some weaknesses were detected so it was decided to implement the following improvements:

5.1.3.2.1.1.

A design modification will be implemented for handling the loss of cooling or water inventory of the spent fuel pool. This modification includes an installation of a 4'' pipe from the pool to outside of the Service Building. This pipe will have an isolation valve and a coupling for the fire fighting system hose that will allow, in case of inaccessibility to the building, the provision of water from the fire fighting network through hydrants or fire engines. Besides, it is expected to have in 2015 a fire engine of 17.000 liters of water volume.

5.1.3.2.1.2.

It was verified the existence of a limited capacity of water displacement, so it was decided to have two mobile tanks of 25.000 litres each one to facilitate the continuous reposition of water to the pool and a portable pump to replace water to the tanks or the fire engine. This is expected to be carried out by December 2013.

5.1.3.2.2. Use of fire fighting equipments (water and foam) for "washing" possible radioactive emissions

The use of fire fighting equipments to remove the radioactive material deposited in vehicles or places where necessary has been provided in the planning of the nuclear emergency response. These activities are habitually carried out in emergency exercises by CNE personnel, and especially by other external organizations, like fire brigade.

5.1.3.2.3. Availability of temporary tanks to store large volumes of radioactive water; sand bags to facilitate the construction of temporary dikes or stabilizing materials to control and retain the flow of used water, etc.

In order to retain the water used in decontamination tasks, CNE and other involved organizations it is provided the construction of small trenches properly waterproofed with plastic membranes.

5.1.3.2.4. Possibility of using compounds that retain or absorb eventual radioactive particles, with the purpose of facilitating the subsequent washing and decontamination tasks, as well as the purification of aerosols

CNE has 15 HEPA portable filters with 99,97% efficiency to be used eventually in purification of aerosols.

Anti-dispersing agents (synthetic resin) are stored in safe and accessible places.

5.1.3.3. Revision of Procedures

5.1.3.3.1. Scope of the current procedures applicable to the strategy of emergency management

As result of Fukushima Accident, CNE has begun a re-evaluation process of all the current procedures applicable to the strategy of emergency management (still not finalized), including the evaluation of the necessity to elaborate other procedures. It is expected to be concluded by the end of 2013.

5.1.3.3.2. Interferences between the components of physical safety and the necessary mobility and accessibility during emergency situations

In order to avoid interferences between the physical safety measures and the necessary mobility and accessibility during emergency situations, the procedures indicate that during general alarm, physical safety must free up the gates and doors in order to facilitate the mobility of equipments and people. The personnel that evacuates the plant leaves its card in a basket by crossing the safety gate before arriving the meeting point, the security personnel introduces the cards through a card reader so the system allows the exit, reporting which person could not still leave the building and the area where is found. The security personnel also examine the areas of CNE with video cameras, make the notifications and are the responsible for controlling the communication system.

5.1.3.4. Personnel Training

5.1.3.4.1. Exercises and Practices

The training for emergency situations is under the training plan of the CNE personnel.

Members of the On-Site Emergency Organization are annually re-trained according to their responsibilities and missions in the emergency, following the personnel categories:

- Emergency Chief.
- Members of Internal Emergency Control Centre
- Radiological Evaluation and Control Personnel.
- Fire Brigade Personnel.
- Control Room Personnel.
- Physical Safety Personnel.
- Medical Assistance Personnel.

The personnel without specific missions during an emergency situation are daily re-trained about the support functions during this situation.

The theoretical-practice re-training of the Personnel of the On Site Emergency Organization is carried out by doing exercises, where specific activities from the Emergency Plan are tested. These activities are included in the Emergency Plan.

An emergency drill is annually performed, in accordance with the provisions of the Emergency Plan. Details about the preparation, implementation and evaluation of the drill are described in “Annual Report of the Application Exercise from the Emergency Plan”.

5.1.3.4.2. Familiarization program with the plant

Every person who is incorporated to work in Embalse Nuclear Power Plant receives an initial training (familiarization program with the plant) which is divided in two parts: first part is a course on “Introduction to Nuclear Power Plants”, duration of 247 hours, and are considered the following topics:

- Introduction – Personnel Induction
- Positioning
- Safety (Introduction) – Temporary Workers Course
- Electricity Market
- Industrial and Institutional relationship
- Mathematics
- Nuclear Physics
- Reactor Physics
- Radiological, Nuclear and Conventional Safety
- Nuclear Power Plants
- Quality Assurance
- Complementary topics
- Human Performance and Operational Experience.

The second part of this training describes the most important systems of the plant during 200 hours.

5.1.3.4.3. Coordinated exercises between external organizations and personnel from operations, maintenance recuperation, etc.

An emergency drill is annually performed, in accordance with the provisions of the Emergency Plan. Details about the preparation, implementation and evaluation of the drill are described in “Annual Report of the Application Exercise from the Emergency Plan”.

5.1.3.4.4. Joint training with external organizations

The Emergency Exercise is annually performed, according to the Operation License of CNE, which has two aspects: the Internal Exercise with the plant personnel and Nuclear Regulatory Authority (ARN), and every two years the External Exercise with the plant personnel, ARN, Local Boards of Civil Defence, Civil Defence Direction of Cordoba, National Guard, Police and Fire Brigade and other organizations.

The external exercise is carried out within 10 km around the Plant, and it involves one or more neighbouring cities (Embalse, La Cruz, Villa del Dique and Villa Rumipal).

In the scope of the Emergency Exercise, the training and re-training of the personnel involved in the relevant actions during a emergency case is performed:

- Sheltering
- Distribution of Potassium Iodide tablets.
- Evacuation.
- Roadblock and access control.
- Evaluation, identification of initial events and categorization.
- Communications and notifications.
- First aids, rescues and decontamination.
- Evaluation and internal / external radiological monitoring.
- Sampling procedure.
- Fire extinction.

CNE Personnel, ARN, Local Boards of Civil Defence, Civil Defence Direction of Cordoba, National Guard, Police and Fire Brigade take part in this training.

It should be noted that during the mentioned exercises, people around 10 km from the Plant carry out the Automatic Countermeasures (Sheltering, reception of iodide tablets and ingestion – when indicated).

5.1.3.4.5. Procedure for the use of equipment and special materials

CNE has the following procedures for the use of equipment and special materials that face the necessities in a severe accident scenario:

- Procedure PS-070: Personal protection equipment.
- Procedure PS-026: Methodology for the implementation of the effluent monitoring system.
- Procedure PS-013 Filter spectrometry evaluations for iodine and aerosols in laboratory in case of nuclear emergency.
- Procedure PS-099: Filter spectrometry evaluation for iodine and aerosols in trailer in case of nuclear emergency.
- Procedures PS-039 a PS-054: they detail the calibration of measurement equipments
- Procedure PS-076: Operation of tritium in air monitors.
- Procedure PS-081: Operation of superficial contamination monitors.
- Procedure PS-086: Operation, calibration and verification of portal monitors.
- Procedure PS-088: Guide for the operation and calibration of liquid scintillation equipment.
- Procedure PS-090: Calibration and operation of the body counter equipment.
- Procedure PS-092: Use and calibration of portable multi-channels.
- Procedure PS-096: Operation of exposure meters and intelligent probe.

5.1.3.4.6. Training program of personnel on site with more than one unity

No applicable, given that CNE has only one unit and there are no plans to install a second one.

5.1.3.4.7. Training program of the personnel from external organizations

The theoretical-practical On Site re-training of the personnel of the Emergency Organization is carried out by doing exercises, where specific activities from the Emergency Plan are tested. The re-training of the personnel from external organizations is done as part as the preparations tasks for drills.

5.1.3.4.8. Training program in equipment coupling and alternative instrumentation to power supplies and water, as well as device performance in critical o degraded situations

Training on equipment coupling and alternative instrumentation to power supplies and water, as well as device performance in critical o degraded situations is performed through the Triennial Training Program from the Operation Department (DOP-1-2012). The objective is to establish a systematic method of permanent training, in order to maintain the operation group sufficiently trained and skilled to deal all the abnormal conditions.

The implementation of this program divides the operation personnel in two groups, control room personnel and field operators. Both personnel receives at least 40 annual hours in a classroom and 48 hours for the control room personnel in the Simulator of the plant Gentilly II. In this training program are incorporated all the new equipments and the operative experience from the plant and from other CANDU nuclear power plants.

5.1.3.4.9. Training program in the use of devices, accessories and special clothing

According to the Planning of the SAT Program (Systematic Approach to Training) the topic “Preparation for Emergencies” is included in the Annual Course of Safety, with the objective of updating the personnel. In these programs personnel of CNE, Central Services Management, Life Extension Project Management and National Guard is included.

5.1.3.5. Equipments

5.1.3.5.1. Availability of connectors for the coupling of the necessary auxiliary equipments, as well as the procedures for its effective alignment

Connectors for the coupling of the necessary auxiliary equipments are available, for which there are operation procedures. The personnel in charge of its utilization perform periodic practices. This is also contemplated for the new installations and equipment detailed in this document.

5.1.3.5.2. Capacity of power supply and cooling required to maintain the safety functions of the plant, the availability of equipments and components to interconnect easily the power supplies and auxiliary cooling circuits, and to power the instrumentation and the devices required

About the power supply, after identifying the consumption to be powered with the mobile diesel generator (see point 5.1.3.1.14), connections boxes will be installed in order to connect it. The necessary training will be implemented for the maintenance and operation personnel, for the use in emergency situations.

The operation personnel practices annually the connection of the fire engine to the of emergency core cooling system (ECCS) for breaking down the rupture discs of the system in case of unavailability of the high pressure emergency core cooling system (HPECCS). According to the Severe Accident Management Guidelines (SAMGs), that are currently being reviewed, the personnel will be trained for their implementation together with the operation of the auxiliary cooling systems that will be installed during the life extension of CNE.

5.1.3.5.3. Possible impact on other essential areas of the plant, that may cause the loss of power supply

In the operative procedures for abnormal conditions (POEAs) and particularly in SAMGs that are under revision, the impact of the loss of power supply on essential areas of CNE is contemplated.

5.1.3.5.4. Availability of auxiliary recovery equipments in safety areas

The available recovery equipment are installed or stored in safety areas. Also additional recovery equipment to be installed, as part of improvements or tasks associated to the extension life of CNE will be properly stored or installed.

5.1.3.5.5. Equipments of the mitigation strategy

The equipments provided to be used in the mitigation strategy are subjected to a maintenance program, test and inspection in accordance to the mandatory documentation defined in the Operation License of CNE.

5.2. ACTIVITIES PERFORMED BY THE REGULATOR

5.2.1. SAFETY INTEGRAL EVALUATION (STRESS TEST)

According to was indicated in Annex I, ARN has sent a Regulatory Requirement (RQ-NASA-38) to Nucleoeléctrica Argentina S. A. (NA-SA), the Licensee of CNA I, CNA II and CNE, asking to perform a Comprehensive Safety Assessment (Stress Test) of those plants, in order to detect eventual weaknesses and implement the corresponding improvements.

The main topics referred to the emergency management, that includes this evaluations are the following:

- Measures of prevention, recuperation and mitigation of the accidents characterized as mentioned above, considering the automatic actions and the actions of the operators that are in the operation procedures for abnormal conditions, emergencies and severe accidents management.
- Availability of resources in the nuclear power plant for the response to internal and external emergencies in case of accidental scenarios.
- Previsions for the planning and management for these emergencies, including the protection of the public and communications.

5.2.2. EMERGENCY MANAGEMENT

ARN continues dealing with prevention of accidents at nuclear installations as its regulatory function. The criteria that must be adopted by the Licensee are established in the ARN document “Criteria for the intervention in nuclear emergencies with off-site radiological consequences”. Its compliance has been required to CNA I, CNE and must be applied in CNA II, in the final construction stage.

In order to accomplish what is set in Act Nº 24.804 and Decree Nº 1.390, ARN’s - Nuclear Emergency Response System (SIEN) was created by ARN Resolution Nº 25 on November 1999. The SIEN is the system that ARN uses to respond in cases of nuclear emergencies and coordinates the national, provincial and local response organizations (National Civil Protection Direction, Provincial Civil defence and Local Civil Defence of every Local Government within a 10 km radius around each NPP) to effectively manage nuclear emergencies in preparedness, intervention and recovery stages.

In particular, the operational capacity of the ARN in the management of the implementation of actions to protect the members of the public in the surroundings of the NPPs from the radiological consequences is currently checked by carrying out emergency exercises.

ARN, in addition to its main role as head of the Incident Command System for off-site consequences, performs the nuclear and radiological assessments, the radiological protection of intervening teams and the environmental monitoring. Representatives of all the intervening organizations (as established in the Emergency Plan) integrate a Response Command System and ARN coordinates the response teams that belong to civil organizations (Fire fighter brigades, Civil Defence, etc.), law enforcement (Police, Gendarmerie and Coast Guard), and military institutions (Army, Navy and Air Force). These organizations apply the precautionary measures with their response teams. All these groups have procedures to deal with nuclear emergencies, under ARN coordination.

In order to conduct the actions within the 10 km radius established as the “precautionary action zone”, a Nuclear Emergency Operative Chief (JOEN) from the ARN is designated and integrated to the Local Emergency Operative Centre (COEM). The ARN-JOEN is the officer to whom civil organizations and law enforcement must report to. An Emergency Control Center (ECC) has been set up at ARN’s Headquarters in order to coordinate the response and support to ARN representative in the COEM. ARN´s strategy to respond to a nuclear emergency consists in establishing expert teams and a decision making team at Headquarters.

ARN is the National Competent Authority according to the “Convention on Early Notification of a Nuclear Accident” and in the “Convention on Assistance in the case of a Nuclear Accident or Radiological Emergencies”, and the ECC is the National Warning Point according to the IAEA - Emergency Notification Assistance Technical Operations Manual (ENATOM).

5.2.3. EMERGENCY PLANS

The emergency plans of the nuclear power plants, approved by ARN agree with the international recommendations (IAEA, GS-R-2, 2002). Besides, the planning and response preparation in emergency situation are ruled under the regulatory standards AR 10.1.1, AR 3.7.1 and AR 4.7.1, under the operation licenses and the requirements asked to the Responsible Entity (NA-SA) and to the Primary Responsible of the facilities.

The Emergency Plan includes the application of protection actions to prevent and/or mitigate radiological consequences in accidental situations. This plan contemplates all the aspects related to the strategy to control an eventual accidental situation and to avoid or limit the consequences on the people, on the installations and on the environment of a nuclear accident. This plan includes the actions to be carried out by the operator, and also the notification and communication procedures to the external organizations such as Civil Defence, Security Forces, Fire Brigade, hospitals, etc. Furthermore, this plan contemplates the logistic and human resources such as involved staff, communication media, measurement equipments, protection equipments, etc.

In case of a nuclear emergency, ARN assigns to the Manager of the Nuclear Power Plant the responsibility of carrying out the urgent and immediate actions of protection off site, contemplated in the Emergency Plan, so the Manager is the initial Responsible of the Emergency outside the building and manages the actions until ARN comes. Then Local Operative Emergency Centre (COEM) is created on site. ARN designs an Operative Chief of Nuclear Emergencies (JOEN) who arrives to the plant and manages the COEM, in relation to the radiological protection measures of the people and environment.

To prevent the mechanisms that ensure the permanent availability of human, material and financial resources is also a responsibility of the Responsible Entity, in order to provide operative ability in emergency situations.

The Emergency Plans must accomplish with the following requirements:

- Provide an organization to deal with abnormal conditions, classifying them according to the severity
- Define and determine responsibilities to the involved staff. After the accident occurred, the Manager of the Plant must carry out the urgent protective actions contemplated in the Emergency Plan.

5.2.3.1. Severe Accidents Exercises

After Three Mile Island accident, ARN considered to reduce the residual risk of severe accidents that are included in the Criteria Curve for the Argentinean Standard AR 3.1.3, so since 1978 the accidents beyond the design basis were always considered.

5.2.3.2. Implementation of improvements

5.2.3.2.1.

The lessons of Fukushima conveyed the importance of carrying out prolonged emergency exercises. So these exercises are planning to be prolonged for 2014; the objective is to have operative emergency centres during an extended period of time to improve the transfer of emergency management, sustainability of resources, provision of technical advice, communications and reliable data.

5.2.3.2.2.

The revision of source term has been required to the nuclear power plants, in order to update the implementation of pre-calculated scenarios with radiological consequences for different types of severe accidents.

5.2.3.2.2.

To facilitate an effective response in accidental situations, information in real time is necessary, for which a network of environmental monitoring was implemented (it includes fixed and portable meteorological and radiological stations).

5.2.4. EMERGENCIES RESPONSE

In the emergency plans of Argentinean nuclear power plants, groups focused on monitoring and decontamination are contemplated. In each exercise, they work together with the Army and Coast Guard on the external decontamination of people and vehicles. Besides, the Armed Forces are involved in the management of multiple victims of natural disasters in the frame of the national response, so they have a relevant function after the accident and during the recovery stage.

In local hospitals, the installations and trained personnel to assist the contaminated / irradiated victims are identified. They work together with a network of national hospitals (by signed agreements between the National Health Ministry, the mentioned hospitals and ARN) that according to its complexity and capacity they are designated to assist contaminated / irradiated victims. These abilities are tested in every emergency exercise.

In Argentine, the coordination of radiological surveillance is in charge of ARN. In case of nuclear emergency, the radiological monitoring is expected to be carried out through monitoring groups of the nuclear power plant and ARN on the basis of local meteorological data provided by the National Meteorological Service and the meteorological towers in each installation.

Since 2003, the Emergency Control Centre (ECC) from ARN has a Geographic Information System (GIS) for the response planning in case of nuclear emergency. GIS has a data basis with demographic, economic, geographic and environmental information and incorporates the results of prevision models and environmental impact measures during the emergency.

The results of the evaluation consequences that are included in GIS for a more comprehensive analysis (integrated with the actual data basis) correspond to the results of the Accident Dose Evaluation System (SEDA), from the International Exchange Program (IXP) from U.S. National Atmospheric Release Advisory Centre (NARAC), and the results of the models of atmospheric diffusion from World Meteorology Organization (WMO) provided by the Argentinean National Meteorological Service (SMN).

SEDA is a evaluation code based on a Gaussian model of atmospheric dispersion, applicable to radioactive material releases in a nuclear power plant, relevant up to 20 km distance. The data entry includes: meteorological information (speed, direction and intensity of wind, cloud cover or vertical temperature gradient and atmospheric stability) and information from the release of pollutants (qualitative and quantitative). The results are iso-dose and soil deposit iso-concentration lines that define the reach of the protection actions measures.

IXP is a prediction system for the radioactivity atmospheric transport of a nuclear accident in real time. It is based on a web version of a NARAC computer system, where the users carry out calculations of the atmospheric dispersion and the associated dose depending on time. It employs real time information from meteorological stations that allow to know the weather conditions, as well as the forecast for the next hours. The output data is: expose rate contours, iso-lines of activity for ground deposition, and recommended protection actions.

The third model is regional (it reaches thousands of kilometres). WMO has a network of Regional Specialized Meteorological Centres (RSMC) that includes tools for the calculation of the atmospheric dispersion. When requested by ARN, the National Meteorological Service call the WMO for assistance, receives the results of the models and sends them to the Emergency Control Centre (ARN).

All these tools have been incorporated to the Emergency Control Centre and help to improve the Emergency Preparation and Response system

5.2.5. PLANNING ZONES FOR PROTECTION ACTIONS IN CASE OF EMERGENCIES

According to the international requirements for the planning and response in case of nuclear emergencies, the following planning zones are created:

- a zone of primordial preventive actions, ZPP, applicable from 3 km (in all directions) up to 10 km radius, with an angle of 45° according to the wind direction. This zone is called “key hole”. The response time contemplates the first 10 hours from the beginning of the accident.
- a zone of evaluated primordial actions, ZPE, that includes the rest of the directions within the 10 km radius. The response time goes from 12 to 24 hours.

- a zone of complementary evaluated actions, ZCE, from 10 km up to 100 km radius around the nuclear power plant. In this zone, depending on the magnitude of the accident, the completion of the application of the correspondent protection actions may require days, weeks or months.

From the re- evaluation of the external extreme events and the safety studies carried out by the nuclear power plants, it is noted that the emergency planning zones are adequate and there is no need to extend them.

5.2.6. PUBLIC PROTECTION MEASURES

5.2.6.1. Before release

For the Nuclear Power Plants, it is expected that the implementation of automatic protection actions reduce the radiation dose in the people within the planning zones. The urgent protection actions that will be executed, on the basis of the plant situation and the meteorological conditions without waiting for the results of radioactivity measurements in the environment are the followings:

- Early evacuation within 3 km radius in all directions
- Access control to the emergency zone. The control points are defined according to the situation.
- Sheltering inside the houses. This measure may be extended for a few hours; it is expected that through the local radios the people is informed about its finalization and about other instructions respect to the posterior ventilation of the houses. This measure allows reducing the cloud dose, the inhalation of radioactive material and the dose by ground deposition.

Stable Iodine Prophylaxis: the National Guard (managed by the General Manager of the Plant at the first moment of an emergency) is designated by ARN to distribute iodine (as potassium iodide tablets) to the people involved.

5.2.6.2. After release

After finalizing the first stage of the accident and the release of radioactive material, urgent measures are ratified or rectified in order to start with the non-urgent protection measures.

Personnel from Civil Defence and from other external organizations carry out the non-urgent protection measures contemplated in the Emergency Plan and managed by JOEN.

The implementation of the non-urgent protection measures will depend mainly from the results of measurements. The more significant non urgent protection measures are the following:

- Evacuation of the zones affected by radioactive deposition. This measure must be “mandatory” or “optionally” implemented depending on the dose projection and taking into account the international recommendations (ICRP 60, TECDOC 1432 and TECDOC 953).
    - Mandatory, in case that the radiation level that comes from the material deposited on the ground reaches or exceeds 100 mSv, integrated in the first 6 hours after the radioactive release, or;
    - Optionally if the same dose (100 mSv) is reached in the first 24 hours after the accident.
- Intervention in relation with the food. The levels of intervention adopted by ARN for the substitution of contaminated food are estimated on the basis of optimization analyses, in which the expected effects by the consumption of these foods and the problems of replacement were taken into account.
- Ground decontamination. Given its high cost, the execution of this action is expected to be decided on the basis of specific analyses for each situation.

5.2.7. ORGANIZATION OF A NUCLEAR EMERGENCY RESPONSE

The organization in case of a nuclear emergency requires an inter-institutional response. In Figures 1 and 2 are shown the relations of the involved areas in a nuclear emergency response, at local, regional and national level.

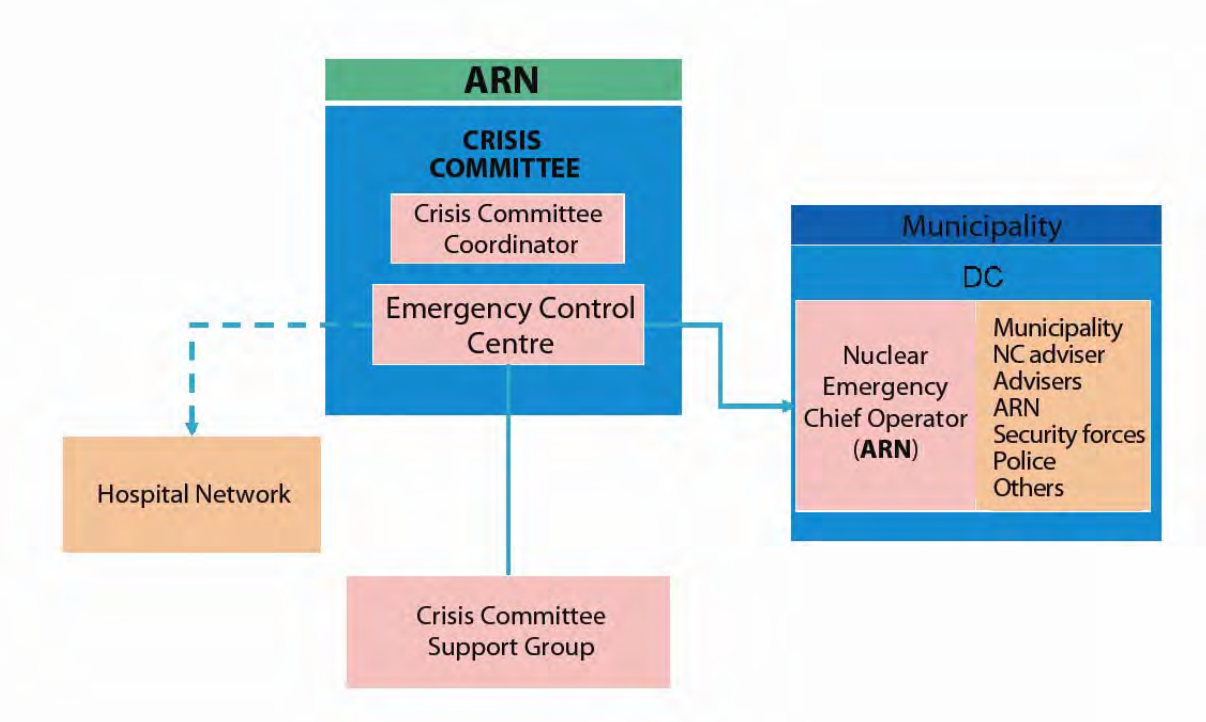


Figure 5-1: Local Organization

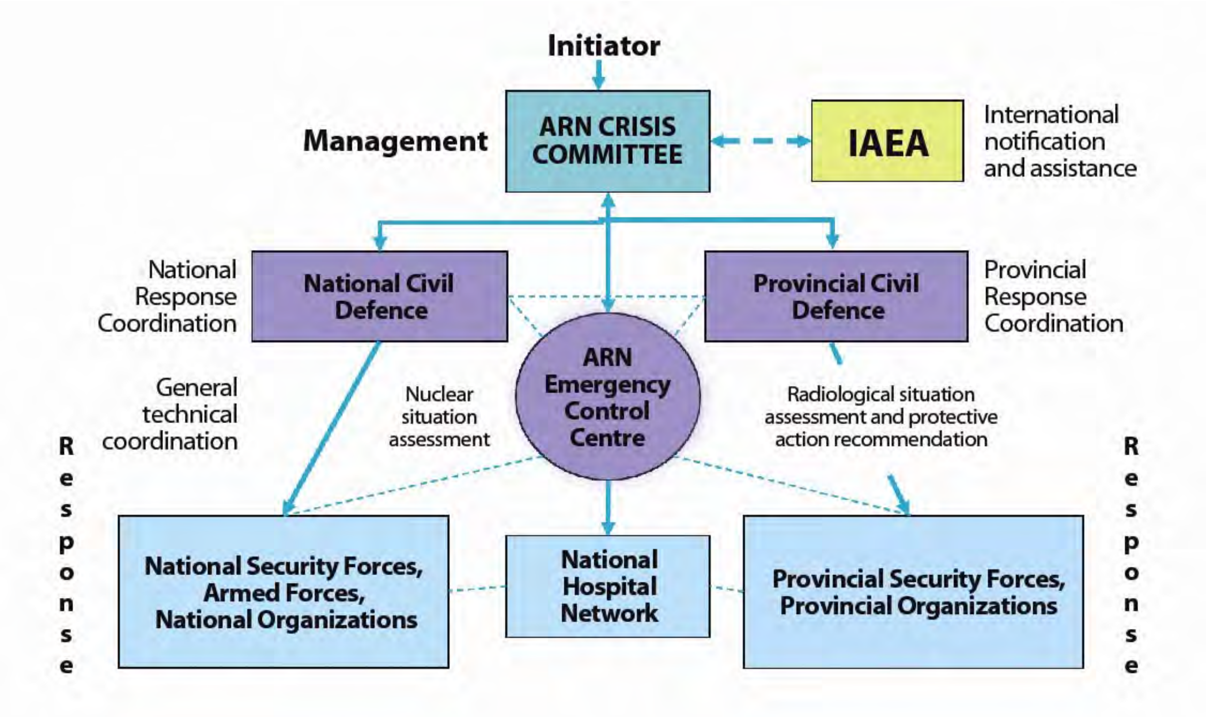


Figure 5-2: Regional and National Organization

5.2.8. TRAINING AND EXERCISES

ARN together with the responsible for the management of emergencies in the nuclear power plant carry out the training of the response organizations (Fire Fighter, Civil Defence, National Guard, police and coast guard, etc.). The functions of these actors are tested in the emergency exercises.

ARN has decided that emergency drills are performed once a year covering all the internal aspects and every two years also including the external aspects. The exercises include all the aspects of an Emergency Plan, and involve all the organizations. In these exercises, ARN has the double function of watching and executing the activities contemplated in the action plan (for example, managing the off-site protection actions)

During the exercises, the staff of the nuclear power plant and the external organization (such as Civil Defence; Police; National Guard; Coast Guard; Fire Fighter; hospitals, etc.) are involved under ARN coordination. In the last twenty years the people from Embalse, Villa del Dique, Villa Rumipal and La Cruz (near CNE) and from Lima (near CNA I) were involved.

Those drills are carried out in order to verify the performing of the urgent protection measures (automatic application) and those that require more time for the implementation. During the drills the behaviour of the plant personnel and the involved organizations is evaluated, while on the other hand the people of the surrounding zones are instructed about the actions to follow in case of a radiological emergency. In this sense, during the previous months of the exercise the divulgation of information to the people is mandatory, about how to proceed and what they should know.

5.2.9. COMMUNICATIONS IN THE PLANNING ZONE

A revision to the communications on and off-site were performed, among all the government organizations that may be involved, taking into account the lessons from other non-nuclear events.

As result of this revision, the following improvements were proposed:

5.2.9.1.

The operator of nuclear power plants has initiated under requirement from ARN a study for the construction of local emergency control centres, outside the planning zone of the nuclear emergency (beyond 10 km). These centres will have infrastructure, instrumentation and communication systems in accordance to the lessons learned from the Fukushima accident.

5.2.9.2.

In collaboration with other response organizations (mentioned in point 5.1.3.1.6), the operator has reviewed the robustness of the communications outside the site during a severe accident. The result of this review allowed to corroborate that the communication system is accordingly robust to deal with such scenarios.

5.2.10. COMMUNICATION IN THE EMERGENCY CONTROL CENTRE

The communication network from in ARN has the necessary technology to perform teleconferences and/or videoconferences via internet or Integrated Services Digital Network (RDSI). It has a GIS server to transmit information processed in ECC, via internet, to the Local Emergency Operative Centres (COEMs).

A satellite communication system is also available to send information (video, data and voice) between the site of environment monitoring, COEM and ECC. This system includes mobile radiological and meteorological stations, with the possibility to transmit data using a wired network, wireless and satellite communication to dedicated mobile and landline servers to process radiological and meteorological data.

6

INTERNATIONAL COOPERATION

Regarding the safety conventions, the Argentine Republic has signed all of them and devotes the highest efforts to rigorously comply with them, i.e. the Vienna Convention on Civil Liability for Nuclear Damage and its Amendment Protocol, the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, the Convention on Early Notification of a Nuclear Accident or Radiological Emergency, the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Moreover, the “National Law of the Nuclear Activity” (Act Nº 24.804, 1997) states that in the implementation of the nuclear policy the obligations assumed by Argentina under the international legal instruments subscribed, must be strictly observed.

During 2011, the Argentine Republic received for its consideration and duly analysed the ideas for amendments to the Convention on Nuclear Safety and to the Convention on Early Notification of a Nuclear Accident or Radiological Emergency. Our country will actively participate in and contribute to the efforts aimed at improving the effectiveness of the international legal framework, taking into consideration the Fukushima accident.

The Fukushima events have in a very dramatic manner shown again that a nuclear accident has a global impact. When considering the central issue of emergency preparedness and response, special attention must be given to the mechanisms for communicating with neighbouring countries and the international community, in the case of an event.

Argentina has always attached great importance to this issue. As an example of this, a chapter named “Cooperation and Mutual Assistance in Cases of Nuclear Accidents and Radiological Emergencies” was included in the Argentine-Brazilian Cooperation Agreement as early as 1986.

Moreover, Argentina has signed agreements related to the peaceful uses of nuclear energy with Paraguay, Bolivia, Uruguay and Chile. These agreements facilitate bilateral communications in the case of a severe accident affecting a Nuclear Power Plant (NPP).

The ARN has always been actively involved in the performance of voluntary activities and good practices related to safety. However, the incident at Fukushima NPP has contributed to highlight the need for greater interaction with neighbouring countries. In connection with this, in 2011, representatives from the government of Paraguay participated in the practical emergency plan exercises at Atucha I NPP. As was already customary, representatives from Brazil and Uruguay also took part as observers in the exercises carried out in Argentina.

Furthermore, in the frame of the international cooperation between Argentina and Brazil, a Binational Nuclear Energy Commission (Comisión Binacional de Energía Nuclear - COBEN) was established on March 3rd 2008, with the aim of implementing specific cooperative projects in the field of peaceful uses of nuclear energy.

Against the backdrop of the accident at Fukushima NPP in March 2011, and bearing in mind the need to review all the relevant aspects related to nuclear safety, emergency preparedness and response and radiation protection of people and the environment; the 11th meeting of the COBEN (April 2011 - Buenos Aires) decided to assign priority to the project “Strengthening Response Capacities for Radiological and Nuclear Emergencies”. Its main objective is to promote the exchange of experiences and practices towards the strengthening of the emergency systems.

With respect to the cooperation with international organizations, it is worth mentioning the following issues. After the Fukushima accident on March 11th, Argentina participated in many meetings in connection with this event as well as it offered support to Japan in different areas.

In June 2011, a Ministerial Conference on Nuclear Safety was held by the International Atomic Energy Agency (IAEA). Argentina attended, and by its statement encouraged an atmosphere of cooperation with Japan and all IAEA's Member States.

Particularly, our country pointed out that in Argentina the regulatory responsibilities have always been effectively separated from the promotional activities, and for more than sixteen years it has perfected its full legal and administrative independence. The regulatory body in Argentina depends directly from the country's highest authority, that is, the President, and is endowed with extensive powers to control the nuclear activity throughout the country.

During the Ministerial Conference, Argentina highlighted its continuous support to IAEA's safety-related statutory functions in the framework of strict compliance with these statutory requirements. Our goal is not passive acceptance, but active improvement of the implementation of these functions by the Secretariat. Additionally, Argentina emphasized that, since the establishment of the international standard system half a century ago, it has been fully involved, not only in the development of the standards, but also in their implementation.

In the mentioned meeting, Argentina claimed that both IAEA's standards and their application should be quantitative, objective, measurable and comparable, and that all qualifying subjectivism whether in the formulation of the standards or in their implementation should be avoided. In the same line, Argentina stressed the need for the IAEA´s standards to be co-sponsored by the competent organs of the United Nations and the related specialized agencies, as required by the Statute.

Argentina also expressed that it continues to support, as it has for the past century, the international nuclear safety regime, but emphasises that it should not depart from the following essential characteristics:

- Standards should be objective, measurable and comparable, adopted by consensus among all sovereign Member States, non-binding and prepared with the joint participation of pertinent organizations of the United Nations System.
- The mechanisms, whereby the Secretariat provides for the implementation of standards, must be rigorous and strictly not qualitative.
- They should not promote industrial interests of technology-supplier States.

On the subject of cooperation in the frame of international working groups, Argentina is one of the founding States of the Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (Foro Iberoamericano de Organismos Reguladores Radiológicos y Nucleares - FORO), which was constituted in 1997. FORO is an association of regulatory agencies from the Ibero-American region, whose objective is to promote radiation and nuclear safety and security, through the strengthening of regulatory actions, in the member countries, as well as in other countries of the region.

As a matter of specific relevance to this report, during the FORO´s plenary in July 2011, which took place in the city of Santiago de Chile, the four regulatory bodies of the countries with operating NPPs (Argentina, Brazil, Mexico and Spain) decided to analyse every NPP in operation. Despite the fact that none of these States have had any sort of accident throughout their lifetime operation, they agreed to carry out "Stress Tests" at every operating facility.

The "Stress Test" will focus on making resistance evaluations similar to those implemented by the Western European Nuclear Regulators Association (WENRA) and the European Nuclear Safety Regulatory Group (ENSREG), among others, and is fully consistent with main action "Safety assessments in the light of the accident at TEPCO's Fukushima Daiichi Nuclear Power Station" of the IAEA Action Plan on Nuclear Safety, endorsed by Member States on the occasion of the 55th General Conference.

The objective of these evaluations is to determine safety margins, analysing the behaviour of the plant considering its response to the extreme events that could cause effects beyond the design basis, such as the total loss of power and ultimate heat sink, and the capacity to manage these accidents. The results will be ready in 2012.

Argentina's regulatory body, as a member of the CANDU Senior Regulators Group, participated in its annual Technical Meeting (Vienna, October 31st to November 4th), which was devoted to exchange information and consider lessons learned from the Fukushima nuclear accident. Recognizing that countries with CANDU-type reactors share a common technology, and in preparation for the Second Extraordinary Meeting of the Contracting Parties to the Convention on Nuclear Safety (CNS), this Technical Meeting focused on six of the topics covered in the Guidance for National Reports developed for the Second Extraordinary Meeting of the CNS. These topics are external events, design issues, severe accident management and recovery (on site), national organizations, emergency preparedness and response and post-accident management (off site) and international cooperation.

The meeting was a constructive forum for national nuclear regulatory bodies of the Member States that operate CANDU-type reactors. It was also useful to exchange information in preparation for the upcoming Extraordinary Meeting on lessons learned from safety reviews (or "stress tests") in response to the Fukushima nuclear accident.

As expressed at the Ministerial Conference and throughout the process of discussing and adopting IAEA Action Plan on Nuclear Safety, Argentina will always join consensus on courses or action plans that are technically sound, relevant and adopted following a comprehensive and informed international discussion.

In this line, Argentina is especially keen on the main action "IAEA peer reviews" of the Action Plan, with its objective of strengthening such reviews in order to maximize the benefits to countries, and will duly contribute to such process.

In particular, Argentina notes that the IAEA Secretariat's effort towards strengthening the existing IAEA peer reviews will aim at guaranteeing that these reviews correctly address regulatory effectiveness, operational safety, design safety, and emergency preparedness and response. While our country welcomes the recognition by all Member States of the Agency of the voluntary nature of such reviews, is also well aware of the increased global interest in the nuclear activities of any State and on the need to provide adequate information, in a transparent manner, on Argentina´s activities in the area of nuclear safety.

Our country sees the process of strengthening the peer reviews as twofold: on one hand, by adequately incorporating lessons which may arise from the Fukushima accident, as it relates to the scope of the reviews; on the other, by trying to maximize the benefit to the country involved, in terms of the relevance to its national situation and duly having in mind its legal and regulatory frameworks.

While we will follow closely these important developments, Argentina is advancing on its reflection on how to best make use of the IAEA peer reviews, reflection which is certainly enriched by its own experience with a number and variety of peer reviews that it has received in the past.

Argentina will also contribute by providing experts for peer review missions.

With respect to sharing international operating experience, there are three areas to focus on.

1) Regarding the International General Ageing Lessons Learned Program (IGALL), which was initiated by the IAEA in 2009, Argentina obtained its membership in this program in 2012. Its aim is to develop a document in order to provide:

- A guide for ageing mechanisms and effects based on both research results and accumulated operational experience.
- An international agreement on what an acceptable Ageing Management Program involves for standard plant components, structures, material and environments.

The Program also intends to facilitate the exchange of experience accumulated in Member States on identification, establishment and implementation of Ageing Management Programs. It will also provide a knowledge base for the design of new plants, the design reviews of existing plants, among others.

There are three Working Groups which are focused on mechanical components and materials; electrical components, instrumentation and control; and structural components and structures.

2) On the matter of providing feedback of the operating experience of other NPPs, it is worth underlining that at the beginning of the operation of CNA I, its designer (Siemens - Kraftwerk Union AG) played an important role in the transmission of operating experience of the German PWR applicable to the Argentinean NPPs.

CNE has had, since the beginning of its operation, a fluent communication with other CANDU plants of similar design -such as Point Lepreau, Gentilly-2, Wolsung-II-, in order to exchange operating experience. Moreover, it is member of the CANDU Owners Group since its creation.

Presently, both CNAI and CNE NPPs receive information from the following databases:

- CANDU Owners Group (COG).
- World Association of Nuclear Operators (WANO).
- IAEA's International Reporting System (through the ARN coordination).

CNE uses COG databases as part of its usual working activities. Several corrective actions have been implemented, as a result of the information received via COG. On the other hand, CNE provides COG a periodic report of its significant events.

Since 1996, CNA I has been using WANO's database. The collection, selection and classification of information have been systematised. ARN examines the effectiveness of operating experience feedback using information coming from national and international databases.

This information is examined by a team of analysts using models to identify the relevant parts that need a deeper investigation into the process. The team is directly involved in:

- Events screening,
- Definition of scope of events to be analysed,
- Application of root cause methodologies,
- Corrective actions,
- Corrective action follow-up.

As a consequence of a requirement issued by ARN in 1998, the Licensee started a formal and systematic process of evaluation of the operating experience, with the purpose of obtaining feedback so as to improve reliability and availability of the NPPs.

According to a requirement by ARN, the Licensee prepared an "Operating Experience Management Program", in order to analyse events (at National and International level) to be used as a feedback of Operating Experience from domestic NPPs. The major actions required by ARN include:

- Use of international and national databases,
- Use of root cause analysis methodologies in the cases where an event is applicable in domestic plants,
- Taking immediate corrective actions to avoid event occurrence or recurrence,
- Corrective action follow-up,
- Lessons learned from analysis.

The Licensee constituted three working groups: two within each plant site and the third one within the Licensee headquarters, to obtain feedback to improve plant systems (modifications) and optimize maintenance activities (through the execution and follow-up of corrective actions).

To fulfil the Operating Experience Management Program, a set of activities to be carried out was defined, and trend analysis workshops to share experience and training were also included. The Licensee prepares a quarterly report, including the results obtained from the application of the program. In addition, NPP's senior teams evaluate "low level events" and "near misses" creating their own database.

The Licensee working group performs a screening analysis using international databases selecting the applicable events for the domestic plants. After screening, those events are analysed in detail and they are presented to the plants for evaluation. The evaluation could include recommendations, proposed design modifications, changes in procedures and training courses for operators, if necessary.

The program coordinator reviews more than 1000 reports per year from different international sources. However, due to design, procedures, systems and components or low safety significant actions, approximately only 10% of the events are applicable.

The ARN has performed audits to the CNA I's and CNE's Operating Experience sectors that showed improvements on corrective actions, implementation, organization of training meetings and discussions, as well as an increasingly experienced operating personnel.

3) In connection with activities with the World Association of Nuclear Operators (WANO), the operator of CNA I and CNE (NA-SA) is its member.

Both NPPs under operation, received WANO Peer Review (PR) Missions and the corresponding Follow-Up (FU) (CNA I NPP: PR in 2006, FU in 2009; CNE: PR in 2007, FU in 2009). NA-SA participated in WANO programs through the WANO – PC (Paris Centre): Peer Reviews, Technical Support Missions, Assist Visits, Operating Experience Programs, Workshops and Seminars. Argentina has also provided specialists from both NPPs to participate in every WANO Program.

In order to improve some particular areas in CNA I and CNE, NA-SA requested during 2011, WANO's assistance through four technical support missions, as well as it participated in seven technical support missions made in others plants in the world.

There will be a WANO Peer review for CNA I in December 2012, with a pre-visit in May, as well as a technical support mission on operational focus in CNE. Also, it is expected that NA-SA experts participate in eight WANO peer reviews, along with several technical support missions and workshops.

In regard to the International Reporting System for Operating Experience (IRS), Argentina continues to participate in it, as a way of contributing to the identification of actions that can be applied to domestic NPPs.

A final issue warranting particular consideration concerns the utilization of IAEA safety standards. As it was mentioned in the Fifth National Nuclear Safety Report, ARN has begun a harmonization process of its standards against IAEA's safety documents. The detailed comparative analysis comprises the criteria established in ARN standards and the corresponding ones indicated in IAEA´s safety requirements.

In a first stage, a comparison and analysis of ARN Regulatory Standards / IAEA Standards was performed. As a result of this analysis, some possibilities for improvement in some non-essential aspects of the ARN standards were identified.

It was possible to conclude from this exercise that Argentine standards are consistent with IAEA's corresponding documents.

The second harmonization stage foresees the performance of a comparative analysis of the recommendations contained in ARN's Safety Guides vis-à-vis the IAEA recommendations.

During 2011, this process was still underway, but was paused after the Fukushima accident of March. One issue which arose at the time was whether the IAEA standards themselves (and the way they were implemented) needed some sort of re-assessment. As a result of this discussion, the IAEA Action Plan does include a main action related to the standards, aimed at reviewing them in the light of the lessons learned from Fukushima NPPs accident. This very important process is under way, so it was established that ARN (while contributing to it with its technical expertise) would wait for that review to be completed before continuing with its own harmonization-related evaluation of the national standards.

ANNEX I

REGULATORY REQUIREMENT RQ-NASA-38

COMPREHENSIVE SAFETY EVALUATION OF THE ARGENTINE NUCLEAR POWER PLANTS

(STRESS TESTS)

I.1. REGULATORY REQUIREMENT RQ-NASA-38

The Nuclear Regulatory Body (ARN) requires to Nucleoeléctrica Argentina S.A (NA-SA) fulfil the following regulatory requirement:

Send to ARN, before 04/27/12 for CNA I / CNE and before 05/31/12 for CNA II, a report containing a Comprehensive Safety Evaluation of these NPPs that, complying with what is specified in the attached Annex, includes:

1. The extreme initiating events conceivable at each NPP site.
2. The loss of safety functions caused by each one of the extreme initiating events considered.
3. The severe accident management corresponding to each one of the extreme initiating events considered.

The Comprehensive Safety Evaluations requested should include:

a) The long-term evolution of severe accidents (referred to in point 3) caused by extreme events (referred to in point 1) and the ability to recover / repair the supplies of cooling water and electricity until a stable NPP condition is achieved, in order to identify the more suitable strategies for repair and recovery as well as the elements that have to be available for their implementation. Indicate the expected schedule to have the above mentioned elements available.

b) The safety implications resulting from the existence of multiple reactors at the Atucha site in case of extreme events (listed in point 1), identifying and implementing the appropriate measures that eventually must be adopted and the procedures to use the existing resources from one unit for the other one. Indicate the corresponding schedule when this would be achievable.

c) Spent fuel management strategy (SF) and the design and the performance of the SF storage systems in case of extreme-event occurrences (referred to in point 1). Determination of the foreseen actions and the schedule for their implementation.

d) Disposal of equipment and components belonging to safety systems and safety-related systems to ensure that, in case of extreme-event occurrences (such as those listed in point 1), those systems can continue performing the function for which they were designed. In the case it were necessary to implement modifications, indicate the planned measures and the corresponding implementation schedule.

e) Measures for prevention, recovery and mitigation of accidents characterized as indicated in items 2 and 3. Reassess both the automatic and the operators' actions established in the operating procedures for abnormal conditions, emergency and severe accident management. Indicate the foreseen schedule for the implementation of the improvements eventually decided on.

f) Availability of the existing resources in the nuclear power plants in response to internal and external nuclear emergencies in case of accident scenarios characterized as described in points 1, 2 and 3. Furthermore, during the time that elapses between the accident occurrence until the ARN assumes the responsibility for the emergency management, the planning and the management actions must be foreseen including the communications and the protection of the public. For each of the above issues, the necessary improvements should be identified, and their implementation schedule given.

I.1.2. ANNEX TO REGULATORY REQUIREMENT RQ-NASA-38: SPECIFICATIONS FOR THE COMPREHENSIVE SAFETY ASSESSMENT THAT SHALL BE CARRIED OUT BY THE NUCLEAR POWER PLANTS

I.1.2.1. Comprehensive safety assessment definition

The Comprehensive Safety Assessment (stress-tests) is aimed to determine the nuclear power plants safety margins considering extreme events such as those that occurred in Fukushima, which may endanger the safety functions of the NPPs and may lead to a severe accident situation. This assessment should include:

- An analysis of the response of each NPP in case of the occurrence of a set of extreme situations as considered in item 2 of this Annex, and;
- The verification of the prevention and the mitigation measures based on the defence-in-depth concept.

In the aforementioned extreme situations, - the sequential loss of the existing defence-in-depth lines must be analyzed -under a deterministic approach-, independently of the likelihood of such a loss. It should be considered that the loss of safety functions and severe accident situations can only occur when numerous design provisions have failed. Furthermore, it should be assumed that the available measures to properly manage these situations were not successful.

The Stress-Tests shall include for each NPP, the plant response and the effectiveness of the preventive measures, highlighting any potential weaknesses and any cliff-edge-effect identified in the analysis. These cliff-edge effects are defined by limit values of critical parameters where by a small change leads to a disproportionate increase in the accident consequences (these conditions may correspond, for example, when a significant flooding begins when a level exceeding the design considerations is surpassed, or the battery capacity is depleted during a total loss of alternating current - AC power event).

The purpose of the aforementioned is to assess the robustness of the defence-in-depth philosophy, the suitability of the accident management measures, and to identify the potential to implement safety improvements, both technical and organizational, such as procedures, human resources, emergency response organization or use of external resources.

The Stress-Tests should focus on such measures that may be taken after the loss of safety functions in case of accidents already considered in the design. Furthermore, the hypothesis relating to the operation of such systems must be re-evaluated. Finally, all measures taken to protect the integrity of the fuel elements in the reactor or in the spent fuel storage systems (wet and dry) should be considered as an essential part of the defence-in-depth.

I.1.2.2. Scope of the stress tests

The technical scope of the stress-tests were defined considering the problems caused by the events that occurred at Fukushima, which included a combination of extreme initiating events and multiple failures. Therefore, the following situations should be considered:

a) Conceivable initiating events at the plant site during extreme meteorological conditions:
   - Earthquakes.
   - Flooding / low water level (river / lake).
   - Other extreme natural events.

b) Loss of safety functions:
   - Loss of electrical power.
   - Loss of the ultimate heat sink.
   - Combination of both of the above.

c) Severe accident management issues. Measures to manage the loss of:
   - Core cooling function.
   - Spent fuel cooling function.

- Containment integrity.

d) Considerations about the internal emergency management:

  - Direction and Control.
  - Fuel damage mitigation.
  - Radioactive releases reduction.
  - Procedures review.
  - Equipment.

The following defines the general information and issues to be considered by NA-SA for each extreme situation.

I.1.2.3. General aspects

I.1.2.3.1. Stress-Test Report Content

Considering that the Stress-Test Report shall be submitted to other institutions for their consideration, the report should include the following basic information for each facility:

- Details of the site location.
- Number of units.
- Reactor features.
- Brief description of the systems involved in the required analysis and that of the existing heat sinks, and the corresponding heat removal chains.
- Date of first criticality.
- Type and storage location of spent fuel.
- Significant safety differences between the units.
- The scope and main results of the available Probabilistic Safety Assessments -PSA- shall be provided. If applicable, indicate the use of APS in the evaluation.

Each of the situations included in the Scope (item 2) and in the Content (items 4, 5, 6 and 7) shall be assessed following the indications given below.

I.1.2.3.2. Hypothesis

For existing plants, the assessments shall refer to the current state of each power plant, as built and / or operated on July 30, 2011.

The approach should be essentially deterministic, i.e. when analysing an extreme scenario, a progressive approach shall be followed, in which protective measures are assumed to be lost sequentially during a prolonged period.

The initial plant conditions should represent the operational states of the plant at full power and unpressurized cold shutdown.

It is presumed that all reactors and fissile material or waste facilities are affected at the same time. The possibility of degraded conditions surrounding the NPP, characterized by the lack of external electrical power, difficulty in accessing it, etc. should also be considered.

Besides, consideration should be given to:

- Automatic actions.
- Operator actions specified in emergency operating procedures.
- Any other planned measures for the prevention, recovery or accident mitigation.

I.1.2.3.3. Information to be included in the report

The following main aspects must be included in the stress-test reports:

- Provisions taken in the design basis of the plant.

- Plant robustness beyond its design basis. For this purpose, the robustness of the structures, systems and components (SSC) relevant for safety and the defence-in-depth preservation must be assessed (available design margins, diversity, redundancy, structural protection, physical separation, etc.). Regarding the installations robustness and the available measures, one of the approaches of the review is to identify the possible cliff-edge-effects which could lead to a relevant change in the sequence of events and the existing measures to avoid extreme conditions.

- Possibility of implementing modifications to improve the present defence-in-depth level.

The existing protective measures aimed at avoiding the extreme scenarios considered in the assessment scope must to be described. The analysis should be complemented, where necessary, by the results of the dedicated plant walk-downs performed.

In this regard, NASA should identify the measures to maintain the three fundamental safety functions (control of reactivity, fuel cooling, confinement of radioactivity) and support functions (electrical power supply and cooling), taking into account the probable damage done by the initiating event as well as any available measure to face the event even if they are not considered in the plant safety analysis.

In addition, NASA should identify the possibility of having mobile external equipment and the conditions of their use; the availability of alternative sources of coolant supply; any existing procedure to use resources from one reactor to help another reactor and the dependence of safety functions of one reactor on other reactors on the same site.

Regarding severe accident management, NA-SA must determine, when relevant, the time interval before damage to the fuel becomes unavoidable. If the fuel is in the reactor core, the time available before the water begins to leave the fuel exposed should be indicated as well as the time before fuel degradation initiates (the start of fast cladding oxidation with hydrogen production). If the fuel is in the spent fuel pool, the time before pool boiling commences, the time during which an adequate shielding against radiation is maintained, the time before the water level reaches the top of the fuel elements and the time before fuel degradation starts should be indicated (all of these situations without considering operator intervention).

I.1.2.4. Initiating external events

The stress-test report should include the information detailed in sections 4.1., 4.2. and 4.3.

I.1.2.4.1. Earthquakes

I.1.2.4.1.1. Design basis

I.1.2.4.1.1.1. Earthquakes against which the plant is designed

- Level of the design basis earthquake (DBE) expressed in terms of peak ground acceleration (PGA) and reasons for the choice.
- Indicate if the current DBE is different from the DBE taken into account in the original licensing basis.
- Methodology used to evaluate the current DBE (return period, past events considered and reasons for their choice, margins added, etc.), the validity of data with time.
- Conclusion on the adequacy of the design basis, considering the state of the knowledge to date.

I.1.2.4.1.1.2. Provisions to protect the plant against the DBE

- Identification of the key structures, systems and components (SSCs) which are needed for achieving a safe shutdown state, defined as the state of the plant in which the long term core sub-criticality, the core cooling and the confinement of radioactive material are guaranteed.
- Main existing operating provisions (including emergency operating procedures, mobile equipment, etc.) to prevent damage to both the reactor core and spent fuel after the earthquake, assuming the unavailability of such SSCs.
- Indicate the earthquake´s potential indirect effects, including:

- Failure of SSCs that are not designed to withstand the DBE and that, in losing their integrity, could cause the consequential damage of the SSCs that need to remain available (e.g. fire, explosions or pipe breaks that can affect the performance of safety functions, including the action of the operation staff).
- Loss of external electrical power supply.
- Situation outside the plant, including preventing or delaying access of personnel and equipment to the site.

I.1.2.4.1.1.3. Plant compliance with its licensing basis

– General process to ensure compliance (e.g., periodic maintenance, inspections or testing).

I.1.2.4.1.2. Other considerations

– Activities and specific verification tests already started by the plant after the Fukushima accident.

I.1.2.4.1.3. Evaluation of the margins

I.1.2.4.1.3.1. Severity of the earthquake

– Based on the available information (seismic margin assessment or other engineering studies), an estimation must be made of the range of earthquake severity above which the loss of fundamental safety functions or severe damage to the fuel (in the core or in storage) is not guaranteed.
– Indicate if any provisions can be envisaged to prevent these cliff-edge-effects or to increase the robustness of the plant (hardware modifications, procedure modifications, organisational provisions, etc.).

I.1.2.4.1.3.2. Maximum earthquake that the plant could withstand without losing the containment integrity.

– Based on the available information (seismic margin assessment or other seismic engineering studies) analyse what is the severity range of the worst earthquake that the plant could withstand without losing confinement integrity (containment and fuel pool integrity).

I.1.2.4.1.3.3. Earthquakes exceeding the DBE causing flooding or downspouts that exceed the design basis flooding or downspout level.

– Indicate if this scenario is physically possible considering the NPP location. To this end, identify if potential serious damages to the structures located inside and outside the site (such as dams or buildings and structures of the plant) could have an impact on the safety of the installation.
– Indicate the weaknesses and failure modes that could lead the NPP to unsafe conditions, identifying any potential cliff-edge-effect and the buildings and equipment that would be affected.
– Indicate whether additional measures can be anticipated to avoid the cliff cliff-edge-effects identified or to increase the plant´s robustness (equipment modifications, changes in procedures, organizational arrangements, etc.).

I.1.2.4.2. Flooding / low water level (river/ lake)

I.1.2.4.2.1. Design Basis

I.1.2.4.2.1.1. Flooding / low water level against which the plant is designed

– Level of the flood / low water level design basis and reasons for their choice.
– Indicate if the level of the current flood / low water level design basis is different from that postulated in the original licensing basis.

- Methodology used to assess the current flood / low water level design basis (return period, past events considered and the reasons for his choice, possible margins added, etc.) and the validity of the data with time.
- Conclusions on the adequacy of the current design basis, considering the state of knowledge to date.

I.1.2.4.2.1.2. Provision to protect the NPP from the flood / low water level design basis

- Identification of the critical SSCs which are necessary in order to achieve a safe shutdown condition and are supposed to remain available after the flooding / low water level, including provisions to maintain:
    - The water intake function.
    - Emergency electrical power supply.
- Identification of the main design requirements to protect the site against flooding / low water level (platform levels, etc.), indicating the associated monitoring programs.
- Main planned measures foreseen to mitigate the effects of floods / low water level, including emergency operating procedures, mobile equipment, flood / downspouts monitoring, warning systems, etc., to warn of flood / downspout and facilitate their effects mitigation, indicating the associated monitoring programs.
- Verify that the possible effects associated with the flood / low water level have been taken into account, including:
    - Loss of external power supply.
    - Situation outside the plant, including those problems that might prevent or delay the access of the personnel and the required equipment to the site.

I.1.2.4.2.1.3. Plant compliance with the current licensing basis

- Process to ensure its compliance, for example, periodic maintenance, inspections or tests.

I.1.2.4.2.2. Other considerations

- Verification activities and specific tests already initiated by the plant after the Fukushima accident.

I.1.2.4.2.3. Evaluation of the margins

- Based on the available information (including engineering studies), indicate the level of the flood / low water level for which the loss of the fundamental safety functions, or serious damage to the fuel (in the core or the spent fuel storage facility) is not guaranteed.
- Depending on the available time between the warning and the flood / low water level, indicate whether additional protective measures can be foreseen and implemented.
- Indicate which are the weak points and identify any potential cliff-edge-effects, identifying the buildings and equipment that the flood / low water level might affect first.
- Indicate if additional measures can be foreseen to prevent the identified cliff-edge-effects or to increase the robustness of the plant (hardware modifications, procedures modification, organisational provisions, etc.).

I.1.2.4.3. Other natural extreme events

Extreme weather conditions (storms, torrential rainfalls, tornadoes, etc.): It should be indicated:

- Events and combinations of events considered and the reasons for their selection.
- Which are the weak points and the cliff-edge-effects. Identify buildings and equipment that could be affected.
- Provisions to avoid the cliff-edge-effects above mentioned or to increase the NPP´s robustness (design changes, modification of procedures, organisational provisions, etc.).

I.1.2.5. Loss of safety functions

The impact on plant safety functions resulting from the loss of electrical power and the loss of the ultimate heat sink (UHS) should be analyzed.

I.1.2.5.1. Loss of electrical power

The electrical alternating current (AC power) sources to be considered should be:

- Off-site power sources (electrical grid).
- Plant generator.
- Backup generators (diesel generators).
- Other alternative backup sources (diesel generators, water turbines, gas turbines, etc.).

The analysis must assume the occurrence of the sequential loss of these electrical supplies.

I.1.2.5.1.1. Loss of off-site power (LOOP)

In this case it must be presumed that the off-site power is lost for a long time and that the site will remains isolated for 72 hours from the possibility of receiving heavy supplies by any transport means, although it is assumed that light mobile equipment could arrive to the site within 24 hours from the event occurrence. The following aspects should be considered:

- Describe how the plant design has taken into account the LOOP as well as the internal electrical power supply systems designed to deal with it.
- Indicate how long the above mentioned internal power supplies could run without any external support.
- Indicate what actions are needed and foreseen to extend the internal power supply equipment´s operating time (for example, diesel generators fuel tanks refilling, etc.).
- Identify possible measures to be taken to increase the plant´s robustness, such as systems modification, procedures changes, organizational arrangements, etc.

In order to clarify this issue, it must be indicated which systems (such as hydraulic turbines, pumps driven by engine / turbine, etc., systems with stored energy in compressed air tanks, etc.) may be considered available.

I.1.2.5.1.1.1. Loss of off-site and on-site backup power sources (SBO, Station Black Out).

In these analyses the following two situations must be considered:

- LOOP + loss of the backup power supplies (backup generators).
- LOOP + loss of the backup power supplies + loss of any other alternative backup source.

For each one of these situations, NA-SA must:

- Provide information on the measures considered in the design for these accident scenarios.
- Provide information on the battery capacity and duration. Analyze the consequences of their total loss.
- Indicate how long the plant can withstand an SBO without any external support before severe damage to the fuel becomes unavoidable.
- Indicate the foreseen external actions to prevent fuel damage, considering:
    - Equipment already present on site, e.g. equipment from another reactor.
    - Equipment available off-site or near-by power stations (e.g.: gas turbine or water turbine) that can be aligned to provide power via a secure direct connection; assuming that all units on the site have been damaged.
    - Time necessary to have each of the above systems operating.
    - The availability of competent human resources to make these connections (considering their exceptional characteristic).

- Identification of potential cliff-edge-effects that could occur, indicating the additional measures that could be incorporated to avoid their effects or to increase the robustness of the plant (system modifications, procedure changes, organizational arrangements, etc.).

I.1.2.5.1.1.2. Loss of the ultimate heat sink (UHS)

The UHS is the mean where the residual heat from the reactor is ultimately transferred. Consideration should be given to the sequential loss of these available sinks needed to cool the reactor and the spent fuel pool under any circumstances.

It should be assumed that the functionality of the various existing heat sinks are successively lost and the site will remain isolated for 72 hours with regard to the possibility of receiving heavy supplies by any transport means, although it is assumed that light mobile equipment could arrive at the site 24 hours after the start of the event.

In this regard, a description of the existing design basis provisions to prevent the loss of the various sinks should be provided. For example, diverse water intakes located in different places, etc.

For these scenarios, it should:

- Indicate for how long the plant could withstand the situation without external support before severe fuel damage is unavoidable.
- Provide information on the existing provisions taken in the design basis and the internal actions foreseen for each of the above-mentioned scenarios.
- Indicate the external actions foreseen to prevent fuel damage, considering:
    - The equipment already existing on-site, for example, equipment from another unit, etc.
    - Available off-site equipment, assuming that all reactors at the site have also been damaged.
    - The time required for these systems to be operating.
    - The availability of competent human resources to perform the necessary actions (considering their exceptional characteristic).
- Identify the possible cliff-edge-effects which could occur, indicating the additional measures that could be incorporated to mitigate their effects or to increase the robustness of the plant (hardware modifications, procedure changes, organizational provisions, etc.).

I.1.2.5.1.1.3. Loss of the heat sinks with SBO

The following information should be provided:

- Indicate for how long the plant can withstand a loss of all heat sinks with SBO without any external support and before severe damage to the fuel is unavoidable.
- The existing design provisions and the internal actions foreseen for each of the above-mentioned scenarios.
- Indicate the external actions foreseen to prevent fuel damage considering:
    - The equipment already present on- site, e.g. equipment from another unit.
    - The off-site available equipment, assuming that all reactors in the site are equally damaged.
    - The necessary time to have these systems operating at the site.
    - The availability of competent human resources to carry out the necessary actions (considering their exceptional characteristics).
- Identify the cliff-edge-effects that could occur, and when they could occur, indicating the additional measures that might be incorporated to prevent their effects or to increase the plant´s robustness (hardware modifications, procedure modification, organizational provisions, etc.).

I.1.2.6. Severe accident management

The mitigation measures should be mainly considered, even if the event occurrence probability is very low. In this evaluation it should be deterministically assumed that a severe accident occurs.

I.1.2.6.1. Describe the accident management measures currently available for protecting the reactor core during the various stages of a loss of cooling function scenario

- Before occurrence of fuel damage in the reactor pressure vessel, indicating:
  - Whether there are means to prevent fuel damage for primary high-pressure sequences in the primary heat transport system (PHTS), in case the capability to depressurize the PHTS is not available.
  - The last-resource measures foreseen to prevent damage to the fuel.
- After the occurrence of damage to the fuel in the reactor core.
- After the failure of the reactor pressure vessel or the pressure tubes.

I.1.2.6.2. Describe the accident management measures and the plant design features to protect the containment integrity after the fuel damage occurrence

- Prevention of hydrogen deflagration or hydrogen detonation (inerting, recombiners, or igniters), taking into account the capacity of the contention venting processes.
- Prevention of the containment over-pressurization. If a radioactive material release to the environment to protect the containment were needed, it should be assessed whether this release could be filtered or not. In this case, the available means to estimate the amount of radioactive material released into the environment should also be described.
- Prevention of re-criticality.
- Prevention of basemat melt through (containment flooding at different levels to prevent the reactor pressure vessel failure or to limit the interaction of the molten core with the concrete).
- Need of supply of electrical AC power, DC power and compressed gas to the necessary equipment to protect the containment integrity.

I.1.2.6.3. Describe the accident management measures currently in place to face the various stages of a scenario of loss of cooling function in the fuel storage pools

- Before and after losing adequate shielding against radiation (loss of water column shielding).
- Before and after occurrence of uncovering / exposure of the top of the fuel.
- Before and after occurrence of the fuel degradation (fast cladding oxidation with hydrogen production).

I.1.2.6.4. Additional aspects

For I.2.6.1., I.2.6.2. and I.2.6.3. items, if applicable, it should:

- Identify any cliff-edge-effect that might occur and evaluate the available time before it occurs.
- Assess the adequacy of the existing management measures, including guides and procedures, to cope with a severe accident, and evaluate the potential for additional measures. In particular, the following should be considered:
  - the suitability and availability of the required instrumentation.
  - the accessibility and habitability of the plant´s essential areas (main and secondary control rooms, emergency response centres, local controls and sampling points, repair possibilities etc.).
  - potential hydrogen accumulations in other buildings than that of containment.

The following aspects should also be considered:

- Organization to adequately manage the situation, including:
  - staffing, resources and shift management.
  - use of off-site technical support for accident management and a place from where the accident is managed, including contingencies if it becomes unavailable).
  - Procedures, training and exercises.
- Possibility to use existing equipment.
- Provisions to use mobile devices. Availability of such devices, appropriate connectors for coupling, time required for them to be available on-site and in operation, and the site accessibility.
- Provisions for and management of supplies (fuel for diesel generators, water, etc.).
- Management of possible radioactive releases and provisions to limit them.
- Management of workers' doses and provisions to limit them.
- Communication and information systems (internal and external).
- Foreseen long-term activities (post-accident).

The envisaged accident management measures shall be evaluated considering that the situation could cause:

- Extensive destruction of the existing infrastructure around the plant, including the communication facilities (this would make the technical and personnel support from outside more difficult).
- Impairment of work performance, including the impact on the accessibility and habitability of the main and secondary control rooms and the plant emergency centre, due to high dose rates, radioactive contamination and the possible destruction of some on-site facilities.
- The need to analyse the feasibility and effectiveness of accident management measures under the conditions of external hazards (earthquakes, floods etc.).
- Unavailability of power supply.
- Potential failure of instrumentation.
- Potential effects from other neighbouring plants at site.

Besides, the conditions that would prevent or make difficult for the staff, to work in the main or secondary control room, as well as in the plant emergency/crisis centre should be identified, as well as the measures that could be taken to avoid the occurrence of these conditions.

I.1.2.7. On-site emergency management

In order to meet the requirements for this assessment, it is required to report the status of the issues considered in the following items, as well as any measures deemed necessary to be taken.

I.1.2.7.1. Management and Control

Verify the existence of:

a) The procedures in place to relocate in safe areas the personnel that should remain in the plant (including operating personnel).

b) The procedures related to agreements with external organizations and their activation protocols.

c) Plans for evacuation and personnel assistance.

d) The procedures for preparation and coordination for the reception of materials, equipment and additional personnel, in particular operating personnel from other plants.

e) The necessary professional personnel resources, taking into account the possible loss of operating personnel, making it essential to have a minimum number of qualified persons to perform these tasks.

f) Identification of external organizations and facilities that might have the knowledge and resources to help manage the emergency. For this case, the relevant protocols/agreements should be included to ensure this aid, if it is required, from the start of the accident until the ARN takes charge of the emergency management.

g) The procedure should consider the possibility of sharing personnel in places with more than one unit, as well as the resources required for the accident management in all units of the site.

h) The safe areas to store materials and equipment needed to implement the emergency response plan. The most important aspects to consider in the evaluation include the location of these facilities and the availability of both the equipment and the necessary means.

i) The establishment of safe areas protected from potential fires and radiation exposure.

j) The establishment of appropriate areas, free of obstacles, to facilitate the external assistance arrival of light air vehicles (helicopters).

k) The procedure and the means to ensure the effective guarantee of the communication among the emergency management, the control room, the safe areas, the operation shift, the recovery team, the fire fighting equipment, other external organizations, ARN, etc., as well as the communication among the different groups acting throughout the mitigation process (fire-fighters, recovery personnel, operation personnel, external organizations, etc.). In particular matters relating to education and training of such personnel should be considered.

l) The assessment of the availability and compatibility of the necessary communications equipment, as well as the need of additional emergency equipment, batteries and chargers for this equipment, etc.

m) The availability of sufficient portable fire extinguishers, ventilation equipment, breathing apparatus, personal protective equipment, etc., conveniently located in diverse and accessible places.

n) The planning and availability of appropriate emergency lighting equipment to facilitate performing the activities inside and outside the different plant buildings. The existing capacity to feed and recharge such equipment, so as to ensure their operation for long periods, must be analyzed.

o) The availability of cables, prefabricated connections, adapters, hoses, etc., necessary for the cooling and venting systems operation, as well as to ensure the instrumentation operability and the critical elements required according to the mitigation strategy.

p) The availability, in the appropriate locations, of the applicable plans and procedures.

q) The analysis will take into account aspects related with the personnel radiation protection as well as the use of dispersion models suitable for calculating the involved doses and reducing them to the minimum. The possibility of estimating doses in areas where intervention is necessary will be analyzed.

I.1.2.7.2. Fuel Damage Mitigation

The analysis of the strategies to maintain or restore the effective fuel cooling of both the core and the spent fuel storage pool, as well as to maintain the radioactive material confinement function, taking into account the personnel resources needed to deal with recovery actions, the accessibility and the means to ensure communication among the different acting groups, the procedures and the necessary equipment to develop the planned strategies and the suitable training of both the plant personnel and the external support personnel, should be considered.

The fuel damage mitigation strategy should consider:

a) The review and assessment of the scope of the guidelines for severe accident management.

b) The procedure for filling tanks or water storage reservoirs used in the plant cooling systems (e.g.: water supply tank, condensate tank, etc.).

c) The procedures for manual operation of systems and components, including the use of motor/turbine driven pumps, etc., provided for the case of absence of the corresponding electrical power supply.

For the fuel stored in pools, the water cooling should be optimized to delay fuel damage should the pools be emptied by adopting measures that consider the following criteria:

d) Availability of an alternative supply of emergency cooling water.

e) Leakages reduction and recovery action implementation.

f) Measures to face adverse radiological conditions in the fuel storage pool building.

Additionally, as a long-term measure, the possibility should be considered of:

g) Providing natural and forced air circulation.

h) Optimizing the spent fuel arrangement in its housing to distribute the heat load adequately.

I.1.2.7.3. Reduction of Radioactive Emissions

In case the actions included in the severe accident management guidelines and the special fuel damage mitigation measures are not successful, the release of radioactive material to the outside should be minimized. For this, measures should be taken prior to the start of such a release, considering that once it begins, the surrounding areas may be inaccessible due to the existing radiation levels. For this purpose, the following should be performed:

a) Analyse the availability of a suitable water supply to be used in the most convenient way.
b) Detail the ability to use the fire fighting equipment (water and foam) for washing of possible radioactive emissions.
c) In the case of washing during an extended time or any other actions that could lead to a significant quantity of liquid spills, the availability of temporary deposits to store large volumes of radioactive water in a controlled way, sand bags to facilitate temporary dike construction or stabilizing materials to control and retain the flow of the water used, etc., should be analyzed.
d) Consider the possibility of using compounds that retain or absorb any radioactive particles present, in order to facilitate subsequent cleaning and decontamination tasks as well as aerosol depurations.

For each of the above, their relationship with the following issues (listed in 7.4., 7.5. and 7.6.) should be indicated.

I.1.2.7.4. Revision of Procedures:

It should:

- Reassess the scope of the existing procedures applicable to emergency management strategy, including fire fighting and the corresponding recovery actions.
- Analyse the interferences between the security components and the necessary mobility and ease of access during emergencies.

I.1.2.7.5. Personnel Training

Suitable education and training for both the plant personnel and the external support personnel is a basic element of the mitigation strategy. In this regard NA-SA must verify that it has:

a) The program of exercises and practices.
b) The program of familiarization with the plant.
c) Coordinated exercises among external organizations and personnel from operation, maintenance, recovery, etc..
d) The joint training with external organizations (assistance protocols).
e) The operating procedure to use both special equipment and materials.
f) The training program for shared personnel in sites with more than one unit.
g) If applicable, the training program of the personnel belonging to external organizations to the plant or to other similar plants.
h) The training program for the equipment and instrument connection to alternative electrical power and water supplies, as well as the training program in the operation of devices under critical or degraded situations.
i) The training program in the use of devices, accessories and special clothing.

I.1.2.7.6. Equipment

Information must be given on:

a) The availability of appropriate connectors used in the required auxiliary equipment (e.g: alternative supplies for electricity, water or air), as well as the procedures necessary for their effective alignment.
b) The capacity of both the required electrical power and the cooling to maintain the plant´s safety functions, as well as the availability of the easy interconnection of the equipment and

the aligned components to the different electrical power supplies and to the auxiliary cooling loops and the power for the instrumentation and other devices required.

c) The possible impact on other key plant areas (control room, switch room, cables, relays, breakers, etc.) that can cause the loss of electrical power.

d) The availability of auxiliary equipment used for recovery actions in diverse and safe areas.

e) The existence of an appropriate maintenance, testing and inspection program for all equipment to be used in the mitigation strategy to.

ANNEX II

MAIN TECHNICAL FEATURES OF THE ARGENTINE NUCLEAR POWER PLANTS IN OPERATION

II.1. ATUCHA I NUCLEAR POWER PLANT

II.1.1. INTRODUCTION

In 1964 CNEA initiated the feasibility study for the construction of Atucha I Nuclear Power Plant (CNA I) which would be the first nuclear power plant in Argentina and Latin America designed for electric power generation, and in 1967 entrusted its design and construction to the Siemens Aktiengesellschaft Company of Erlangen, Germany. The construction began in June 1968 and the commercial operation in June 1974.

CNA I is located by the right side of Paraná River, some 7 km from Lima, Province of Buenos Aires, and near 100 km to the north-west of Buenos Aires city. ***Figure II.1-1*** shows its geographic location.

The owner of CNA I is Nucleoeléctrica Argentina S.A., and the plant provides a net electric power of 335 MWe to the interconnected national system.

The station contains a reactor of the pressure vessel type, and is fuelled with natural uranium as well as slightly enriched uranium; it is heavy water moderated and cooled (being of the PHWR type); it is periodically refuelled on power.

Besides, the plant also has the big systems which characterize the classic electric power generating plants, steam turbine and electric generator, as well as components, equipment and sub-systems required for the functioning of the big systems located at its "nuclear' and "conventional" sections.

Figure II.1-2 shows schematically the main systems of the nuclear power plant.

CNA I's original design considered only natural uranium as fuel, being its electric power of 340 MWe and its thermal power of 1100 MWt at that time.

Nevertheless, afterwards, the station suffered two essential modifications that improved its performance:

- In 1977 the electric power was increased to 357 MWe (335 MWe net) and, correspondingly, its thermal power to 1179 MWt.
- Since 1995 until 1999 a progressive loading with slightly enriched uranium (0.85% w) was done, so that at present the core contains only slightly enriched fuel elements.

As regards the technological precedents associated with CNA I, it should be mentioned that in 1955 Siemens AG began developing reactors fuelled with natural uranium and moderated with heavy water, since free access to enrichment services was not envisaged for the immediate future. Two design principles for the pressure boundary of the reactor core were investigated: the pressure vessel type (PHWR) and the pressure tube type (PTR).

The PHWR was realised at the beginning of 1962 as a Multi-Purpose Research Reactor (MZFR) of 57 MWe which went into operation at the Nuclear Research Center Karlsruhe in 1966. The PTR was realised as a demonstration reactor of 100 MWe near Niederaichbach, Bavaria. This reactor line was given up due to poor operating economics.

On the basis of the MZFR the reactor of Atucha I Nuclear Power Plant was designed and up to date this reactor has shown excellent operating performance with high rates of availability and demonstrating its full operating reliability.

II.1.2. OVERALL PLANT LAYOUT

The overall layout of Atucha I Nuclear Power Plant on the site is governed by the following basic considerations:

- Clear separation of nuclear and conventional systems.
- Clear energy flow paths.
- Short piping and cable runs.
- Good transport conditions and access for construction, installation and operation.

Building and structure arrangements of CNA I are shown in ***Figure II.1-3***. As it can be seen from the site plan, the main buildings and structures of the plant are:

(1) Reactor building.
(2) Reactor auxiliary building.
(3) Fuel storage building.
(4) Turbine building.
(5) Switchgear building.
(6) Secondary installations building.
(7) Cooling water intake channel.
(8) Cooling water intake structure and service cooling water pump house.
(9) Cooling water seal pit.
(10) Water turbine building.
(11) Cooling water outfall channel.
(12) High-voltage station service transformer.
(13) Generator transformer.
(14) Off-site system transformer.
(15) Secondary Heat Sink.

In the CNA I, a clear physical separation exists between the nuclear and conventional sections of the plant. The reactor building, the reactor auxiliary building and the fuel storage building constitute the "controlled area" in which all systems assigned to the nuclear section are installed. In this way the potential radioactivity is limited to defined regions. There is only one controlled access to the “controlled area".

The reactor building is linked with the reactor auxiliary building through a personnel airlock and this auxiliary building is, in turn, linked to the fuel storage building through an underground tunnel. The rest of the buildings are located in the "conventional" section of the nuclear power plant.

The reactor building ***(Figure II.1-4)*** contains the reactor, the reactor coolant system, the moderator system and associated equipment. Its outer structure is formed by a cylindrical reinforced concrete shield with a hemispherical top enclosure and is founded on a base slab.

All high-pressure-retaining components of the plant are arranged within the spherical full-pressure steel containment. The containment sphere (diameter: 50 m) is constructed as a pressure vessel and designed for the maximum pressure associated with the worst event which has to be taken into account.

The ancillary and low-pressure-leading auxiliary systems and components (e.g. residual heat removal system, safety injection system or heavy water storage system) are accommodated in the reactor building annulus, the annular space between the cylindrical part of the concrete containment and the spherical steel containment.

A special ventilation system for the annulus ensures that even under accident conditions small radioactive leakages from the containment are retained by charcoal filters, thus preventing any radiation hazards to the environment.

The low-level arrangement of the reactor building and the heavy internal concrete structures, as well as the massive outer concrete shield provide good protection against seismic and other external loads. At the same time, they subdivide the interior of the reactor building into operating and plant compartments. Due to special ventilation systems the former is accessible for Inspection and maintenance work during reactor operation without restriction and without any special protective measures.

The plant compartments for reactor, steam generators and pumps are provided with removable covers, so that all heavy components can be serviced by the polar crane.

The systems necessary for on-load refuelling are also housed in the containment structure.

The reactor auxiliary building adjoins the reactor building, and surrounds a part of it, thus allowing short connections to the equipment located in the reactor building annulus.

On top of the building the vent stack is situated.

The fuel storage building is linked with the reactor building by the fuel transfer system. Personnel access is possible from the reactor auxiliary building. The spent fuel assemblies are transferred from the reactor to the fuel storage pools with the aid of the fuel transport system, consisting of refuelling machine, tilter, transfer tube, tilting device and manipulating bridges. The new fuel assemblies are supplied to the reactor in the reverse way.

The turbine building is of a two-bay design. Its dimensions are governed to a large extent by the dimensions of the turbine generator and its auxiliaries. The main bay houses the turbine generator set and the feedwater tank.

The lower bay houses the condenser, the feedwater pumps and other equipment associated with the steam/feedwater cycle. All these compartments are free of radioactive media. Besides the turbine building contains the emergency diesel generator and the high-voltage station service transformer.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine. Vertical moisture separators are installed on both sides of the high-pressure casing.

The turbine operating floor is clear of off pumps and piping so that floor' space is available to set down components removed when the turbine generator unit is opened yap for major overhauls.

Floor space has been left clear for a clamping fixture. The feedwater tank with the deaerator is installed on the level of the turbine-operating floor. The feedwater pump units and the start-up and shutdown pumps are installed underneath.

An overhead travelling crane for transporting and erecting plant components is installed in the turbine building.

The controlled access to the "controlled area" is located in the same floor of the switchgear building as the control room.

Ventilation of individual floors is provided by the ventilation systems through redundant intake and exhaust air shafts and smokes vents.

This is accomplished by the extreme leak-tightness of the primary systems, adequate shielding and physical separation of equipment, and by the ventilation which enforces a definite continuous airflow from rooms with lower activity to rooms with higher activity.

II.1.3. CNA I MAIN SYSTEMS

In what follows the main safety and process systems that are part of the station, are briefly summarised.

II.1.3.1. Reactor

The reactor ***(Figure II.1-5)*** is of the pressure vessel type, natural and slightly enriched uranium fuelled and heavy water moderated and cooled. The bulk thermal power is 1179 MWt.

The reactor core is approximately cylindrical in shape and consists of 252 enriched uranium fuel assemblies located in the same number of coolant channels. The fuel assemblies are bundles of 36 closely packed fuel rods which are arranged in 4 concentric rings having 1, 6, 12 and 17 fuel rods each, plus an additional structural rod located in the external ring. Each fuel rod consists of a stack of uranium dioxide pellets enclosed by a thin walled zircaloy 4 canning tube, which is both gas and pressure tight. Each fuel assembly, together with the filler body and the closure plug, forms the fuel bundle column. The coolant channels are arranged vertically in a trigonal lattice within the moderator tank. Each coolant channel contains one fuel bundle column.

The coolant channels are surrounded by the moderator, which is enclosed in the moderator tank. For reactivity reasons, the moderator is maintained at a lower temperature than the reactor coolant. This is accomplished by the moderator system, which extracts the moderator from the core, cools it down in the moderator coolers, and feeds it back into the core.

The heat removed from the moderator is used for pre-heating the feed-water. This is one of the reasons for the high net efficiency of the power plant.

The reactor coolant system and the moderator system are connected by the pressure equalisation openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor' pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

In order to control reactivity, and thus the power output of the reactor, various methods are applied. The reactor contains 24 "black" (absorbers made of hafnium) and 5 "grey" (steel) control rods. The control rods are used to control reactivity and power distribution, to compensate the build-up of xenon poisoning after a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor.

In addition to the control rods, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to bum-up during the first period of operation. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

In addition to these reactivity control systems, reactivity can also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel constitutes the pressure boundary of the reactor core and encloses core components and reactor pressure vessel internals. The reactor pressure vessel consists of a lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

Most of the reactor pressure vessel internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant, and, in conjunction with the reactor pressure vessel, forms the annulus for the in-flowing coolant. The moderator tank bottom serves as the lower fixing level for the coolant channels and the control rod guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head forms the upper plenum for the reactor coolant leaving the coolant channel slots. The closure head and the moderator tank jointly form a unit which keeps the coolant channels and the control rod guide tubes in position firmly and without displacement during all operating modes, as well as during postulated accidents. The moderator tank and its closure head are suspended from the flange of the reactor pressure vessel and are field tightly in position by the pressure vessel closure head.

The coolant channels consist of vertically-arranged tubes which contain the fuel bundle columns, direct the reactor coolant flow and separate the reactor coolant from the surrounding moderator.

The reactor coolant flows inside the coolant channels in an upward direction. After passing through the fuel assembly, it leaves the coolant channel through slots and enters the upper plenum formed by the moderator tank closure head.

The coolant channel closure head, together with the coolant closure plug, forms the pressure-tight cap of the coolant channel. It can be opened by the refuelling machine during reactor operation in order to exchange the fuel bundle column located inside the coolant channel.

The moderator piping serves for supply, distribution and extraction of the moderator inside the moderator tank. The moderator piping essentially encompasses down-comers, the sparger ring on the moderator tank bottom, and the suction boxes with nozzles in the moderator tank closure head.

The moderator flows downwards through the down-comers to the sparger ring, where it is distributed at the moderator tank bottom. After rising and heat-up in the moderator tank, the moderator flows to the suction boxes and leaves the moderator tank through two nozzles.

II.1.3.2. Reactor coolant system and moderator system

The reactor coolant system ***(Figure II.1-6)*** removes the heat generated in the reactor core and transfers it via the steam generators to the turbine generator plant.

The system is structured similar to that of a pressurised light water reactor and consists of two identical loops, each comprising a steam generator, a reactor coolant pump and the interconnecting piping, as well as one common pressurizer.

The heat is carried by the reactor coolant, which flows from the reactor pressure vessel to the steam generators, where it is cooled down and then pumped back to the reactor pressure vessel by the reactor coolant pumps.

The pressurizer system is connected to one reactor coolant loop and basically comprises the pressurizer with the electric heaters, the surge line, the spray lines with valves, and the safety valves.

Besides pressure control by sprays in the pressurizer, protection against over-pressure in the reactor coolant system is provided in accordance with international codes for pressure vessels and systems. Protection is afforded by independent, self-actuating safety valves.

When the safety valves open, the steam discharged from the pressurizer is directed into the pressurizer relief system, where it is condensed to water.

The moderator system consists of two identical loops operating in parallel. Each loop comprises a moderator cooler, a moderator pump, and the interconnecting piping with valves.

The moderator system performs various functions depending on the operating mode of the reactor.

During normal operation the moderator system maintains the moderator at a lower temperature than that of the reactor coolant. The moderator leaves the top of the moderator tank flows to the moderator pumps, is pumped to the moderator coolers and flows back to the bottom of the moderator tank. The heat transferred in the moderator coolers is used for pre-heating the feedwater.

For residual heat removal the moderator system is switched over to the residual heat removal position by means of the moderator valves. Under this mode of operation, the moderator is extracted from bottom of the moderator tank by the moderator pumps and fed into the cold legs of the reactor coolant loops, and also directly into the reactor coolant inlet annulus of the reactor pressure vessel via the moderator coolers. The moderator system forms the first link of the residual heat removal chain. The residual heat is transferred from the moderator system to the residual heat removal system and thin to the service cooling water system.

During emergency core cooling the moderator serves as a high-pressure core re-flooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

An essential feature of the moderator system together with the residual heat removal system is the allowing of the hot shutdown condition of the reactor to be maintained for as long as required, or the cool-down at a pre-set gradient, as well as emergency core cooling without main steam blow-off and thus without an additional heat sink.

All systems of the residual heat removal chain are of a consistent two-loop design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

II.1.3.3. Refuelling system

The slightly enriched uranium reactor makes it possible and desirable, with a view to obtaining a high burn-up, to shuffle and replace the fuel assemblies during power plant operation. The refuelling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pools, and the corresponding auxiliary systems and maintenance installations.

The refuelling procedure is fully automated and monitored from the control room.

The refuelling machine is moved from a maintenance position in the refuelling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor well in which the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refuelling machine to form a watertight seal between the machine and the coolant channel. Pressure equalisation takes place between the refuelling machine and the reactor before opening the isolation valve of the refuelling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refuelling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt-up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refuelling machine a check for leak-tight closure is performed. Then the refuelling machine is removed from the reactor pressure vessel and positioned above the vertically arranged tilter. The tiIter has the following functions in the indicated sequence:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with heavy water.
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the tiIter with heavy water.
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the tilter via the transfer tube, and later from there into the refuelling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The task of the fuel transfer tube is to establish a connection between the tilter within the reactor building and the tilting device in the fuel pool building, while both components are in the horizontal position.

The tilting device takes the fuel bundle column from the fuel transfer tube and swivels it from the horizontal into the vertical position.

Besides the main components mentioned above, important auxiliary systems, tools and maintenance and service equipment necessary inside the fuel pool building are provided.

The great advantage of the refuelling system is that it handles the fuel assemblies with only one refuelling machine and that it has one seal ring for each coolant channel. A silver-clad seal ring with good material flow properties is used as sealing material. In this established design, the coolant channel seals are almost perfectly tight.

Using the gas drying and cooling circuit in the tilter, the heavy water humidity is completely removed from the tilter and then recovered by cooling and condensing the extracted gas stream. This Is one of the reasons for the very low heavy water -in Atucha I Nuclear Power Plant. With this fuel transport system, it is also possible to transport semi-burnt fuel assemblies from the fuel pool building to the reactor.

II.1.3.4. Reactor auxiliary and ancillary systems

The auxiliary systems are basically organized in the same way as the auxiliary systems in PWR plants. The auxiliary systems work together with the reactor coolant system and moderator system to ensure the specified chemical conditions of the coolant and moderator. The systems containing heavy water are strictly separated from the systems containing light water in order to avoid downgrading the heavy water. The main tasks of the auxiliary systems are:

- Storage of heavy water.
- Volume control, seal water supply.
- Treatment and upgrading of heavy water.
- Boric acid dosing and chemical feeding into the primary circuit.

- Fast boron injection.
- Nuclear component cooling.
- Fuel pool cooling.
- Supply of refuelling machine with auxiliary fluids.
- Compensation of leakages.
- Removal of decay heat from the core, emergency core cooling.

The auxiliary and ancillary systems are located mainly in the auxiliary building and partly in the annulus of the reactor building.

Based on the primary system concept, the number of auxiliary systems in CNA I is minimized. This is the result of simple water chemistry in the primary system, of the same heavy water quality and enrichment in the reactor coolant and moderator system, and is also a logical consequence of the material concept for the primary system and for the auxiliary systems.

II.1.3.5. Main control room

The main control room of the nuclear power plant contains the operating and information equipment for the control and monitoring of the plant systems. This means that manual control, set-point adjustment and monitoring of the reactor, important reactor auxiliaries, the feedwater/steam cycle, the turbine, the generator and the auxiliary power equipment are controlled from the main control room.

Additional enunciator panels are located in the main control room. These include the fire alarm system, area monitoring, etc.

II.1.3.6. Instrumentation and control systems

The instrumentation and control equipment includes measurement, control, protection and monitoring systems.

The control room is, through the automation and plant interface equipment, connected to the drives and signal transmitters in the plant. Plant conditions and operational transients are transmitted via analog and binary signal transmitters to both the operator in the control room and to the automatic equipment. The command signals to the drives in the plant are transmitted via the control interface as required for maintaining proper operation.

Measured-value and status signals from the entire plant are continuously monitored by means of a process computer. The computer provides the data for trend logging and fault analysis and transmits the information to the operating personnel in the control room and other places via data display terminals and tele-printers.

Automatic functional group controls are provided to minimise the operating errors and to obtain a higher degree of automation.

II.1.3.7. Electric power system

The Atucha I nuclear power plant has two physically independent grid connections ***(Figure II.1-7)***. One of them is the 220 kV grid and the other is the 132 kV grid. In addition, the basic concept enables CNA I auxiliary power supply from the generator in case of a grief disturbances after load rejection. Only in the case of common outage of all three power supply possibilities, the emergency power system with the diesel generators will be required. Definite loads, mainly of the control and instrumentation field, are power supplied by rectifiers and converters or by means of batteries with direct current.

The generator feeds into the 220 kV network via one generator transformer and supplies the station auxiliary service requirements by means of one high-voltage station service transformer.

The high-voltage station service transformer or the off-site system transformer feed into two separate medium (each 6.6 kV) high voltage bus sections, to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the station service power system fails, certain equipment (pumps, etc.) are needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put into operation and must therefore be supplied with emergency power.

The emergency power system is like the other safety equipment divided into redundant separate trains. Under normal operating conditions, the auxiliary switchgears of the auxiliary power system feed the emergency power system. To avoid loss of power in case the auxiliary power system fail, each of the redundant trains in the emergency power system is equipped with a quick-starting diesel set.

CNA I 's electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 220 kV and 132 kV transmission lines connecting CNA I with two sub-stations belonging to the national electric grid.

With the generator load-breaker in the "off" position the station service power for "start up" and "shut down" of the CNA I can be drawn from both 220 kV grid and 132 kV grid. Upon simultaneous failure of the main grid and the turbine generator set the 132 kV grid provides power for shut down operation of the plant down to the "hot-subcritical" condition.

The onsite power system consists, in turn, of two subsystems: the auxiliary power system (“normal system") and the emergency power system.

The auxiliary power system provides power for the loads of the nuclear power plant, which are necessary during normal operation, start-up and shutdown operation. It is subdivided into two trains {6.6 kV buses BA and BB} which are supplied normally by the high-voltage station service transformer. This transformer is fed either from the generator or from the 220 kV grid via the generator transformer. For shutdown operation or after loss of the normal power supply grid and generator, it may be fed by the off-site power supply via the off-site system transformer. The offsite power supply system is available via automatic changeover.

The emergency power system provides the power required for safe shut-down of the reactor to maintain it in the shut-down condition, for removal of residual heat and to prevent release of radioactivity during normal operation and accident conditions, and for some loads important for plant availability. It is subdivided into two trains -6.6 kV uninterruptable BU and BV buses-, which are usually supplied by 6.6 kV buses BA and BB as well as the water turbine driven generator (located in the water turbine building).

During emergency situations only safety related loads are fed. For those situations in which one or more out of the three power supply possibilities before mentioned are available, buses BU and BV continue to be fed by buses BA and BB. In this last case, the condition of uninterruptible voltage in BA and BB is initially achieved by the emergency diesel generators supply.

As the water turbine-driven generator was connected to bus BV before the emergency situation, this turbo-generator continues supplying energy after the emergency signal, during a time period of 40 seconds; during such period, the emergency diesel generators start and they are connected to buses BU and BV.

The emergency diesel generators are three redundant units and it is enough that any two of them are functioning, for the supply of all the loads fed from buses BU and BV. Each emergency diesel generator is, in turn, constituted by different main and auxiliary subsystems, such as the starting subsystem, the lubrication subsystem, etc.

II.1.3.8. Safety systems

The safety philosophy, on which the design is based, fulfils, In all conceivable plant conditions, the following basic requirements:

- The reactor can be safely shut down and kept shut down over prolonged periods (the decay heat can be reliably removed).
- Any release of radioactivity is within the limits established by the radiation protection regulations.

In order to meet these requirements, safety measures against damage to the systems or components are provided. Safety measures can be classified in three safety levels according to the possible plant conditions:

Components and systems necessary for normal operation (including startup, partial load and full load operation, load changes, shutdown) are of such design as to preclude failure. The safety measures provided are:

- Conservative and careful design.
- Stringent quality assurance and control.
- Regular examinations and inspections.

According to general engineering experience, it must be considered that systems and components can fail during their service life despite adequately high quality. It is assumed that operational disturbances (e.g. reactor coolant pump failure, load rejection) can occur. In order to prevent faults and operational disturbances and to mitigate their consequences the following safety measures are provided:

- Inherently safe operational characteristics.
- Alarm annunciation.
- Reactor protection limitation.

Despite the safety measures of the first and second safety levels, theoretically assumed accidents are postulated. In order to prevent these accidents and to mitigate their consequences, safety systems are provided. The design of the safety systems is based on the assumption that parts of the safety systems (sub-systems) can fail simultaneously with the accident. As a consequence, safety systems are of redundant design.

The basic safety systems provided are:

- Fast Reactor Shutdown System.
- Emergency Core Cooling System.
- Containment System.
- Emergency Electric Power System (safety related system).

In order to protect the environment against the release of radioactivity, the following radioactivity barriers are provided as passive safety measures:

- The fuel matrix of the uranium dioxide pellets.
- The seal welded cladding tubes enclosing the fuel.
- The closed and seal-welded reactor coolant system and moderator system.
- The full-pressure gas-tight steel containment structure.
- The concrete secondary shield.

The components of the radioactivity barriers act according to their mechanical properties, without auxiliary energy. In case of damage to one of these barriers the next one will act and thus retain the radioactivity.

The accidents considered in the plant design are the plant internal and external accidents. The internal accidents are, above all, loss of coolant accidents (LOCA), with the whole spectrum of pipe ruptures including the break of the largest connection pipe to the reactor coolant loops or to the moderator system. The external accidents considered are aeroplane crash, explosion pressure wave, floods, tornadoes, etc.

In order to meet the safety requirements even during the considered internal and external accidents, the following design principles were established:

- Multiplicity of safety features.
- Redundancy of safety systems and of their auxiliary systems.
- Diversity of important parts of the reactor protection system.
- Physical separation and/or protection by concrete walls of the redundant sub-systems.
- Protection of safety systems against external accidents.
- Periodic testing of safety systems.

The task of the safety systems is to prevent any damage to the radioactivity barriers during operational malfunctions and during accidents in order to fulfil the safety philosophy requirements.

The fast reactor shutdown safety system consists of two separate sub-systems: the shutdown control rod system (first independent shutdown system) and the boron injection system (second independent shutdown system). The emergency core cooling safety system consists of the following basic sub-systems: the moderator system, the residual heat removal system, the service cooling water system for the secured plant, the nuclear component cooling system and the safety injection system.

The containment safety system consists of several basic sub-systems: the concrete containment, the steel containment, the containment isolation system and the reactor building annulus air extraction system.

The safety systems are supported by a high degree of quality assurance and quality control measures, regular inspections during operation of the plant and in-service inspection programs. Through these measures, a high safety standard can be ensured.

II.1.3.9. Technical data

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	Pressurised heavy water (PHWR)
Net nominal electric power	335 MWe
Bulk nominal electric power	357 MWe
Authorized thermal power	1179 MWt

Reactor Core Data	
Type of fuel	Slightly enriched uranium (0.850 weight)
Shape of fuel assembly	37 - rod cluster
Number of coolant channels or fuel assemblies	252
Cladding material	Zircaloy 4
Fuel assemblies length	6180 mm
Refuelling	On load
Coolant and moderator	Heavy water
Thermal and Hydraulic Data	
Pressure at reactor vessel inlet	12.2 MPa
Pressure at reactor vessel outlet	11.6 MPa
Coolant channel inlet temperature	264 °C
Coolant channel outlet temperature	303.3 °C
Maximum temperature on the fuel assembly cladding surface	325 °C
Coolant flow in coolant channels	20210 t/h
Average coolant speed in central channel	9 m/s
Mean heat-flux density	67.7 W/cm^2
Average specific thermal power of fuel roofs	232 W/cm
Heavy water concentration	99.75 (weight)

Steam and Power Conversion System Data	
Live steam pressure at steam generator outlet	4.46 MPa
Live steam temperature at steam generator outlet	254.9 °C
Live steam flow	1856 t/h
Live steam moisture	0.3%
Turbine rated speed	3000 rpm
Condenser pressure	4.56 kPa
Cooling water inlet temperature of condenser	22 °C
Cooling water flow of condenser	62500 m^3/h
Generator apparent power	425 MVA
Generator power factor	0.8
Generator voltage	21 kV
Generator transformer rated power	400 MVA
Generator transformer transformation ratio	21 kV / 245 kV
High-voltage station service transformer rated power	35 / 20 / 20 MVA
High-voltage station service transformer transformation ratio	21 kV / 6.95 kV
Generator off-site system transformer rated power	35 / 20 / 20 MVA
Oft-site system transformer transformation ratio	132 kV / 6.95 kV

SITE LOCATION

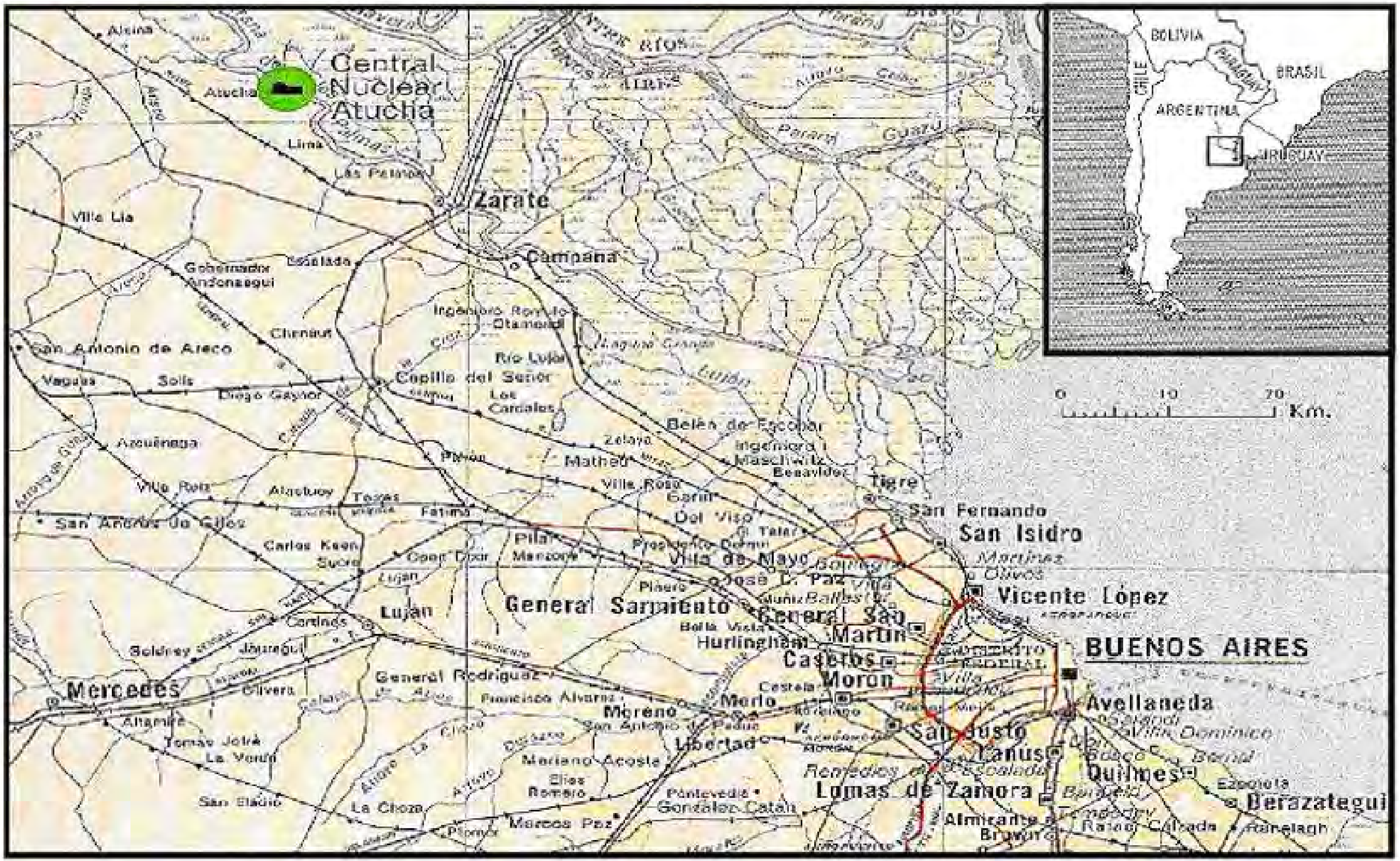


Figure II.1-1 - Atucha I Nuclear Power Plant - Geographic Location

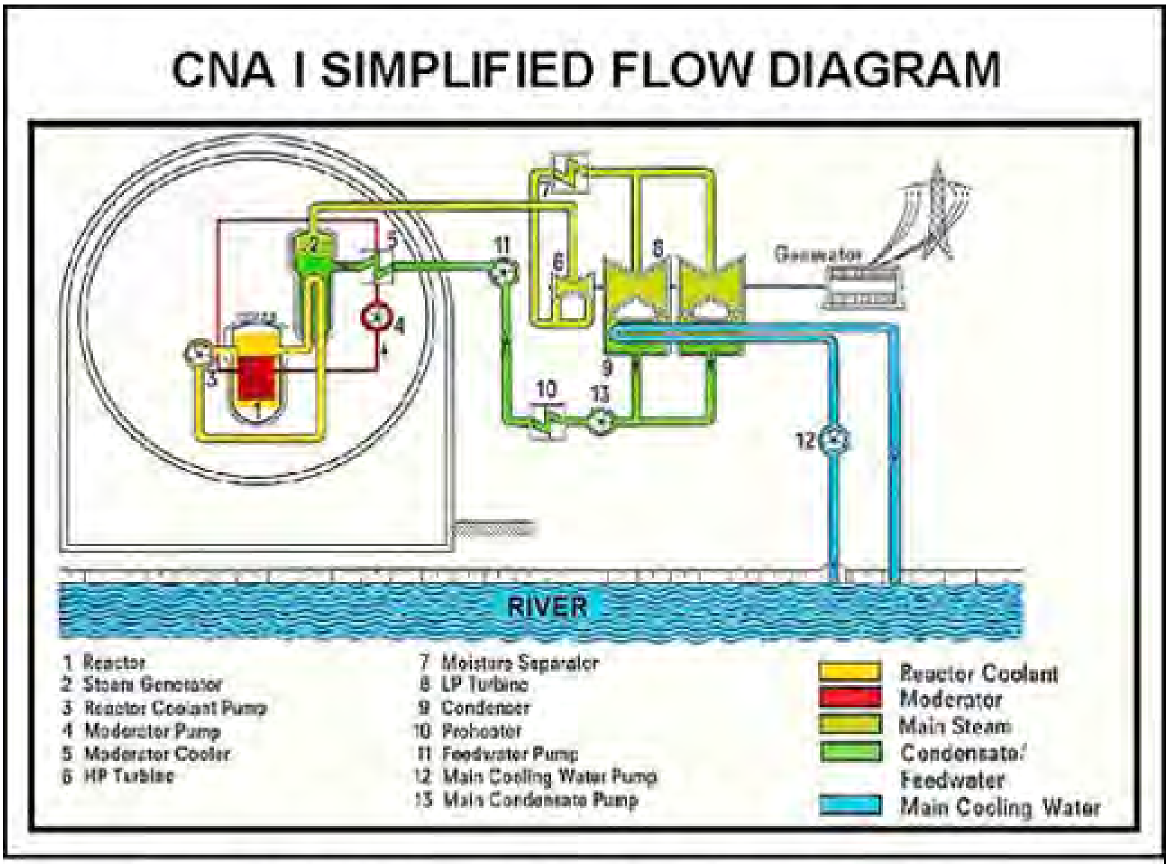


Figure II.1-2 - Atucha I Nuclear Power Plant - Main Systems

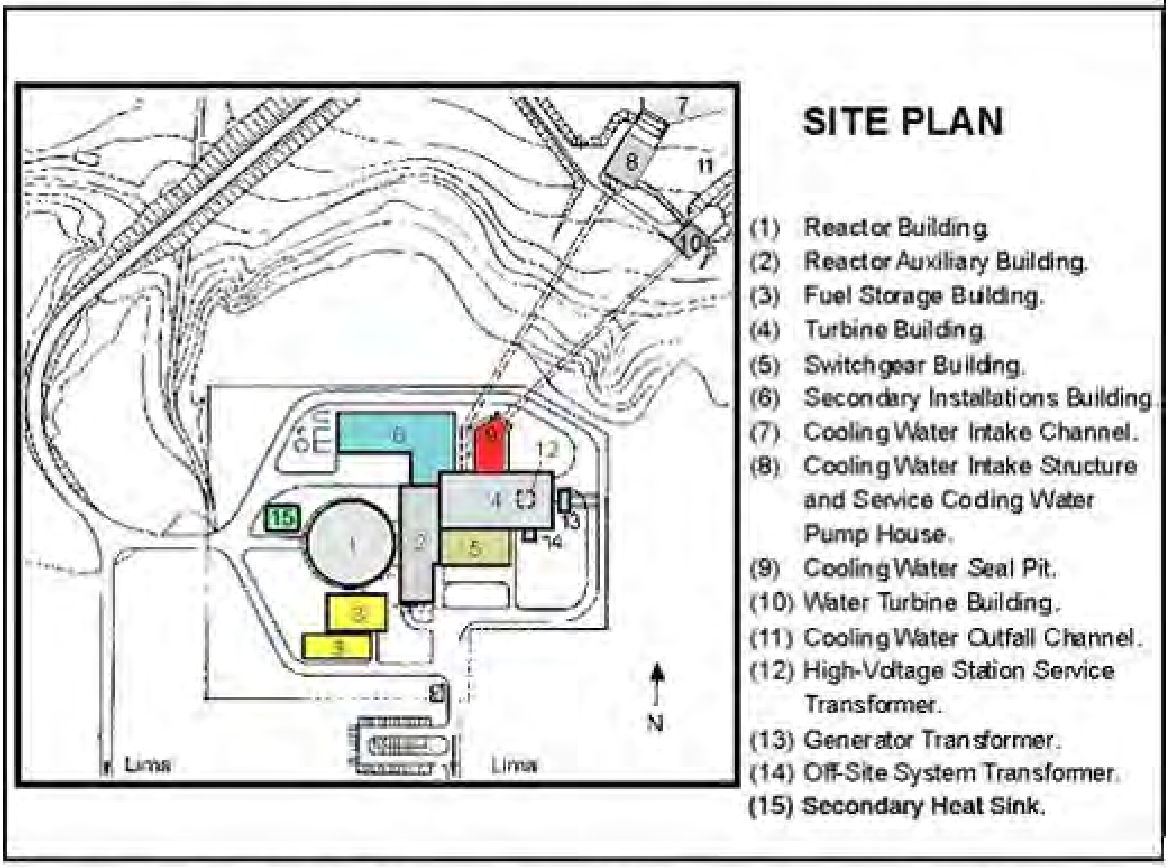
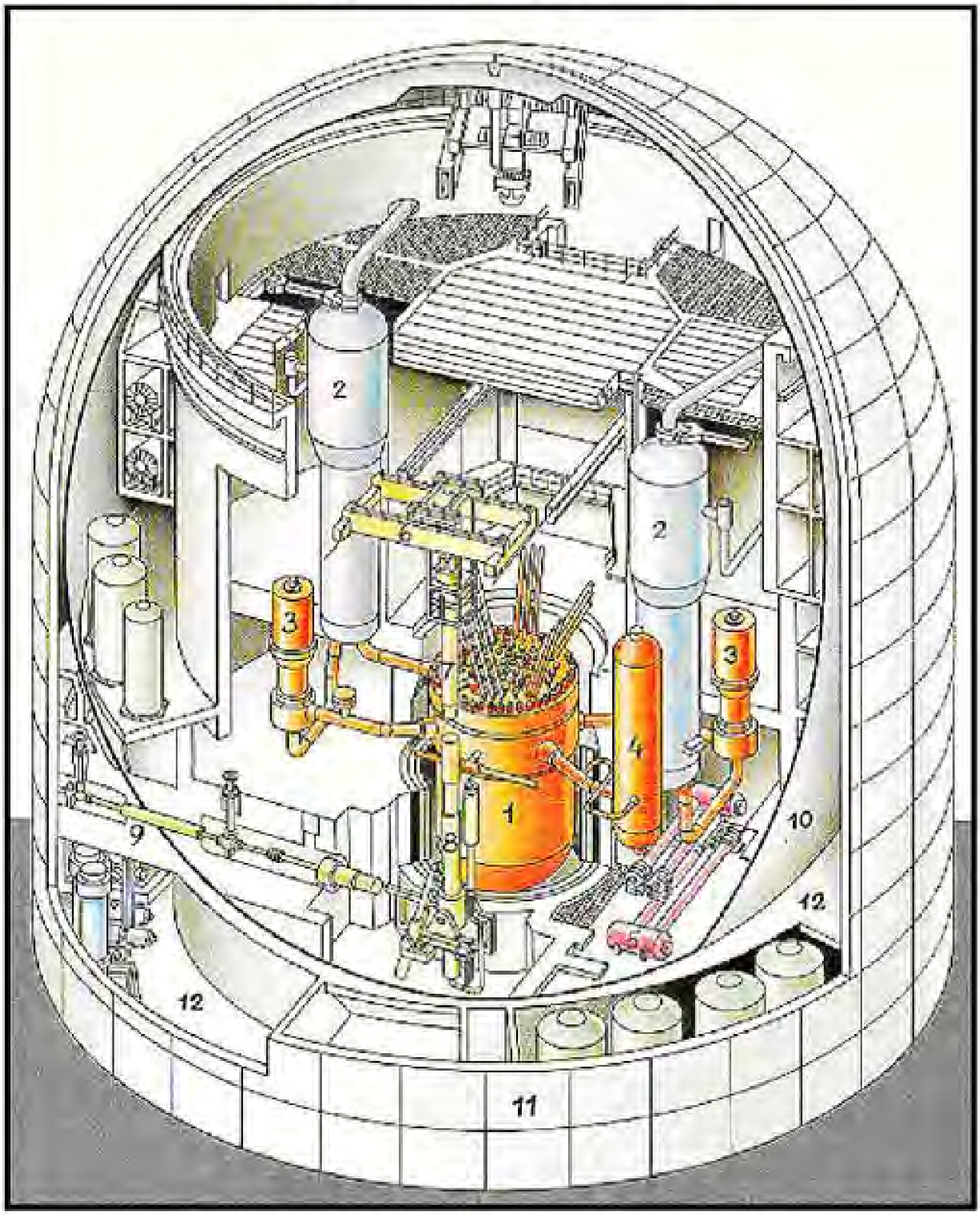


Figure II.1-3 - Atucha I Nuclear Power Plant - Main Building and Structures

REACTOR BUILDING



1 - Reactor pressure vessel
2 - Steam generator
3 - Reactor coolant pump
4 - Pressurizer
5 - Moderator cooler
6 - Refueling machine travelling gear
7 - Refueling machine
8 - Tilter
9 - Fuel transfer tube
10 - Containment
11 - Reactor Building.
12 - Annulus

Figure II.1-4 - Atucha I Nuclear Power Plant - Reactor Building

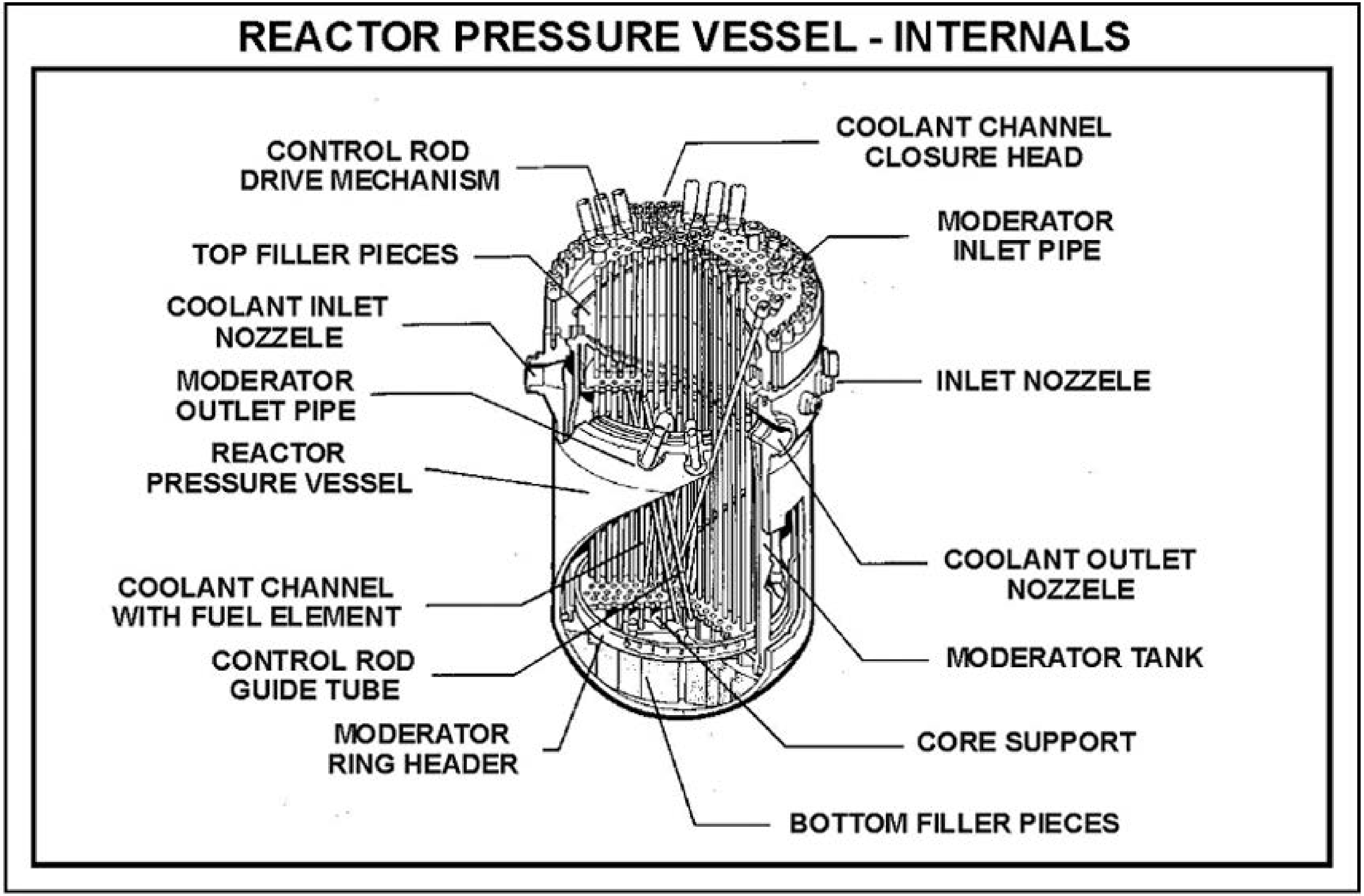
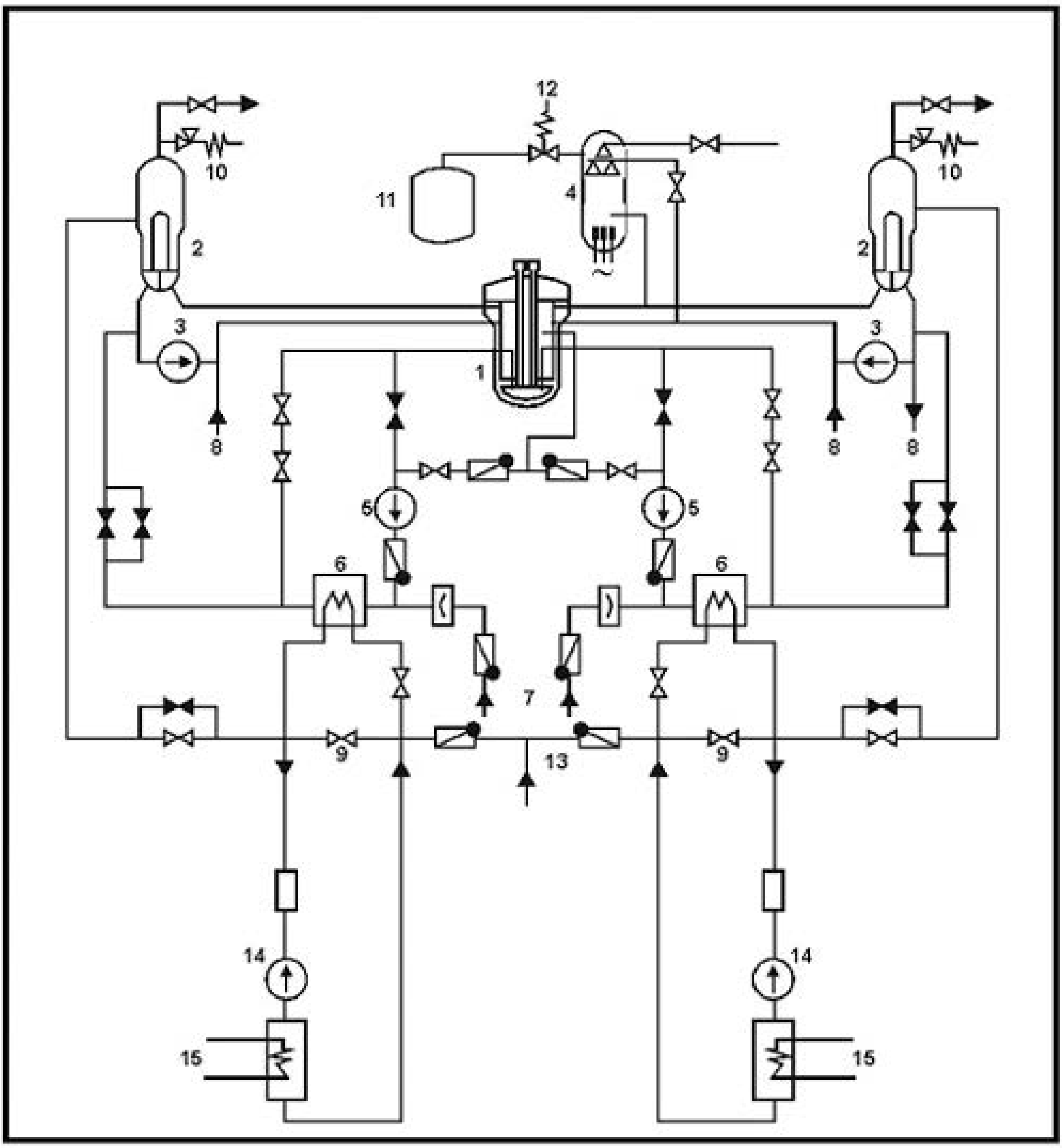


Figure II.1-5 - Atucha I Nuclear Power Plant - Reactor Pressure Vessel and Internals

REACTOR COOLANT SYSTEM AND MODERATOR SYSTEM



1 - REACTOR PRESSURE VESSEL
2 - STEAM GENERATORS
3 - REACTOR COOLANT PUMPS
4 - PRESSURIZER
5 - MODERATOR PUMPS
6 - MODERATOR COOLERS
7 - EMERGENCY COOLING SYSTEM INLET
8 - PRESSURE AND INVENTORY CONTROL SYSTEM
9 - SHUTDOWN COOLING SYSTEM (MODERATOR)
10 - SECONDARY SIDE SAFETY VALVES
11 - PRESSURIZER RELIEF TANK
12 - PRIMARY SIDE SAFETY VALVES
13 - SECONDARY INLET LIGHT WATER
14 - RESIDUAL HEAT REMOVAL SYSTEM
15 - SERVICE COOLING WATER SYSTEM FOR PLANT SECURED

Figure II.1-6 - Atucha I Nuclear Power Plant
Reactor Coolant System and Moderator System

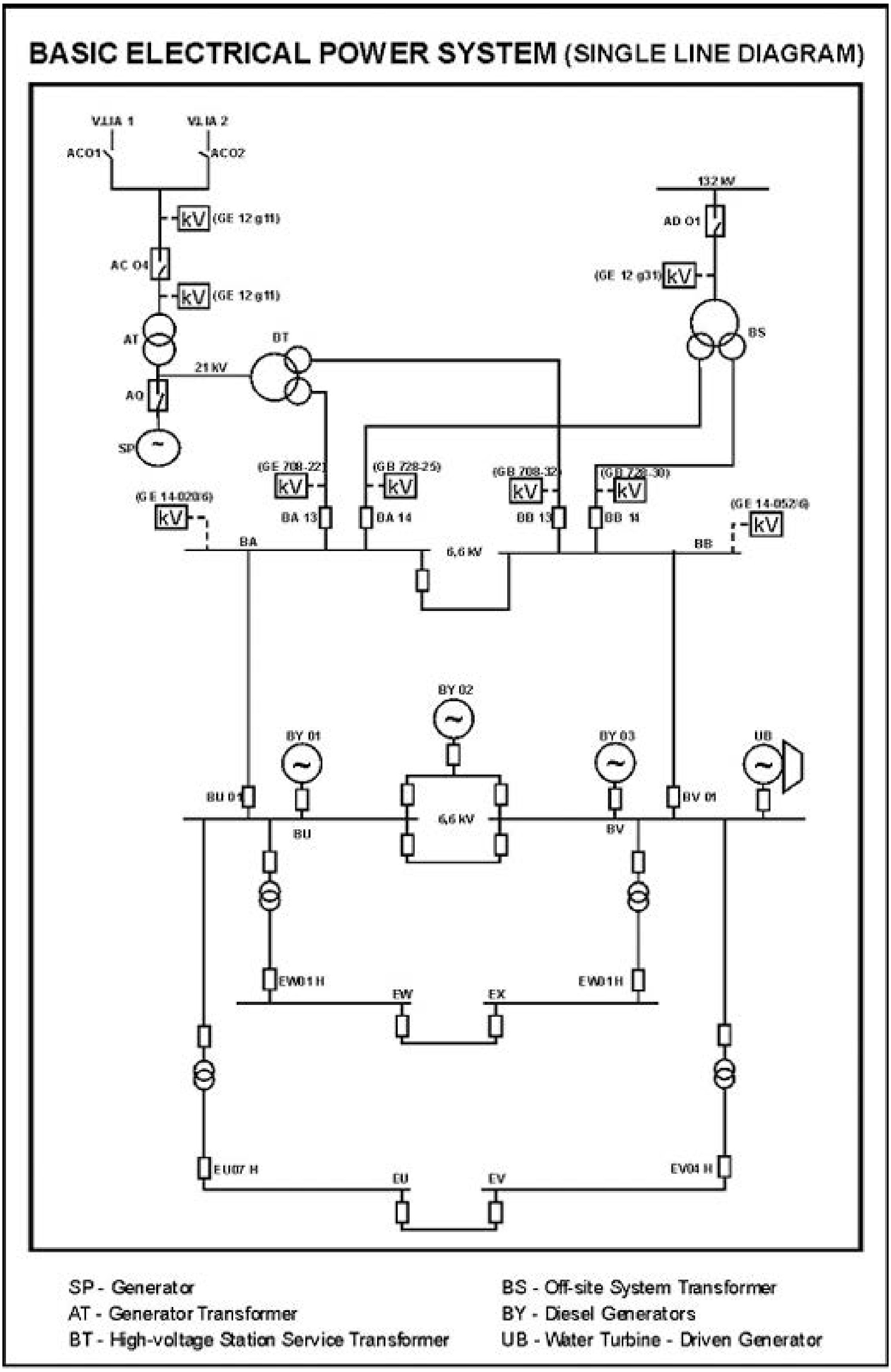


Figure II.1-7 - Atucha I Nuclear Power Plant
Basic Electrical Power System

II.2. EMBALSE NUCLEAR POWER PLANT

II.2.1. INTRODUCTION

In 1967 the Comisión Nacional de Energía Atómica (CNEA) of Argentina initiated the feasibility study for the construction of Embalse Nuclear Power Plant (CNE) and in 1973 signed a contract with Atomic Energy of Canada Limited (AECL) and Societa Italiani Impianti P.A. (IT) for a 600 MWe CANDU-PHW (pressurized heavy water) type nuclear power plant at the Embalse site in the Province of Córdoba, Argentina, on the Almafuerte Peninsula just out from the south shore of Río Tercero Lake, as shown in ***Figure II.2-1***.

The construction of the station began in May 1974 and the commercial operation in January 1984.

At present, the owner of CNE is Nucleoeléctrica Argentina S.A., and the plant provides a net electric power of 600 MWe to the interconnected national system.

The plant is designed for commercial base-load operation. It contains a turbine generator set, with steam supply from a CANDU-PHW type nuclear reactor. This design has been used in all Canadian designed nuclear power plants built to date, with the exception of Gentilly-1.

Besides, the plant also has components, equipment and sub-systems required for the functioning of the big systems located at its "nuclear" and "conventional" sections.

The CANDU-PHW type reactor uses heavy water as moderator and as a heat transport medium. The fuel is natural uranium supplied in the form of bundles loaded into and removed from the reactor during "on power" operation. Its thermal power is 1987 MWt. A closed loop cooling circuit is provided to transfer the heat from the fuel and to produce light water steam in the steam generators. The turbine cycle is similar to that which has been used for other plants of this type.

Figure II.2-2 shows schematically the main systems of the Embalse Nuclear Power Plant.

II.2.2. OVERALL PLANT LAYOUT

Building and structure arrangements of CNE are shown in ***Figure II.2-3***.

CNE's main buildings and structures may be classified into nuclear steam plant and balance of plant. The nuclear steam plant includes the reactor building, service building, emergency water supply building, high-pressure emergency core cooling building, and their contents except for balance of plant equipment in the control room. The balance of plant includes all other buildings and their contents.

The reactor building ***(Figure II.2-4)*** houses the reactor, fuel handling systems, the heat transport system, including the steam generators, and the moderator system, together with their associated auxiliary and special safety systems.

The reactor building is divided into three major structural components: the containment structure, the internal structure, and the reactor vault structure.

The containment structure is the main component of the containment system. This structure is a pre-stressed concrete building comprising three structural components: a base slab approximately 1.74 m thick; a cylindrical wall approximately 41.5 m diameter with a minimum wall thickness of about 1.07 m, and a spherical segmental dome with a thickness at the crown of about 0.60 m.

Beneath the outer dome there is a second dome having an opening in the crow, which together with the perimeter wall forms a container to provide storage for 2170 m^3 of water for dousing and emergency core cooling.

The internal structure is a reinforced concrete building dividing the reactor building into two areas as follows: the "accessible area” to which operating and maintenance personnel have access during normal plant operation, and the "inaccessible area" which is not accessible during plant operation, but to which access can be obtained after plant shutdown. The internal structure is separated from the containment structure. All those system and items of equipment to which access is routinely required for operation, servicing or maintenance, are housed in rooms within the accessible area. Outside of the accessible area, the remainder of the reactor building forms the inaccessible area containing the reactor and its vault, the heat transport and moderator system, the fuelling machine operating areas, steam generator room, and the areas for auxiliaries. Service cranes are provided as required in this area.

The reactor vault structure is a reinforced concrete, carbon steel-lined, light water-filled tank which contains and supports the calandria and end shields. Adequate shielding is provided by the concrete vault for access within the reactor building during plant operation. The vault is independent of other structural units within the reactor building.

The service building is a conventional reinforced concrete structure with concrete floors. It contains the following main facilities: control room, spent fuel transfer and storage facilities, and heavy water treatment and radioactive waste treatment facilities. It also contains conventional and nuclear service facilities such as stores, workshops, charge rooms, a decontamination centre and laboratories.

The turbine building, consisting of a turbine hall and the turbine auxiliary bay, has a reinforced concrete main structure. The turbine hall houses the turbine generator and some associated auxiliary equipment. Other auxiliary equipment and electrical power distribution equipment are contained in the turbine auxiliary bay.

The auxiliary bay is adjacent to and structurally independent from the service building which forms part of the plant. The main access leading to the loading bay In the turbine building is at the end of the turbine hall.

The building complex has reinforced concrete foundations and structures. The turbogenerator pedestal is a reinforced concrete structure rising from the foundations slab. Only the roof of the turbine building is structural steel work.

The other main structures of the station are: diesel building, emergency water supply pumphouse, and water supply structures.

The diesel generator and emergency steam generator building has reinforced concrete slabs. The diesel building consists of one single building formed of concrete walls and roof. This building is divided into five rooms, four of which contain generator sets together with the related control panels. The fifth room is used for storage purpose. Partition walls between these rooms are full height reinforced concrete. The auxiliary steam generators are located outdoors near the diesel building.

The emergency water supply pumphouse is a reinforced concrete structure with a floor elevation of 97.0 m. Two removable hatches in the concrete roof slab are located over the diesel-driven pumps.

The water required for the different services of the station is taken from the reservoir of Embalse by means of the water supply structures. These structures include: pump house, water intake structure, and the water circulation piping.

II.2.3. CNE MAIN SYSTEMS

In what follows the main safety and process systems that are part of the station, are briefly summarized.

II.2.3.1. Reactor

The Canadian heavy water-moderated, natural uranium-fuelled, pressurized heavy water reactors utilize the “pressure tube" concept. This consists of an array of pressure tubes, containing the reactor fuel, passing through a large cylindrical vessel (the calandria) containing the heavy water moderator and reflector.

Pressurized heavy water coolant is pumped through the pressure tubes, cooling the fuel and conveying heat from the fuel to the outlet header and to the steam generator. Each pressure tube is isolated and insulated from the heavy water moderator by a calandria tube. The annular space between concentric pressure and calandria tubes is filled with a gas.

It should be noted that this type of design results in a partially redundant structure, insofar as any localized failure of the moderator boundary will not result in the failure of the structure as a whole, and is therefore tolerable from a safety standpoint.

The reactor assembly ***(Figure II.2-5)*** comprises the calandria assembly within the calandria vault, fuel channel assemblies and reactivity control units. The calandria vault is an ordinary carbon steel-fined concrete structure, and is filled with light water. The water serves as a thermal shield and as a cooling medium.

The calandria assembly comprises the calandria, two end shields, and an embedment ring at each end shield (the embedment rings are grouted into the concrete wall of the calandria vault). This assembly forms an integral multi-compartment structure which provides containment for the heavy water moderator and reflector, the fuel channels (less end fittings), the reactivity control units, and the reactor shielding.

The calandria comprises a horizontal, cylindrical, single-walled, stepped shell, enclosed at each end by the tubesheet of an integral end shield, and spanned horizontally by 380 integral calandria tubes. The functions of the calandria are the following:

- Contains the heavy water moderator and reflector enveloping the in-core portions of the fuel channels.
- Helps support the in-core components of the reactivity control units.
- Helps support the fuel channels.
- Helps support the moderator piping, and any other piping, attached to it.

The calandria is designed for a postulated pressure tube/calandria tube rupture. To limit the pressure resulting from such an accident, four pressure relief pipes are provided. These pipes extend from the top of the calandria through the shield light water in the calandria vault, and terminate at the rupture discs located in the top of the calandria vault adjacent to the reactivity mechanism deck.

The end shields are horizontal, cylindrical shells enclosed at each end by tubesheets, and spanned horizontally by 384 lattice tubes. They contain biological shielding material in the form of carbon steel balls and ordinary light water. The functions of the end shields are as following:

- Shields the fuelling machine areas from the reactor during reactor operation and during shutdown.
- Helps support the calandria.
- Helps support and align the fuel channels.
- Provides a gas-filled annulus between the hot end fittings and lattice tubes in order to minimize the heat loss.

Two end shields are integral parts of the calandria assembly, one end shield being welded to each end of the calandria. Outside of each end shield, and concentric to it, is the end shield embedment ring which is grouted into the calandria vault wall. The end shield support structures are designed to accommodate the differential movements between the reactor assembly and the calandria vault which result from thermal and loading effects.

Each fuel channel assemblies consists of a zirconium-niobium alloy pressure tube expanded at each end into the hub of an alloy steel end fitting. Each assembly with its fuel and heavy water coolant is supported by the end shield lattice tubes through sliding bearings and, partially, by the calandria tube/pressure tube annular spacers. The end fittings are designed to allow relative axial movement between the fuel channel assemblies and the lattice tubes to cater for thermal expansion and pressure tube creep.

The inlet and outlet end fittings are designed to meet the following requirements:

- To provide a suitable high pressure closure that can be operated by the fuelling machine to allow insertion and removal of fuel.
- To provide shielding in the end shield penetrations to allow service access to the fuelling machine operating areas and to the face of the end shields at shutdown.
- To provide a transition between the pressure tubes and the primary circuit piping.
- To provide support for the pressure tubes and their contents.

The channel closure consists of a flexible seal disc mounted on a body witch locks firmly into the end fitting by means of a set of extendable jaws. The seal disc bears against a seal face in the end fitting to prevent leakage and is nickel plated to improve leak tightness. A shield plug is locked into each end fitting where the end fitting passes through the end shield. Both the channel closure and the shield plug can be removed and reinserted by the fuelling machine during refuelling.

The fuel is designed to be compatible with the operating conditions imposed on it by the heat transport system, the fuel handling system and reactor nuclear design.

The reactor is fuelled with natural uranium in the form of compacted and sintered cylindrical pellets of uranium dioxide. About thirty-five uranium dioxide pellets are stacked end-to-end and are sealed in a zirconium alloy sheath to form a fuel element. An interlayer of graphite between the pellet stack and the sheath is used to reduce the pellet/sheath interaction. These fuel elements are also fitted with zirconium alloy spacers and bearing pads. Thirty-seven fuel elements are welded to two zirconium alloy end plates to form the cylindrical bundle. The end plates maintain separation among the fuel elements at the bundle element extremities.

The separation among the fuel elements at the bundle mid-length is maintained by the spacers which are brazed to the fuel elements. The spacers are positioned on each individual fuel element such that the contact between any two mating elements is spacer-to-spacer. Bearing pads are brazed to the outer ring of fuel elements. The three planes of bearing pads maintain proper clearances between the bundle and the fuel channel during fuel handling operations and during the bundle's residence in the reactor.

Concerning reactivity control units it should be mentioned that neutron absorbing devices, both liquid and solid, are provided to control reactivity. During operation, reactivity is controlled by adjuster units, control absorber units, and liquid zone control units. Under emergency or abnormal conditions, reactor shutdown is quickly achieved by dropping shutoff absorbers into the reactor core, or by injecting liquid poison into the heavy water moderator.

Twenty-one vertical adjuster units are provided, each comprising an assembly of zircaloy clad cobalt absorber elements, a vertical guide tube and a drive mechanism. The absorber shape the neutron flux for optimum reactor power and fuel burnup when inserted in the calandria, and upon removal from the calandria allow excess reactivity for overriding xenon poison following a power reduction.

Four control absorber, mounted vertically, adjust the flux level at times when greater reactivity rate or depth is required than that provided by the zone control system. The design is essentially the same as that of the shutoff units, except that the shutoff unit accelerator spring is omitted from the design.

The liquid zone control units are tubular members divided into compartments within the reactor core, each capable of being filled to any desired level with light water. There are six vertically oriented zone control units in the reactor. The units are used to adjust the flux level in any one of fourteen zones in the reactor. This is accomplished by introducing a continuously controlled amount of light water into the zones to provide a local control of neutron absorption.

On the other hand the reactor has two shutdown systems: the shutoff units and the liquid poison injection system; these systems are discussed in section II.2.3.9.

II.2.3.2. Heat transport system

The heath transport system circulates pressurized heavy water (reactor coolant) through the reactor fuel channels to remove heat produced by fission of uranium fuel. The heat is carried by the reactor coolant to the steam generators where it is transferred to light water to form steam, which subsequently drives the turbine generator.

The major components of the heat transport system are the reactor fuel channels, four vertical steam generators, four motor driven pumps, four reactor inlet headers, four reactor outlet headers, one electrically heated pressurizer, and all necessary interconnecting piping and valving. The fuel channels are horizontal and allow access to both ends by the fuelling machines. The header, steam generators and pumps are located above the reactor. The normal operation flowsheet for the heat transport system is show in ***Figure II.2-6***.

The main features of the transport system are as follows:

- Circulation of the reactor coolant is maintained at all times during reactor operation, shutdown and maintenance.
- Each heat transport pump has sufficient inertia in rotating components to prevent a sudden decrease in the flow if power to the pump motor is lost.
- Adequate heat transport system flow for shutdown heat removal is maintained by natural convection flow following pump rundown.
- Heat transport system pressure is controlled at an acceptable value for all normal modes of operation.

- System components are protected from overpressure by instrumented relief valves and suitable reactor regulating and/or safety system action.
- A separate shutdown cooling system is provided to remove reactor shutdown heat, thus permitting the draining of steam generators and pumps in the heat transport system, for maintenance.
- Purification by filtering, ion exchange and degassing is provided to control the chemistry of the reactor coolant.
- Potential heavy water leak sources are kept to a minimum by using welded construction and bellows sealed valves wherever practicable. Where potential leak sources exist, they are connected to closed collection and recovery systems.

The heat transport system has two loops in order to reduce the rate of positive coolant void reactivity insertion in the event of a loss-of-coolant accident. The two loops each contain two steam generators, two pumps, two reactor inlet headers, two reactor outlet headers, one set of inlet feeders and one set of outlet feeders. Feeders flow are matched to individual fuel channel powers to give an equal heavy water steam quality for each channel at the reactor outlet headers when the reactor is at full power. Pressure drop causes the heavy water steam quality to increase at the inlet to the steam generator.

The two figure-of-eight loops provide bi-directional flow through the core such that the flow is in opposite directions in adjacent channels. Each loop removes the heat from half of the fuel channels in the reactor core. Each loop has one inlet and outlet header at each end of the reactor core. Heavy water is fed to each of the fuel channel through individual feeder pipes from the horizontal reactor inlet headers, and heavy water is returned from each fuel channel through individual outlet feeder pipes to the horizontal reactor outlet headers. Individual feeder piping sizes depend on the coolant flow to the particular channel.

The pressure in the reactor outlet headers is controlled by a common pressurizer connected to a line linking the outlet headers at one end of the reactor. Valves in these lines provide isolation between the two loops in the event of a loss-of-coolant accident.

Two pipes connect each reactor outlet header to one steam generator. As the reactor coolant passes through the four steam generators, heat is removed and the reactor coolant at the outlet of the steam generator is sub-cooled. Each steam generator is connected to the pump suction of one heat transport pump by one pipe, and each heat transport pump delivers coolant to one reactor inlet header through two pipes.

II.2.3.3. Moderator system

Neutrons produced by nuclear fission are moderated by the heavy water in the calandria. The heavy water is circulated through the moderator system of cooling, for purification and for control of the concentration of substances used for reactivity adjustment. ***Figure II.2-7*** is a simplified flow diagram for the moderator system. The system consists basically of two 100% capacity pumps (connected in parallel) which are connected in series with two 50% capacity heat exchangers (connected in parallel). The series/parallel arrangement permits the operation of either pump with the two heat exchanger. Main moderator system connections are provided for the purification, liquid poison addition, heavy water collection, heavy water supply and heavy water sampling systems.

The heavy water in the calandria functions as a heat sink in the unlikely event of a loss-of-coolant accident coincident with failure of emergency core cooling. The capacity of the heat sink is assured by controlling the heavy water temperature in the calandria at a constant value.

Potential heavy water leak sources are kept to a minimum by using welded construction, seal welding, and bellows seals wherever practical. Where potential leaks sources do exist in the moderator system, the leak sources are connected to the heavy water collection system. The reliability of the moderator system is assured by appropriate component, Instrument and power supply redundancies.

The main moderator system pumps, valves and heat exchangers are in compact arrangement at approximately grade elevation to one side of the calandria vault. The pump suction lines and heat exchanger outlet lines are anchored to a rigid penetration seal where they pass through the calandria vault concrete to eliminate any possibility of loss of vault shielding water.

The moderator pump motors are connected to the high voltage Class III power supply. In addition, each pump has a pony motor capable of driving the pump at 25% speed and connected to the low voltage Class III power supply. In the event of a loss of Class IV power the power to the main motors is lost until the diesel generators can supply Class Ill power. The cooling water supply to the heat exchanger is also re-established after three minutes at a lower flow following a total failure of Class IV power. The rate of heat removal is sufficient to limit the increase of moderator temperature in the calandria to an acceptable value during a failure of Class IV power and subsequent reactor shutdown.

The heavy water in the calandria is maintained at relatively uniform temperature and circulated to eliminate hot spots. The circulation is promoted by pumping the heavy water from the bottom of the calandria and, after cooling, returning it through nozzle jets inside both sides of the calandria at the horizontal diameter.

Live-loaded double-packed stem seals are used on large valves in the moderator system to reduce leakage and maintenance. Bellows stem seals are used on small valves. All of the equipment in the moderator system is accessible for isolation and maintenance when the reactor is shutdown. Space for heat exchanger shell removal and other provisions for maintenance are features of the equipment arrangement.

II.2.3.4. Fuel handling system

The fuel handling system comprises equipment for storage of new fuel for fuel changing and for temporary storage of spent fuel. Reactor fuel is changed on a routine basis with the reactor operating at full power. Space and lifting facilities are provided for shipping spent fuel. The new fuel storage room, the fuelling machine decontamination and service rooms and the spent fuel storage rooms are located in the service building. The fuelling machines, which load and unload the fuel discharge equipment are normally operated remotely and automatically from the control room of the plant. Personnel are only required to enter the reactor building to load new fuel into the new fuel transfer mechanism and for maintenance of the fuel handling system components. These access areas are provided with full biological shielding.

Storage and handling facilities are provided to accommodate bulk storage of fuel in the service building, safe transfer of fuel to the reactor building and easy manual loading of new fuel bundles into the motorized new fuel ports. New fuel is received in packages in the new fuel room in the service building. This room can accommodate 9 month's fuel inventory and can store temporarily the fuel for the initial loading. When required, the packages with new fuel are transferred to the new fuel loading area in the reactor building. Here the bundles are identified, inspected and loaded manually into the magazines of the two fuel ports which penetrate into the fuelling machine maintenance locks. Mechanisms of the ports are motorized and can be controlled remotely. To load a fuelling machine with new fuel, the machine locks on to the port and normally accepts up to 10 bundles into a magazine within the head. The spent fuel discharge and storage equipment is sized for the accumulation of fuel over a period of 10 years at the average fuelling rate of the reactor, with provision for loading a shipping flask with fuel underwater.

After the minimum decay period established in 6 years, spent fuel elements are transferred to special dry storage silos, also located inside the nuclear power plant site. The fuel elements are introduced in stainless steel baskets, each of them containing up to 60 fuel elements vertically arranged in a circular grid; this operation is carried out under water. Later on, the baskets are sent to the transfer building, where the lid is weld. Finally they are introduced in a special container providing enough shield and containment (transfer 'flask") to be transported to the silo field where they are stored. Each silo contains 9 baskets.

Handling and storage space is also provided underwater for irradiated parts of the reactivity control mechanisms which may be removed from the reactor, and for shipping irradiated cobalt adjuster bundles from the station.

The spent fuel handling system consists of: discharge and transfer in the reactor building, reception, storage bays in the service building, and dry storage system. The transfer of spent fuel between buildings is under water through a transfer channel. The discharge and transfer operations are controlled remotely, while operations in the storage bays are carried out manually underwater using long tools, and aided by powered cranes and hoists. The equipment incorporates devices for canning failed fuel bundles and is arranged to reduce the radiation exposure of personnel when handling failed fuel to acceptably low levels. The discharge equipment comprises two valved spent fuel ports located above the water level, while the transfer equipment is located in a shielded room and extends down under the water and into a reception bay in the service building.

The on-power fuel changing equipment is located in the reactor building and consists of two identical, unshielded fuelling machines, which are operated remotely. The fuelling machines are normally stored In two fuelling machine maintenance locks and are suspended by tracks. Each set of tracks connects with a bridge at each face of the reactor. Powered shielding doors separate the maintenance locks from the reactor and, when closed, allow access to the fuelling machines while the reactor is at full power. While in the maintenance locks the fuelling machines can lock on to the new fuel port to accept new fuel, to the service port for maintenance or service, or on to the spent fuel port discharge spent fuel.

The fuel loading is based on the combined use of the two remotely controlled fuelling machines operating at each end of a fuel channel. New fuel bundles, from one fuelling machine, are inserted into a fuel channel in the same direction as the coolant flow and the displaced irradiated fuel bundles are received into the second fuelling machine at the other end of the fuel channel. Either machine can load or receive fuel. The direction of loading depends upon the direction of coolant flow in the fuel channel being fuelled, which alternates from channel to channel. The fuelling machine receive new fuel while connected to the new fuel port and discharge irradiated fuel while connected to the discharge port. The entire operation is directed from the control room.

II.2.3.5. Auxiliary systems

The main auxiliary systems of the Embalse Nuclear Power Plant are the heat transport auxiliary systems and the moderator auxiliary systems.

The heat transport auxiliary systems are the following:

- Heat transport system purification circuit: this system minimizes the accumulation of radioactive corrosion products in the circuit, controls the fission products concentration (iodine) released by defective fuel elements, and contributes to a proper control of the coolant pD.
- Gland seal system: it supplies filtered heavy water at high pressure to the heat transport pump glands.
- Shutdown cooling system: it cools the heat transport system from 170 °C down to 54 °C and holds the system at that temperature indefinitely.
- Heat transport pressure and inventory control system: it provides the pressure and inventory control for each heat transport circuit, and provides overpressure protection and a controlled degassing flow.
- Heat transport heavy water collection system: it collects leakage from mechanical components, and receives heavy water sampling flow, and heavy water drained from equipment prior to maintenance.
- Heat transport heavy water sampling system: it is used to obtain samples of heavy water from various points in the heat transport system. The samples are tested in the laboratory for pD, conductivity, chloride, tritium, lithium, dissolved gases, fission products and corrosion products.
- Steam and feedwater systems: they enable the live steam supply to the plant turbine generator, the control of the feeding water level and the vapour pressure in the steam generators, the steam release to the atmosphere under certain situations of the station, and an adequate protection against overpressures in the steam generator secondary circuit.

The moderator auxiliary systems are the following:

- Moderator purification system: it maintains the heavy water purity, thereby minimizing radiolysis which may cause excessive build-up of deuterium in the cover gas; minimizes corrosion of components and crud activation by removing impurities present in the heavy water and by controlling the pD; removes soluble poisons, boron and gadolinium, used for reactivity control in response to reactivity demands; removes the gadolinium, after initiation of the liquid injection shutdown system.
- Moderator cover gas system: it prevents the accumulation of gaseous deuterium and oxygen produced by water radiolysis of the moderator in the calandria. The system recombines deuterium and oxygen catalytically, generating heavy water. The cover gas used in the moderator system is helium, because it is an inert gas and is not activated by neutron irradiation.
- Liquid poison system: this system adds negative reactivity to the moderator to allow for excess reactivity in new fuel; adds negative reactivity to the moderator to allow for loss of xenon reactivity after a poison-out or long shutdown; provides a means of decreasing reactivity together with other reactivity control devices; provides a means to guarantee enough poison in the moderator to prevent criticality during shutdown.
- Moderator heavy water collection system: this system collects heavy water leakage from the moderator pump seals, from the interpacking space of the main moderator gate valves, and from the intergasket of the main moderator heat exchangers.

II.2.3.6. Control centre

The control centre is a clean air conditioned area comprising the main control roam and adjacent auxiliary control areas. The control centre is divided into four basic areas. The centre portion contains the main control panels, the operator's desk and the fuelling machine and fuel handling console. A control equipment area containing the bulk of the control and instrumentation equipment for the unit is located behind the main control panels. The plant computers are located in a room behind the switchyard and electrical services panel. Also located in the control centre are a shift supervisor's office, a technical office, a washroom and a work control area.

The control room contains a unit control panel, an electrical services panel, an operator's desk with two high speed line printers, and a fuel handling and fuelling machine control console. The control room instrumentation is based on the philosophy of having sufficient information displayed to allow the unit to be controlled safely from the control room. To achieve this goal, all indications and controls essential for operation (start-up, shutdown and normal) are located on the control room panels. Also located there are controls for any systems requiring attention within 15 minutes of an alarm occurrences. For system no requiring attention within 15 minutes, local control may be provided.

Most information is presented to the operator via the station computer system. However, sufficient conventional display, annunciation and recording of plant variables is included to allow the plant to be properly run in the shutdown condition with both computers out of service.

In case the control room becomes uninhabitable, enough display and control instrumentation is provided at a location remote from the control room (the secondary control area) to allow the plant to be shut dawn and maintained in a safe shutdown condition.

II.2.3.7. Instrumentation and control systems

The instrumentation in the CNE encompasses a variety of equipment, designed to perform a number of monitoring, control and display functions. Nuclear instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor local core behaviour. Conventional instrumentation provides signal for control and display of other plant variables. Central to the instrumentation and control system is a dual digital computer system. The plant is automated to require a minimum of operator actions during all phases of operation. All major control loops use the two computers as direct digital controllers, giving a redundant and highly reliable system which is powerful and flexible. Conventional analog control instrumentation is used on smaller local loops.

Instrumentation and control systems are designed to a large variety of detailed requirements, depending on their function, importance and physical environment. However, all the systems are designed according to the following general criteria:

- The maximum practical amount of automatic control is incorporated in the design, to allow CNE to be operated safely with a minimum staff, and to leave operating staff free for high level monitoring of overall plant status.
- Adequate redundant information is designed to be available to the operator at all times, in order to allow him to assess the status of the plant quickly, and to intervene with manual actions if necessary.
- Equipment is designed for a minimum of regular maintenance. Any necessary maintenance operations are kept as simple and speedy as possible.
- The instrumentation and control systems are designed for a high reliability and availability, to maximize both plant availability and safety. This reliability is achieved through a combination of component selection and design and through redundancy.
- The control systems are designed to make CNE tolerant to expected and unexpected transients in order to prevent unnecessary plant outages.
- Where possible, the control system are designed to prevent or minimize damage to plant equipment.
- The control systems are designed to minimize the number of unnecessary reactor trips by strong control action. A design objective is to make the intervention of the shutdown systems unnecessary (although not necessarily avoidable) in all cases except real accidents in which public safety is threatened.
- The instrumentation and control design complies with the applicable standards.

II.2.3.8. Electric power system

The Embalse Nuclear Power Plant has two physical independent grid connections ***(Figure II.2-8)***. One of them is the 500 kV grid and the other is the 132 kW grid (offsite power system). The generator supplies power to the 500 kV grid through the main output transformer. During normal station operation, the auxiliary service is supplied through the unit service transformers, which are supplied by the generator output. On start-up, the station auxiliary service power supply is provided by the system service transformers, which are supplied from the electrical grid via the switchyard at 132 kV.

A load switch is provided on the 22 kV generator bus bars. The purpose of the load switch is to make possible station start-up having the auxiliary services supplied from the 500/22 kV main transformer and from the 22/6.6 kV transformers as alternative to the 132/6.6 kV transformers. In this eventuality the parallel between the 500 kV grid and the generator is achieved using the load switch.

To provide electrical power with higher than usual reliability to the Class IV and Class III loads, an automatic transfer scheme is incorporated which ensures continuity of supply in the event of a failure of the unit or a failure of the system supply. Standby diesel generators and station batteries are provided.

The electrical system of CNE is similar to that found in conventional large thermal stations, with modifications introduced to satisfy the increased redundancy requirements. This results in a more selective bus arrangement and more standby and redundant equipment.

The station service system is designed to meet the following major design criteria from safety and reliability point of view:

- Following a fault resulting in the severance of the unit from the grid, the unit must be able to supply its own station services.
- Dual bus or better reliability must be provided.
- The system must be stable under fault conditions.
- The design must meet the requirements of all classes of power and lend itself to automatic and emergency transfer schemes.
- Simplicity and economy are to be maintained.

The safety and reliability requirements of the CNE onsite power system are realized by applying two main concepts:

- The subdivision of power according to reliability of supply (classes of power and emergency power supplies to safety related systems).
- The odd and even supply concept which relates to redundancy of supplies and loads.

As regards the subdivision of power according to reliability of supply, it should be mentioned that:

1. The CNE service system buses are classified in order of their four levels of reliability to provide power during the routine operating states of the plant. The lowest number classified buses are the most reliable. These are as follows:

   - Class IV power supply: normal ac supplies to auxiliaries which can tolerate long duration interruptions without affecting personnel or equipment safety. Complete loss of Class IV power initiates a reactor shutdown. Class IV power is the normal source of power to Class I, II, and III systems. The voltages for the supply Class IV are as follows: 6.6 kV ac – 380 V ac - 50Hz.
   - Class III power supply: ac supplies to essential auxiliaries which can tolerate the short interruption required to start up and load the on-site standby generators, after the interruption of the normal (Class IV) supply sources. These essential auxiliaries are necessary for an orderly safe shutdown of the reactor. The voltages for the supply Class III are as follows. 6.6 kV ac – 380 V ac - 50 Hz.
   - Class II power supply: ac supplies for safety related and other essential loads. Power is normally provided through dc/ac inverter systems from the Class I dc buses. In the event of inverter system trouble, alternative power is supplied automatically from the Class Ill buses via appropriate transformers. Upon interruption of the normal power source (Class Ill) to the Class I bus the on-site battery supplies power without break until Class III is restored and hence the supply is termed "uninterruptable". The voltages for the supply Class II are as follows: 380 V ac – 220 V ac -- 50 Hz.
   - Class I power supply: dc supplies for safety related and other essential loads. Power is provided from on-site batteries when the normal power source (Class III via rectifier) is interrupted. The transfer of power supply from one source to another is without break and hence the supply is

termed "uninterruptable". The on-site batteries are continuously charged from Class III sources. The voltages for the supply Class I are as follows: 220 V dc - 48 V dc.

2. The standby power for the Class Ill loads is supplied by four diesel generator sets. These are housed in four separate rooms with fire resistant walls. Two diesel generators together are sized to supply the total shutdown of the unit with the exception of Class IV loads. The Class Ill shutdown loads are duplicated, one complete system being fed from two diesel generator sets. On loss of Class IV power the four diesel generators are required to start automatically. When they come up to speed and voltage, an automatic sequencing system will connect all necessary safety-related loads in a few seconds.
3. CNE emergency power supply system is provided as an independent backup electrical supply for certain safety related loads. It supplies power to facilitate safe shutdown of the reactor and decay heat removal in the event that the Class I, II, III, and IV power systems are unavailable or the main control room is uninhabitable due to a design basis earthquake. The system is seismically qualified and is also able to supply power to emergency core cooling valves to ensure that the emergency water supply system can supply makeup to the heat transport system after an earthquake. The voltages supplied by the emergency power supply system are as follows: 380 V ac-220 V ac-50 Hz and 48 V dc.

As regards the basic aspects of the odd and even supply concept, it should be mentioned that:

- The distribution systems for all classes of power at all voltage levels are divided into odd and even buses so that the dual bus, or better, reliability is provided.
- Loads and redundant auxiliaries are connected wherever practical such that half of any process is supplied from an odd bus, and the other half from an even bus.
- Auxiliaries supplied at a lower voltage than the associated primary element are connected to an odd or even bus to match the source for the primary element.
- The odd and even concept is also applied to the cable tray system, junction boxes, etc. in order to maintain physical separation between the odd and even systems.

II.2.3.9. Safety systems

Safety related systems are incorporated in the plant design to perform the following functions:

- Shutdown the reactor and maintain it shut down.
- Remove decay heat and thus prevent subsequent process failures which might lead to accidental releases of radioactivity to the public.
- Supply necessary information for post accident monitoring to permit the operator to assess the state of the nuclear steam supply system.
- Maintain a barrier to limit the release of radioactive material to the environment.

The systems included under the general term "safety related systems" are classified as special safety systems and safety support systems.

The special safety systems are incorporated in the plant to limit radioactive releases to the public for two classes of events: the single failure of a process system, and the single failure of a process system combined with the coincident unavailability of one of the special safety systems (a dual failure).

The CNE contains the following special safety systems:

- Shutdown system Nº 1 (shutoff units).
- Shutdown system Nº 2 (liquid poison injection).
- Containment system.
- Emergency core cooling system.

These systems are independent in design and operation and free from operational connection with any of the process systems, including the reactor regulating system, to the greatest possible extent.

The purpose of shutdown system Nº 1 is to rapidly and automatically terminate reactor operation under emergency conditions. Twenty-eight vertical shutoff units are provided, each comprising a stainless steel sheathed cadmium absorber, vertical guide tube, and a drive mechanism. The system shuts down the reactor by releasing the cadmium absorber elements of the shutoff units, introducing negative reactivity. This release is initiated when any two of the three independent trip channels are actuated. When a reactor trip occurs, the reactivity control units of the regulating system automatically take a safe attitude.

Typically, the liquid zone control compartments are flooded, the control absorbers are dropped, and the adjuster drives are shut off (the adjusters remain inserted or as is).

The purpose of shutdown system Nº 2 is to rapidly and automatically terminate reactor operation independently of shutdown system Nº 1. The system trips the reactor by injecting liquid poison into the bulk moderator when any two of the three independent trip channels are actuated. The system comprises injection nozzles, thimbles, bellows assemblies, gadolinium pressure vessels (poison tanks), a helium supply tank, a poison mixing tank, valves and piping.

The containment system is an envelope around the "nuclear" components of the heat transport system where failure of these components could result in the release of a significant amount of radioactivity to the public. Because of the large amount of energy stored in the heat transport system, the envelope must withstand a pressure rise. The criterion for determining the effectiveness of the envelope is the integrated leak rate for the period of the pressure excursion. To meet the design leakage requirements two approaches are taken. The first involves the detailed design of the envelope to minimize the leak rate. The envelope comprises a primary containment, and systems to filter and monitor the gas removed from the primary containment after a loss-of-coolant accident following dousing. The second approach involves the addition of a system that will absorb the energy released to the envelope, thus reducing the peak pressure and the duration of the pressure excursion. This energy absorbing system is composed of a source of dousing water, spray headers and initiating valves, and building air coolers.

The emergency core cooling system has three stages of operation: high, medium and low pressure. System operation is triggered, on a loss of coolant accident (LOCA), when the heat transport system pressure drops to 55.25 kg/cm^2 and a circuit isolation system (independent of emergency core cooling system logic) closes the applicable valves to isolate the ruptured circuit.

The safety support systems provide reliable services, such as power and water, to the special safety systems, but may also perform other normal process functions in addition to their safety support roles. Because of the reliance on these systems for both normal plant operation and continuing operation of the special safety systems, special measures are taken in their design to assure reliability.

Two of the CNE safety support systems are the emergency water supply system and the emergency power supply system.

The emergency water supply system ensures that there is always sufficient water available to establish an adequate heat sink for decay heat removal when the normal source of such water is not available. The emergency power supply system is designed to act as an alternative source of electrical power for certain safety related loads when the normal source of supply is unavailable; this system was discussed in section II.2.3.8.

II.2.3.10. Technical data of Embalse Nuclear Power Plant

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	CANDU-PHW horizontal pressure tube. Model: CANDU 6
Net nominal electric power	600 MWe
Bulk nominal electric power	648 MWe
Authorized thermal power	2015 MWt

Reactor Core Data	
Type of fuel	Natural uranium
Shape of fuel bundle assembly	37 - rod cluster
Length of fuel bundle assembly	495 mm
Number of fuel channels	380
Cladding material	Zircaloy 4
Fuel bundles per channel	12
Refuelling	On load
Coolant and moderator	Heavy water

Primary Heat Transport System Data	
Pressure in the reactor inlet header	11.24 MPa
Pressure in the reactor outlet header	9.99 MPa
Temperature in the reactor inlet header	268 °C
Temperature in the reactor outlet header	310 °C
Primary coolant flow	32.750 t/h
Heavy water concentration	More than 99.75% (weight)

Turboset Data	
Stages	1 high pressure; 3 low pressure
Speed outlet	1500 rpm
Steam pressure	46.2 kg/cm^2
Steam flow	3.366 t/h
Condenser coolant flow	163.800 m^3/h
Generator type	Direct coupled, three-phase, four poles, hydrogen/water cooled
Generator power factor	0.85
Generator voltage output	22 kV
Generator frequency	50 Hz

SITE LOCATION

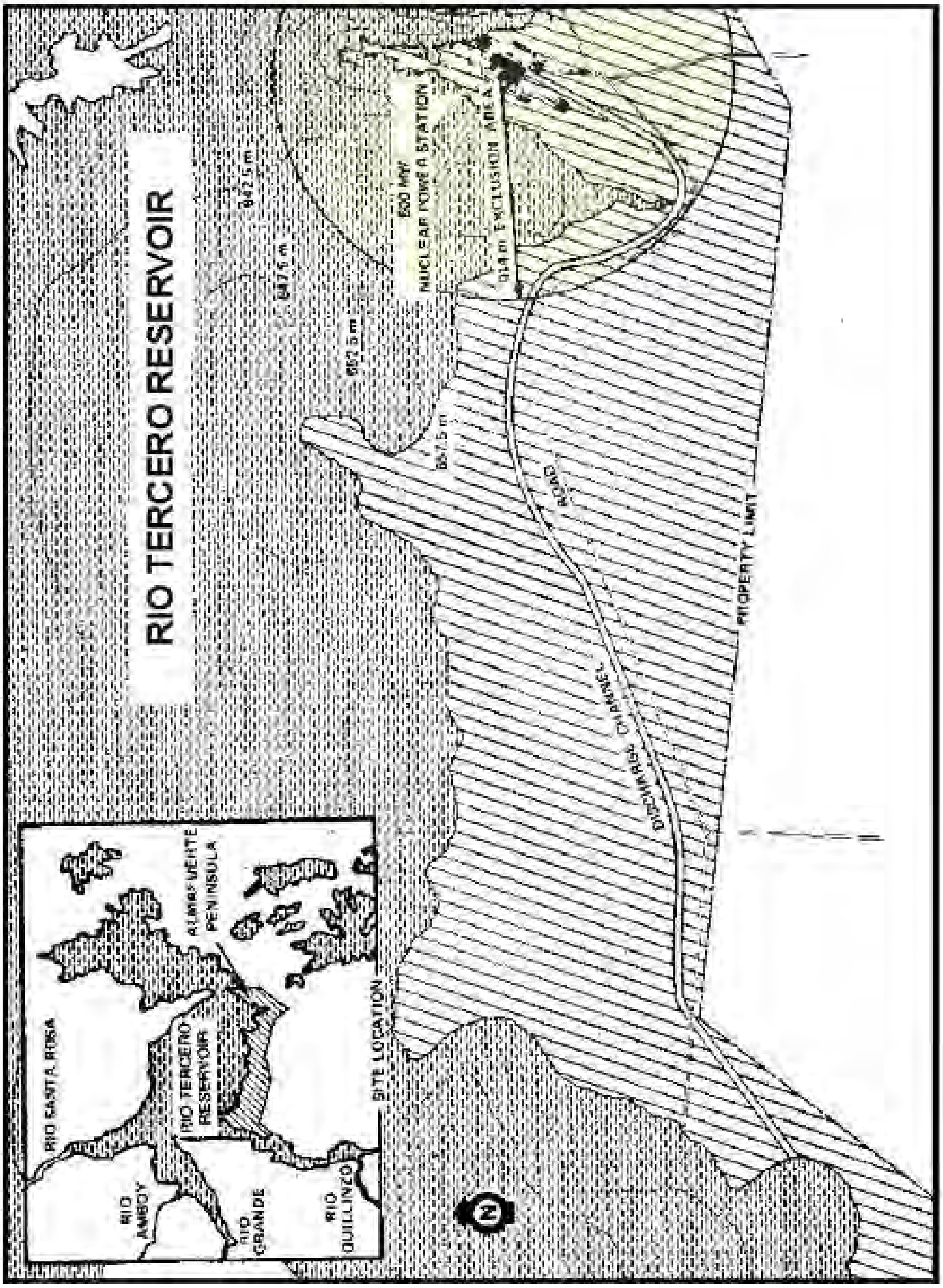


Figure II.2-1 - Embalse Nuclear Power Plant - Site Location

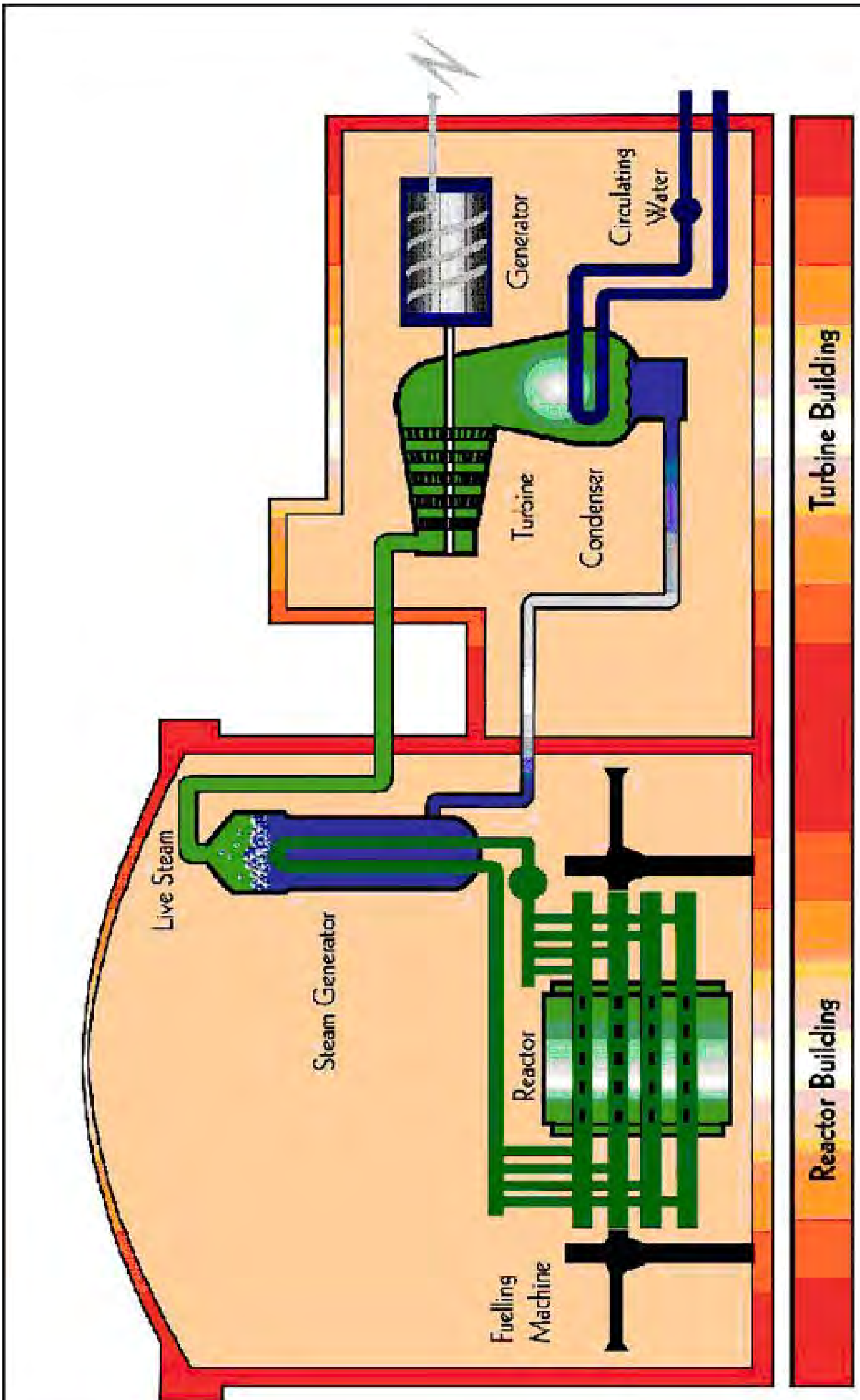
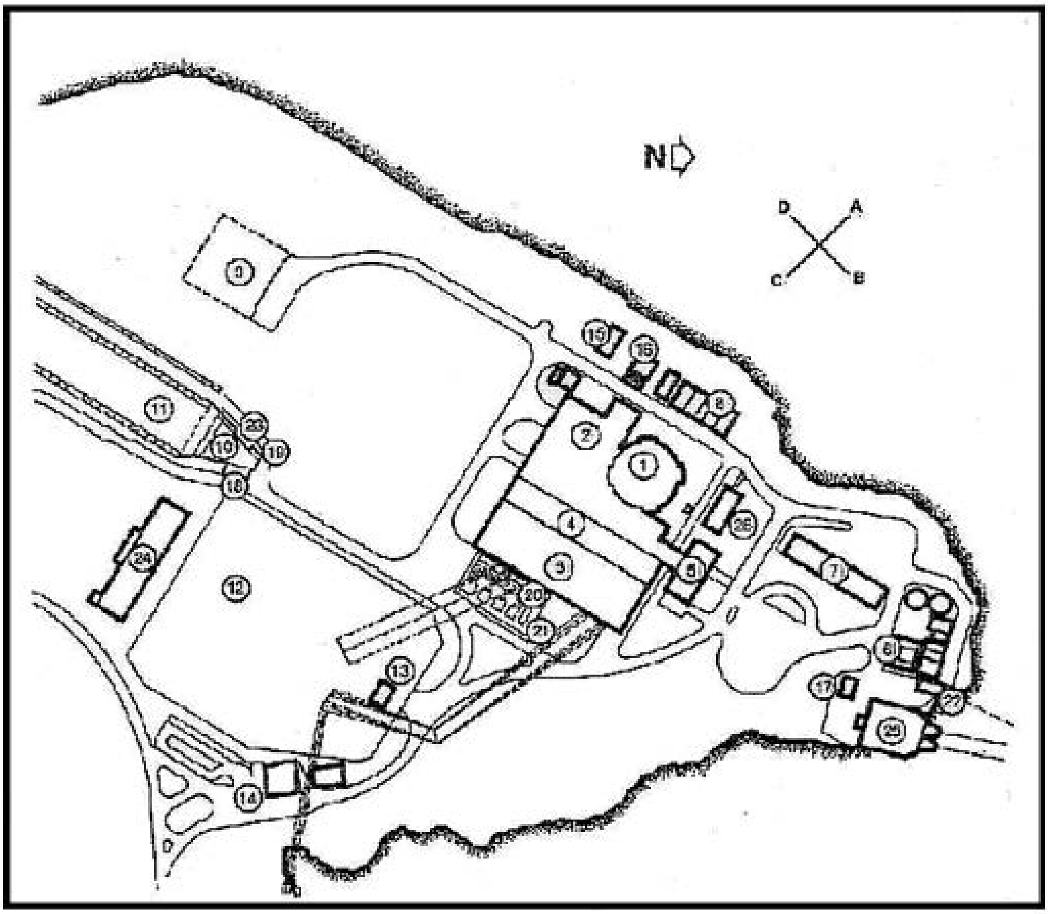


Figure II.2-2 - Embalse Nuclear Power Plant - Simplified Flow Diagram

SITE PLAN



1 REACTOR BUILDING
2 SERVICE BUILDING
3 TURBINE BUILDING
4 AUXILIARY BAY
5 ADMINISTRATION BUILDING
6 WATER TREATMENT PLANT
7 GARAGE
8 STANDBY GENERATOR
9 SOLID WASTE STORAGE
10 DISCHARGE WEIR
11 DISCHARGE CHANNEL
12 SWITCHYARD
13 COMMAND STATION
14 MAIN GATE
15 AUXILIARY BOILER
16 FUEL TANKS
17 SELF CLEANING FILTER
18 FIRE FIGHTING PUMPS
19 DRAIN PUMPS
20 TRANSFORMER AREA
21 HYDROGEN STORAGE
22 EMERGENCY WATER SYSTEM PUMP HOUSE
23 PROCESS WATER POOL
24 GENERAL WAREHOUSE
25 PUMPHOUSE
26 HIGH PRESSURE EMERGENCY CORE COOLING BUILDING

Figure II.2-3 - Embalse Nuclear Power Plant - Site Plan

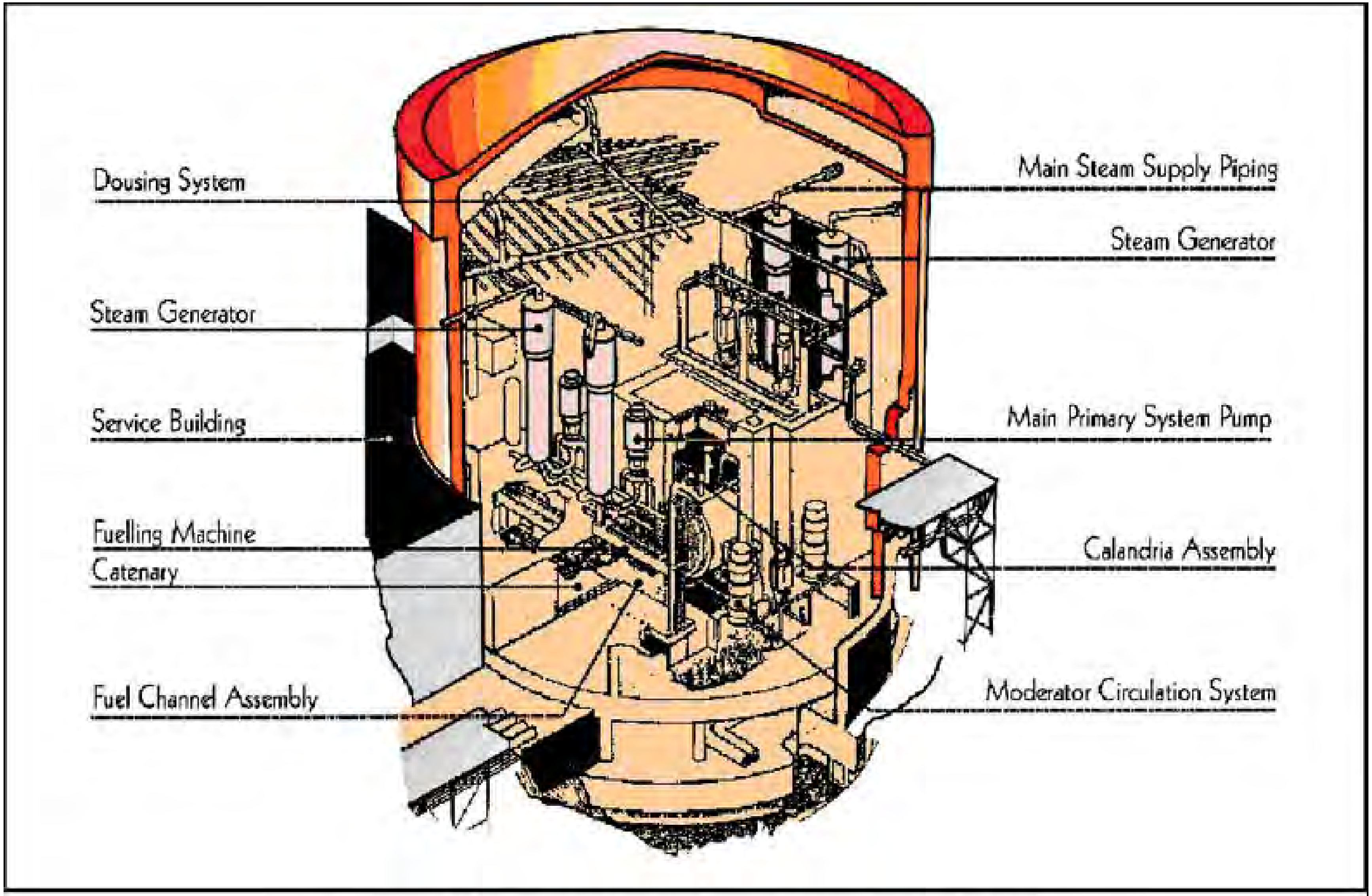
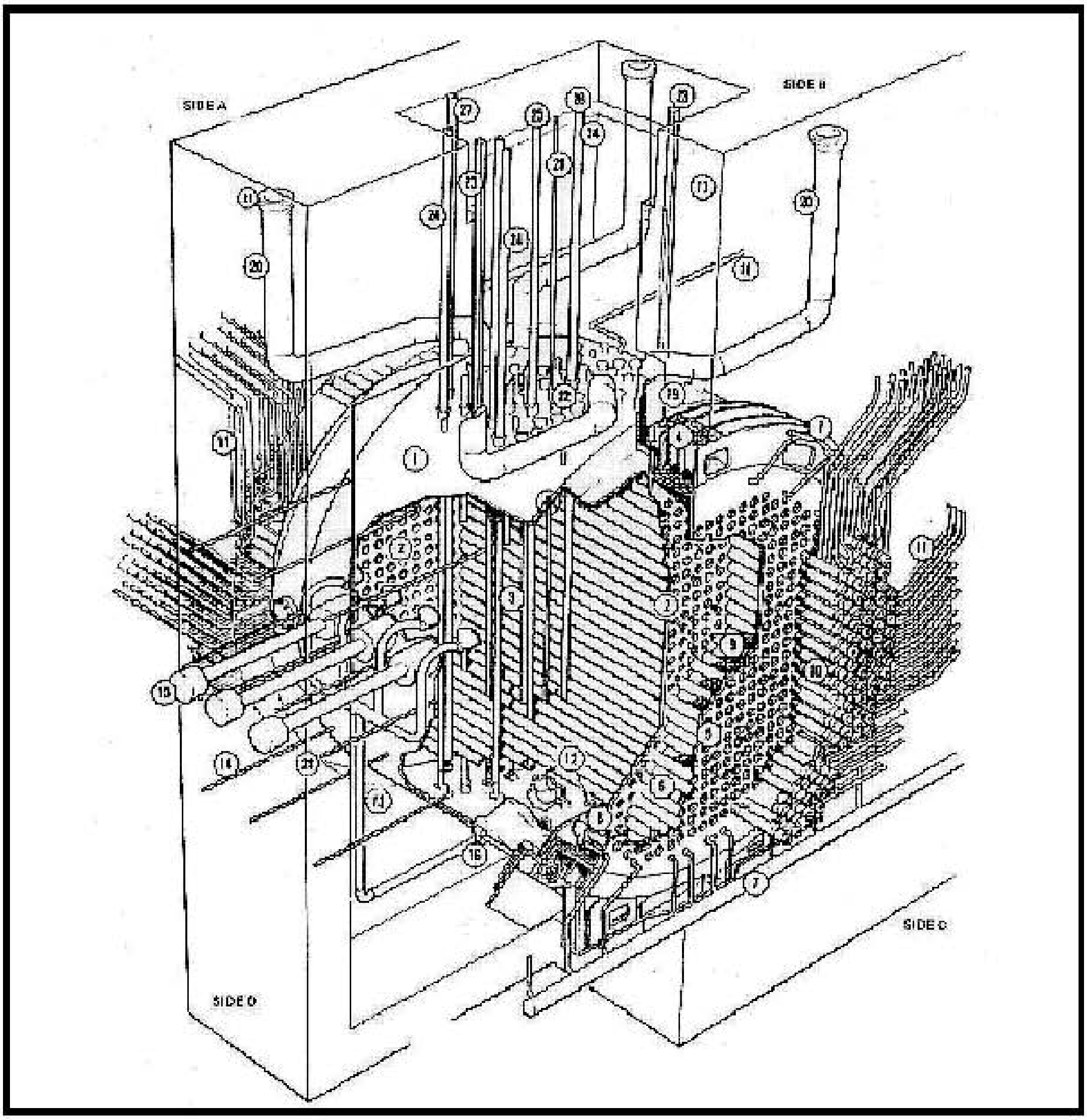


Figure II.2-4 - Embalse Nuclear Power Plant - Reactor Building

REACTOR ASSEMBLY



1. CALANDRIA
2. CALANDRIA-SIDE TUBESHEET
3. CALANDRIA TUBES
4. EMBEDMENT RING
5. FUELLING MACHINE - SIDE TUBESHEET
6. END SHIELD LATTICE TUBES
7. END SHIELD COOLING PIPES
8. INLET-OUTLET STRAINER
9. STEEL BALL SHIELDING
10. END FITTINGS
11. FEEDER PIPES
12. MODERATOR OUTLET
13. MODERATOR INLET
14. HORIZONTAL FLUX DETECTOR UNIT
15. ION CHAMBER
16. EARTHQUAKE RESTRAINT
17. CALANDRIA VAULT WALL
18. MODERATOR EXPANSION TO HEAD TANK
19. CURTAIN SHIELDING SLABS
20. PRESSURE RELIEF PIPES
21. RUPTURE DISC
22. REACTIVITY CONTROL UNIT NOZZLES
23. VIEWING PORT
24. SHUTOFF UNIT
25. ADJUSTER UNIT
26. CONTROLABSORBER UNIT
27. LIQUID ZONE CONTROL UNIT
28. VERTICAL FLUX DETECTOR UNIT
29. LIQUID INJECTION SHUTDOWN NOZZLE

Figure II.2-5 - Embalse Nuclear Power Plant - Reactor Assembly

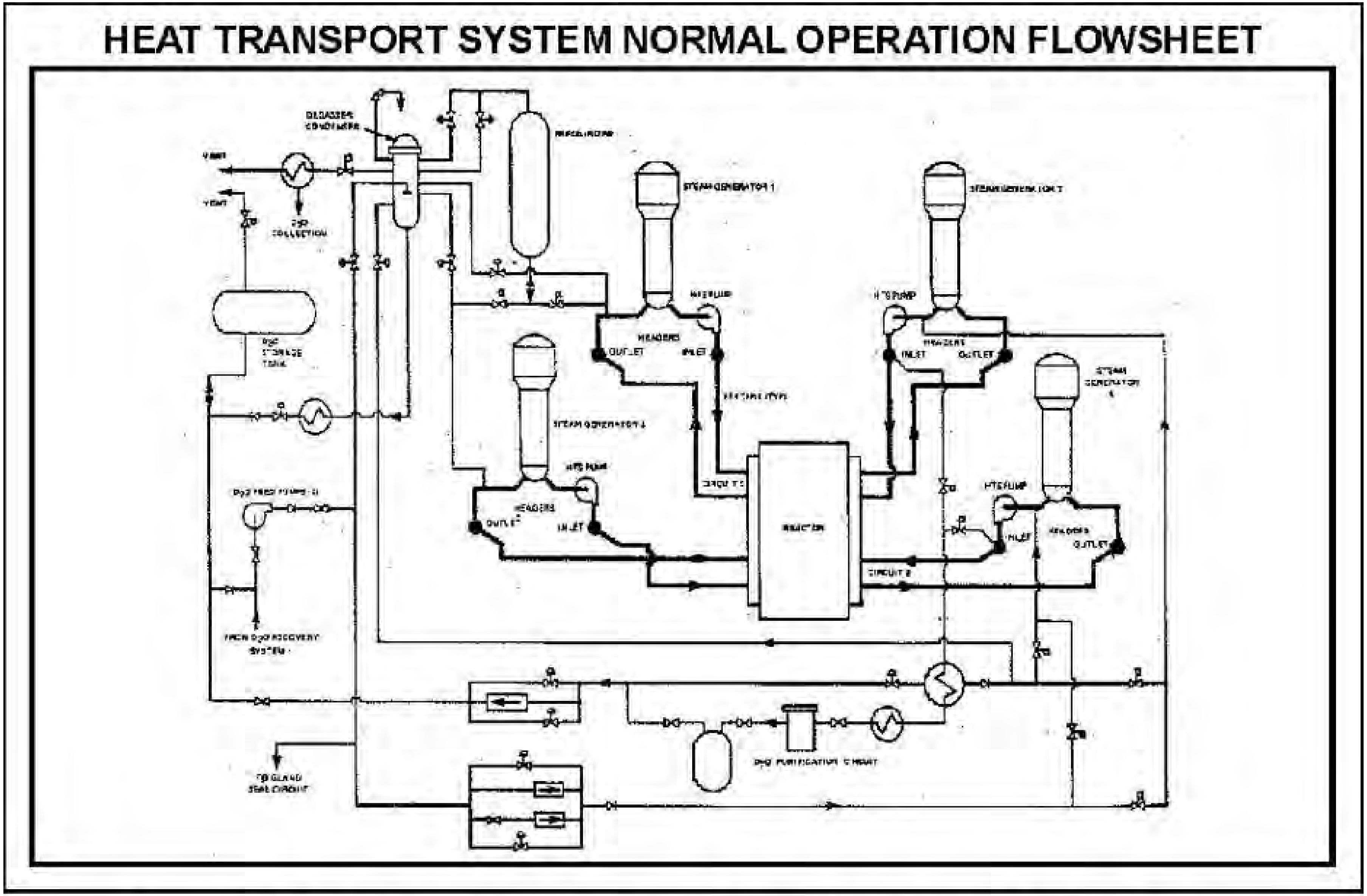


Figure II.2-6 - Embalse Nuclear Power Plant - Heat Transport System Normal Operation Flowsheet

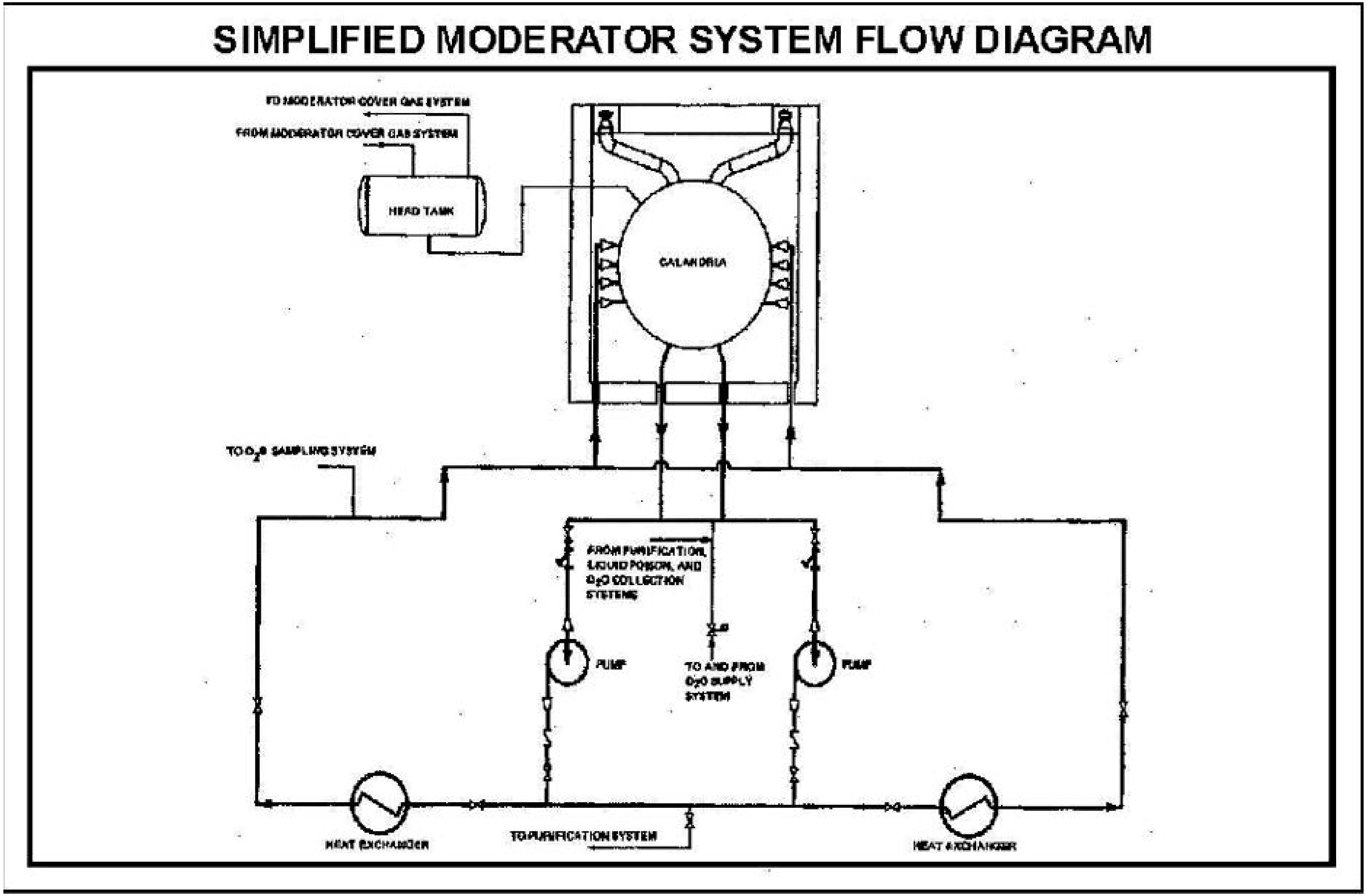


Figure II.2-7 - Embalse Nuclear Power Plant - Simplified Moderator System Flow Diagram

SIMPLIFIED SINGLE LINE ELECTRICAL DISTRIBUTION DIAGRAM

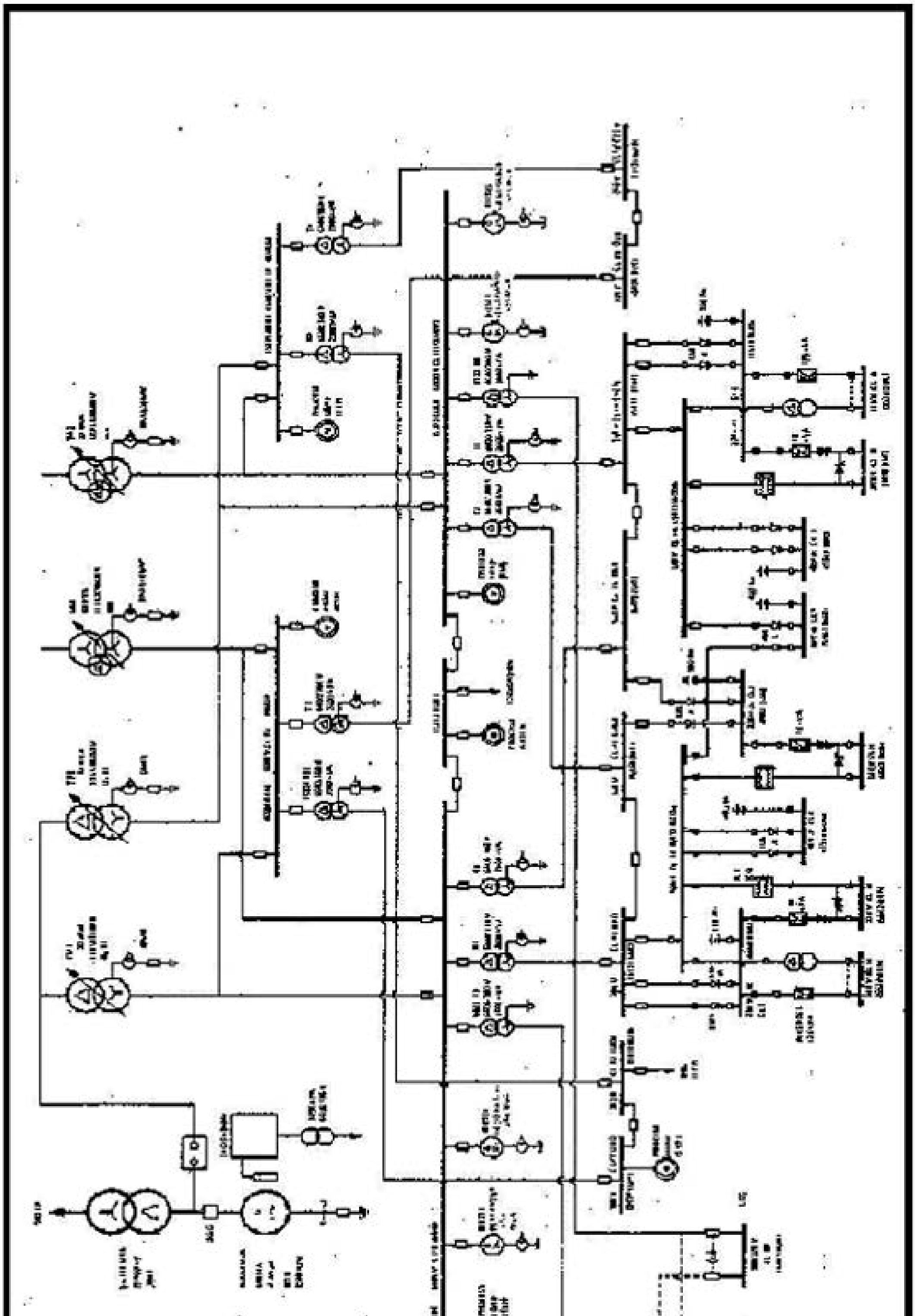


Figure II.2-8 - Embalse Nuclear Power Plant - Simplified Single Line Electrical Distribution Diagram

ANNEX III

PRINCIPAL TECHNICAL CHARACTERISTICS OF ATUCHA II NUCLEAR POWER PLANT

III.1. INTRODUCTION

The CNA II - a PHWR type of 700 MWe - was designed by the Siemens, with the participation of ENACE as architect engineer at the time the project began. CNA II construction license was issued in July 14, 1981. The construction and commissioning process is, since 1994, under the direct responsibility of the State enterprise NA-SA, as the Licensee.

The 700 MWe PHWR makes full use of KWU's experience in the light and heavy water reactor fields and the operating experience of CNA I, a station that has shown excellent operating performance with high rates of availability and thereby furnished proof of its full operating reliability.

The pressure vessel type PHWR implemented in the 700 MWe PHWR is derived from CNA I and the 1300 MWe KWU standard PWR. Thus, the heavy water specific components such as moderator pumps, moderator coolers, moderator valves, coolant channel closures, refueling system, heavy water upgrading columns, etc. used are of almost identical design with CNA I; all other components of the nuclear and conventional part of the plant, almost all main and auxiliary systems and the layout of the buildings are derived from the KWU standard PWR design.

CNA II plant is located 110 km north-west of Buenos Aires city, on the southern bank of the Río Paraná de las Palmas, in 9 km distance of the Community of Lima, Zárate County, Province of Buenos Aires. The plant is located adjacent to the east side of CNA I ***(Figure III-1)***.

The station contains a reactor of the pressure vessel type, fuelled with natural uranium (like CNA I, it could also be fuelled with slightly enriched uranium); it is heavy water moderated and cooled. The moderator heat is used for preheating of steam generator feedwater and its temperature for reactivity control of the reactor. The reactor is on-load refueling with a single refueling machine arranged on top of the reactor pressure vessel.

Four redundant trains are installed for every safety system, thus enabling repair work on one train during plant operation. In every conceivable accident condition, the reactor plant can be kept "hot subcritical" or cooled down with the help of the high pressure residual heat removal system or with the emergency feeding of water on the secondary side of the steam generators as in PWR technology.

Figure III-2 shows schematically the main systems of CNA II.

III.2. OVERALL PLANT LAYOUT

The overall layout ***(Figure III-3)*** and arrangement of the CNA II on the site is governed by the following basic considerations:

- Clear separation of the nuclear and conventional systems.
- Clear energy flow paths.
- Short piping and cable mans.
- Good transport conditions and access for construction, installation and operation.

Buildings and structure arrangement of the CNA II are shown in Figure III-3. As it can be seen from the site plan, the main buildings and structures of the plant are:

- Reactor building, incorporating containment structure and annulus.
- Reactor auxiliary building with heavy water enrichment tower and vent stack.
- Fuel store building, with storage areas for new and spent fuel assemblies.
- Switchgear building, including the plant control room.

- Turbine building.
- Switchgear building.
- Fuel storage building.
- Emergency power and chilled water supply building.
- Main steam and feedwater valve compartment.

CNA II has a clear physical separation between the nuclear and conventional sections of the plant. The reactor building, along with the annulus, reactor auxiliary building and the fuel storage building, constitutes the “controlled area” in which all systems assigned to the nuclear section are installed. In this way the radioactivity which arises is limited to defined regions. There is only one controlled access to the "controlled area".

All pressure retaining components of the nuclear steam supply system such as the reactor, the reactor coolant system, the moderator system and associated equipment are arranged inside the reactor building, which is enclosed by the inner spherical steel containment and the outer concrete shield. The containment structure is designed for the maximum pressure associated with the worst event which has to be taken into account.

A special ventilation system for the annulus ensures that even under accident conditions small radioactive leakages from the containment are retained try charcoal filters, thus preventing any radiation hazards to the environment. The systems necessary for on-load refueling are also housed in the containment structure.

In the lower part of the annulus between the containment sphere and the concrete shield various auxiliary and ancillary systems are accommodated, such as: residual heat removal system, safety injection system, heavy water storage system and components of the reactor cooling system.

The reactor auxiliary building adjoins the reactor building, and surrounds a part of it, thus allowing short connections to the equipment located in the reactor building annulus.

On the upper floors of the building there are active and inactive sanitary rooms, the laundry with ancillary rooms, the controlled access area ventilation system, the radiochemistry laboratory, the areas for radiation protection and the respiration apparatus room. From this area of the reactor auxiliary building there is an access to the reactor sphere via a personnel airlock.

The lower floors accommodate different auxiliary systems, such as: volume control system, heavy water purification and degassing system, heavy water treatment and enrichment system, boric acid and chemical control system and the gaseous, liquid and solid waste processing systems.

The fuel storage building is linked with the reactor building by the fuel transfer system. Personnel access is possible from the reactor auxiliary building.

Inside the building, there are four fuel storage pools, a manipulating pool, a small pool for the spent fuel shipping cask, a new fuel store and the necessary auxiliary equipment. The spent fuel assemblies are transferred from the reactor to the fuel storage pool with the aid of the fuel transport system, consisting of: refuelling machine, tilter, transfer tube, tilting device and manipulating bridge. The fresh fuel assemblies are supplied to the reactor in the reverse way.

The switchgear building has nine floors. They are used as follows:

- Cable ducts.
- Cable basement.
- High voltage switchgear.
- Cable race below D.C. systems.
- Battery, rectifiers, D.C. distribution boards.
- Cable race, instrumentation and control.
- Cabinets for instrumentation and control.
- Ventilation ducts, cable race below control room.
- Main control room, computer room, ventilation systems.
- Vent air system.

Access to the switchgear building is from the staff facilities and office building via a personnel passageway. Access to the reactor auxiliary building is at the same level. The personnel passageway between staff facilities and office building/ turbine building allows passage between switchgear building and turbine building.

The off-site power transformer is located in front of the longitudinal side of the building facing the turbine building.

The turbine building is located adjacent to the reactor building with the turbine axis pointing in the direction of the reactor building. This gives maximum protection of the reactor building should the highly unlikely event of a turbine rotor burst occurs.

The building is of a two bay design. The main bay houses the turbine generator set and the feedwater heating equipment. The lower ancillary bay houses the feedwater tank, deaerators and feedwater pumps and other equipment associated with the water/steam cycle. All these compartments are free of radioactive media.

The main steam lines coming from the reactor building enter the turbine building along the shortest route leading to the area of the high pressure casing of the turbine, where the main steam flows through the steam strainers into the high pressure turbine, Vertical moisture separators are installed on both sides of the high pressure casing.

The basement of the turbine building is used mainly to accommodate pipes and cables. The heat exchangers for the low pressure feed heater drains, the closed circuit cooling water system and the associated pumps are also installed in the basement.

The generator busbars are routed from the generator to the generator transformers installed against the wall outside the turbine building, and to the high voltage station service transformers.

The circulating water pipes enter and leave the turbine building on the same side.

The emergency power and chilled water supply building has two service floors. The building is further subdivided into four equal sections of similar construction which house redundant systems and equipment.

The diesel fuel storage tanks, pumps, secured component cooling heat exchangers, air recirculation system and the cable and pipe spreading rooms are installed on the lower floor.

The emergency power generators with their switchgear and the water-chilling units are installed on the upper floor. The diesel fuel day tanks and the start-up air supply system for each of the emergency power generators are installed on a gallery structure.

III.3. CNA II MAIN SYSTEMS

III.3.1. Reactor

The reactor ***(Figure III-4)*** is of the pressure vessel type, natural or slightly enriched uranium fuelled and heavy water cooled and moderated. The total thermal power is 2160 MW.

The reactor core is approximately cylindrical in shape and consists of 451 natural uranium fuel assemblies located in the same number of coolant channels. Each fuel assembly consists of 37 fuel rods arranged in three concentric circles, the rod supporting plate, the spacers for lining up the fuel rods, and the linkage with a coupling for connection to the filler body. Each fuel rod consists of a stack of uranium dioxide pellets enclosed by a thin walled zircaloy 4 canning tube, which is both gas and pressure tight. Each fuel assembly, together with the filler body and the closure plug, forms the fuel bundle column. The coolant channels are arranged vertically in a triangular lattice within the moderator tank. The fuel bundle columns can be removed from the coolant channels during reactor operation by the refueling machine. The filler bodies serve to reduce the volume of the coolant in the reactor coolant system.

The heat generated in the fuel assemblies is transferred to the reactor coolant, which flows through the coolant channels and transports the heat to the steam generators.

The coolant channels are surrounded by the moderator, which is enclosed in the moderator tank. For reactivity reasons, the moderator is maintained at a lower temperature than the reactor coolant. This is accomplished by the moderator system, which extracts the moderator from the core, cools it down in the moderator coolers, and feeds it back into the core. The heat removed from the moderator is used for pre-heating the feed-water. This is one of the reasons for the high net efficiency (approx. 32%) of the NPP.

The reactor coolant system and the moderator system are connected by the pressure equalization openings of the moderator tank closure head. Therefore, the pressure differences in the core are comparatively small, which results in thin walls for the reactor pressure vessel internals. This allows a very high burn-up to be attained. Furthermore, the connection between the reactor coolant system and the moderator system permits the use of common auxiliary systems to maintain the necessary water quality. The number of auxiliary systems can therefore be reduced to a minimum.

For control of the reactivity, and thus of the power output of the reactor, various methods are applied. The reactor contains nine "blacks" (absorbers made of hafnium) and nine "grey" (steel) control elements arranged in 3 groups. The control elements are used to control the reactivity and the power distribution, to compensate the build-up of xenon poisoning following a reactor power reduction, to provide damping of azimuthal xenon oscillations, and to shut down the reactor. The reactivity value of all control elements is sufficient to shut the reactor.

In addition to the control, elements, reactivity control is provided by the boric acid dosing system. The injection or extraction of boric acid serves to compensate slow reactivity changes due to the burnup during the first period of operation and to maintain the reactor in a safe subcritical condition at zero power. Extraction of the boric acid is performed by anion exchangers.

Additionally, a boron injection system, as a second independent shutdown system is provided, which injects boric acid into the moderator.

The reactivity can, in addition to these reactivity control systems, also be controlled by varying the moderator temperature within a certain range, which is advantageous for some operating modes.

The reactor pressure vessel constitutes the pressure boundary of the reactor core and encloses the core components and the reactor pressure vessel internals. The RPV consists of the lower part, the closure head and the studs and nuts which connect both sections. The connection is made leak-tight by means of a welded lip seal.

The lower part of the RPV consists of the hemispherical bottom section, two shell courses and a shell flange which carries the coolant inlet and outlet nozzles and the support pads located between them. The reactor coolant inlet and outlet nozzles are arranged on one plane; there are no penetrations or pipe connections below this plane. The reactor core is housed below the plane of the inlet and outlet nozzles.

The closure head consists of a flange and a dome plate connected by a circumferential weld. The closure head dome carries the nozzles for coolant channels, moderator pipes, and control element drives and for in-core instrumentation. The nozzles are screwed into holes in the closure head dome and sealed by an overlay weld.

Most of the RPV internals form the structure of the reactor core. The moderator tank accommodates all core components, separates the moderator from the coolant and, in conjunction with the reactor pressure vessel, forms the annulus for, the in-flowing coolant. The bottom of the moderator tank serves as the lower fixing level for the coolant channels and the control element guide tubes. The moderator tank shell serves as thermal shielding.

The moderator tank closure head form the upper plenum for the reactor coolant leaving the coolant channel slots. The closure head and the moderator tank jointly form a unit which keeps the coolant channels and the control element guide tubes in position firmly and without displacement during all operating modes, as well as during postulated accidents. The moderator tank and its closure head are suspended from the flange of the reactor pressure vessel and are held tightly in position by the pressure vessel closure head.

The coolant channels consist of vertically arranged tubes which contain the fuel bundle columns, direct the reactor coolant flow and separate the reactor coolant from the surrounding moderator.

The reactor coolant flows inside the coolant channels in an upward direction. After passim through the fuel assembly, it leaves the coolant channel through slots and enters the upper plenum formed by the moderator tank closure head.

The coolant channel closure head, together with the coolant closure plug, forms the pressure-tight cap of the coolant channel. It can be opened by the refueling machine during reactor operation in order to exchange the fuel bundle column located inside the coolant channel.

The moderator piping serves for supply, distribution and extraction of the moderator inside the moderator tank. The moderator piping essentially encompasses four down-comers, the sparger ring on the moderator tank bottom, and the suction boxes with nozzles in the moderator tank closure head.

The moderator flows downwards through the down-comers to the sparger ring, where it is distributed at the moderator tank bottom. After rising and heat-up in the moderator tank, the moderator flows to the suction boxes and leaves the moderator tank through two nozzles.

The filler pieces are provided in the reactor pressure vessel in order to displace the heavy water and thus reduce the heavy water inventory rewired. The upper filler pieces are adapted to fit the reactor pressure vessel closure head. The lower filler pieces are divided into several interlocking rings and adapted to fit the bottom head of the pressure vessel.

III.3.2. Reactor coolant system and moderator system

The reactor coolant system ***(Figure III-5)*** removes the heat generated in the reactor core and transfers it via the steam generators to the turbine generator plant.

The system is structured similar to that of a pressurized light water reactor and consists of two identical loops, each comprising a steam generator, a reactor coolant pump and the interconnecting piping, as well as one common pressurizer and pressurizer relief system.

The heat is carried by the reactor coolant, which flows from the reactor pressure vessel to the steam generators, where it is cooled down and then pumped back to the reactor pressure vessel by the reactor coolant pumps.

The pressurizer system is connected to one reactor coolant loop and basically comprises the pressurizer with the electric heaters, the surgeline, the spray lines with valves, and the safety valves.

The function of the pressurizer system is to maintain the appropriate pressure in the reactor coolant system in order to prevent boiling of the coolant under all operating conditions (principle of the pressurized water reactor), and to avoid or limit the pressure variations caused by volume fluctuations during load changes. The pressurizer is partly filled with saturated water and partly with steam. If the pressure drops, water is evaporated by switching on the electric heaters, raising the pressure to its set point. In the event of a pressure rise, steam is condensed by spraying water into the steam space.

Besides pressure control by sprays in the pressurizer, protection against overpressure in the reactor coolant system is provided in accordance with international codes for pressure vessels and systems. Protection is afforded by independent, self-actuating safety valves.

When the safety valves open, the steam discharged from the pressurizer is directed into the pressurizer relief system, where it is condensed to water.

The moderator system consists of four identical loops operating in parallel. Each loop comprises a moderator cooler, a moderator pump, and the interconnecting piping with valves.

The moderator system performs various functions depending on the operating mode of the reactor.

During normal operation, the moderator system maintains the moderator at a lower temperature than that of the reactor coolant. The moderator leaves the top of the moderator tank, flows to the moderator pumps, is pumped to the moderator coolers and flows back to the bottom of the moderator tank. The heat transferred in the moderator coolers is used for preheating the feedwater.

For residual heat removal, the moderator system is switched over to the residual heat removal position by means of the moderator valves. Under this mode of operation, the moderator is extracted from bottom of the moderator tank by the moderator pumps and fed into the cold legs of the reactor coolant loops, and also directly into the reactor coolant inlet annulus of the reactor pressure vessel, via the moderator coolers. The moderator system forms the first link of the residual heat removal chain. The residual heat is transferred from the moderator system to the residual heat removal system and then to the service cooling water system.

During emergency core cooling, the moderator serves as a high pressure core reflooding and cooling system. The emergency core cooling position is similar to that of the residual heat removal, but additionally, water is injected into the hot legs of the reactor coolant loops and into the upper plenum of the reactor pressure vessel. The residual heat removal chain connected to the moderator coolers during emergency core cooling is the same as during residual heat removal.

The steam generators transfer heat from the reactor coolant on the primary side to the feedwater/ steam cycle on the secondary side. The transferred heat raises the feedwater temperature and generates the saturated steam which drives the turbine generator unit. The steam generator constitutes the barrier between the radioactive reactor coolant and the non-radioactive feedwater/steam cycle, preventing the carry over of radioactive matter.

The steam generator is a vertical U-tube heat exchanger with natural circulation of the feedwater on the secondary side. The primary side of the steam generator consists of the channel head (primary plenum) and of the heating tube bundle. On the secondary side, the feedwater enters through a nozzle located in the steam dome and is distributed by a feedwater ring manifold in the annulus, (down-comer) formed by the secondary side shell and the tube bundle wrapper. The feedwater flows downwards to the tube sheet and enters the tube bundle region (riser). The feedwater is heated and partly evaporated around the U-tubes in this region. The generated steam-water mixture leaves the riser region and flows through cyclones and steam driers arranged in the steam dome. The dried steam is discharged through the main steam outlet nozzle, the separated water flows downwards into the down-comer where it is mixed with the incoming feed-water.

All systems of the residual heat removal chain are of a consistent “four loop” design. The residual heat removal system acts as a barrier between the active moderator and the service cooling water and prevents the escape of radioactivity into the service cooling water in the event of leakages in the moderator coolers.

III.3.3. Refuelling system

The natural (and slightly enriched) uranium reactor makes it possible and desirable, with a view to obtaining a high burnup, to shuffle and replace the fuel assemblies during power plant operation. The refueling procedure is carried out by a single refuelling machine. The fuel assembly transport system is located in the reactor building and in the fuel pool building. The main items of the fuel transport system are: refuelling machine, tilter with supporting structure, fuel transfer tube, fuel pool, and the corresponding auxiliary systems and maintenance installations. The refueling procedure is fully automated and monitored from the control room.

The refueling machine is moved from a maintenance position in the refueling machine maintenance room, by remote control, to a previously selected coolant channel position in the reactor vessel. in which the machine is centered. The seat-on seal is pressed on to the coolant channel closure head by the dead weight of the refueling machine to form a water-tight seal between the machine and the coolant channel. A pressure equalization takes place between the refueling machine and the reactor before opening the isolation valve of the refueling machine and opening the coolant channel closure. Following this, the fuel bundle column is withdrawn into an empty position in the refueling machine magazine. The magazine is then rotated in such a way that a fuel bundle column with a partially burnt up fuel assembly or with a new fuel assembly is positioned above the open coolant channel. This fuel bundle column is lowered into the coolant channel position and the coolant channel closure is locked again. After closing the isolation valve of the refueling machine a check for leak tight closure is performed. Then the refueling machine is removed from the reactor pressure vessel and is positioned above the vertically arranged titter. The titter has the following functions in the sequence indicated:

- Take-over of the fuel bundle column with the spent fuel assembly.
- Removal of the decay heat by cooling with D_2O.
- Drying and cooling the spent fuel assembly with gas.
- Flooding and cooling of the titter with H_2O.
- Tilting to the horizontal position and connecting with the fuel transfer tube.
- Transfer of the fuel assembly into the fuel transfer tube.

When a new fuel bundle column is transported from the fuel pool building into the titter via the transfer tube, and later from there into the refueling machine, the process of cooling and change of cooling medium takes place in the reverse order.

The task of the fuel transfer tube is to establish a connection between the tilter within the reactor building and the tilting device in the fuel pool building, while both components are in the horizontal position.

The tilting device takes the fuel bundle column from the fuel transfer tube and swivels it from the horizontal into the vertical position.

A silver-clad seal ring with good material flow properties is used as sealing material. In this established design, the coolant channel seals are almost perfectly tight.

III.3.4. Reactor auxiliary and ancillary systems

The auxiliary systems are basically organized in the same way as the auxiliary systems in PWR plants and work together with the reactor coolant system and moderator system to ensure the specified chemical conditions of the coolant and moderator. The systems containing heavy water are strictly separated from the systems containing light water in order to avoid downgrading the heavy water. The main tasks of the auxiliary systems are:

- Storage of heavy water.
- Volume control, seal water supply.
- Treatment and upgrading of heavy water.
- Boric acid dosing and chemical feeding into the primary circuit.
- Fast boron injection.
- Nuclear component cooling.
- Fuel pool cooling.
- Supply of refueling machine with auxiliary fluids.
- Compensation of leakages.
- Removal of decay heat from the core, emergency core cooling.

The auxiliary and ancillary systems are located mainly in the auxiliary building and partly in the annulus of the reactor building.

III.3.5. Instrumentation and control systems

The instrumentation and control equipment includes the measurement, control, protection and monitoring system.

The control room is, through the automation and plant interface equipment, connected to the drives and signal transmitters in the plant. Plant conditions and operational transients are transmitted via analog and binary signal transmitters to both the operator in the control room and to the automatic equipment. The command signals to the drives in the plant are transmitted via the control interface as required for maintaining proper operation.

Measured-value and status signals from the entire plant are continuously monitored by means of a process computer. The computer provides the data for trend logging and fault analysis and transmits the information to the operating personnel in the control room via data display terminals and teleprinters.

Automatic functional group controls are provided to minimize the operating errors and to obtain a higher degree of automation.

III.3.5.1. Main control room

The main control room of the nuclear power plant contains the operating and information equipment for the control and monitoring of the plant systems. This means that manual control, setpoint adjustment and monitoring of the reactor, important reactor auxiliaries, the feedwater/steam cycle, the turbine, the generator and the auxiliary power equipment are controlled from the main control room.

The main control room is situated on the top floor of the switchgear building above the electronic equipment rooms. For security reasons, it shall only be entered through monitored entrances.

III.3.6. Electric power system

The CNA II has two physically independent grid connections. One of then is the 500 kV grid and the other is the 132 kV grid.

There is additionally, auxiliary power supply from the generator in case of grid disturbance after load rejection. Only in the case of a common outage of all three power supply possibilities, the emergency power system with the diesel generators will be required. Definite loads – mainly of the control and instrumentation system – are power supplied by rectifiers and converters or by means of batteries with direct current.

The generator feeds into the 500 kV network via one generator transformer and supplies the station auxiliary service requirements by means of two auxiliary transformers.

The four secondary windings of the auxiliary transformers or the two secondary windings of the offsite system transformer feed into four separate medium high voltage bus section (each two 6.6 kV and 13.2 kV), to which the large auxiliary loads and the transformers for the low voltage switchgears are connected.

If the station service power system fails, certain equipment (pumps, etc.) are needed to remove residual heat and to run the plant into safe conditions. This equipment must remain in operation or has to be put in operation and must therefore be supplied with emergency power. Here a distinction must be made between two groups with regard to safety related requirements: loads allowing a voltage interruption while the diesel run up and loads which must remain in operation without interruption or which must be put into operation immediately should the normal station service system fail.

The diesel emergency power system is like the other redundant safety equipment, divided into for separate trains. Under normal operating conditions, the auxiliary switchgears of the normal power system feed the emergency power system. To avoid loss of power in case the auxiliary power system fails, each of the four trains in the emergency power system is equipped with a quick-starting diesel set.

The CNA II electric system may be divided into two main subsystems: the offsite power system and the onsite power system.

The offsite power system is constituted by the 500 kV transmission line, which is linked to the substations Rosario Oeste (113 km), Colonia Elía (160 km) and Ezeiza (67 km), and the 132 kV transmission line, source that is connected to the Zárate substation (23 km) and, in addition, to the 220 kV switchyard at the power plant Atucha via a 150 MVA coupling transformer.

With the generator load-breaker in the “off” position the station service power for “start up” and “shut down” of the power plant can be drawn from the 500 kV grid. Upon simultaneous failure of the main grid and the turbine generator set the 132 kV grid provides power for shut down operation of the plant down to the “hot-subcritical” condition.

The onsite power system consists of two subsystems: the auxiliary power system (“normal system”) and the emergency power system.

The auxiliary power system provides power for the loads of the nuclear power plant which are necessary during normal operation, start-up and shut-down operations. It is subdivided into four trains which are supplied by the 13.2 kV and 6.6 kV windings of the two unit auxiliary transformers. The transformers are fed in via single phase totally enclosed leads either from the min generator or from the grid via the external generator transformer. For shut-down operation or after loss of the normal power supply grid and the generator, it may be fed by the offsite power supply. The offsite power supply is available via automatic changeover.

The A.C. emergency power system provides the power required for safe shut-down of the reactor to maintain it in the shut-down condition, for removal of residual heat and to prevent release of radioactivity during normal operation and accident conditions resulting from system faults, and for some loads important for plant availability. It is subdivided, according to the safety system to be power supplied, into four redundant independent trains, each capable of supplying 50% of the power required to perform the safety function.

Normally the A.C. emergency power supply system is connected with the two 6.6 kV buses of the plant auxiliary power system by means of two circuit breakers, connected in series, for each train. Therefore, it can be fed via the plant auxiliary power system or the offsite system. At loss of the plant auxiliary power it is fed by the diesel generator emergency power system supply.

The stand by diesel generator emergency power system is provided for safety related loads. Each diesel generator set is assigned to one train, each with 50% capacity. Each emergency diesel generator is constituted by different main and auxiliary subsystems, such as the compressed air, the fuel supply equipment, the lubrication system, etc.

In the event of the power plant and the high voltage system failing at the same time, an emergency power supply can be obtained from diesel sets to permit proper shutdown of the plant. On failure of the auxiliary voltage, a period of 20 to 30 seconds elapses before the emergency diesel sets can take load. However, certain loads, such as the reactor protection system, measuring systems, and other protective systems must remain operative at all times. These are therefore fed directly from 220 V or 24 V D.C. batteries or from static converters fed from the 220 V breakers.

III.3.7. Safety philosophy and safety systems

The safety philosophy, on which the design is based, fulfills, in all conceivable plant conditions, the following basic requirements:

- The reactor can be safely shut down and kept shut down over prolonged periods.
- The decay heat can be reliably removed.
- Any release of radioactivity is within the limits laid down by the radiation protection regulations.

In order to meet these requirements, safety measures against damage to the systems or components are provided. The safety measures can be classified under three safety levels according to the possible plant conditions:

- Components and systems necessary for normal operation (including startup, part load and full load operation, load changes, shutdown) are of such design as to preclude failure. The safety measures provided are:
  - Conservative and careful design.
  - Stringent quality assurance and control.
  - Regular examinations and inspections.
- According to general engineering experience, it must be considered that systems and components can fail during their service life despite adequately high quality. It is assumed that operational disturbances (e.g. reactor coolant pump failure, load rejection) can occur. In order to prevent the faults and operational disturbances and to mitigate their consequences the following safety measures are provided:
  - Inherently safe operational characteristics.
  - Alarm annunciation.
  - Reactor protection limitation.
- Despite the safety measures of the first and second safety levels, theoretically assumed accidents are postulated. In order to counter these accidents and to mitigate their consequences, active safety systems are provided. The design of the safety systems is based on the assumption that parts of the safety systems (sub-systems) can fail simultaneously with the accident. As a consequence, the safety systems are of redundant design. This multiple-train design is reflected not only in the redundancy of the equipment, but also in the consistent physical segregation of the subsystems. This ensures that a sub-system failure (random failure), postulated in addition to the accident, remains restricted to one subsystem and does not affect the others.

According to the design principles of the safety systems all engineered safety features necessary to control accidents are built as four identical sub-systems. Two of these are sufficient for the control of the accident. Thus, functional availability is assured even when one subsystem is being inspected or repaired and a single failure occurs simultaneously in another subsystem. Each subsystem also comprises the associated power supply and the necessary auxiliary equipment.

The basic safety systems provided are:

- Fast reactor shutdown system.
- Emergency core cooling system.
- Containment system.
- Emergency electric power system.

In order to protect the environment against the release of radioactivity, the following radioactivity barriers are provided as passive safety measures:

- The fuel matrix of the uranium dioxide pellets.
- The seal-welded cladding tubes enclosing the fuel.
- The closed and seal-welded reactor coolant system and moderator system.
- The full pressure gas tight steel containment structure.
- The concrete secondary shield.

The components of the radioactivity barriers act according to their mechanical properties, without auxiliary energy. In case of damage to one of these barriers, the next one will act and thus retain the radioactivity.

The accidents considered in the plant design are the plant internal and external accidents. The internal accidents are, above all, LOCA, with the whole spectrum of pipe ruptures including the break of the largest connection pipe to the reactor coolant loops or to the moderator system. The external accidents considered are aeroplane crash, explosion pressure wave, floods, tornadoes, etc.

In order to meet the safety requirements even during the considered internal and external accidents, the following design principles were established:

- Multiplicity of safety features.
- Redundancy of the safety systems and of their auxiliary system.
- Diversity of important parts of the reactor protection system.
- Physical separation and/or protection by concrete walls of the redundant sub-systems.
- Protection of the safety systems against external accidents.
- Periodic testing of the safety systems.

The task of the safety systems is to prevent any damage to the radioactivity barriers during operational malfunctions and during accidents in order to fulfil the safety philosophy requirements.

The fast reactor shutdown safety system consists of two separate subsystems: the shutdown control rod system (first independent shutdown system) and the boron injection system (second independent shutdown system). The emergency core cooling safety system consists of the following basic subsystems: the moderator system, the residual heat removal system, the service cooling water system for the secured plant, the nuclear component cooling system and the safety injection system.

The containment safety system consists of several basic subsystems: the concrete containment, the steel containment, the containment isolation system and the reactor building annulus air extraction system ***(Figure III-6)***.

The emergency electric power system is not a safety system but a safety related one due the nature of the loads it feeds. It was described in section III.3.6.

The general plant layout features a compact, controlled area with short pipe and cable connections, physical separation of redundant cable and piping trains, minimum exposure of the operating personnel during maintenance and repair work, availability of sufficient shielding and radiation protection measures.

III.4. TECHNICAL DATA

Some of the main technical data are detailed in what follows:

Overall Plant Data	
Reactor type	Pressurised heavy water (PHWR)
Bulk nominal electric power	745 MW
Net nominal electric power	692 MW
Net efficiency	32%
Number of steam generators	2
Number of reactor coolant pumps	2
Number of moderator coolers and moderator pumps	4

Reactor Core Data	
Material of fuel	Natural uranium
Total thermal power	2160 MW
Shape of fuel assembly	37 – rod cluster
Number of coolant channels or fuel assemblies	451
Cladding material	Zircaloy 4

Active core length	5300 mm
Refuelling	On load
Fuel burnup at equilibrium	7500 MWd/MgU
Number/material of control rod elements	9/hafnium and 9/steel

Thermal and Hydraulic Data	
Number of coolant circuits	2
Number of moderator circuits	4
Coolant and moderator	Heavy water
Pressure at reactor vessel outlet	115 bar
Coolant temperature at reactor pressure vessel outlet	312° C
Coolant temperature rise though the core	34° C
Moderator temperature normal/maximum	170°/220° C
Total coolant circulation flow	37080 Ton/h
Total moderator circulation flow	3200 Ton/h
Steam pressure at steam generator outlet	56 bar
Steam temperature	271° C
Total steam flow	3445 Ton/h

Steel Containment Data	
Diameter	56 m
Wall thickness	30 mm
Design pressure	5.3 bar
Design leak rate	0.25 vol %/day

SITE LOCATION

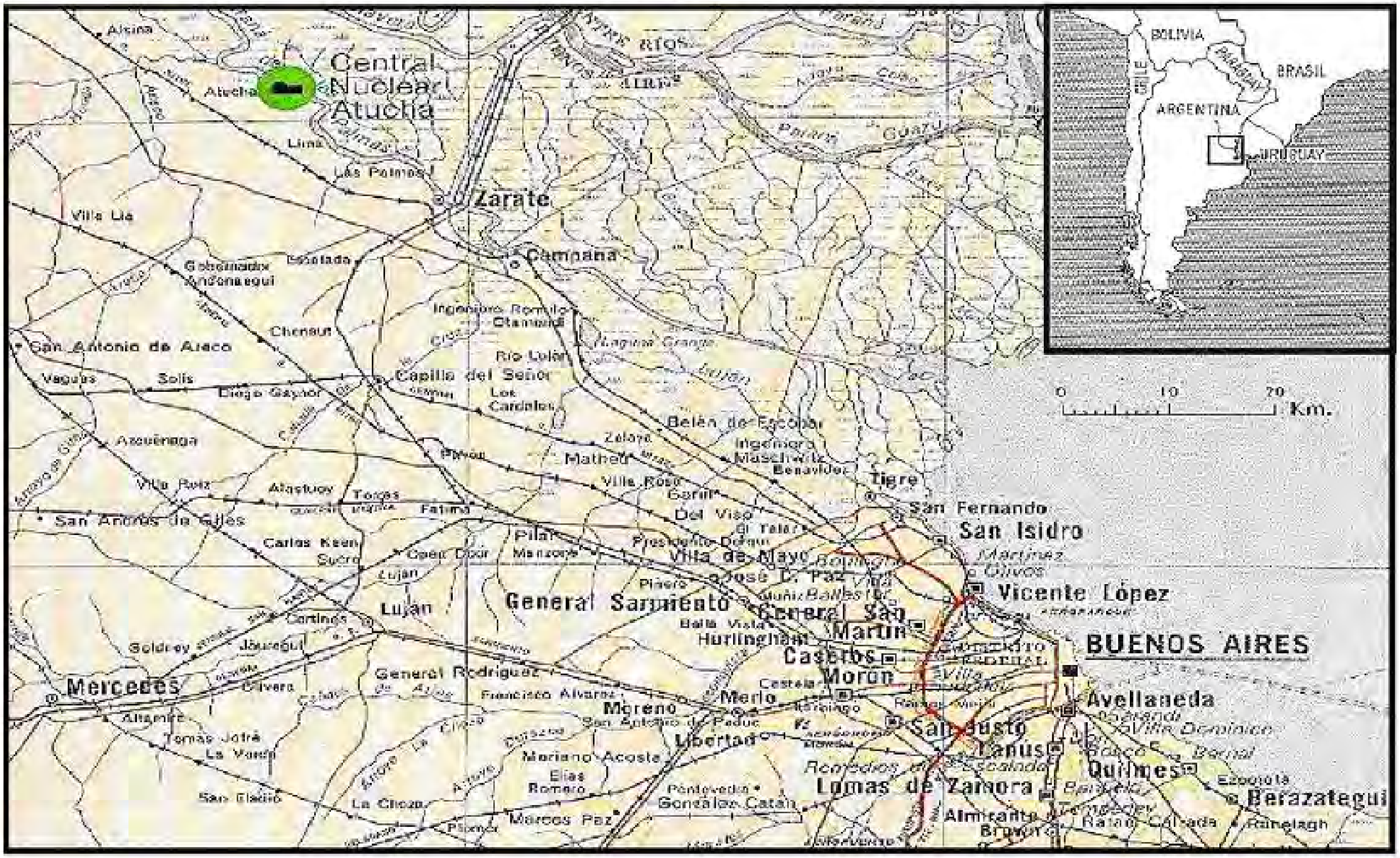


Figure III-1 - Atucha II Nuclear Power Plant - Geographic Location

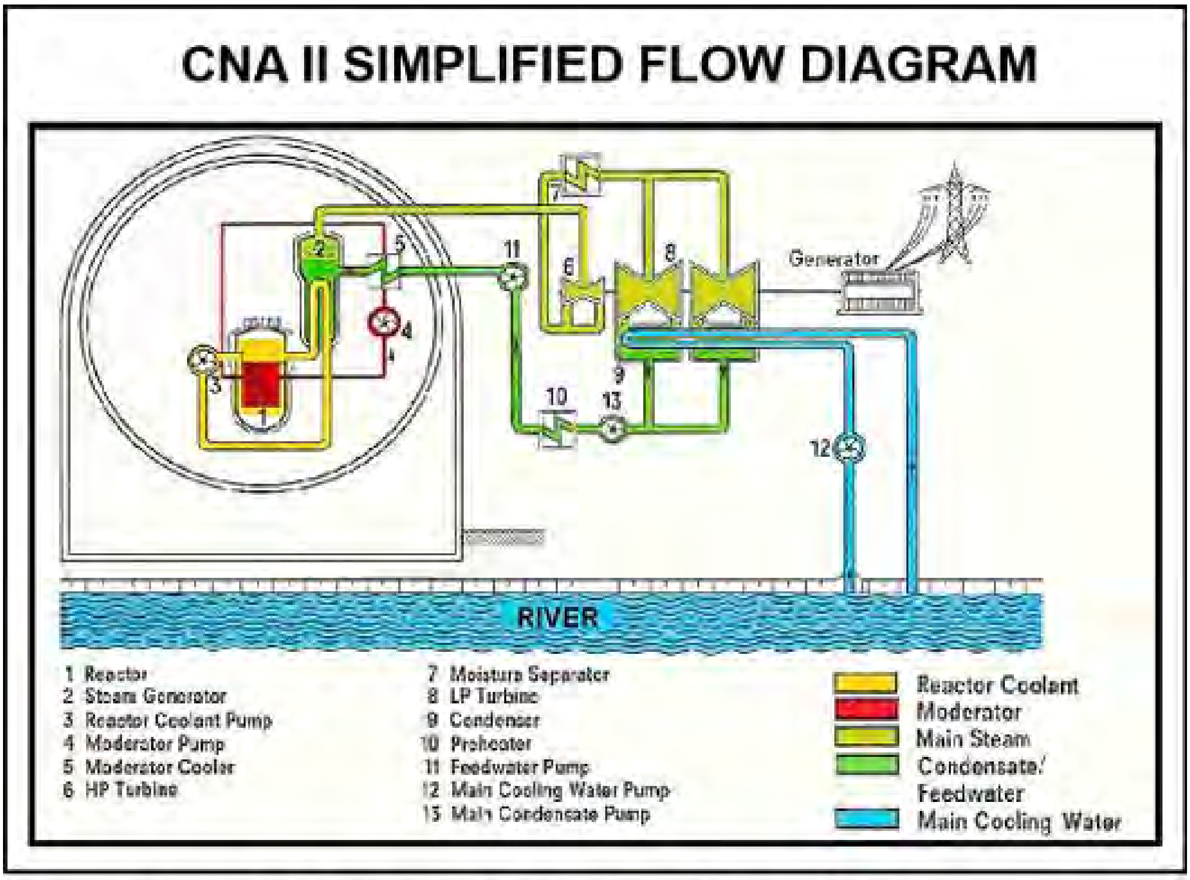


Figure III-2 - Atucha II Nuclear Power Plant - Main Systems

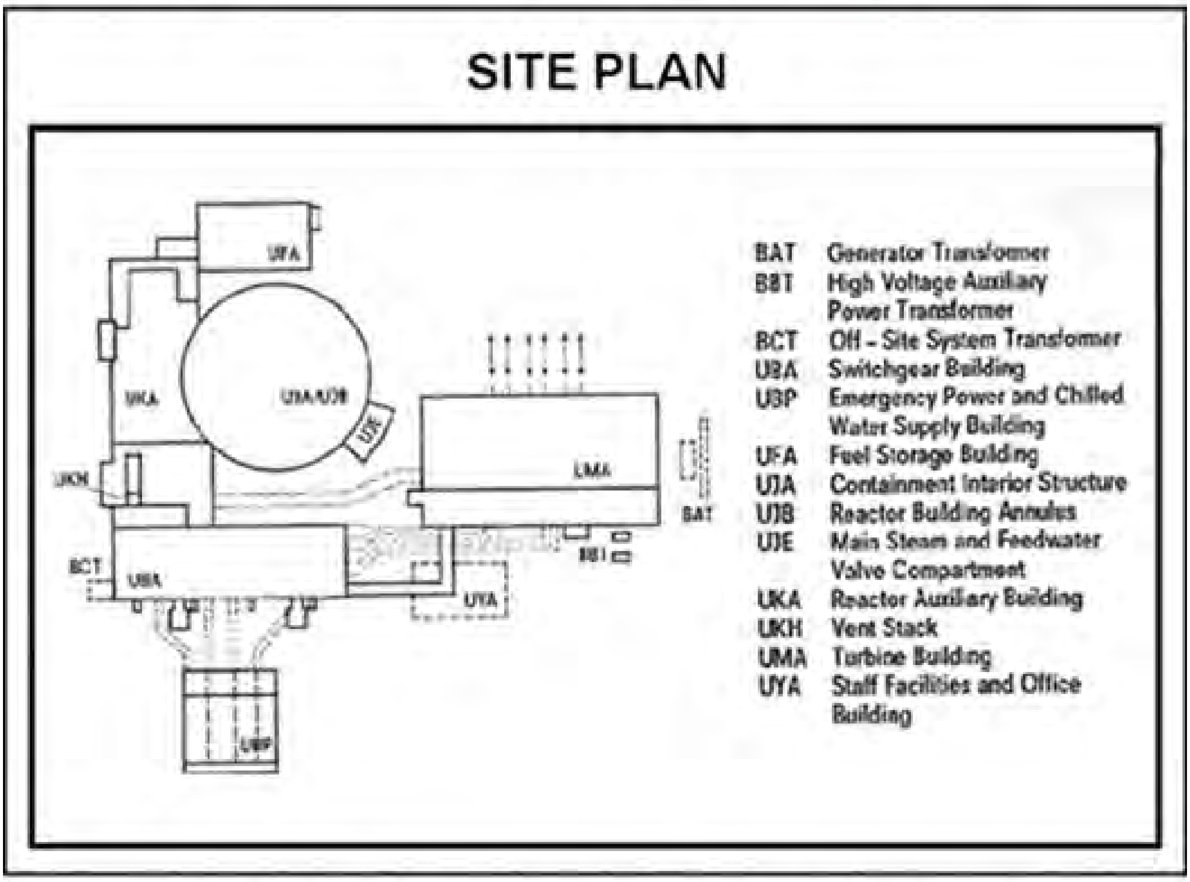


Figure III-3 - Atucha II Nuclear Power Plant - Main Building and Structures

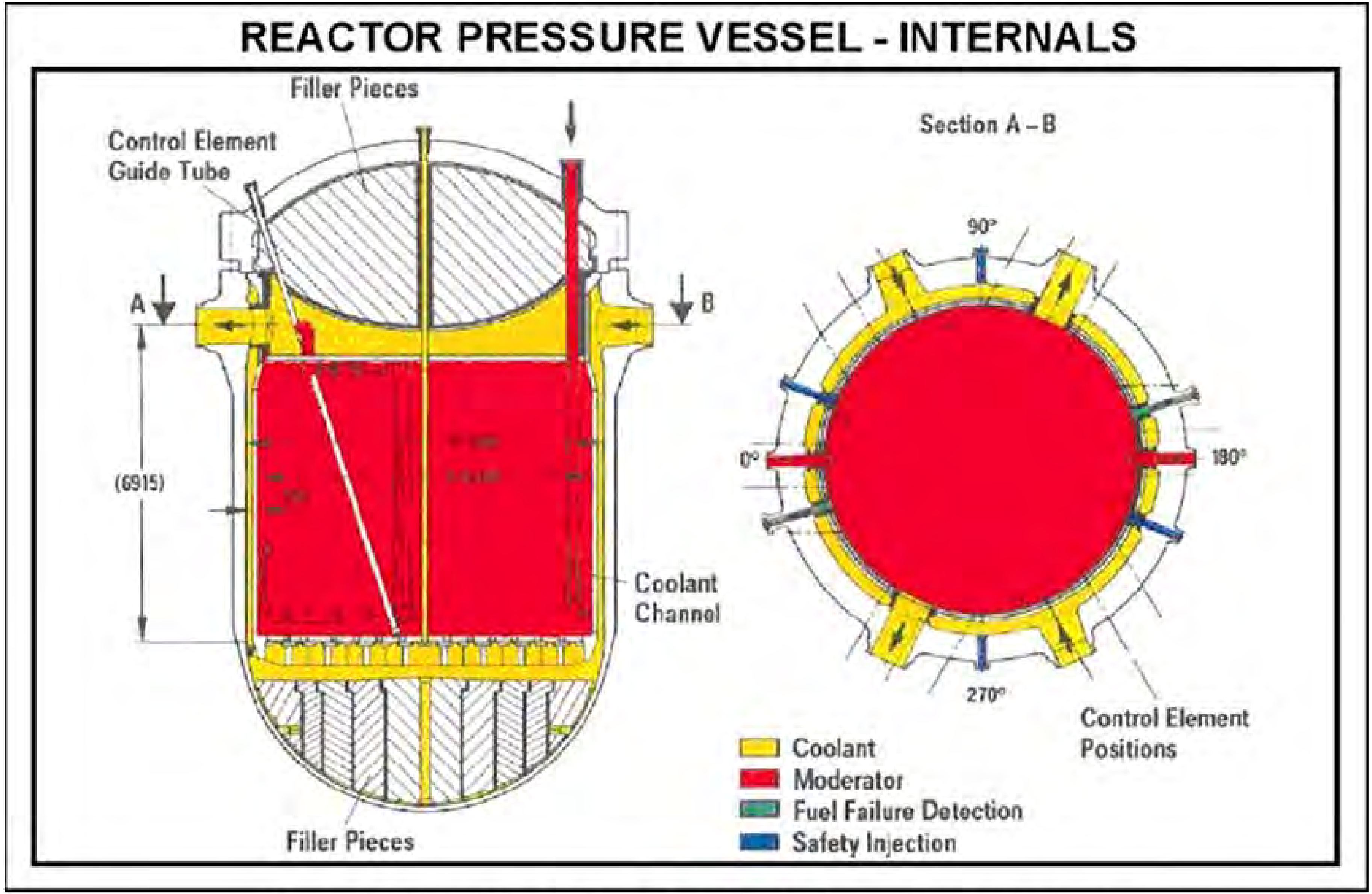


Figure III-4 Atucha II Nuclear Power Plant - Reactor Pressure Vessel with Internals

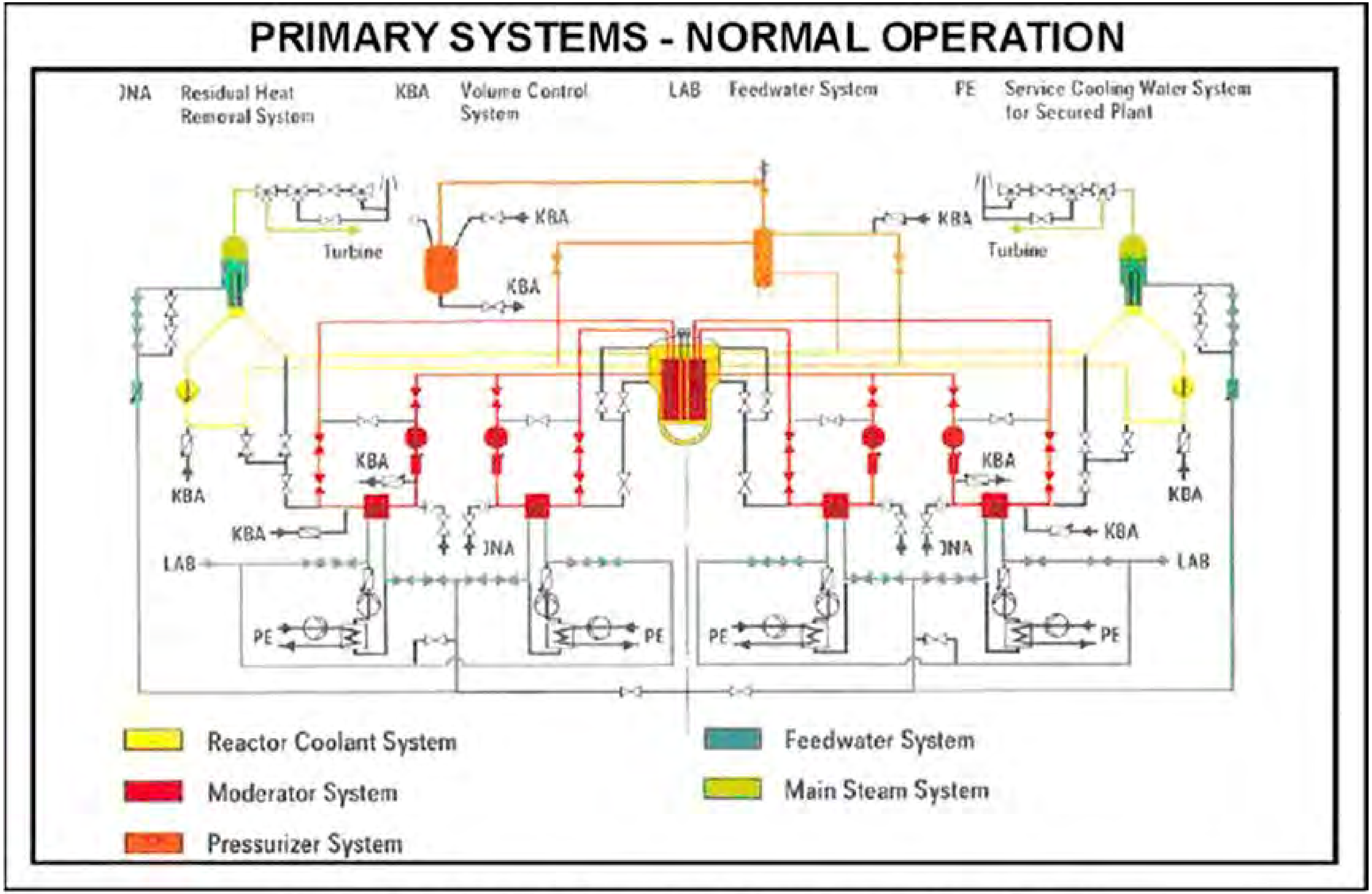


Figure III-5 Atucha II Nuclear Power Plant - Primary Systems - Normal Operation

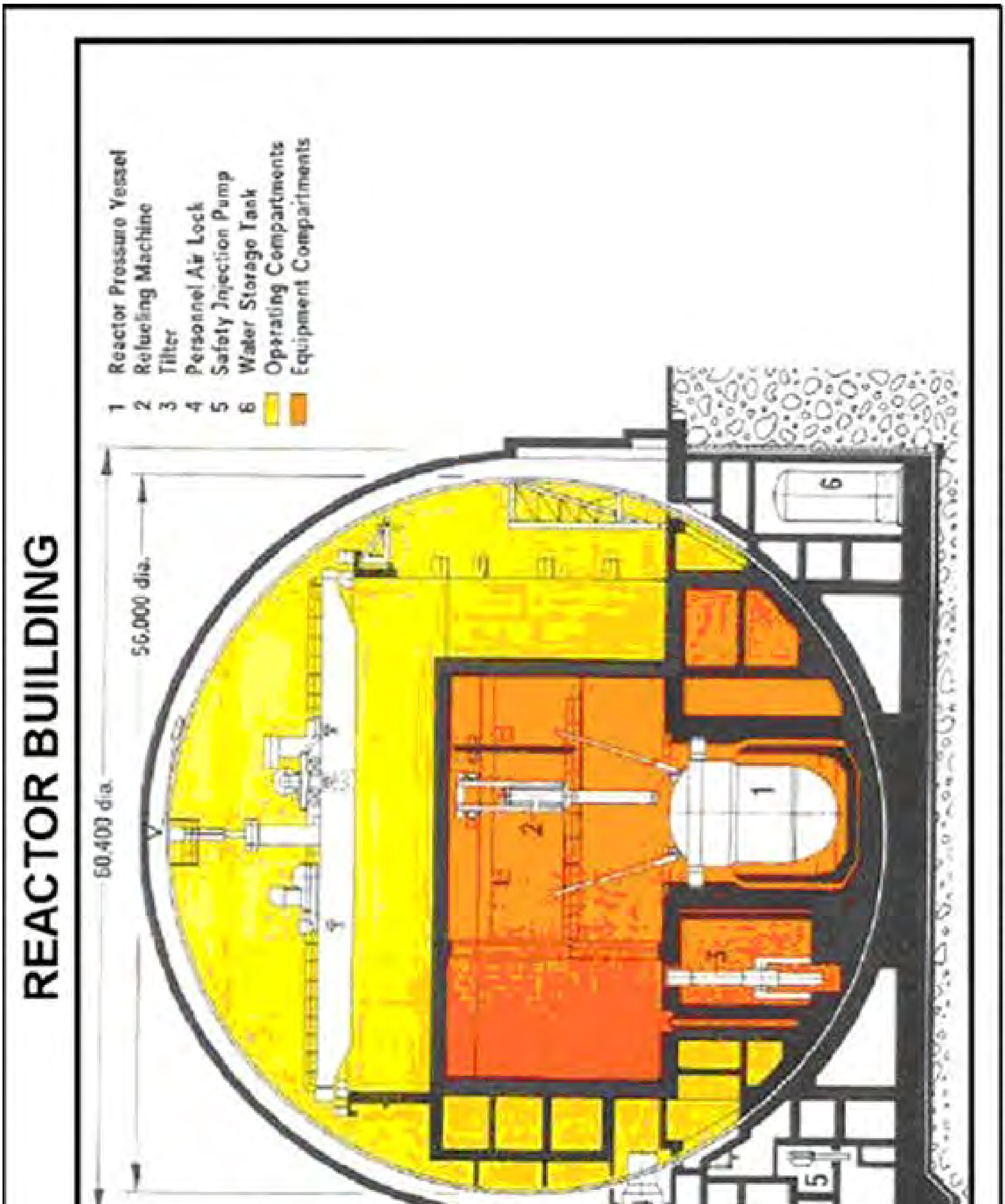


Figure III-6 Atucha II Nuclear Power Plant - Reactor Building

ANNEX IV
SUMMARY TABLE

This Annex presents a summary table containing the activities carried out by the Operator and the Regulator and, the corresponding improvements/modifications resulting from the stress tests, which are foreseen to be implemented in each nuclear power plant as a result of the lessons learned from the Fukushima accident.

This table includes the corresponding schedules and milestones to complete the operator´s planned activities.

Topic 1 – External Events

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 1 – External Events						
1.1 Resistance Assessment of the Argentinean NPPs (stress tests), to determine the performance to face initiating events of natural origin, including the corresponding proposal of improvements and modifications (RQ-NASA-038).	Taken	2012	Yes	Taken	2011	No
1.1.1 CNA I: seismic evaluation program. It consists of five phases: • Phase 1: Scope of the study and preliminary inspection walkdown.	Taken	2012	Yes	Taken	2011	Yes
1.1.2 CNA I: seismic evaluation program. • Phase 2: Development of the Safe Shutdown Equipment List (SSEL) and Systems walkdown, • Phase 3: Seismic response and capacity evaluation of SSC.	Ongoing	2012 2nd half	No	Taken	2011	No
1.1.3 CNA I: seismic evaluation program. • Phase 4: Inspection walkdown of the seismic capacity and screening process.	Ongoing	2012 2nd half	No	Taken	2011	No
1.1.4 CNA I: seismic evaluation program. • Phase 5: Detailed analysis and evaluation for seismic qualification.	Planned	Subject to the results of phase 4	No	Taken	2011	No
1.1.5 CNA I: Easy fix to avoid equipment damage (in core and spent fuel pool): • Electrical and I&C cabinets in raised floor: an anchor will be implemented to the bottom and / or upper floor slab, reinforced in two horizontal directions. • Batteries: additional restrictions will be installed to the racks to prevent slippage.	Ongoing	2012 2nd half	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 1 – External Events						
• Panels of the control room: the panels will be reinforced from above to the concrete wall behind. • Emergency diesel generators: a small wall/dam of concrete is going to be built around the pit to prevent flood damage.						
1.1.6 CNA I: It is being analyzed the need for the installation of seismic instrumentation inside the plant and in the free field, in a radius of 20 km to 40 km around the facility.	Ongoing	2012 2nd half	No	Taken	2011	No
1.1.7 CNA I: Based on studies of boundary conditions of operation for decreases in the river levels values, a procedure for systematizing the outage of the plant in case of extreme low-water-levels is being developed.	Ongoing	2012 2nd half	No	Taken	2011	No
1.1.8 CNA II: Addition of passive auto-catalytic recombiners	Ongoing	2013	No	Taken	2011	No
1.1.9 CNA I: Installation of a mobile diesel generator (MDG) equipment 680 KVA.	Planned	2013	No	Taken	2011	No
1.1.10 CNA II: Procedures for cooling the plant via the SGs in case of SBO and loss of the assured service water cooling system PE (as a result of the analysis performed in the PSA).	Ongoing	2013	No	Taken	2011	No
1.1.11 CNA I: Completion of the new Emergency Power System EPS (Diesel Generators).	Ongoing	2013	No	Taken	2011	No
1.1.12 CNA I: Installation of an independent groundwater pump to feed the storage pools of irradiated fuel elements. Pump electrical supply: from the secured bus bar (through a new electric panel) and from the MDG (easy manual connection). Implementation of an alternative system to feed the fuel storage pools that allows the monitoring of relevant parameters from outside the pool building.	Planned	2013	No	Taken	2011	No
1.1.13 CNA I: Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) capable of withstanding the maximum level given by the breaking of the Yaciretá dam upstream.	Ongoing	2013	No	Taken	2011	No
1.1.14 CNA II: Easy fixes arisen from the plant walkdown.	Planned	2013	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 1 – External Events						
1.1.15 CNA II: Seismic Margin Assessment - Identification of SSC needed to achieve a safe shutdown condition.	Planned	2013	No	Taken	2011	No
1.1.16 CNA I: Manual water refilling process to the spent fuel storage pools, so that an operator monitoring the level and temperature of the pools will operate the process.	Planned	2013	No	Taken	2011	No
1.1.17 CNA II: Procedures for the removal of residual heat via SG for flooding and low water level.	Planned	2013	No	Taken	2011	No
1.1.18 CNA I: The original conditions of impact of objects thrown by tornadoes shall be re-evaluated for the plant buildings.	Planned	2013	No	Taken	2011	No
1.1.19 CNA II: Implementation of an additional system to refill the fuel elements storage pools from an alternative reservoir (water inlets from the underground water-table, existing tanks, etc.).	Planned	2014	No	Taken	2011	No
1.1.20 CNE: Seismic margin assessment (SMA) based on probabilistic safety analysis (PSA).	Planned	2014	No	Taken	2011	No
1.1.21 CNA II: Improvements for extending the water supply to the SG for a longer time consisting in the mentioned MDG and the water supplied from an alternative reservoir. Is being evaluated the possibility that this reservoir will be constituted by water intakes from the groundwater.	Planned	2014	No	Taken	2011	No
1.1.22 CNA I: Installation of passive auto-catalytic recombiners (PARs).	Planned	2014	No	Taken	2011	No
1.1.23 CNA I / CNA II: A new hydrologic and hydraulic study, which includes a review of background studies of the Atucha site, in order to supplement and update, and also make a prospecting considering possible future scenarios. This study reassess both high water level and low water level of the design basis and takes into account the combination of the maximum flow of the tributaries, broken dams located upstream as well as the boundary condition at the mouth of Parana river given by the levels of the Rio de la Plata.	Ongoing	2014	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 1 – External Events						
1.1.24 CNA II: Flooding of the RPV cavity, with water reposition from the demineralized water tank (GHC), the fire fighting water system or from an external reservoir.	Planned	2015	No	Taken	2011	No
1.1.25 CNE: Upgrade of the Emergency Power Supply (EPS) system. Replacement of the existing 50 kW / 75 kVA emergency power supply diesel generators with diesel generators of increased capacity. The new generators will be rated at approximately 1 MW and will be able to feed both the ECC pumps and the new electrical EWS.	Planned	2015	No	Taken	2011	No
1.1.26 CNE: Upgrade of the Emergency Water Supply (EWS) system: replacement of the existing diesel engine pumps with two new 100% higher capacity pumps to be driven by the new EPS generators. The piping and valves from the EWS pump house to the service building will also be replaced, and the valves that feed emergency water to the steam generators and to the reactor will be duplicated. The higher capacity pumps will be able to feed the ECC heat exchanger.	Planned	2015	No	Taken	2011	No
1.1.27 CNE: To improve the seismic capacity of MP/LP ECCS based on the SMA walkdown and fragility analysis findings.	Planned	2015	No	Taken	2011	No
1.1.28 CNE: Addition of passive auto-catalytic recombiners (PARs) in the reactor buil-ding.	Planned	2015	No	Taken	2011	No
1.1.29 CNE: Addition of a make-up water supply line from outside the reactor building to the calandria vault.	Planned	2015	No	Taken	2011	No
1.1.30 CNE: Addition of a seismically qualified rupture disc assembly to the existing inspection port of the calandria vault.	Planned	2015	No	Taken	2011	No
1.1.31 CNE: Reinforcement of the secondary control room door to preserve it in case of flooding in the turbine building.	Planned	2015	No	Taken	2011	No
1.1.32 CNE: Re-evaluation of the consequences of earthquakes on the dam downstream of Embalse NPP.	Planned	2015	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 1 – External Events						
1.1.33 CNE: The gateway to the secondary control room located in the level 100 m of the turbine building shall be reinforced, if for any eventuality the water that floods the E/T exceeded that level.	Planned	2015	No	Taken	2011	No
1.1.34 Re-evaluation of the risk of tornadoes for the sites of Atucha and Embalse.	Ongoing	2015	No	Taken	2011	No

Topic 2 – Design Issues

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 2 – Design Issues						
2.1 Resistance Assessment of the Argentinean NPPs (stress tests), to determine the performance to face the loss of safety functions including the corresponding proposal of design modifications and improvements (RQ-NASA-038).	Taken	2012	Yes	Taken	2011	No
2.1.1 CNA I: Emergency procedure and/or accident management: • SBO1. Manual action to inject the SSC in a short period of time with a cooling ramp of 100°C/h and manually deactivate of the TB (2013). • Inventory reposition of the SSC with increase in the capability of the SSC feed water tanks, using the UA10, D20 and D21 pumps and replace water in those pools with groundwater using one of the pumps of the potable water supply (UJ) (2013). • Outage of the plant due to an extremely low river level based on some studies of the operative limit conditions in a decreasing river level. This will allow systematic manoeuvres to do a plant outage (2012 2nd half).	Ongoing	2012 2nd half / 2013	No	Taken	2011	No
2.1.2 CNA I: Procedure related with passive components control, for example the piping for void/siphon rupture in the fuel element pools, and the frequency increase of tests and inspection.	Ongoing	2012 2nd half	No	Taken	2011	No
2.1.3 CNA I: Preventive strategies, related with the loss of cooling function, plan to avoid core damage. The next cases are analyzed:	Ongoing	2012 2nd half	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 2 – Design Issues						
• Coolant inventory reposition by the "pressure and inventory control system" (TA) in small LOCA conditions with different changes of design alternatives. • Water supply to the SGs through the second heat removal system (SSC/RX) in different accidental scenarios.						
2.1.4 CNE: Water reposition to the spent fuel elements pool by means of a connection from outside the pool building (installation of a 4" pipe with an isolation valve and a connection with the fire fighting system hose).	Ongoing	2012 2^{nd} half	No	Taken	2011	No
2.1.5 CNE: Instrumentation and control of spent fuel elements pool will be installed in the secondary control rooms (level and temperature measurements), independent from the ones that the existing ones in the main control room. The measurements will be repeated in the main control room.	Ongoing	2012 2^{nd} half	No	Taken	2011	No
2.1.6 CNE: Procedure of abnormal event in response to the loss of cooling / inventory in the fuel elements pool. It will include actions and contingencies to monitor the coolant level / temperature from the secondary control room when the main control room and the pool building are inaccessible.	Ongoing	2012 2^{nd} half	No	Taken	2011	No
2.1.7 CNA I: Installation of a mobile diesel generator equipment (MDG) 680 KVA.	Planned	2013	No	Taken	2011	No
2.1.8 CNA I: Completion of the new Emergency Power System EPS (Diesel Generators).	Ongoing	2013	No	Taken	2011	No
2.1.9 CNA I: EPS additional modification: the supply of the emergency current bus bars by the normal bus bar will be improved due to the duplication of the coupling breakers between normal and secured bus bars.	Ongoing	2013	No	Taken	2011	No
2.1.10 CNA I: Installation of an independent groundwater pump to feed the spent fuel storage pools. Pump electrical supply: from the secured bus bar (through a new electric panel) and from the MDG (easy manual connection). Implementation of an alternative system to feed the fuel storage pools that allows the monitoring of relevant parameters from outside the pool building.	Planned	2013	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 2 – Design Issues						
2.1.11 CNA I: Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) capable of withstanding the maximum level given by the breaking of the Yaciretá dam upstream.	Ongoing	2013	No	Taken	2011	No
2.1.12 CNA I: Evaluation of the fuel elements integrity in the management process or the fuel elements exchange inside the Refuelling Machine during a SBO event.	Ongoing	2013	No	Taken	2011	No
2.1.13 CNA I: Modification to make up water inventory to the SGs using groundwater, through the SSC when losing the feed water tank, the residual heat removal and the secured system of water injection to the SG. That system tank inventory can be replaced in those cases where the SSC integrity has not been affected. To do that it is foreseen to inject water from the storage pools to the SGs. The groundwater will be pumped using one of the pumps that supply potable water. In case of a SBO coincident with SSC diesel generators unavailability, the proposal considers the possibility to feed the involved components by means of an external generator.	Planned	2013	No	Taken	2011	No
2.1.14 CNA I: To assure the energy supply to the instrumentation involved with repre-sentative variables signals that are used for monitor its state and evolution, in case of a SBO.	Planned	2013	No	Taken	2011	No
2.1.15 CNA I: To implement a reduction of the loads that are feed by batteries to extend the availability of the batteries until it is provided an alternative energy supply.	Planned	2013	No	Taken	2011	No
2.1.16 CNA II: Procedure revision to extend the use of the GD utilizing additional fuel tanks. All the maintenance and test programs will be revised. It has to be guarantee that the minimum necessary previsions are kept in the inspection and test system.	Planned	2013	No	Taken	2011	No
2.1.17 CNA II: Availability analysis of the external electric supply including the high voltage of the 220 kV and 500kV lines.	Planned	2013	No	Taken	2011	No
2.1.18 CNA II: Evaluation of the fuel elements integrity in the management process or the fuel elements exchange inside the Refuelling Machine during a SBO event.	Planned	2013	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 2 – Design Issues						
2.1.19 CNE: Protection of the 500 kV switchyard, and bus bar, lines and protections improvements.	Planned	2013	No	Taken	2011	No
2.1.20 CNA II: Heat removal by means of steam generators: Water supply from an alternative reservoir is necessary to maintain the long term feeding of the SGs and the refrigeration of the spent fuel elements pool. There is a possibility that the alternative water reservoir will be constituted with underground water.	Planned	2014	No	Taken	2011	No
2.1.21 CNA II: The provision of an auxiliary 6.6kV Mobile Diesel Generator (MDG) to connect the switch located in the emergency bus bar. A facility to do an easy connection must be provided.	Planned	2014	No	Taken	2011	No
2.1.22 CNA I - CNA II electric interconnection between normal bus bars.	Planned	2015	No	Taken	2011	No
2.1.23 CNA I: Preventive strategies, related with the loss of cooling function, plan to avoid core damage. The next cases are analyzed: • When there is a station blackout (SBO) the strategy is: establish a core cooling mechanism avoiding air entry to the primary circuit from the boron injection system (TB). • Strategy for the failure of 220 VCC direct current supply. • Strategy for the diminution of the 24V direct current.	Planned	2015	No	Taken	2011	No
2.1.24 CNA II: Adequate the current cooling system of two diesel generators, by means of forced air cooling towers with a cooling capacity up to 5MW.	Planned	2015	No	Taken	2011	No
2.1.25 CNA II: To supply energy to CNA II from the CNA I Diesel Generators through manual actions. This will be done by means of two 6.6 kV normal bus bars and four 6.6 kV emergency bus bars of CNAII, in case of no availability of all emergency CNA II DGs, taking advantage of the existing interconnection between normal bus bars of CNA I and II.	Planned	2015	No	Taken	2011	No
2.1.26 CNA II: Procedure for disconnection of unnecessary loads to increase the batteries duration.	Planned	2015	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 2 – Design Issues						
2.1.27 CNE: Upgrade of the Emergency Power Supply (EPS) system. Replacement of the existing 50 kW / 75 kVA emergency power supply diesel generators with diesel generators of increased capacity. The new generators will be rated at approximately 1 MW and will be able to feed both the ECC pumps and the new electrical EWS.	Planned	2015	No	Taken	2011	No
2.1.28 CNE: Replacement of the class III (emergency power) diesel generators (DG) and improvements in the building where the DG are located.	Planned	2015	No	Taken	2011	No
2.1.29 CNE: Relocate the plant light system to avoid flooding during a loss of condenser water event occurrence.	Planned	2015	No	Taken	2011	No
2.1.30 CNE: To install a connection for fire truck through a hose to the ECC lines. This improvement and the operation of some valves will allow the possibility of adding water to the dousing, to refill the SGs. This improvement will permit the core cooling during at least the required 72 hours.	Planned	2015	No	Taken	2011	No
2.1.31 CNE: Fire truck will be available with a capacity of 17000 litres.	Planned	2015	No	Taken	2011	No
2.1.32 CNE: Availability of a 550 kVA mobile diesel generator (MDG) for electrical supply to essential loads during a SBO condition.	Planned	2015	No	Taken	2011	No
2.1.33 CNE: It is foreseen to provide the necessary systems to extend the electrical supply from batteries beyond the first 8 hours under SBO condition as well as its corresponding recharge through the emergency systems.	Planned	2015	No	Taken	2011	No
2.1.34 CNE: An operative procedure for abnormal events (POEA), to increase up to 7 days the reposition of water to the dousing and to the SGs. This water will be delivered by the emergency water supply pumps (EWS).	Planned	2015	No	Taken	2011	No
2.1.35 CNE: Evaluation of the impact of a SBO occurrence during the refuelling machine operation with spent fuels inside.	Planned	2015	No	Taken	2011	No

Topic 3 – Severe Accident Management

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 3 – Severe Accident Management						
3.1 Resistance Assessment (stress tests) of the Argentinean NPPs by analyzing the mitigation actions foreseen if severe damage in both the reactor and the spent fuel pool occurs, in order to prevent large radioactive releases, including the corresponding proposal of improvements and modifications. (RQ-NASA-038).	Taken	2012	Yes	Taken	2011	No
3.1.1 CNA I: The procedure for "Operation in Perturbations and Accidents" was modified in order to include the control of critical parameters of the spent fuel storage pools.	Taken	2012	Yes	Taken	2011	No
3.1.2 CNA II: cooling the RPV external side once significant damage to the core has been observed. This strategy and its effectiveness, including the guides for severe accident management, are currently under development and analysis.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.3 CNA I: To complete the development of specific model with MELCOR code.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.4 CNE: Automation of Service Water System to switch from normal cooling mode to an alternative one in case of service water lost event.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.5 CNA I: Analysis to determine the specifications, location and quantity of the required PARs.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.6 CNA I: The proper functioning of the program for the verification of the vacuum breakdown / siphons associated with cooling was checked, finding them in adequate condition. Besides, it is foreseen to add to the periodic inspections program, the control of the functionality of the breaking vacuum / siphon system associated with the pipes of the cooling systems or the inventory control of the spent fuel storage pools.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.7 CNA I: Dose rate remote measurement system including a circle of 10 km radius in all directions, connected on line to both the internal centre of emergency management (CICE) and the external centre of emergency management (CECE).	Ongoing	2012 2nd half	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 3 – Severe Accident Management						
3.1.8 CNE: Operating Procedure for Abnormal Events that covers response to loss of cooling in the spent fuel storage pool and/or loss of inventory. This procedure shall include actions to verify the coolant level and temperature of the pool from the secondary control room in the event that the main control room and the pool room were unavailable. It shall include actions to replenish water from alternative systems (eg. fire hydrant system or fire truck) in the event of sustained loss of cooling or loss of inventory.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.9 CNE: Facility to connect a fire-truck from outside the pool building, which will replenish water to the pools in the events of loss of cooling, loss of circulation or SBO.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.10 CNE: Instrumentation and control of spent fuel elements pool will be installed in the secondary control rooms (level and temperature measurements), independent from the ones that the existing ones in the main control room. The measurements will be repeated in the main control room.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.11 CNE: A new item will be incorporated to the Operations Manual: "Check once per shift the functionality of break vacuum pipes / siphon belonging to fuel pools ". Such control is planned to be included in the “walkdown form” that is filled by the operations assistant when monitoring the pools.	Ongoing	2012 2nd half	No	Taken	2011	No
3.1.12 CNA I: Emergency procedure and/or accident management: • SBO1. Manual action to inject the SSC in a short period of time with a cooling ramp of 100°C/h and manually deactivate of the TB (2013). • Inventory reposition of the SSC with increase in the capability of the SSC feed water tanks, using the UA10, D20 and D21 pumps and replace water in those pools with groundwater using one of the pumps of the potable water supply (UJ) (2013). • Outage of the plant due to an extremely low river level based on some studies of the operative limit conditions in a decreasing river level. This will allow systematic manoeuvres to do a plant outage (2012).	Ongoing	2012 2nd half / 2013	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 3 – Severe Accident Management						
3.1.13 CNA I: Strategies to reduce the containment pressure during severe accidents. Its implementation will be defined after the results of accident progression model be available: • From the outside by means of the annular ground ventilation. • From the inside by recirculaing air using their own ventilation systems.	Planned	2013	No	Taken	2011	No
3.1.14 CNA I: detail engineering and the required procedures related with the PARs implementation.	Ongoing	2013	No	Taken	2011	No
3.1.15 CNA II: Procedures to cool down the plant through SGs in case of SBO.	Planned	2013	No	Taken	2011	No
3.1.16 CNA I: Emergency operative procedure for responding to an event of loss of cooling or water inventory of the spent fuel storage pools foreseen to provide the level and temperature monitoring of the pools during an emergency and the possibility to restore inventory even under conditions of loss of control room; SBO; earthquakes and flooding.	Planned	2013	No	Taken	2011	No
3.1.17 CNA I: Installation of an independent separate pump to supply water to the spent fuel storage pools and an electrical panel for manually connecting the pump to the auxiliary emergency mobile diesel. Besides, it is foreseen to perform a modification of the electrical connection of the pump to power it from the assured bar of the new Emergency Power System (EPS)	Planned	2013	No	Taken	2011	No
3.1.18 CNA II: Review of procedures for extending the use of the Diesel Generators (DG) using additional fuel tanks: the maintenance and testing programs were reviewed, including verification of fuel tank level and provisions of water and lubricants.	Planned	2013	No	Taken	2011	No
3.1.19 CNA II: Analysis of the availability of external power supply lines including the interconnection with the 220 kV and 500 kV high voltage lines.	Planned	2013	No	Taken	2011	No
3.1.20 CNA II: Use the fuel of the auxiliary boiler to increase the DGs autonomy.	Planned	2014	No	Taken	2011	No
3.1.21 CNA I: Installation of passive autocatalytic recombiners (PARs).	Planned	2014	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 3 – Severe Accident Management						
3.1.22 CNA II: Connection to mobile diesel generator (MDG).	Planned	2014	No	Taken	2011	No
3.1.23 CNA I: Evaluate the need for cooling the RPV external side in conditions where the core is severe damaged and the eventual design changes that are necessary to supply cooling water (in case to be necessary it is foreseen to be implemented in 2015).	Planned	2015	No	Taken	2011	No
3.1.24 CNA II: Ventilation closure of the distribution building (UBA, switchgear building) and use of the portable purifycation equipments.	Planned	2015	No	Taken	2011	No
3.1.25 CNE: Installation of a Containment Atmosphere Filtered Release System.	Planned	2015	No	Taken	2011	No
3.1.26 CNE: Improvements to instrumentation that measure the following parameters: • PHTS Subcooling margin. • Moderator level. • Calandria Vault Water Level. • Containment pressure. • Plant Radiation measurement. • Containment hydrogen flammability. • R/B Basement water level.	Planned	2015	No	Taken	2011	No
3.1.27 CNE: Improvement in the safety system trip parameters coverage. New trip parameters will be added and some of the existing will be improved for the defence in depth of accidental situations already covered.	Planned	2015	No	Taken	2011	No
3.1.28 CNE: Improvements in the ECCS reliability: • Changes meant to guarantee the injection. • Changes meant to increase the system reliability operation. • Changes meant to avoid coolant leaks to the ECCS (containment by-pass).	Planned	2015	No	Taken	2011	No
3.1.29 CNA II: The possibility of connecting one of the three new CNA I DGs (EPS, 3.4 MW each) to CNA II is being evaluated. Having this connection available will allow the following options for residual heat removal: • Primary side: main cooling chain (RHR) including auxiliary components, and / or	Planned	2015	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 3 – Severe Accident Management						
• Secondary side with residual heat removal by steam generators (SG) connected to a startup and shutdown pump with the corresponding station relief valves. The reposition of water for maintaining the long term cooling must be analysed in this case.						
3.1.30 CNA II: Maintain current cooling towers as an alternative mode of cooling of two of the CNA II diesel.	Planned	2015	No	Taken	2011	No
3.1.31 CNA II: Disconnect unnecessary electrical loads to increase battery life.	Planned	2015	No	Taken	2011	No
3.1.32 CNE: Installation of filtered vent system from containment atmosphere.	Planned	2015	No	Taken	2011	No

Topic 4 – National Organizations

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 4 – National Organizations						
4.1 ARN delivered tailored-made courses for the Security Forces and the Armed Forces, which have specific roles in emergency response activities.				Taken	2011	Yes
4.2 Agreement between the ARN and the Joint Chief of Staff (Estado Mayor Conjunto); National Bureau of Civil Defence of the Ministry of the Interior (Dirección Nacional de Protección Civil) and Ministry of Health of the Buenos Aires city (including all its hospitals), for the training, preparation and intervention during a nuclear emergency.				Ongoing	2012 2nd half	No
4.3 CSN Extraordinary Meeting.				Ongoing	2012 2nd half	No

Topic 5 – Emergency Preparedness and Response

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 5 – Emergency Preparedness and Response						
5.1 To facilitate an efficient response in accidental situations it is necessary to have real time information. That was done with the implementation of an environmental monitor net around the nuclear power plants which consist of mobile and fix radiological and meteorological stations.	Taken	2011	Yes	Taken	2011	Yes
5.2 CNA I – CNE: It was required to the nuclear power plants to perform the revision of their reactor source term to actualize the calculated scenarios implementation with radiological consequence for all types of severe accidents.	Ongoing	2012 2^{nd} half	No	Taken	2011	No
5.3 CNA I – CNE: It was demanded to nuclear power plants the construction of a municipal emergency control centre far from the planning zone of the nuclear emergency (more than 10 km). Those centres must have infrastructure, instrumentation and communication systems according to what was learnt in Fukushima accident.	Ongoing	2012 2^{nd} half	No	Taken	2011	No
5.4 Review of the activities related to preparedness for post-accidental management (off-site), including the corresponding proposal of improvements and modifications. (RQ-NASA-038).	Taken	2012.	Yes	Taken	2011	No
5.4.1 CNE: Water reposition to the spent fuel elements pool by means of a connection from outside the pool building (installation of a 4" pipe with an isolation valve and a connection with the fire fighting system hose).	Ongoing	2012 2^{nd} half	No	Taken	2011	No
5.4.2 CNA I: Remote measurement of the dose rate in a 10 km radius circle, in every direction. The system has 13 measure stations to evaluate the field conditions. These send online information to the Internal Emergency Control Centre (CICE) as well as to the External Emergency Control Centre (CECE).	Ongoing	2012 2^{nd} half	No	Taken	2011	No
5.4.3 CNA I: A foam and air system against fire (cannon for long distance action) will be installed in a mobile foundation to mitigate fires in the heliport zone and can eventually contribute to do washing and the retention radioactive particles.	Ongoing	2012 2^{nd} half	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 5 – Emergency Preparedness and Response						
5.4.4 CNA I: Revision of current procedures applicable to the strategy of emergency management, including fire fight and recuperation actions.	Ongoing	2012 2nd half	No	Taken	2011	No
5.4.5 CNE: Fix and mobile satellite phones will be available to be used as a back-up of the existing communication systems.	Ongoing	2012 2nd half	No	Taken	2011	No
5.4.6 CNA I: Installation of a mobile diesel generator equipment (MDG) 680 KVA.	Planned	2013	No	Taken	2011	No
5.4.7 CNE: The re-evaluation of all the actual procedures applicable in the emergency management strategy EOPs (not finished yet) is taking place including the evaluation of the need to generate other procedures.	Ongoing	2013	No	Taken	2011	No
5.4.8 CNA I: Installation of an independent groundwater pump to feed the storage pools of irradiated fuel elements. Pump electrical supply: from the secured bus bar (through a new electric panel) and from the MDG (easy manual connection). Implementation of an alternative system to feed the fuel storage pools that allows the monitoring of relevant parameters from outside the pool building.	Planned	2013	No	Taken	2011	No
5.4.9 CNA I: Installation of an additional (fourth) pump to the river Water Cooling Ensured System (UK) capable of withstanding the maximum level given by the breaking of the Yaciretá dam upstream.	Ongoing	2013	No	Taken	2011	No
5.4.10 CNA I: Modification to make up water inventory to the SGs using groundwater, through the SSC when losing the feed water tank, the residual heat removal and the secured system of water injection to the SG. That system tank inventory can be replaced in those cases where the SSC integrity has not been affected. To do that it is foreseen to inject water from the storage pools UA00B03/B04 to the SGs by means of the pumps UA10D20 /D21.The groundwater will be pumped using one of the pumps that supply potable water. In case of a SBO coincident with SSC diesel generators unavailability, the proposal considers the possibility to feed the involved components by means of an external generator.	Planned	2013	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 5 – Emergency Preparedness and Response						
5.4.11 CNA II: Modify the PS-101 procedure “Conformation and function of the control of emergency internal centre – CICE” to include plant staff.	Planned	2013	No	Taken	2011	No
5.4.12 CNE: Due to a limited capacity of water movement, it was decided to implement two mobile cisterns, 25.000 litres minimum capacity each one to facilitate a continuous feed water to the spent fuel storage pool. Besides, it is foreseen a mobile pump to supply water to the cisterns.	Planned	2013	No	Taken	2011	No
5.4.13 CNE: For attend the needs to perform different works outside the Service Building, it was foreseen to incorporate: • Three 5.5 kW portable electric generators with light columns. • Three 6 kW mobile diesel generators that are located in trailers with its mobile light columns.	Planned	2013	No	Taken	2011	No
5.4.14 CNE. An electric panel will be installed outside the service building with connection facilities to the emergency light equipment; powered by batteries (Class I).	Planned	2014	No	Taken	2011	No
5.4.15 CNA I – CNE: From the experience of Fukushima accident it is clear the need to do emergency exercises extended in time. These exercises are being planned. The aim of them is to have operative emergency centres during a long term to improve the emergency direction transfer mechanism, the resources sustainability, the technical support provision, the communications and trustworthy data acquisition.	Ongoing	2014	No	Taken	2011	No
5.4.16 CNA I - CNA II: Relocation the Emergency Control Internal Centre (CICE) in the new unique lodge for the site will have secure energy supply, a system of positive pressure ventilation and a filter chain to allow extended stays of the emergency conduction personnel inside this building.	Planned	2015	No	Taken	2011	No
5.4.17 CNA I: A new building will be constructed to house a meeting point with a capacity of 1300 people; it will have secure energy supply, a system of positive pressure ventilation and a filter chain.	Planned	2015	No	Taken	2011	No
5.4.18 CNE: Availability of a 550 kVA mobile diesel generator (MDG) for electrical supply to essential loads during a SBO condition.	Planned	2015	No	Taken	2011	No

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 5 – Emergency Preparedness and Response						
5.4.19 CNE: The construction of small trenches with plastic membranes that make them waterproof to retain the water used in decontamination tasks.	Planned	2015	No	Taken	2011	No
5.4.20 CNE: Installation of an air recirculation filter system in the secondary control room (SCR) that allows its habitability when a smoke or radioactive emission to the atmosphere occurs during the nuclear accident. The SCS ventilation system will be improved installing a 100% air recirculation system by means of active carbon filters and absolute filters. The systems will be powered from the secure electric supply system (EWS).	Planned	2016	No	Taken	2011	No
5.4.21 CNE: The Emergency Internal Control Centre (CICE) building will be modify to improve seismic aspects and a ventilation system with HEPA / active carbon filters and emergency electric energy will be installed. In addition to these, the communication system will be improved to stand severe accident conditions for long term.	Planned	2016	No	Taken	2011	No

Topic 6 – International Cooperation

Activity	Activities by the Operator			Activities by the Regulator		
	Activity	Schedule or milestones	Results available	Activity	Schedule or milestones	Conclusion Available
Topic 6 – International Cooperation						
6.1 Request to Argentine NPPs to perform a Resistance Assessment (stress tests) agreed by the FORO, Ibero-American Forum of Radiological and Nuclear Regulatory Agencies (RQ-NASA-038).	Taken	2012	Yes	Taken	2011	Yes
6.2 Practical emergency plan exercises with the participation of neighbouring countries	Taken	2011	Yes	Taken	2011	Yes
6.3 There will be a WANO Peer review for CNA I as well as a technical support mission on operational focus in CNE. Also, it is expected that NA-SA experts participate in eight WANO peer reviews, along with several technical support missions and workshops.	Ongoing	2012 2^{nd} half	No			
6.4 Harmonization process of Argentinean standards against IAEA's safety documents.				Ongoing	2013	No